A Path Forward for the LMFBR

Revision 6

D. Clark Gibbs

In the mid to later part of the 20th Century, the Liquid Metal Fast Breeder Reactor was envisioned by many as the technology that could supply all the nation's energy needs for the foreseeable future – thousands of years if necessary. The Liquid Metal Fast Breeder Reactor failed in the United States in 1983 because it was considered too expensive. This perception was based on the preliminary design of a demonstration plant, the Clinch River Breeder Reactor Plant that was then in the latter stage of construction permit licensing. The purpose of this monograph is to describe how that happened and to propose a different method that could lead to a more favorable outcome. The design and institutional approach described is one of many that could be devised. It is not intended to be a blueprint. It is only intended to show that it is possible to capitalize on the inherent features of liquid metal and breeder reactor technology in such a way that economic outcomes are achievable. There are undoubtedly many other such approaches.

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Abstract

Advantage can be taken of the inherent characteristics of sodium cooled reactors to achieve major reductions of the plant capital cost making the LMFBR highly competitive with existing designed LWRs. First, it is proposed to adopt a reactor core design that capitalizes on the breeding principle so as to allow continuous full power operation without refueling for periods of up to ten years. Doing so enables the adoption of a much lower cost (but slower) refueling system, a shortened reactor vessel, and simplifies the containment system. Second, the cylindrical containment is abandoned in favor of a rectilinear containment with no requirement for a single elevation basemat so that space utilization is improved. A related measure is to replace the operating floor concept common to LMFBRs designed to date with a refueling cell that connects the reactor and the adjacent fuel storage. This reduces containment volume and the need for much of the reactor vessel head shielding. Third, the elimination of the elevated loop primary system piping concept similarly reduces containment volume. Fourth, a major reduction in core pressure drop enables the elimination of hydraulic hold-down of core assemblies, probably eliminates the need to consider control rod ejection accidents in the design basis, and facilitates adoption of more compact and trouble free EM pumps for both the primary and intermediate circuits. Fifth, the adoption of naturally circulating Decay Heat Removal Systems permits reduction of the number of primary loops to two, eliminates the requirement for the Intermediate Heat Transport System and Steam Generating System to be safety related, and reduces 1E loads dramatically, likely eliminating the requirement for emergency diesel generators. Finally, auxiliary systems are carefully reviewed for unnecessary features that have a tendency to creep undetected into plant designs.

Included is a potential institutional plan for the path forward.

1 Prelude

The country which first develops the breeder reactor will have a great competitive advantage in atomic energy. Those words spoken by Enrico Fermi in 1945 may appear dated and no longer meaningful in 2024. The many difficulties experienced by the promoters of nuclear power with vigorous coordinated opposition from environmental groups, excesses and misplaced judgment on the part of regulators, horrible decisions by politicians and policy makers, selective blunders by utility industry management, poor performance by certain architect engineers/constructors, failure to implement an acceptable solution to waste disposal, and three accidents worldwide at nuclear power plants in the intervening years have definitely taken the bloom off the rose of what was once a promising technology. Today, it would appear that the future of electric energy production rests with renewables, particularly with wind turbines and solar plants while filling in gaps with combined cycle gas plants, and the continuing use of coal and nuclear plants that were constructed in the 1960s and 1970s until the total fleet of renewable power plants is put in place.

This forecast may or may not reflect the reality of the renewables, depending on whether acceptable and economic energy storage can be developed and whether a reliable electricity supply can be maintained through sustained periods of darkness and calm. In 2020, renewables accounted for 20% of U.S. generation and 12% of U.S. consumption. The disparity between generation and consumption is a reflection on the availability of the renewables. If hydro and geothermal are removed from the picture, the difference probably becomes greater. Rather than moving toward complete reliance on renewables, it is more likely that the U.S. energy supply will continue to be diversified in the interest of both system reliability and economics.

In 2004, natural gas was extracted for the first time from the Marcellus shale, opening floodgates of supply, particularly to electric utility companies deploying combined cycle plants. The combined cycle is somewhat unique among power generating options in that it achieves thermodynamic efficiencies approaching 60% and contributes up to 70% less CO_2 to the atmosphere than an equivalently sized coal fired plant. At about $650/KWe capital cost, installed in very manageable 400 MWe increments, and with fuel cycle costs as low as 3¢/KW-hr., what's not to like? The electric power industry currently represents 40% of the national consumption of about 33 trillion ft^3 of natural gas per year. The proven U.S. reserves are about 580 trillion ft^3. So each time an electric company puts a combined cycle plant in service that is designed to operate for 40 years, it is betting that the proven reserves will increase by a factor of about 3 from their current values over the life of the plant. That doesn't count the new combined cycle plants yet to be placed into service, which will deplete reserves faster. There will inevitably be a point in time when U.S. natural gas is in short supply, prices will increase dramatically, and electric utility companies will be compelled to curtail usage. Will those companies have another generating source they can put into service to replace their combined cycle plants? This nightmare has probably visited several electric utility company executives.

It appears likely that electric utility companies will soon be reassessing their positions on nuclear power, if they aren't already. The use of nuclear power is easily justified by a resource argument and if the plants are properly designed, nuclear power can be justified by its superior economics and reliability. Nuclear power includes not just Light Water Reactors (LWRs) but also Liquid Metal Fast Breeder Reactors (LMFBRs). Fermi's assertion is most likely as correct today as it was when he made it, as long as his word "develops" is assumed to imply that the development leads to a plant design that is economic and therefore actually deployed.

This monograph is intended to describe the technology and outline the history of LMFBRs for audiences generally familiar with LWRs. Most particularly, it is intended to identify a pathway for deployment of LMFBRs through a conceptual design approach which draws upon collective experience, recent innovation, and re-visitation of approaches previously considered that have been long lying dormant. The paper is also intended to suggest areas where further creative thought could be directed to improve the economic and operational performance of the concept to make it highly competitive with LWRs.

The climax of development of the breeder reactor in the United States was the Clinch River Breeder Reactor Plant (CRBRP) project, which was terminated by congressional action in 1983 primarily on economic grounds. The plant, as designed, was perceived to be too expensive and, unfortunately, it probably was. This raises the question of whether breeder reactors are inherently expensive or was the CRBRP design rendered in a fashion which led to excessive costs. Although engineers perform the actual design work, much of the way design develops is a direct result of the decisions of policy makers and upper level management. Ultimately, it is managers who are called upon to make many of the key decisions. Many decisions were made within a temporal context that was fleeting. At the time of the Clinch River Breeder Reactor Plant (CRBRP) project, the emphasis was reliability, not economics. When there was a choice between conservatism and cost reduction, conservatism always won. Economics was left for "commercialization", i.e. some plant in the future. There is a more extensive treatment of the historical underpinnings in Appendix 7.

Although conservatism contributed to the plant's demise, there was considerable useful work performed on the project, much of which continues to be available on the internet. The accessibility of CRBRP Preliminary Safety Analysis Report (PSAR) on the internet results in the availability of detailed CRBRP engineering descriptions and performance data and creates the opportunity to use CRBRP as a point of departure for many of the discussions contained herein. All references to CRBRP data that are drawn from the PSAR are not separately identified by reference in the text. There are some references to CRBRP data that are drawn from sources other than the PSAR and those have been footnoted.

The next LMFBR designed in the U.S. must be designed with the minimization of its capital cost in the forefront and a focus on plant operability that will be appealing to prospective electric utility company owners and operators. It would be desirable for it to make use of the breeding concept that is unique with the LMFBR in a fashion that dramatically improves the plant's operability when compared to LWRs. The LMFBR has so many inherent advantages over the LWR there is good reason to expect that its capital cost would be equal to, less than, or even significantly less than a comparably sized LWR. Existing designs of LMFBRs fail to meet this objective. If the LMFBR has any possibility of attracting interest from skeptical potential users, it is essential that this objective be attained if deployment is to be accomplished within any reasonable time frame.

The purpose here is to propose a "design approach" which could possibly then be used as a basis for more serious design activity. There needs to be a context for this "design approach", viz. a plant size and key parameters. Since the Superphénix reactor, completed in France near Lyon, is the largest LMFBR built and operated to date worldwide and comes perhaps the closest to realizing commercial application, its key parameters will be used – 3000 megawatts thermal (MWth) and superheated steam at a pressure of 2400 psig. Certain features of the Superphénix design proved to be a qualified success – notably the steam generators – and those will be carried forward into this proposed plant approach. As will be seen, these boundary conditions will prove to be sufficient for the purposes of this discussion. Meanwhile, there is more ground to be plowed with preliminaries.

2 Source of the idea of the breeder reactor and a brief history

Naturally occurring uranium is composed of two isotopes[1], U^{235} and U^{238}. U^{238}, being the longer lived isotope comprises 99.28% of the total in naturally occurring uranium. In current generation Light Water Reactors (LWRs) the U^{235} is the fissile isotope that sustains the chain reaction. While there is U^{238} fission, it is relatively less common in a LWR. Some excess neutrons are absorbed in U^{238} converting it to U^{239} which then beta decays first to Np^{239} then Pu^{239}. The Pu^{239} is fissile, and can continue the reactor's chain reaction. In LWRs, the amount of Pu^{239} produced per fission is less than the amount of U^{235} consumed, so only a small fraction of the available U^{238} is used as a fuel. This fraction can be improved with fuel reprocessing, but the best that can be achieved with LWRs is about 3-4%. Another feature important to LWRs is that the uranium must be enriched in the U^{235} isotope for the chain reaction to be sustainable.[2] In typical commercial LWRs, the uranium enrichment typically runs in the range of 2-5%. Enrichment is accomplished in uranium enrichment plants that are committed for this purpose.

Breeder reactors are called such because they produce more fuel than they consume. They typically use Pu^{239} as the fissile isotope and U^{238} as the "fertile" isotope. Since less Pu^{239} is consumed in power production than U^{238} that is converted to Pu^{239}, the Pu^{239} inventory gradually increases in the reactor core during operation of the plant. The "breeding ratio" is the ratio of fertile atom conversion to fissile atom consumption and is typically around 1.2-1.3 in a well designed reactor. The "doubling time" is the number of years of full power operation necessary to double the inventory of the fissile isotope, and is approximately 10-20 years. As a result of breeding, breeder reactors can effectively make use of at least 60% of the U^{238} in natural uranium rather than the 3-4% that is used in LWRs.[3] Thus about 20 times more power can be extracted per pound of natural uranium in a breeder reactor than can be extracted in a LWR. The fact that each pound of uranium becomes more valuable means that uranium bearing ores that cannot be economically utilized in LWRs become a resource for breeder reactors. There are vast resources of low grade uranium ore available in the U.S. The same Marcellus shale that is currently being exploited for natural gas production contains about 25 ppm of uranium,[4] far below the concentration that would be economic for LWRs, but potentially useful for breeder reactors. Using the domestic shale, there is sufficient uranium in the U.S. to power fleets of breeder reactors supplying basically the entire nation's energy needs for thousands of years. This uranium resource picture is further elaborated upon in Appendix 8.

Demonstration is not a problem for the LMFBR. The world's first LMFBR, the Experimental Breeder Reactor 1 (EBR-1) in Idaho, was also among the first nuclear plants of any kind to produce usable electric power on December 20, 1951. EBR-1 was followed by EBR-2, the Southwest Experimental Fast Oxide Reactor (SEFOR), the Enrico Fermi Atomic Power Plant (Fermi-1), and the Fast Flux Test Facility (FFTF) in the US. All preliminary design, extensive detailed design, licensing through to the award of a Limited Work Authorization, and major component fabrication was completed on the Clinch River Breeder Reactor Plant (CRBRP) prior to its termination in 1983. In addition, four sodium-cooled reactors that were not LMFBRs were built and operated in the U.S., the Sodium Reactor Experiment (SRE) in southern California, the

1There is a small fraction (0.006%) of U^{234} in natural uranium.

2It is possible to design reactors that can operate with natural uranium as the fuel, but they must be moderated with an isotope that has a low absorption cross section such as carbon or heavy water. The Canadian reactors which are both moderated and cooled with heavy water are examples of this approach.

3Some fraction of U^{238} is inevitably lost in reprocessing and fuel fabrication.

4Bank, Tracy L., *Trace Metal Chemistry and Mobility in the Marcellus Shale*, University of Buffalo

Hallam Nuclear Generating Station in Nebraska, and two intermediate spectrum naval reactors, the S1G prototype built in Schenectady and the S2G installed in the Sea Wolf submarine.

Abroad there were two LMFBRs built in the U.K., three in France, two in Germany, two in Japan, and six in Russia. The range of sizes includes one, the Superphénix in France that was a full commercial sized plant at over 1200 MWe. The Russians completed a 600 MWe plant in 1980 that continues to operate and an 880 MWe plant in 2016 for approximately $2B that is also operating. The design of a follow-on 1200 MWe plant is underway in Russia. Extensive development of LMFBR technology has occurred at the Idaho National Engineering Laboratory and the Liquid Metal Engineering Center in the U.S. as well as at similar centers in the U.K., France, Russia, and Japan.

Since a key advantage of breeder reactors is that they produce more fissionable material than they consume, it is possible to design them so that they require only infrequent refueling. This is a capability that has not been well capitalized on by worldwide breeder reactor development to date and will be treated extensively in this paper. It is a capability that may be of considerable interest to utility company users and it opens a door for alternative design approaches.

Currently LMFBR development in the U.S., France, the U.K., and Japan is all but halted. There is continuing activity in Russia, India, Korea and China, but the energetic worldwide development of the technology so much in evidence in the 60s and 70s has all but ceased.

3 *The unique properties of sodium cooled reactors*

LWRs are considered "thermal" spectrum reactors because the neutrons that are involved in the chain reaction are slowed down by collisions with a "moderator", typically the hydrogen in the water coolant so that they (the neutrons) are in thermal equilibrium with their surroundings.[5] Since the hydrogen nucleus has an atomic number of one, it has a mass about equal to a neutron, and as a result neutrons can lose a great deal of energy with each hydrogen collision. This slowing down of neutrons is desirable since U^{235} has a very high cross section for fission with thermal neutrons.[6] Thus the chain reaction can be made to occur with a relatively low concentration of the fissile material, U^{235}. However, fission with thermal neutrons comes at a price. The number of secondary neutrons produced per fission is lower with thermal neutrons than is the case with fission resulting from fast neutrons. While there are enough neutrons produced per thermal fission in a thermal reactor to sustain the chain reaction, there are not enough left over for breeding. While some transformation of U^{238} occurs in a thermal reactor, the number of transformations per fission is less than one.[7]

5Some thermal spectrum reactors have used carbon as a moderator. Beryllium is also a suitable moderator.
6The "cross section" of a nucleus is a measure of its rate of reaction with neutrons.
7The other "price" that is paid is much greater absorption by fission products, structural materials, and the coolant during slowing down and while at thermal energies.

If it is desired to breed, viz. to produce more transformations of U^{238} than there are fissions, a coolant must be used that does not slow down the neutrons appreciably on the occasions of collisions. Sodium with an atomic number of 23 meets this requirement. The law of conservation of momentum requires that elastic collisions between neutrons and sodium atoms at rest result in just a 2% neutron energy loss on average compared with over 60% neutron energy loss per collision that occurs with hydrogen atoms.

At ambient conditions, sodium is a soft metal that is chemically very reactive, particularly with water or oxygen. In its elemental form, it does not exist in nature, although its compounds are common. In its elemental form, it has a melting point of 208° F. and a boiling point of 1616° F. which are in the operating range of a thermal steam plant. If the reactor is cooled with sodium[8] and the heat is then transferred to water generating steam, the reactor can be used as a heat source for an electric power generating station.

Sodium has certain features that make it very desirable for use as a reactor coolant.

- Since its boiling point is 1616° F. the reactor coolant system can be operated at low pressure greatly simplifying the design of the reactor vessel, the primary coolant system components, and piping. The wall thickness of sodium piping, even at sizes up to 48 in. in diameter, will typically be no greater than ½ in.
- The reactor can be operated at much higher temperatures than water-cooled reactors, greatly improving thermodynamic efficiency. Reactor outlet temperatures in the 1000-1100°F range are readily obtainable in contrast to maximum reactor outlet temperatures of around 600°F for water-cooled reactors.
- Superheated steam can be produced from a sodium cooled reactor plant in contrast to LWRs which produce only saturated steam.[9] Superheat of approximately 250-300°F is readily achievable. Because of the steam conditions that are obtainable with sodium cooled reactors, they can operate with a thermodynamic efficiency in excess of 40% -- much better than LWRs can attain. Since much of a power plant's size is determined by its thermal power and greater efficiency translates into lower thermal power for a given electric output, greater efficiency is a positive cost driver. Higher efficiency also means that less fuel is consumed for a given quantity of electricity generated and less heat must be rejected to the environment.
- A coolant system breech does not result in the immediate pressurization of the containment system which encloses the reactor coolant system as is the case for water-cooled reactors. As a result, containment systems can be designed to much lower pressures than are typically required in water-cooled reactors. Also, the containment size is dictated by the physical size of the primary system and is not influenced by

8Other liquid metals may be used as a coolant. The EBR-1 used a eutectic mixture of sodium and potassium commonly referred to as NaK as the primary system coolant. NaK has the important advantage of a lower melting point (70°F) than sodium, but it was found to be much more chemically reactive and hazardous to handle. In addition, K^{42}, formed by neutron capture from K^{41} is a strong gamma emitter with a 12.4 hr. half life. The Russians have used a mixture of lead and bismuth as a coolant in some reactor applications. Mercury has been used on at least one reactor.

9The B&W design LWR NSSS uses once through steam generators and produces steam with about 35°F superheat. While this modest superheat contributes somewhat to the performance of the B&W designs, it is small in comparison to that which can be achieved in a LMFBR.

considerations of loss of coolant accidents. Novel containment design concepts such as ice condensers need not be considered.

- There is no need for post accident emergency containment cooling systems such as those typically found on LWRs to cope with flashing steam from a coolant leak.
- The sodium coolant is chemically benign with respect to the metal systems that contain it. At reasonably high purities, sodium is non-corrosive of steels or other metals likely to form any sodium system boundary. This is an important feature since corrosion is a continuing problem in power plant design and must be accounted for in the design and operation of the plants. This feature also makes sodium systems attractive if very long plant design lives are an objective.
- Since the coolant is electrically conductive, electromagnetic (EM) pumps with no moving parts can be used. Such pumps have been used at small scales in sodium auxiliary systems since the early days of LMFBR development and have more recently been demonstrated at large scale for use as the main pumps in the primary coolant system. EM pumps have no penetrations into the sodium system reducing the possibility of coolant leakage and eliminating rotating seals. Moreover, the same principle can be used in reverse for flow meters, further reducing the number of coolant penetrations.
- Coolant boundary breech can be accommodated by incorporation of guard vessels into the design. There is therefore no need for emergency injection systems as are required in LWRs.
- There is no deposition of corrosion products on reactor core components, eliminating the problem of crud[10] deposits and crud bursts in the primary system.
- The steam generators are located outside containment improving their accessibility and maintainability.
- The limitations on scaling up the size of the reactor plant are much less restrictive when sodium is used as the coolant than would be the case for pressurized systems. In 1966, Argonne National Laboratory performed a feasibility study of a 10,000 MWth loop-type LMFBR plant with a net electrical output of 3880 MWe.[11] There were no obvious technical obstacles. The scale-up of pressurized systems to this size is likely to encounter limitations on the wall thickness of the pressurized vessels such as the reactor vessel or the containment vessel.
- Sodium is light in weight having a specific gravity around 0.9. At temperatures of interest for reactor coolant applications, its viscosity is comparable to water. It therefore is relatively easy to pump and does not weigh down structural components the way heavier liquid metals e.g. lead/bismuth or mercury would. Unlike mercury, it does not form amalgams with other metals.
- Sodium is an excellent conductor of heat. Its thermal conductivity is approximately 30 Btu/hr-ft-°F vs. 0.3 Btu/hr-ft-°F for water, about 100 times greater.
- The coefficient of thermal expansion of sodium at operating temperatures is about 0.16×10^{-3} per °F which compares with about 0.9×10^{-3} per °F for water in the 400-500 °F

10Crud is an acronym for "Chalk River Undetermined Deposits", Chalk River being a Canadian plant from the 1950s where these deposits were first observed. They are now known to be corrosion products which tend to selectively deposit on core surfaces in LWRs. The corrosion products typically contain cobalt, an alloying metal used in stainless steel, which becomes radioactive while residing on core surfaces and later deposits on other parts of the primary system following "crud bursts".

11Koch, L. J.; Reactor Engineering Division Annual Report July 1, 1965 – June 30, 1966; ANL-7290; April 1967.

temperature range. While this coefficient is lower for sodium than for water, it is still sufficiently high for the purposes described in the next item.

- The high thermal conductivity of sodium enables it to effectively remove heat without dependence on turbulent flow. Its reasonably high thermal expansion coefficient combined with a ~250 °F temperature rise across the reactor promotes natural circulation provided the heat exchanger removing heat is sufficiently elevated above the reactor core. The resulting natural circulation is adequate for decay heat removal from core surfaces following shutdown.
- When compounded with other elements, sodium is abundant in the earth's crust and relatively inexpensive. Sodium can be manufactured to exacting purity standards.
- Sodium readily removes heat from reactor surfaces. There is no boiling on core surfaces thus any need for concern about departure from nucleate boiling (DNB) which is a major design consideration for LWRs.
- There is no need to inject hydrogen or alkalis to control pH as is required in water-cooled reactors.
- The solubility of most contaminants, mainly sodium oxide (NaO_2), in sodium decreases with decreasing temperature allowing for their removal by cold trapping. Cold trapping the coolant can maintain NaO_2 concentration below 10 ppm, which is well below the concentration that it would become corrosive to the metals forming the system boundaries.[12] The cold traps turn out to be effective in removing most of the fission product contaminants from the coolant. There is no need for filters or resin beds and no concomitant need to change out and dispose of spent filters or resins, although the internals of the cold traps will eventually require replacement, particularly if the traps become obliged to remove high levels of oxide contamination in the coolant.
- The specific heat of sodium is 0.32 at 100°C or about 1/3 that of water. The temperature rise across the reactor is typically 250°F or about five times that of a typical PWR. The reactor flow rate is therefore 3/5 that of a PWR of equivalent thermal power. Accounting for the difference in thermodynamic efficiency the primary system pumping power is less than half that of an equivalently sized PWR given equal system head losses.
- Since the Control Rod Drive Mechanisms are not pressurized, it is possible to provide features that eliminate control rod ejection accidents from the design basis (see Appendix 2D)..

For the above reasons, sodium was seriously considered for use in thermal spectrum reactors during the 1950s and 1960s. There were two naval reactors that were sodium cooled. However, sodium cooled reactors were considered inappropriate for use in submarines and their development for land based applications could not be justified as a competitor with sodium cooled fast reactors.[13]

Sodium does introduce challenges into the design.

12One exception to this is zirconium, for which oxide concentrations need to be maintained lower than 10 ppm. In plants where zirconium has been used, hot traps are installed which operate around 1200°F and use zirconium as a sacrificial material.

13The Hallam reactor, which was the last sodium cooled thermal reactor in the US, experienced difficulties with the canning material surrounding the graphite moderator blocks. Development of a solution to this problem couldn't be justified in light of the breeder option that did not need a moderator. It turned out that the contractor for Hallam had independently developed a solution to prevent that failure mode, but it was too late to save the Hallam project.

- Since sodium will freeze at 208° F., if it is desired to keep it in the liquid state during plant shutdown, piping systems and components must be separately heated. This relatively high freezing temperature has the advantage of permitting freeze seals to be easily used to isolate sodium components for maintenance or repair.
- As a consequence of the relatively high temperature rise across the reactor, transient behavior in sodium cooled systems can be more challenging than in water-cooled plants unless design features are present ameliorating the possible effects.
- In the reactor coolant system, the radioactive isotopes Na^{22} and Na^{24} are formed from interactions between the reactor neutrons and the coolant.[14] Na^{24} has a half life of 15 hours and Na^{22} has a half life of 2.6 years.[15] As a result, the coolant becomes very radioactive during operation. This is a contrast with water, which is not radioactive shortly after shutdown[16] other than through a slow buildup of tritium, which is relatively benign. Although contaminants in the water do become radioactive, they generally do not pose a serious problem – at least not as serious as that posed by Na^{24}, in particular. To put this in perspective, LWRs with no fuel element failures may have a coolant activity of around 10^{-3} $\mu Ci/cm^3$ attributable to some fission product recoil that penetrates the cladding and activated corrosion products, mainly Co^{60}. This number might increase an order of magnitude or two following a large crud burst or a fuel element failure. In contrast, a LMFBR can experience coolant activity as high as high as 50,000 $\mu Ci/cm^3$ attributable to Na^{24}. (Although the coolant is highly radioactive, the heat it produces is less than 0.1% of full reactor power.) Sodium cooled thermal reactors have even higher levels of activation due to the greater absorption cross section of sodium at thermal energies. The SRE was designed for primary coolant activity of twice this number, and actual experience disclosed the coolant activity to be 0.3 Ci/cm^3[17]. Na^{22} activity is much lower, but still somewhat high by LWR standards at around 0.5-1.0 $\mu Ci/cm^3$. As a result, sodium-cooled reactors generally that have been designed and built to date have a shielded operating floor inside containment with sodium containing systems located below the floor. The spaces containing primary coolant sodium generally cannot be accessed by personnel until after adequate Na^{24} decay has occurred, typically 10 days. It is primarily for this reason sodium cooled reactors make poor candidates for marine applications. Were some event to occur in the reactor compartment of a submarine that required personnel entry, it would be unsatisfactory to be obliged to wait 10 days before someone could safely enter. Because of the coolant activation issue, two reactors in separate compartments would have been needed to achieve acceptable reliability for naval applications including submarines. Water reactors don't have this problem.
- The fast fission cross section for fissionable isotopes is two orders of magnitude lower than equivalent thermal fission cross sections resulting in the neutron flux being nearly two orders of magnitude greater in a fast reactor in comparison to a thermal reactor. The higher neutron flux can create problems with the structural members of the Reactor Vessel. Stainless steel is susceptible to radiation induced swelling at levels above about 4×10^{22} n/cm^2 ($E > 0.1$ MEV). Such swelling would introduce dimensional anomalies

14An (n,γ) reaction in the case of Na^{24} and an (n,2n) reaction in the case of Na^{22}.

15The Na^{24} decays to Mg^{24} which is one of the metals that is soluble in sodium. The Na^{22} decays to Ne^{22} which will collect in the cover gas.

16While operating, N^{16} is produced in LWRs, which causes the coolant to be highly radioactive during operation, but its 7.3 second half-life renders it inconsequential five minutes after shutdown.

17 R.E. Durand, *Soduim Reactor Operating Experience*, Chemical Engineering Progress, Vol. 57, No. 3, Mar. 1961

and compromise the material strength of the structures. Structures near the reactor core need to be adequately protected against such fluence levels over their lifetimes by adequate shielding.

- Operating temperatures in the range of 1000°F. require that thermal creep be evaluated as a part of structural design.
- Cesium is very near one of the peaks in the fission yield curve and is also an alkali metal like sodium. While most fission product contamination is removed by the cold traps, cesium is not and will gradually build up in the primary coolant unless some specific means is provided for its removal. At the end of operation of the Russian BN-350 reactor, Cs^{137} contamination was 6-7 μCi/cm^3. Cs^{137} has a half life of 37 years so its build up in the coolant creates an operational and maintenance headache which also serves to discourage any prolonged operation with significant cladding breaches. Potential solutions for this problem, including removal of the Cesium from the coolant, will be addressed later.
- The coolant cannot be exposed to air, which complicates operations, such as refueling, when the primary system must be opened.
- It is necessary to transfer the primary system heat to a water system for power generation. Since water and sodium react violently if they come in contact with one another, it is necessary to take measures to prevent any leakage, even in minute quantities, across the sodium water boundary. When sodium reacts with water, hydrogen is produced along with NaOH. A means must be provided for dealing with potential sodium water reaction products in the steam generators.
- There is no convenient way to temporarily poison the coolant as is routinely done in PWRs, which add a boron compound to the coolant for reactivity control, then reduce its concentration as the core burns down.
- Sodium will remove any oxide layer on the metals that contain it. As a result, if metals are in contact, such as in a valve, there is a tendency for the metals to self weld. The tendency for self welding increases with the contact pressure between the surfaces and may be greater with increasing sodium temperature.
- Certain metals including copper, magnesium, tin, lead, and antimony are soluble in sodium and must be avoided as alloying materials in any metals used for the sodium containment boundary.

The designer's challenge is to accentuate the positives and if it is not possible to eliminate the negatives, at least accommodate them in a fashion that minimizes their economic impact on the plant.

There are additional neutronics considerations needing acknowledgment in the design by virtue of the neutron spectrum in fast reactors. These matters are treated in Appendix 1.

4 Typical design features of LMFBRs

All LMFBRs built to date have invoked either the pool or the loop design. The trade-off between these two concepts is the subject of Appendix 5. Where the word “loop” is used in the remainder of this section, in most cases it can be interpreted to apply to either a pool or a loop plant.

While there have been some exceptions, most sodium cooled reactors built to date have had three or four loops to transfer the core heat out of the reactor, each of which includes an intermediate heat exchanger (IHX) and a pump. This is usually done for the benefit of redundancy, to limit the size of the heat exchange components, and to adapt to the geometry of a circular containment structure. During the early years of LMFBR development, consideration was given to continued operation at reduced power with one loop disabled. Some of the early plants were fitted with loop isolation valves in further pursuit of this objective. Later designs such as CRBRP abandoned disabled loop operation as an objective and had no loop isolation valves, but retained at least three loops to permit redundant decay heat removal with one loop inoperable. The number of loops in the plant is a key design decision, typically made early in the conceptual design stage.

As opposed to water reactors, no method has yet been found to directly extract substantial quantities of useful energy from the sodium working fluid in LMFBRs.[18] It is necessary to first exchange the heat from the sodium to a water system in which steam is generated that then drives turbines. The water/steam system is pressurized typically to the 1500-2500 psi range in the interest of improving thermodynamic efficiency. This necessitates heat exchangers, viz. steam generators, to extract the heat from the sodium and transfer it to the water/steam system. Since the sodium is at near atmospheric pressure, leakage in the steam generators will result in water intrusion into the sodium system, which must be protected against. The steam generating system may involve separate recirculating evaporators with a steam drum and separate superheaters as was selected for the CRBRP design or once through units in which evaporation and superheating occur in the same unit as was selected for Superphénix.

One of the most prominent design features of LMFBRs setting them apart from LWRs is the presence of intermediate loops between the primary loops and the steam generating system. These loops are installed to prevent water leakage from the steam generators from reaching the reactor core where it would produce an excursion as a result of moderating the neutron spectrum. While some spectral softening introduces negative reactivity, if the neutron energy were to be reduced to the point that there became a large population of neutrons near thermal energies, significant positive reactivity would be introduced into the reactor due to the high fission cross section of the fissile isotopes at thermal energies.

Since the sodium passing through the intermediate loops does not enter the reactor vessel, it is not radioactive unless it becomes contaminated by the primary system sodium. To prevent such contamination, the pressure in the intermediate loops is maintained higher than the primary system sodium at the intermediate heat exchangers where heat transfer from the primary to the intermediate sodium system occurs[19]. For the case of pool type plants, intermediate sodium does

18Sodium boilers and turbines have been considered for space applications. See Sodium-NaK Engineering Handbook, Vol. 5, O.J. Foust, editor

19Small amounts of tritium pass through the walls of the IHX into the intermediate system.

enter the pool on the shell side of the IHXs, so it is necessary to provide sufficient shielding between the reactor core and the IHXs to prevent intermediate sodium activation.

The more recent designs do not provide for valves in the primary sodium loops, and in some cases there are no valves in the intermediate loops. Primary loop isolation valves were incorporated in the FFTF but were omitted from the CRBRP design. Both FFTF and the CRBRP original design had isolation valves in the intermediate loops, which were later deleted from CRBRP as the design advanced. For the case of CRBRP, the intermediate loop isolation valves were originally provided to enable isolation of the steam generating system in the event of a water leak into the IHTS – a feature which later was deemed unnecessary. If it is desired to disable an intermediate loop, it can be drained to a tank committed for that purpose located outside containment. Similarly, a single primary loop can be drained to an in-containment tank assuming there is a point between the loop components and the reactor vessel that is elevated above the reactor vessel nozzles. Under either circumstance, the reactor would be shutdown and the circulating pumps would be run at low speed or not at all, relying on natural circulation to remove decay heat from the core. The only purpose for primary loop valves would be to permit reactor operation on a reduced number of loops. Since operation of a commercial reactor with a disabled loop involves risks that are typically deemed greater than the rewards for doing so, there is no incentive for primary loop isolation valves. In pool type reactors, they aren't even feasible. CRBRP had check valves designed on the primary pump discharge to prevent reverse flow in an idle loop.

Sodium valves, where they exist, involve a freeze seal on the stem, i.e. sodium is allowed to pass up alongside the stem where it then freezes and seals the stem. When the valve is operated, the freeze seal breaks while the stem is in motion, refreezing again once stem travel ceases. When freeze seals are used on valves, it is necessary to provide an inert cover gas above the freeze seal to prevent a sodium air reaction with the frozen sodium. Above the cover gas space, there will be a conventional stem seal to isolate the cover gas. Small sodium valves can use bellows seals so long as the sodium is kept molten below the bellows.

A provision is made for a failure of a steam generator tube, which would lead to a sodium water reaction with the production of large volumes of hydrogen gas. This system, referred to as the Sodium Water Reaction Products System (SWRPS) consists of some form of relief mechanism or rupture disks on the intermediate loops, a tank into which reaction products are directed, and rapid means for isolating the water and possibly the sodium sides of the steam generator. The hydrogen produced in the reaction is further directed to a stack where it may be ignited, producing a flare. This system is discussed in Section 12.

As was mentioned earlier, the 0.32 specific heat of sodium requires that the temperature rise across the reactor be large – typically in the 250-300°F range. This large ΔT is necessary in the IHTS anyway, to accommodate feedwater that enters the steam generating system around 500°F. To maintain steam conditions constant with changes in power level, it is necessary to maintain the ΔT between the hot and cold legs so the primary and intermediate sodium flow rate must be correspondingly changed. Therefore, the primary and intermediate pumps are fitted with variable speed drives. Typically, these pumps will operate over flow ranges from 25-100%. Below that range, there may be separate low speed motors to provide for decay heat when the reactor is shut down. The low speed motors, sometimes called "pony motors" will provide about 5-10% flow, sufficient to remove decay heat immediately following reactor shutdown. Alternatively, natural circulation may supply the necessary flow for decay heat removal.

The provision of variable speed pumps in the primary and intermediate circuits results in these plants being suitable for load following. The CRBRP was expressly designed for load following at the rate of about 3% power change per minute. At reduced load, the temperature drop across the steam generators and IHXs will decrease, so if the steam conditions are held constant, the reactor outlet temperature will decrease somewhat with decreasing load despite the variable speed pumps. About 1°F change in the primary hot leg sodium temperature per percent of full power would be expected from this effect. Thus 3% per minute load following capability will result in 3°F per minute change in hot leg temperature. The rate of change in the peak centerline fuel temperature will, of course, be much greater – on the order of 30-50°F per minute. There is relatively little experience with operating fuel in a load following mode. There is no particular reason why the fuel would not be expected to perform satisfactorily under load following conditions however; the fuel system had provided more than its share of unexpected and unpleasant surprises during the time when LMFBR development was actively being pursued. At some point the fuel behavior when in the load following mode, needs to be demonstrated and better characterized. Also, the 3% per minute requirement on CRBRP could potentially be relaxed depending on the remaining power generating units in the system.

Because the primary system hot leg will typically run around 1000°F, most primary and intermediate system components are fabricated from austenitic stainless steel. Since the pressures in the primary and intermediate circuits are generally less than 150 psi, thin wall piping can be used. Primary system piping 48" in diameter may have a wall thickness of just ½ inch. In addition to reducing its cost, thin-walled piping is preferred for thermal shock considerations since stainless steel is a poor conductor of heat and with a 250°F rise across the reactor, thermal shock is a distinct possibility with some transients. In the case of the steam generators, austenitic stainless steels are avoided because of their susceptibility to stress corrosion cracking. Ferritic steels such as 2¼% chrome and 1% molybdenum (2¼ Cr/1 Mo) or 9 Cr/1 Mo are more likely to be used as the preferred material of fabrication for these units. Ferritic steels also have the advantage of being better conductors of heat than their austenitic counterparts, a desirable feature for heat exchanger tubing. The disadvantage of these ferritic steels is they cannot be operated at temperatures as high as the austenitic steels. On CRBRP, the design superheated steam outlet temperature was 900°F., which was acceptable for the 2¼ Cr/1 Mo used. Temperatures in the 950°F range would probably require 9 Cr/1 Mo.

All major primary system components are surrounded by guard vessels as a safeguard against primary leaks. A three loop plant would therefore have seven guard vessels, one for the reactor, one for each primary pump and one for each IHX. The guard vessels are sealed at the top to the vessel they are intended to protect so that a leak in one of the components won't siphon the interconnecting piping, which is elevated above the guard vessels. If there is a leak in elevated piping, it will affect that loop, but not the others. It is for this reason that loop plants must have at least two loops, unless some other feature, such as double walled piping, is incorporated to deal with pipe breaks. Since the guard vessels will be exposed to high temperatures only if they are performing their safety function, they can be fabricated from less expensive ferritic steel.

The reactor, all sodium pumps, and all sodium tanks are filled to a specified level with the volume above that level being occupied by a cover gas, typically argon. Argon is used because it is inert and will not interact with the sodium and being heavier than air, will tend to be held in place by gravity within a tank. The reactor head and sodium pumps are therefore provided with gas seals, which eliminate the need for a mechanical seal on the pump shaft exposed to liquid sodium. In the case of sodium storage tanks, if the sodium is not needed to be held in emergency reserve, it can be allowed to freeze so long as there is a provision for melting it when it is needed. In such cases, the argon can be replaced with nitrogen which will not react with the sodium. Since nitrogen is slightly soluble in sodium, above about 400°F it is necessary to replace the nitrogen with argon to prevent the formation of various metal nitrides with structural materials.[20] One of the problems with argon is its use tends to promote the formation of sodium frost on unwetted surfaces. The use of helium as an alternative cover gas has been considered as a means of rectifying this problem since the frosts tend not to develop when such a light gas is used. Helium was used as a cover gas for both the SRE and Hallam Nuclear Power Facility. Its use on these two plants did not appear to present any particular problems. It is easier to separate the fission gases xenon and krypton from helium than it is from argon. But helium is considered to be difficult to contain and for this reason it has not been adopted for further plants beyond Hallam. This is probably an area that deserves revisiting, particularly if large quantities of fission gas must be dealt with in the cover gas. On the CRBRP Project, the bases for selecting argon cover gas were tenuous at best and seemed primarily founded on the fact that there was more experience with argon.

The reactor cover gas system should be continuously monitored for fission gas activity in the interest of early detection of fuel element failures, since radioactive isotopes of both krypton and xenon are fission products. It is common practice to "tag" fuel assemblies with a unique mixture of gases so as to enable location of any failed fuel assembly. The cover gas system must therefore be provided with detection equipment that enables this capability.

Various steam generating systems have been used in LMFBRs including those with separate evaporators, superheaters, and reheaters, those with only evaporators and superheaters, and those with once through units. The Russian plant BN-600 is representative of one extreme with modular evaporators, superheaters, and reheaters and a total of 72 separate heat exchangers making up its entire steam generating system. This approach was used to permit repair of failed units while continuing to operate the plant at slightly reduced power. At the time BN-600 was designed, there had been a history of intractable problems with sodium heated steam generators, particularly at Fermi-1 and the U.K.'s PFR. Since BN-600 began operation in 1980 and has run since about 1984 with a 75% capacity factor, this modular approach has proven itself quite well. In fact, it was reported in a 2018 conference[21] that BN-600 had experienced 12 steam generator leaks over its entire history, but 11 of those occurred between 1980 and 1985. Since then, they have experienced just one leak in 1991 and none since. The Russians have demonstrably made improvements in the module designs to enhance their reliability. However, outside Russia the trend has been to minimize the number of heat exchangers with a single once through unit per

20Although sodium nitride (Na_3N) is reasonably stable at room temperature it readily dissociates at elevated temperatures. A nitrogen cover gas above solid sodium at room temperatures does not lead to the formation of much Na_3N, if any.
21Pakhomov, Ilia, "BN-600 and BN-800 Operating Experience", Generation IV International Forum, Dec. 19, 2018

loop, as was done on Superphénix. This is a reflection of growing confidence that a steam generator can be designed with acceptable reliability. Although Superphénix did not accumulate anywhere near as much operating experience as BN-600, it experienced no steam generator problems and its approach is generally accepted as the most economic.

Both metal and oxide fuel have been successfully used in sodium cooled reactors. Earlier reactors e.g. EBR-1, EBR-2, SRE and Fermi-1[22] used metal fuel. Early on, the advantages of oxide fueled systems were recognized. The oxide has a much higher melting point than uranium/plutonium metal permitting higher sodium temperatures. Higher fuel temperature with operation means higher negative Doppler feedback with increasing power, promoting reactor stability. Fission product gases pass through the oxide to the gas space at the upper end of the fuel pin rather than distorting the metal matrix, permitting higher burnup. Oxide fuel has been universally adopted by the LWR industry, thus its use in LMFBRs could lead to common reprocessing of both fuels in a single plant. Carbides and nitrides have been explored for possible use in LMFBRs and may have certain advantages, notably improved breeding ratios. At present, oxide is the fuel form of choice with most entities that maintain interest in LMFBRs.

The fuel in an LMFBR differs significantly from a LWR. The fuel assemblies tend to be smaller in cross section with much smaller diameter pins, in the 0.23-0.33 in. range, arranged in a triangular rather than the square array chosen by LWR designers. In an LMFBR, it is necessary to control the coolant flow to each fuel assembly therefore each assembly is ducted and orificed at the bottom. The ducts are hexagonal in cross section permitting closer packing of the pins taking better advantage of the excellent thermal conductivity of sodium. The number of pins per assembly is a compromise between economy, which favors more pins and optimization of coolant flow which favors fewer pins. CRBRP had 217 pins per assembly while Superphénix had one additional row and 271 pins. In a LWR, zirconium is the preferred clad material because of its low cross section for thermal neutrons. In a fast reactor, this incentive does not apply, so the clad and duct material is fabricated from less expensive and stronger stainless steel. In fact, it is desirable to eliminate zirconium because its affinity for oxygen in the coolant requires that the coolant oxygen concentration be maintained at a level not achievable by cold trapping. The Clinch River fuel assembly is shown below as Figure 1. The fuel rod pins were about 9 ½ ft. long, 3 ft. fuel, 14 in. each for upper and lower axial blankets and 4 ft. for gas plenum. The core of an LMFBR is typically surrounded by upper and lower axial blankets and radial blankets containing un-enriched or depleted uranium to aid the breeding process. The complete assemblies including inlet and outlet hardware had a length of about 14 ft.

22The Fermi-1 operator planned to move to oxide fuel when it became available.

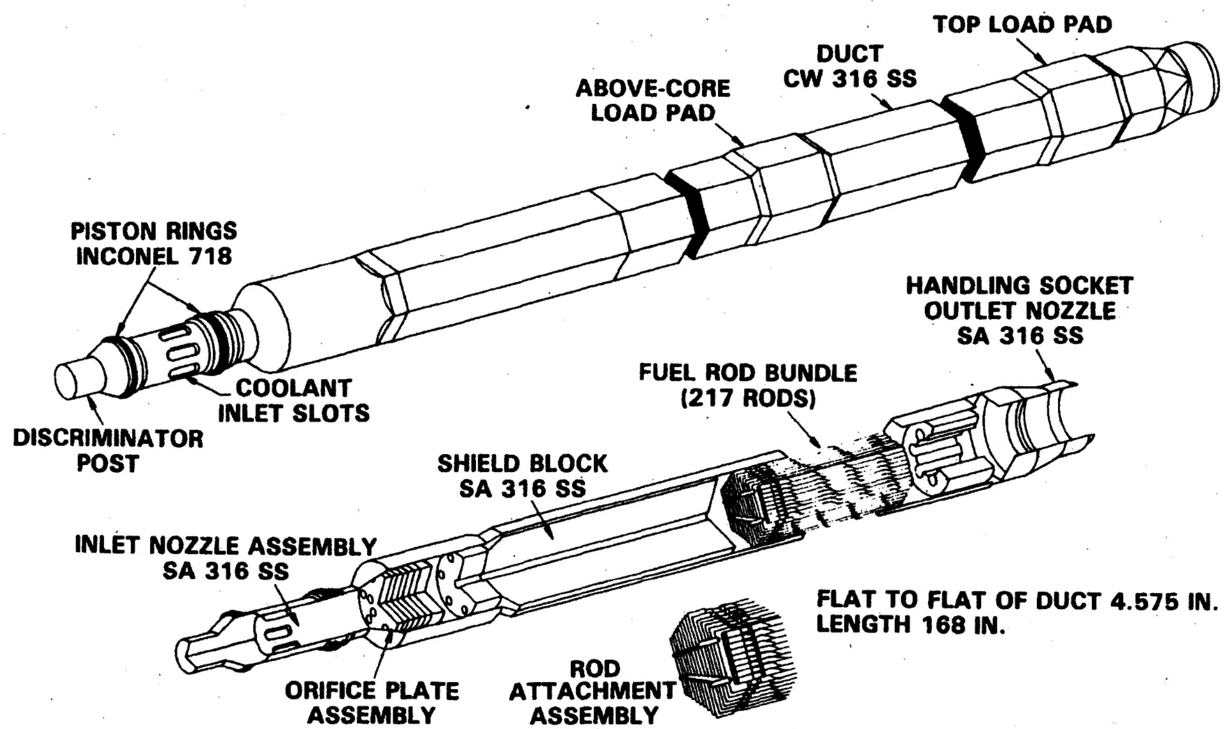


Figure 1 CRBRP Fuel Assembly

Typically, spacing the pins is accomplished by wrapping wire around them along the length of each pin. This approach has the advantages of economy of fabrication and good mixing potential. Grids, as are used in LWRs can be used in LMFBRs and have the advantage of leading to a lower pressure drop. The plutonium/uranium mixed oxide fuel occupies about 3 ft. of the length of the pins. Above and below the fueled region, there is an axial blanket of about 1-1½ ft. in length. Above the upper axial blanket, there is a fission gas plenum of about 4 ft. in length. Typically, the upper and lower axial blankets are of equal sizes. When inlet hardware and handling fixtures are added, the total assembly length is approximately 14 ft. The cladding is 10-15 mils in thickness. The fuel pins on Clinch River are shown below as Figure 2.

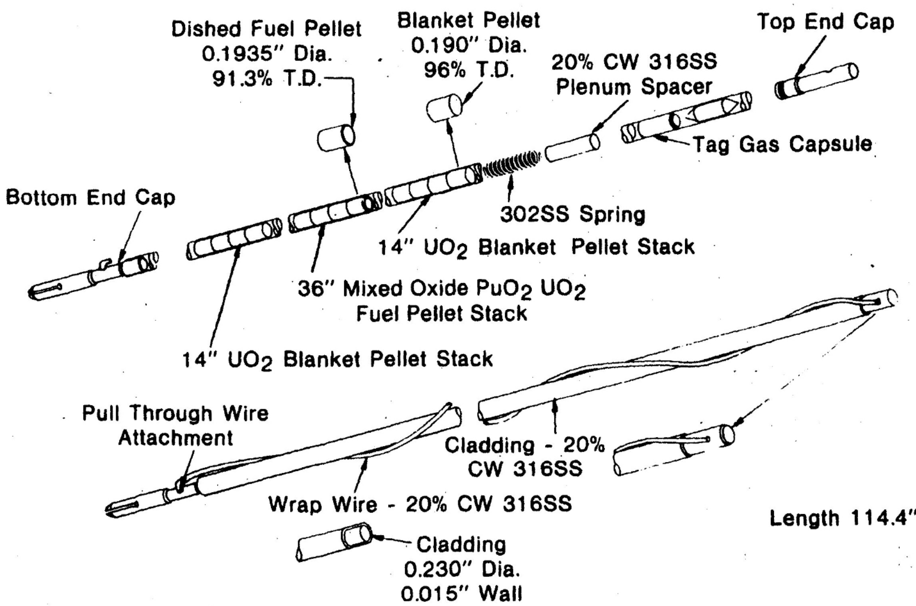


Figure 2 CRBRP Fuel & Blanket Pins

The fuel assemblies are 4-8 in. across the flats of the hexagon in a LMFBR in comparison to about 12 in. in an un-ducted square cross sectioned LWR fuel assembly and the pins are more closely spaced to take advantage of the better heat transfer properties of sodium and to minimize the size of the core and improve the breeding ratio. The CRBRP 217 pin bundle compares with about 144 pins in a ~1 ft. square bundle in a PWR. The active fuel region of just 3 ft. in height corresponds to 12-14 ft. in a similarly sized LWR. This again is a realization of the better heat transfer properties of sodium in contrast to water combined with the absence of any need to moderate the neutrons.

5 Cost reduction approach

Nuclear power plants are composed of the so-called "nuclear island" and a steam plant. With the exception of the steam generators, the steam plant of a nuclear power plant resembles the steam plant of a fossil fired plant. LWR steam plants use saturated or slightly superheated steam while the LMFBR steam conditions are more typical of a fossil-powered plant. Much experience has been accumulated in the power industry building economic steam plants which would carry over to any envisioned LMFBR plant, so any cost reduction initiative need only consider the nuclear island.

The prime factors driving the cost of the nuclear island are size and complexity. A third factor, congestion, can also become a major contributor, and is best handled by avoiding design concepts that are over-integrated, i.e. that place excessive and sometimes competing demands on single components. Congestion lengthens construction adding carrying costs and it tends to occur late in the construction process when the carrying costs are the greatest. We shall see how this problem of congestion appears in certain LMFBR design approaches as we proceed with this discussion, and what steps can be taken to avoid it. A fourth factor is the extent of the plant that is safety related, i.e. necessary for safe shutdown. On CRBRP, the safety related portion of the plant extended to the steam system. Modern design approaches typically attempt to reduce the safety related envelope to the primary system and associated decay heat removal system(s). The two approaches shall be described. A fifth factor is the degree of required on-site fabrication. On-site fabrication has two adverse effects -- it delays construction contributing to carrying cost and is performed in a makeshift environment with typically less skilled personnel. A sixth factor could occur if the size of fabricated components requires barge shipment.

Standards and regulation also significantly impact cost. Design organizations typically participate in standards setting committees in the interest of ensuring standards do not unnecessarily impose onerous requirements that will be unnecessarily expensive. Potential impacts of regulatory action will be treated in subsequent sections where CRBRP experience suggests that trouble may be imminent.

When considering design features, the CRBRP design shall be used as a point of departure for the reasons given in Section 1. Designs that predated CRBRP, foreign designs, and more recent concepts are evaluated for their economic potential and if desirable, are incorporated into a "design approach", mentioned in Section 1. Other promising concepts are evaluated and incorporated where applicable.
For the case of CRBRP, a 58 ft. long and 20 ft. diameter reactor vessel housed a reactor core that was just 5 ft. 4 in. in height and about 8 ft. in diameter. The vessel was centrally located in a cylindrical containment 186 ft. in diameter. The prime drivers impacting the reactor vessel height were the fuel element design and the refueling system design. The Primary Heat Transport System (PHTS) design was the driver for the containment design. The Steam Generating System (SGS) introduced complexity which can be eliminated in a straightforward way. This is related to the Decay Heat Removal System (DHRS) that was selected, for which simpler and possibly more reliable alternatives are available. The containment design approach was borrowed from PWR practice when a lower cost alternative is available and would be appropriate. The extensive 1E electric power system is also a candidate for elimination or at least, major cost reduction.

Before getting into the details, it is necessary to provide an overview of the LMFBR Heat Transport System. Figure 3 shows conceptually how heat is transported from the reactor to the Intermediate Heat Exchangers (IHX) via the Primary Heat Transport System (PHTS), then to the Steam Generator via the Intermediate Heat Transport System (IHTS) and ultimately to the turbine. In practice, there will be from two to four primary loops and an equal number of intermediate loops with the steam from all steam generators combining to a single turbine. The IHTS is made necessary by the requirement to eliminate the possibility of water leaking from failed steam generator tubes finding its way to the reactor. The IHTS, although adding complexity and cost in comparison to PWRs, does have the benefit of enabling the steam generators to be located outside containment, where they are more accessible, maintainable, and replaceable.

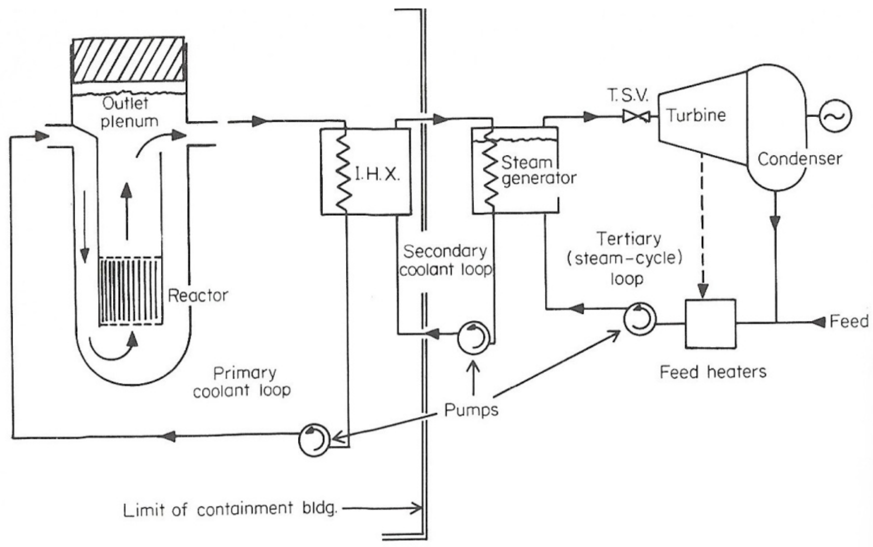


Figure 3 Conceptual LMFBR Heat Transport System

Two sodium cooled submarine reactors, S1G and S2G, avoided intermediate loops by using double-walled steam generators that were located inside containment. It has been reported[23] corrosion was experienced in the steam plant which had been accelerated by gamma radiation, which may or may not have credibility. Double-wall tubes have been deployed in certain liquid metal designs, extensively tested, and may find application in future LMFBRs, independently of whether or not they are used with or without an intermediate loop. Absent double-walled tubes, it is necessary to provide for potential tube failure in the plant design.

Design of a nuclear power plant must begin by setting requirements and objectives. As stated in Section 1, Superphénix parameters will be used, but the overriding objectives are to minimize cost – both capital and operating, while maximizing availability, complying with regulatory requirements and maintaining an acceptable level of safety.

A key objective of this design approach takes advantage of the breeding characteristic in such a way as to maximize the interval between refueling. Since the LMFBR produces more fuel than it consumes, it is possible to design the core so that the newly bred fuel occurs in regions of high importance, minimizing the reactivity swing attendant to burnup. That objective will be the prime driver for the proposed core design. This approach probably penalizes breeding ratio somewhat but maximizing time between refueling creates opportunities for economies in the refueling system, which is more important, particularly for the earlier plants.

It is necessary to next direct attention to the reactor core, proceeding outward to the reactor vessel and refueling system, heat transport system, containment, decay heat removal system, and auxiliaries. It is essential that the reactor core be designed first since it sets the requirements for subsequent systems, but the core design process is lengthy and complicated, even when treated in an overview fashion. Accordingly, it has been moved to the appendix where it can be consulted

23R. G. Palmer, A. Platt, Fast Reactors, Temple Press, 1961 by those who have a particular interest in such matters. For the purposes of the following sections, the core is intended to have; 1) a ten year interval between refueling; 2) features that minimize core pressure drop to about 20 psi; 3) 1% per minute load following capability; 4) capability for power operations in the 15-100% range; 5) steam parameter consistency with Superphénix; 6) capability for 95% capacity factor between refuelings. If capacity factor is lower, the interval between refuelings is correspondingly greater; 7) a core diameter of approximately 22 ft. (including shield assemblies) and core height including the axial blankets of 5 ft. 10 in., which includes a 33% growth allowance (an extra foot in the fueled region) beyond minimum to accommodate the 10 year refueling interval.

The load following capability of the "design approach" is worthy of further comment. Essentially all the nuclear plants currently in operation are base-loaded; i.e. they operate at full power around the clock except when they are being refueled, (or, in some cases, maintained) when they are shut down. This is primarily a reflection of their high capital cost and low fuel cycle cost, which makes base loading the most desirable economic option. If a nuclear plant can be developed with significantly lower capital cost, load following becomes an option, which could be important in systems having a substantial portion of their generating capacity provided by renewables. Such plants therefore can become an enabler for renewables, competing with natural gas combined cycle plants.

6 Reactor Vessel, Internals, and Refueling

In an LMFBR, the refueling system has a major impact on the reactor vessel design, so it is necessary to consider them together. Moreover, when the two systems are treated together, cost reduction opportunities become more apparent. The refueling system is complicated and to communicate its functioning, it is necessary to describe a refueling operation, inevitably causing this section to be the most lengthy in the paper. In the interest of making it more readable, subsection titles have been inserted. As will be the practice throughout this discussion, the CRBRP reactor vessel design, shown below, will be used as a reference point to explore what opportunities for cost reduction exist. On the CRBRP Project, the reactor vessel design and the refueling system design were conducted by two different contractors, which possibly could have impacted a serious search for optimization between the two systems.

CRBRP Reactor Vessel and Guard Vessel

A striking feature of the CRBRP Reactor Vessel (RV) compared to a PWR is its 58 ft. length, particularly considering the fact that the reactor core (including the axial and radial blankets) is just 5 ft. 4 in. high and about 8 ft. in diameter. The 20 ft. diameter RV is also greater than one might expect given that PWRs with four times the electric generating capacity typically have 13-15 ft. diameter reactor vessels. The vessel wall is much thinner than a PWR reflecting the lower pressure of the coolant. It is also fabricated almost entirely of stainless steel and Inconel, which is a reflection of the higher temperature of the coolant. Although it is doubtful that LMFBR reactor vessels will ever be as compact as their PWR counterparts, much of the discussion which follows will be focused on the requirements on the CRBRP that dictated this length and measures that can be taken to reduce it. The vessel length influences the containment size and in the case of CRBRP, sets the elevation of the entire containment floor. There was much unoccupied containment volume in that design.

After passing through the three inlet nozzles at the bottom of the vessel, (CRBRP had three primary loops) the sodium enters the inlet plenum. Flow exits the inlet plenum through the lower inlet modules. There are 61 of these lower inlet modules, each serving seven core assemblies. After leaving the lower inlet modules, the sodium flows upward through the core assemblies (fuel, blanket, control, and removable shield) and the upper internals structure (UIS), enters the outlet plenum then departs through the outlet nozzles. There is a small amount of bypass flow that passes through the annulus between the core barrel and the reactor vessel then up through the annulus between the thermal liner and the reactor vessel. The volume between the core barrel and the vessel wall was used primarily as a transfer position for the refueling machine but also served for interim storage of spent fuel assemblies, when needed. The bypass flow cooled the reactor vessel, core barrel and any assemblies in the storage or transfer positions. There is also a small amount of flow that passes through the interstices between the core assembly ducts. There is a suppressor plate supported from the head located just below the sodium level to prevent excessive surface motion of the sodium and splashing onto the shielding below the head.

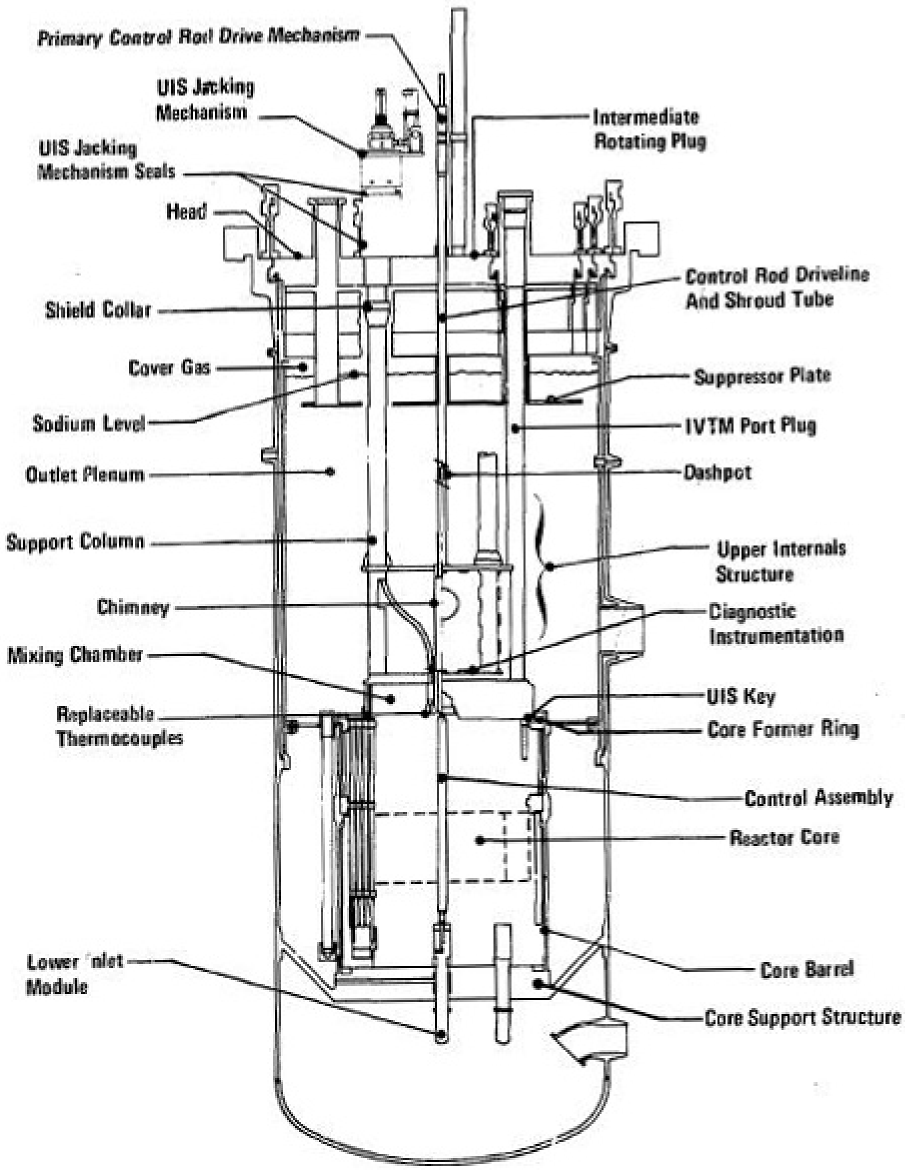


Figure 4 CRBRP reactor vessel

The reactor vessel is surrounded by a guard vessel as shown in Figure 5. The figure has been rotated counterclockwise 90° so the captions are readable. The guard vessel extends above the outlet nozzles and guards the inlet and outlet piping up until all those six pipes turn to the horizontal direction. The primary system piping (discussed in Section 7) runs at a constant elevation between the primary system components. Guard vessels also surround the primary pumps and the Intermediate Heat Exchangers (IHXs). The elevation of the tops of all the guard vessels is uniform.

Guard vessels are a characteristic feature of sodium cooled plants. The idea behind them is that if there is a primary system leak at one of the components, the leaking sodium will fill the guard vessel until the sodium level in the guard vessel is the same as the leaking component. A leak in the elevated piping (See Section 7 for a discussion of this "elevated piping") would theoretically flow to the floor of the vault in which it is located, but much of such a leak would freeze at the leak site, particularly if the leak is small. Basically, the "elevated piping" is the piping that connects the primary system components (RV, primary system pumps, and IHXs). It is above the level of the guard vessels and is unguarded. Such a leak would empty the sodium in the branch of the piping between adjacent components, disabling that loop, but only that loop. In a sense, guard vessels provide much the same protection as the containment in a PWR. Both are intended to protect against major primary system piping failures. This guard vessel concept can only apply to systems where the reactor coolant pressure is close to atmospheric, thus requiring a coolant with a high boiling point. One of the design criteria is to ensure there is sufficient sodium in the reactor vessel above the top of the outlet nozzles to ensure the nozzles remain covered in the event of a reactor vessel leak. The same criterion would not necessarily apply to the IHXs and primary pumps but they must be guarded anyway to prevent siphoning the RV. The guard vessel concept also explains why there were three loops in CRBRP. Each of the primary loops played a role in decay heat removal. If one of the loops were breached, two more loops would be required to meet the single failure criterion, a ground-rule for safe shutdown design since the early days of nuclear power.

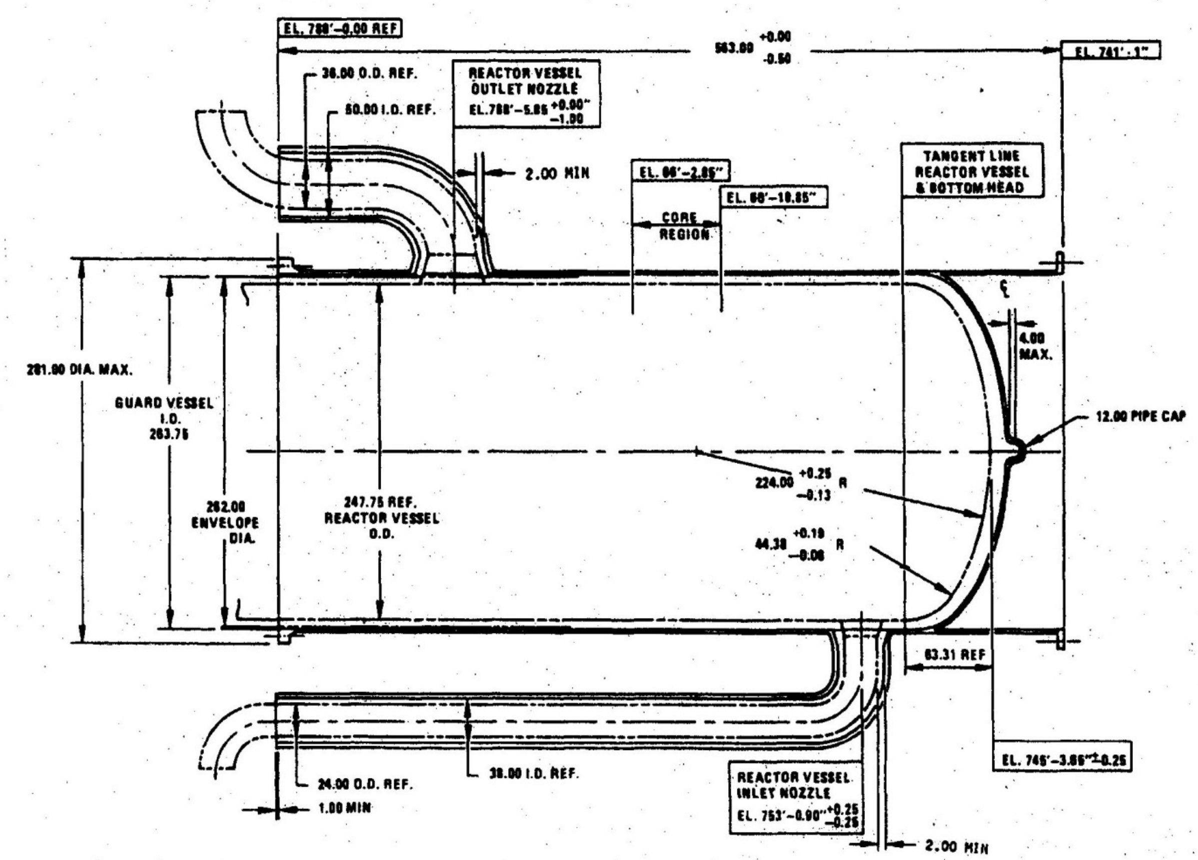


Figure 5 CRBRP Reactor Guard Vessel

The space between the tops of the fuel assemblies and the suppressor plate was provided for horizontal translation of core assemblies by the In-Vessel Transfer Machine (IVTM) during refueling operations, described later. This was at least 14 feet of reactor vessel length on CRBRP and for the fuel design approach proposed in this monograph, it would be in the 11-11 ½ ft. range. The difference between the CRBRP fuel assemblies and the "design approach" assemblies, 2 ½ - 3 ft. (see Appendix 2B for discussion of this feature) represents the first cost reduction measure (CRM) for the "design approach". This difference is achieved through elimination of the fuel assembly gas plenum, shortening of the fuel assembly inlet plenum, and shortening of the upper axial blanket while lengthening the fueled region by one foot. In fact, given that refueling is partially accomplished using an IVTM, this fuel assembly length difference is realized twice, first in the core barrel length and second in the outlet plenum length.

The lower inlet modules were primarily provided to permit shuffling of the blanket assemblies during refueling, since they were orificed for all radial blanket positions. The idea was that when a radial blanket is shuffled to an outer row, it requires less flow. This could be accommodated by the orifices in the lower inlet modules.

Elimination of the lower inlet modules is the second action that is herein being advanced as a cost reduction measure. Since the design approach being proposed does not require blanket shuffling, there is no need for these inlet modules. The core assemblies can be loaded directly into the core plate, which is the bottom forging in the core support structure, which was the approach used on the FFTF, the predecessor plant to CRBRP. In CRBRP, there were 61 of these lower inlet modules and their associated liners. Although they were never procured by the project, the expense of these devices would have certainly registered in the millions aside from their impact on the reactor vessel. A further incentive for eliminating the inlet modules is the elimination of the pressure drop that occurs across them and the coolant flow that is lost by leakage. Inlet module elimination subtracts about 3 feet from the length of the reactor vessel.

The lower inlet modules also added three feet to the length of the core barrel. A figure of one of these lower inlet modules inserted into its module liner is provided below. Dimensions are in inches.

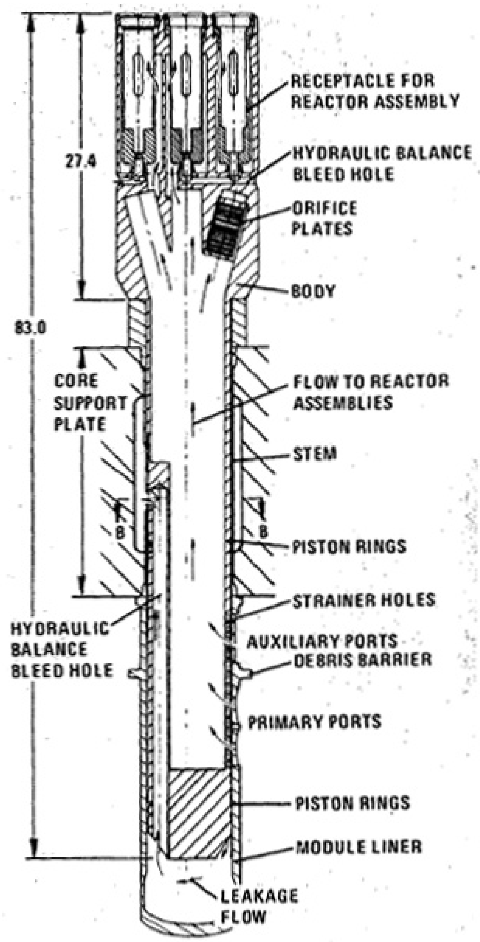


Figure 6 CRBRP lower inlet module

Since it was not possible to support the entire set of removable shield assemblies with these lower inlet modules, it became necessary to design so-called bypass flow modules to accommodate the outer peripheral assemblies. Six large bypass flow modules were intended to be installed in the reactor outside these lower inlet modules. All this hardware came at a high cost both in design and fabrication, added an additional component that could fail, increased the pressure drop across the reactor vessel by up to 8.66 psi, increased hydraulic leakage scavenging coolant away from the core, and turned out to be unnecessary. All of the core assemblies could have been inserted into the core support plate at the bottom of the core support structure, as was done on the FFTF. Passages could have been provided in the core support plate for the bleed flow necessary to make hydraulic hold down work. (Hydraulic hold-down is described in Appendix 2D under "Thermal-Hydraulics Design" and a figure is provided.) In fact, passages were provided in the core plate anyway to enable hydraulic hold down for the lower inlet modules.

The orifice plates provided in the lower inlet modules to permit radial blanket assembly shuffling proved to be unnecessary since the nuclear designers wound up having no intention of shuffling blanket assemblies. This happened on CRBRP because the reactor vessel and internals design was ahead of the core design[24]. The internals designers knew the lower inlet module concept would require lengthening the reactor vessel and needed to commit on a conceptual approach since a RFP was about to be released for reactor vessel fabrication.[25] The internals designers anticipated a nuclear design requirement (shuffling of blanket assemblies) that didn't materialize. Although well intentioned, the whole lower inlet module exercise turned out to be a waste of time and money. It is an object lesson of the need to have high confidence in the nuclear design before making irreversible commitments on other parts of the plant design.

The reduction of the pressure drop across the reactor which is also described in Appendix 2D is a significant cost reduction measure as it reduces the design requirements on the PHTS pump, reduces pumping power making more power available to the grid and eliminates the need for hydraulic hold-down of the fuel assemblies. These steps constitute items three and four of the cost reduction measures. Since the fuel assemblies are proposed to be vented to the coolant, gas tagging is not possible, therefore creating a fifth cost reduction measure, the elimination of gas tagging. It should be pointed out that gas tagging was not required on CRBRP to satisfy any regulatory requirement. There is more on this subject in section 12.

The head shielding on CRBRP was provided to permit head area access shortly after shutdown when Na^{24} activity in the coolant is high. For the refueling approach proposed here, Na^{24} activity is allowed to decay before head area access is needed. The head shield represents another 3-4 feet of RV length. If one adds the reductions that can be obtained from elimination of the transfer space, elimination of the fuel & blanket assembly gas plenum, elimination of the lower inlet modules, and elimination of the head shielding, there is a reduction potential of 25 feet in reactor vessel height. Head shield elimination constitutes CRM 6. Steps that can be taken to eliminate the transfer space are discussed later in this section.

The SRE and Hallam Reactor Vessels

The ellipsoidal lower head on CRBRP is suspect. It possibly provides greater strength, better flow distribution, and a lower pressure drop than a flat bottomed head but it may also be no more than an unnecessary carryover from LWRs, where the high pressure coolant dictates configurations like this. In fact, the pressure drop across the inlet plenum was predicted to be a rather high 5.8 psi, which is either a conservativism or possibly a consequence of the lower head design. The figure below shows an early concept for the Hallam reactor vessel.[26] The designer of Hallam (then North American Aviation, later Rockwell International) had no previous experience designing LWRs. Hallam followed the Sodium Reactor Experiment (SRE), also designed and operated by North American Aviation.

24There did exist a core design at the time, but the nuclear design did not meet project requirements – a fact that may not have been known to the internals designers. It turned out to be necessary to make major changes in the core design to remedy its inadequacies, all of which occurred after the inlet modules had become a fixed feature of the design.
25On the CRBRP project, the reactor vessel fabrication contract was let long before it was needed primarily to keep the fabricator, Combustion Engineering, in business. The concern was, if the Combustion Engineering fabricator went out of business for lack of work, there would be no other domestic fabricator who could have handled the job.
26Drawing from Starr, Chauncey; Dickenson, Robert W.; *Sodium Graphite Reactors*, Addison Wesley; 1958.

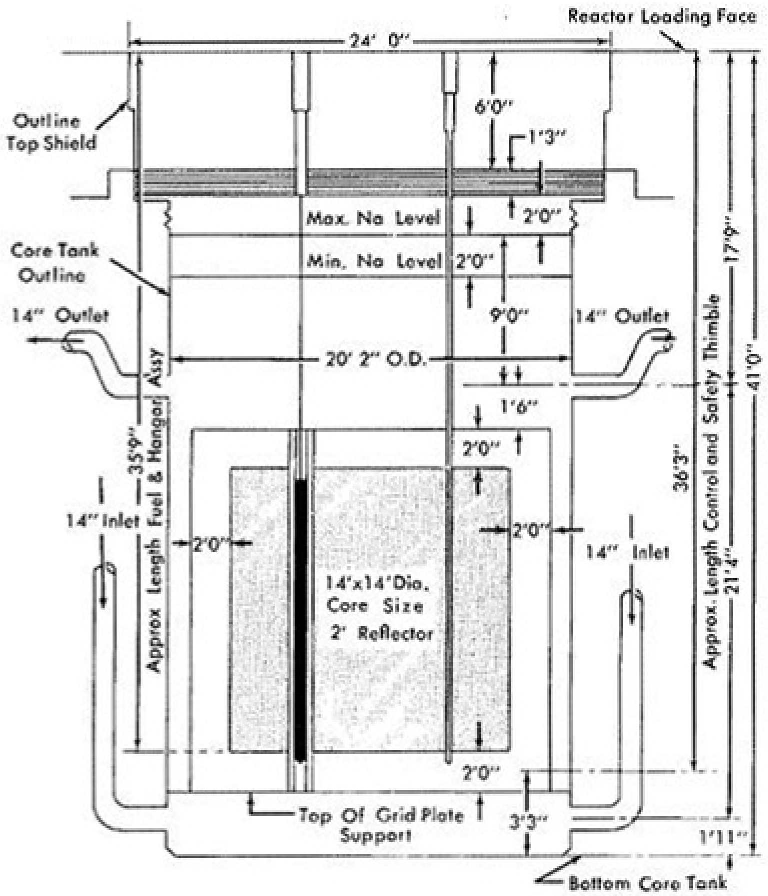


Figure 7 Hallam reactor vessel

The Hallam core had a thermal spectrum that was moderated with graphite so the core dimensions are not relevant to this discussion. However, the flat bottom is quite evident. In addition to reducing vessel height, removing the ellipsoidal lower head could simplify bottom mounting of the vessel. Bottom mounting the reactor vessel removes much of the tensile load off the reactor vessel wall which would permit thinning the vessel wall. A second important difference is the absence of a core barrel. The core assemblies, moderator and fuel, appear to extend to the thermal shield. Presumably, there was some sort of fixed shield between the core assemblies and the thermal shield to accommodate differences in the geometry between the core assemblies and the cylindrical thermal liner. A third important difference is the absence of an Upper Internals Structure (UIS). The UIS served three functions on CRBRP: 1) backup holddown of core assemblies, 2) A place to locate core exit thermocouples, and 3) a place to mix sodium exiting fuel, blanket, control, and shield assemblies having widely different temperatures. It is noted that there is no need for backup holddown in the "design approach" and the need for a structure to promote mixing is not obvious. A fourth difference is a significant shortening of the distance from the top of the core assemblies and the bottom of the RV head.

The reason for this will be explained in the subsequent discussion. A second more detailed rendering of the Hallam vessel is presented below.[27]

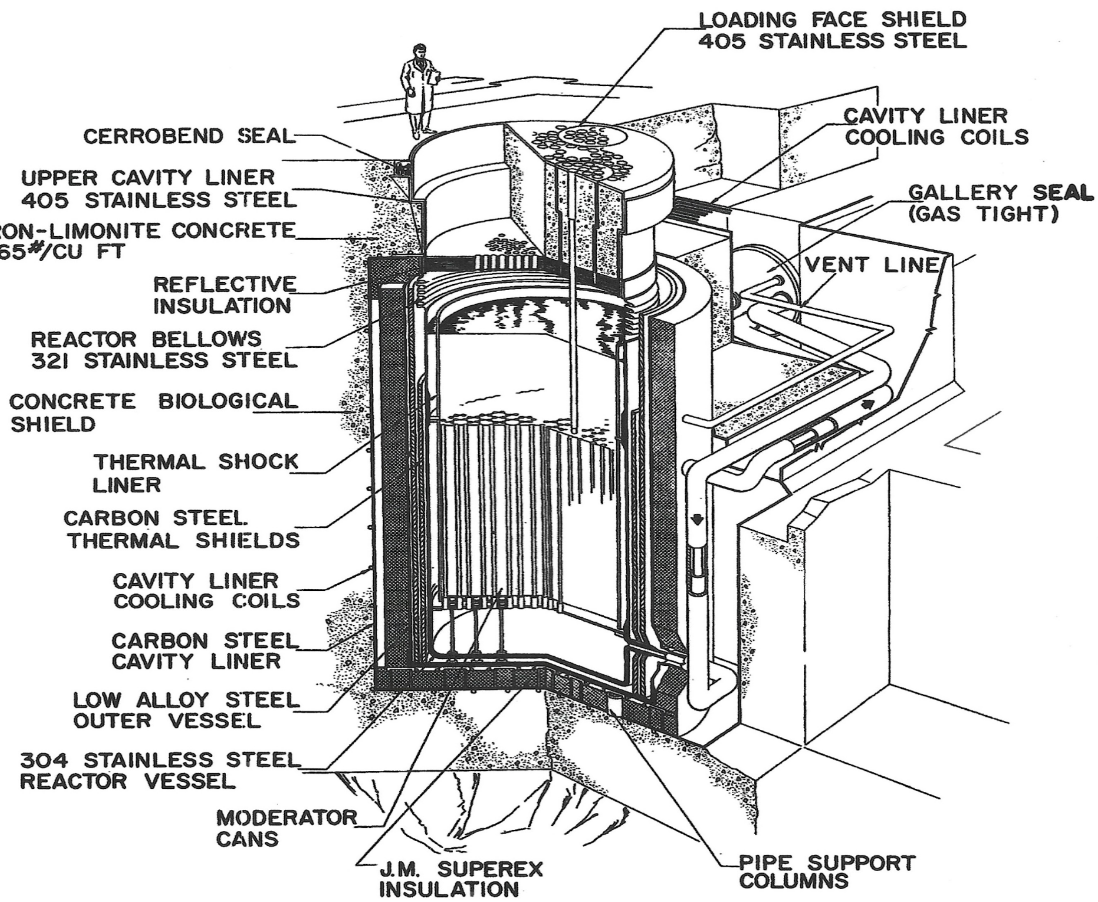


Figure 7a Hallam Reactor Vessel

From the above drawing, there appears to be some sort of standoff between the bottom of the RV and the guard vessel that is aligned with the support posts for the grid plate. This contrasts with the SRE configuration where the reactor vessel rests on the guard vessel (see figure 8). The Guard Vessel appears to be riding on some sort of bearing arrangement mounted on the insulation. The thermal shield (identified as the thermal shock liner) extends down to the grid plate, and there is a bellows at the top of the reactor vessel to accommodate thermal expansion between the fixed reactor vessel bottom and the closure head. Since there is essentially no tension on the vessel wall at the location of this bellows, it can be as thin as necessary to permit operability of the bellows. Tetralin, the same cooling fluid that was used for the seals of the PHTS pump on SRE, was supplied to channels in the surface of the concrete facing the reactor vessel.

27 Beeley, R.J., Mahlmeister, J.E., *Operating Experience with the Sodium Reactor Experiment and its Application to the Hallam Nuclear Power Facility*, Atomics International, 1961

One of the questions that emerges from close inspection of figure 7a is how thermal expansion of the bottom of the guard vessel is accommodated. The wall of the guard vessel is hard up against the surrounding insulation. If the horizontal constraint resulted in the bottom of the guard vessel bowing upward, the result would likely be serious. Another question would pertain to the support structure for the guard vessel. There are 79 support columns each 18 in. in diameter and 25 in. high, fitted with a cap on top.[28] Again, there is the question of whether the vessel moves over the caps attendant with thermal expansion. The outer vessel is low alloy ferritic steel while the RV is 304 SS, so the two will expand with temperature at differing rates (austenitic having the higher thermal expansion coefficient than ferritic) and the temperature of the RV is likely to be higher than the Guard Vessel. How the relative motion between the two is accommodated is not clear.

There is a bellows at the top of the RV wall. This is a consequence of a fixed bottom and a fixed head. The head is supported by the operating floor deck which is a massive shield. It is necessarily heavy to allow access to the head area while Na^{24} levels are high. If the requirement for access to the head area is removed, the head thickness can be reduced to about 2 ft. and can be supported by the RV. This bellows is not necessary if the RV head is supported by the RV and is permitted vertical movement associated with temperature changes. A bellows connecting to the vessel and the vault wall would probably be required to adequately seal the inerted reactor vault.

Of all of the features of the Hallam Reactor which differentiate it from CRBRP, the most important is likely the bottom mounting. Bottom mounting takes the heavy tensile load off the reactor vessel wall and enables supporting the core plate (or core grid as it was called on Hallam) off the bottom. Bottom support of the core plate would permit reduction of its thickness, and the support posts would possibly improve mixing in the inlet plenum. Elimination of the core barrel would permit reduction of the RV diameter, but it may pose a problem for core restraint (not required in a sodium thermal reactor) which can impose heavy loads on the relatively thick core barrel. Possibly the thermal liner could be thickened so as to serve a dual purpose, but such an approach is left for future study. The UIS continues to provide a convenient means for measuring core assembly outlet temperatures. The core barrel and UIS will therefore be retained in the “design approach”, but the flat bottom will be adopted.

While the elimination of the core barrel and UIS are both worthy ideas and should be evaluated further as part of any preliminary design activity, the elimination of the lower ellipsoidal head and bottom mounting the vessel will be carried forward in the “design approach” and constitute CRMs 7 & 8.

One design question regarding this approach is how one accommodates thermal expansion at the point where the bottom of the vessel contacts the support structure. For the case of the reactor plant SRE, which was predecessor and supported similarly to Hallam, there was no space between the reactor vessel bottom and the guard vessel – i.e. the reactor vessel was in contact with the guard vessel while outside the vessel wall there was an annular space, filled with

28 Gronemeyer, F.C., Merryman, J.W., *75,000 Kilowatts of Electricity by Nuclear Power at the Hallam Nuclear Power Faciclity*, ASCE Convention Reno, Nev., e 23, 1960

helium, surrounded by a “thermal shield” which was in contact with the guard vessel. The guard vessel was supported by a bearing plate, which consisted of circular rings of insulating material. A diagram of this arrangement, is shown below.[29]

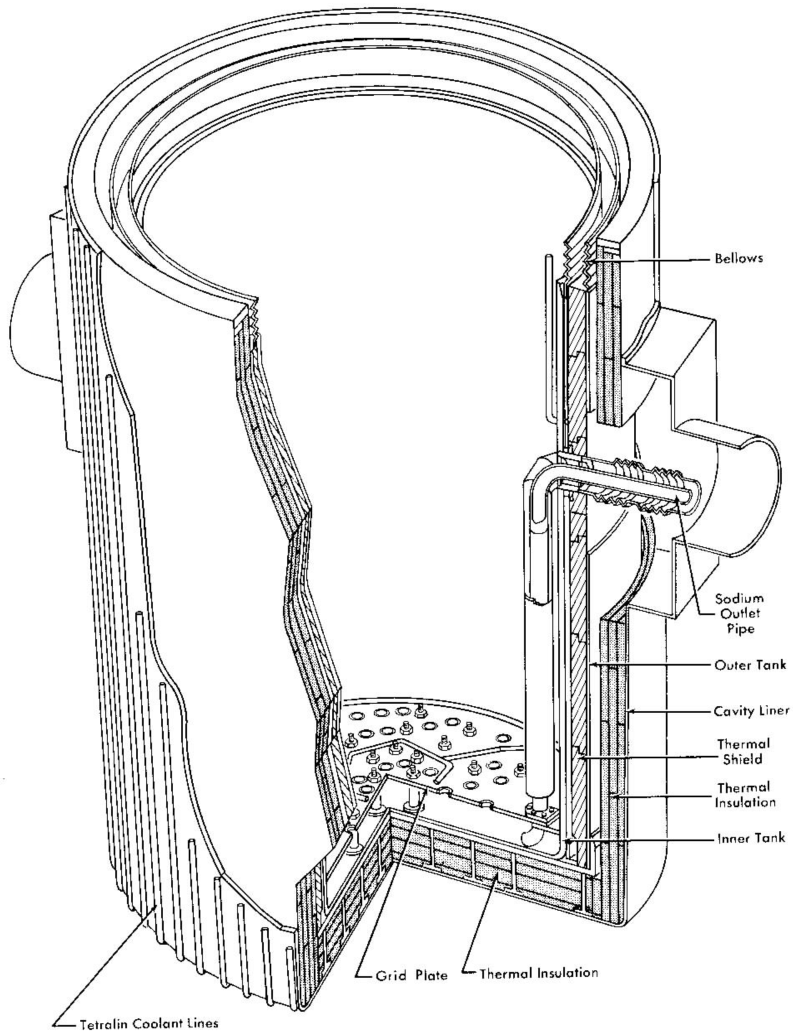


Figure 8 SRE Reactor Vessel Installation

While it is encouraging that the most obvious problem of bottom mounting was addressed on this early design, there remain a host of questions, such as how the bearing rings are configured so as to allow for differential expansion between the vessel bottom and the underlying concrete. Serious consideration of bottom mounting the reactor vessel must begin with a careful review of how SRE and Hallam resolved these problems followed by determination of whether the approaches used will scale up to the sizes envisioned in this paper. Some form of demonstration involving a mockup would likely be required to establish confidence in its suitability for long term application. Despite these uncertainties, the configuration proposed will adopt the bottom mounted approach as a cost reduction measure for reasons that will become clearer as the

29Starr, Chauncey; Dickenson, Robert W.; *Sodium Graphite Reactors*, Addison Wesley; 1958.

discussion in this section advances. Later in this section, a trade-off between the top mounted and bottom mounted approaches will be made. If the bottom mounted concept proves to be unworkable, fallback to the conventional flange mounted concept would be available at an economic penalty. As an aside, it is worthy to note that the BN-600 pool reactor is bottom mounted off a skirt. The bottom of the BN-600 pool is ellipsoidal and not flat.

CRBRP Refueling Approach

At this point, it becomes necessary to consider refueling, which had a significant impact on the CRBRP reactor vessel. Compared with the refueling procedure and equipment required for a LWR, the typical LMFBR refueling system is remarkably complex. This complexity is the combined result of using sodium as a coolant and requiring rapid refueling – on the order of two weeks – and starting the refueling process shortly after shutdown while Na^{24} activity remains high. Although the refueling system for Superphénix is different from CRBRP, it is no less complex so CRBRP provides a decent baseline for evaluation, which will be used here. A composite figure of the CRBRP refueling system is shown below. The Superphénix scheme will be discussed later.

There are four major components that constitute the refueling system: the three rotating plugs that form most of the reactor head, the In-Vessel Transfer Machine (IVTM), the Ex-Vessel Transfer Machine (EVTM), and the Ex-Vessel Storage Tank (EVST). These four components are shown in Figure 9, below. The figure also shows an Auxiliary Handling Machine (AHM), which was used primarily for installing the in-vessel section of the IVTM and a Fuel Handling Cell (FHC) whose main purpose was to receive new assemblies and prepare spent assemblies for shipment.

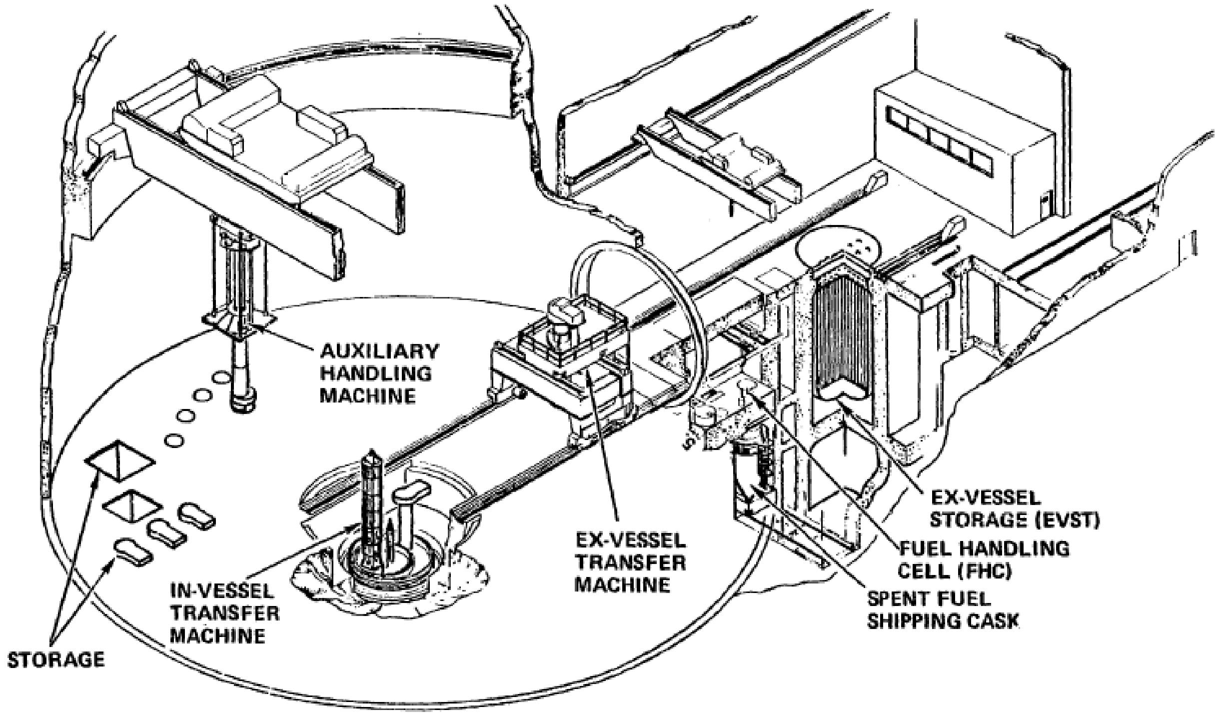


Figure 9 CRBRP Refueling System

The reactor closure head, shown in the figure below, consisted of three rotating plugs mounted in the vessel flange. The rotating plugs were provided to give access for the IVTM to the in vessel assemblies (fuel, blanket, control, and shield) through a port in the small rotating plug and allow the IVTM to transport such assemblies inside the reactor vessel into a Core Component Pot (CCP) which was placed in a transfer position located outside the core barrel. The CCP was provided to keep fuel, blanket, and control assemblies covered with sodium for cooling purposes while they are being withdrawn from the reactor and into the EVTM. After a core assembly had been loaded into a CCP, the large rotating plug could be adjusted to place the EVTM directly over the transfer position, giving the EVTM access to the CCP containing the selected core assembly. This arrangement is rather typical with one important exception – most sodium cooled reactors have just two rotating plugs, which give complete access to all core locations. The reason CRBRP had three harkens back to an early decision to use the FFTF hydro head on the project, fixing the reactor vessel diameter to FFTF's 20 ft. The idea was that the FFTF hydro head could ultimately be used as the material for fabrication of the CRBRP head, an idea that was never proven, since the project was terminated before the head was fabricated. Had the vessel diameter been a little larger, two plugs would have sufficed, but at 20 ft., it turned out to be necessary to have a third plug to give access to all the core assemblies and the transfer position. Eliminating the small rotating plug by giving freedom to the vessel diameter is an obvious and easy cost reduction measure and is identified here as CRM 9.

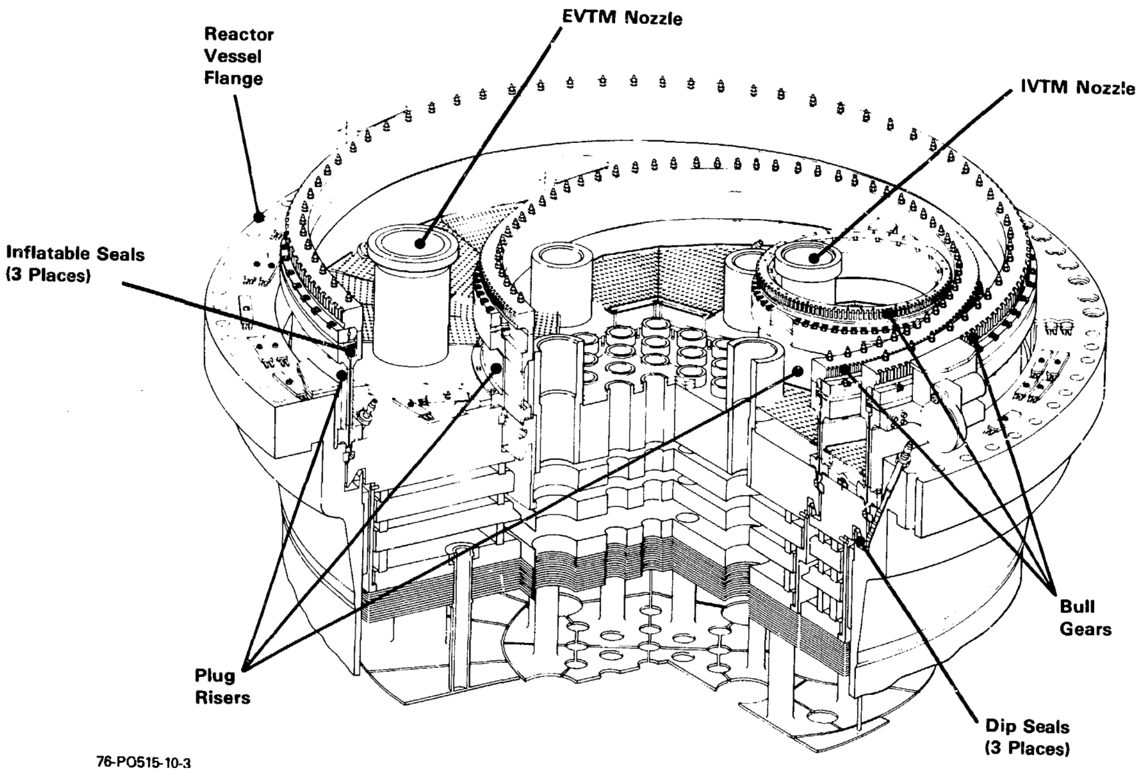


Figure 10 CRBRP Closure Head

Even without the small rotating plug, it is evident from the figure that this head system is complicated and badly congested. To make matters worse, the control rod drive mechanisms are not shown, nor are the UIS jacking mechanisms, nor are any of the gas purge lines, nor are the plug drive motors, nor are required electric power cables. Congestion at the heart of the plant slows and complicates construction, operation, and maintenance.

Consider the removal of a spent fuel assembly. First the head gas seals are deflated, the head freeze seals are melted, the control rod drivelines are disconnected and raised, and the upper internals structure is raised about nine inches by a jacking mechanism on the reactor head to permit rotation of the plugs. Using the polar crane, a floor valve and adapter is installed over the IVTM port. A typical floor valve is shown in the figure below. The Auxiliary Handling Machine (AHM) couples with the IVTM port adapter (not shown) and the space between the AHM valve and the IVTM port valve is evacuated and purged with argon, many times if necessary. The valves are then opened, the AHM then removes the IVTM port plug from the SRP through the floor valves, the valves are then closed, and the AHM stores the plug and transports the in-vessel section of the IVTM to its location over the SRP port. The same valve purging sequence is followed, the in-vessel section installed, the AHM removed and stowed, and the drive section of the IVTM is installed using the in-containment crane. Once the IVTM is fully installed, the IVTM then has access to the in-core assemblies. Next, a core assembly is grappled by the IVTM, the IVTM raises the assembly to clear the core, and the rotating plugs are activated to transfer the assembly from its existing lattice position to a transfer position located between the core barrel and the reactor vessel. This step is taken so the IVTM can lower the core assembly into a core component pot (CCP) which had previously been installed in the transfer position. During these operations, the equipment hatch between the RCB and the RSB is removed.

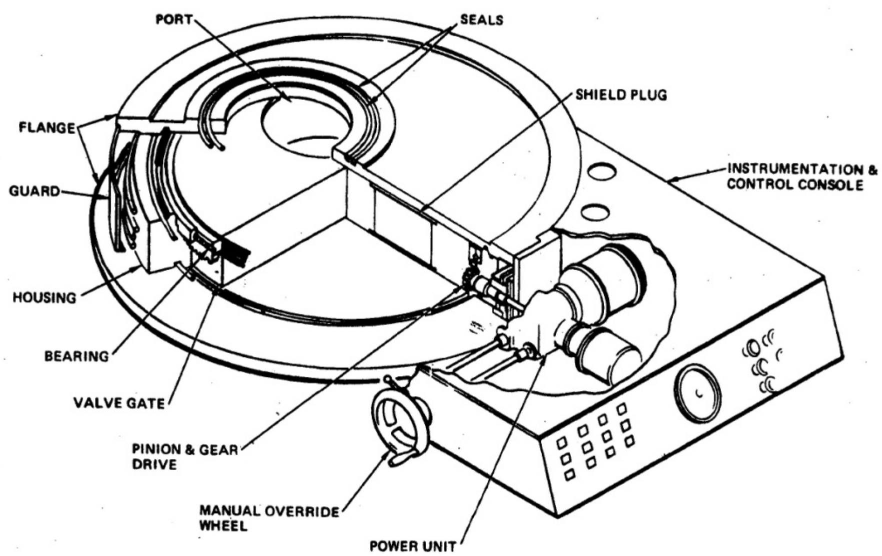


Figure 11 Floor Valve

It should be noted that in vessel transfer machines come in various configurations. The FFTF used a machine with an offset arm to improve its reach to the transfer location. An alternative to the offset arm is the pantograph, which is basically a collapsible offset arm. An offset arm or pantograph could be configured to replace one of the rotating plugs. On the CRBRP project, there was an objection to both the offset arm and the pantograph. Since the offset arm is obliged to remain in the reactor where it is exposed to the full range of reactor transients accompanying operation, there was a concern that it may become stuck requiring removal. To remove the offset arm would require removal of the rotating plug through which it penetrates – a major operation involving a lengthy shutdown. The project participants saw this as a very undesirable prospect and objected to the offset arm. A pantograph could be removed from the reactor after refueling and not be exposed to the reactor operating environment. However, the pantograph was considered undesirable out of concern that it could become stuck in the extended position similarly requiring an extended shutdown for recovery. So, the CRBRP project adopted a straight through in-vessel transfer machine (IVTM).

The spent fuel is transferred to an ex-vessel storage tank (EVST) located outside containment in the Reactor Service Building. To get outside containment, the EVTM must pass through the equipment hatch in the containment building wall. The hatch is about 75 ft. in diameter.[30] After the hatch is removed, it is necessary to install bridge rails for the EVTM. Similar to the reactor, a floor valve is attached to one of many ports in the EVST with an adapter that mates to the EVTM. Assemblies in the EVST, both new and spent, are loaded into receptacles in a carousel

30Preliminary design on this hatch was not completed at the time of project cancellation. It seems likely the hatch could have been smaller.

which rotates inside the tank. The receptacles are arranged in concentric circles which are accessed by the ports at the top of the EVST, one for each concentric circle. Once the EVTM has finished with an entire circle, the floor valve on the EVST is moved to the next circle of interest. The EVST is filled with sodium at some nominal temperature (around 400°F.) and is actively cooled. On CRBRP, there were two levels for storage in the EVST and a total capacity of over 800 assemblies could be accommodated. For CRBRP, this would have been enough storage for approximately three full core loads.

Another refueling facility in the Reactor Service Building is the Fuel Handling Cell (FHC). On CRBRP, this facility did not proceed beyond conceptual design. The purpose of the FHC was to provide for loading spent fuel to a shipping cask and accept new fuel prior to its loading into the EVST. The CRBRP project had never provided for sodium removal of spent fuel or shipment of spent fuel in water cooled casks. There was no shipping cask design that was developed to accommodate the plant's spent fuel. At the time of the project's termination, there was no provision for removal of the hatch between the containment and reactor service buildings. While none of these were intractable problems, it would seem reasonable to expect that any shipment of spent fuel that travels over public roads would be required to occur in water cooled casks so removal of the sodium from the spent fuel would need to be provided for.

Returning to the reactor head, figure 12 shows some of the details of the head riser. With the small plug eliminated, there would remain a large rotating plug supported by the reactor vessel flange and an inner rotating plug supported by the large rotating plug. The IVTM nozzle would be located adjacent to the inner plug riser. The inner plug houses the control rod ports and the support columns for the Upper Internals Structure (UIS). Supports for the suppressor plate are located on both the large and inner plugs. The drive motor for the inner plug is mounted on the large plug. At the top of the risers are located inflatable seals, which are normally inflated but are deflated when the plugs are being rotated. Sealing during plug rotation is provided by sodium seals located at the bottom of the risers. During plug rotation, the seals are purged with argon to prevent sodium oxidation. The weight of the inner plug is transmitted to the large plug by ball bearings at the top of the risers. The plugs are driven by drive motors whose pinions engage teeth on bull gears on the outsides of the tops of the risers.

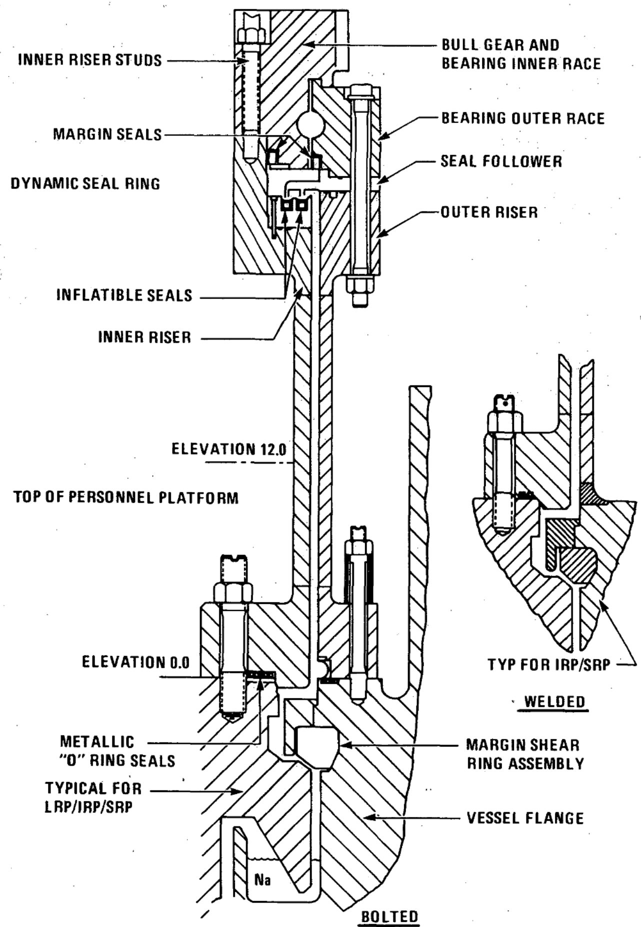


Figure 12 Head Riser system

The EVTM and IVTM are shown below as Figures 13 and 15. It is the task of the EVTM to move fresh and spent fuel back and forth between storage and the reactor. The EVTM is a vertical tube including a cold wall, which serves as a heat sink, into which the fuel assemblies are loaded and is supported by a gantry trolley. The trolley rides on rails with a span of 13 ft. The trolley is supported by a gantry that rides on rails with a 30 ft span. At the bottom of the EVTM tube is a valve, normally referred to as a floor valve, which isolates the environment inside the EVTM from the outside air. At the top of both the reactor and the EVST is mounted yet another floor valve with an adapter that mates to the EVTM. The EVTM begins the refueling operation at the EVST, by obtaining a new fuel assembly contained within a Core Component Pot (CCP). The EVTM then moves to the reactor where the Large Rotating Plug (LRP) has been rotated to be aligned with an unoccupied transfer position. When the EVTM has mated with the adapters and floor valve atop the LRP, the air space between the two floor valves is first evacuated then purged with argon. Once the oxygen has been satisfactorily removed possibly requiring repeated evacuations and purges, the two floor valves are opened, and a hoist located within the EVTM and carrying the new fuel assembly in its CCP is lowered into the unoccupied transfer position. The grapple is then partially retracted.

At this point, it becomes necessary to describe another interesting (and complicating) feature of the CRBRP refueling system. A Rotating Guide Tube (RGT) was installed in the reactor directly above the transfer positions. The Reactor Fuel Transfer Port (RFTP) adapters, when installed, mate with the RGT and include the RGT drive system. The RGT had an offset such that rotation of the RGT positioned the offset at either of two adjacent transfer positions. This RGT is shown in the figure immediately following the EVTM. This feature permits the EVTM and the LRP to remain stationary while a new fuel assembly is being loaded and a spent assembly is being withdrawn. This RGT drive system is one more item to congest the head area.

Following the loading of the new assembly, the RGT is rotated so that the lower offset tube is positioned over the spent assembly. The grapple is lowered and engaged to the CCP holding the spent assembly, the CCP and spent assembly are lifted into the EVTM, the floor valves are closed, the EVTM disconnects from the LRP and proceeds back to the EVST. The engaging and disengaging process is accomplished with machinery installed on the EVTM. Decay heat from the spent fuel assembly is radiated to a "cold wall" which is actively cooled by forced convection of air on its outside surface. The EVTM is self propelled by electric motors as it lumbers back and forth from the reactor to storage. Its total weight including gantry, trolley and the EVTM itself was 270 tons. It was designed to be 35 ft high. All this mass was provided to transport an assembly that was 14 ft long and weighed 450 lb. Electric power is provided to the machine by cables that wind and unwind as the machine moves back and forth.

The SRE/Hallam Refueling Approach

There are two major simplifications that can be made to this system drawing on previous liquid metal designs. One would eliminate the intermediate rotating plug leaving only one large rotating plug and the second; more dramatic simplification would eliminate both rotating plugs and adopt open vessel refueling. Both schemes would eliminate the IVTM, and with it the need for the transfer volume in the reactor vessel with attendant reactor vessel shortening. Both would also eliminate the need for a transfer position outside the core barrel and would eliminate the core component pots. These three items (shortened RV, no CCPs, and no IVTM) constitute CRMs 11, 12, and 13.

At this point, it would be propitious to bring up an important feature described in Appendix 2C under core design. An objective in the core design was to achieve ten years between refuelings. Since the expected average fuel burnup to achieve ten year refueling intervals is about 15% with peak burnup in the neighborhood of 30%, this objective was the driver for proposing vented fuel leading to first cost reduction item of foreshortened fuel assemblies. This ten year refueling interval shall be herewith identified as CRM #10. It makes long refueling outages feasible since they occur so seldom, relaxing refueling requirements that had been imposed on CRBRP in the interest of minimizing refueling shutdown time.

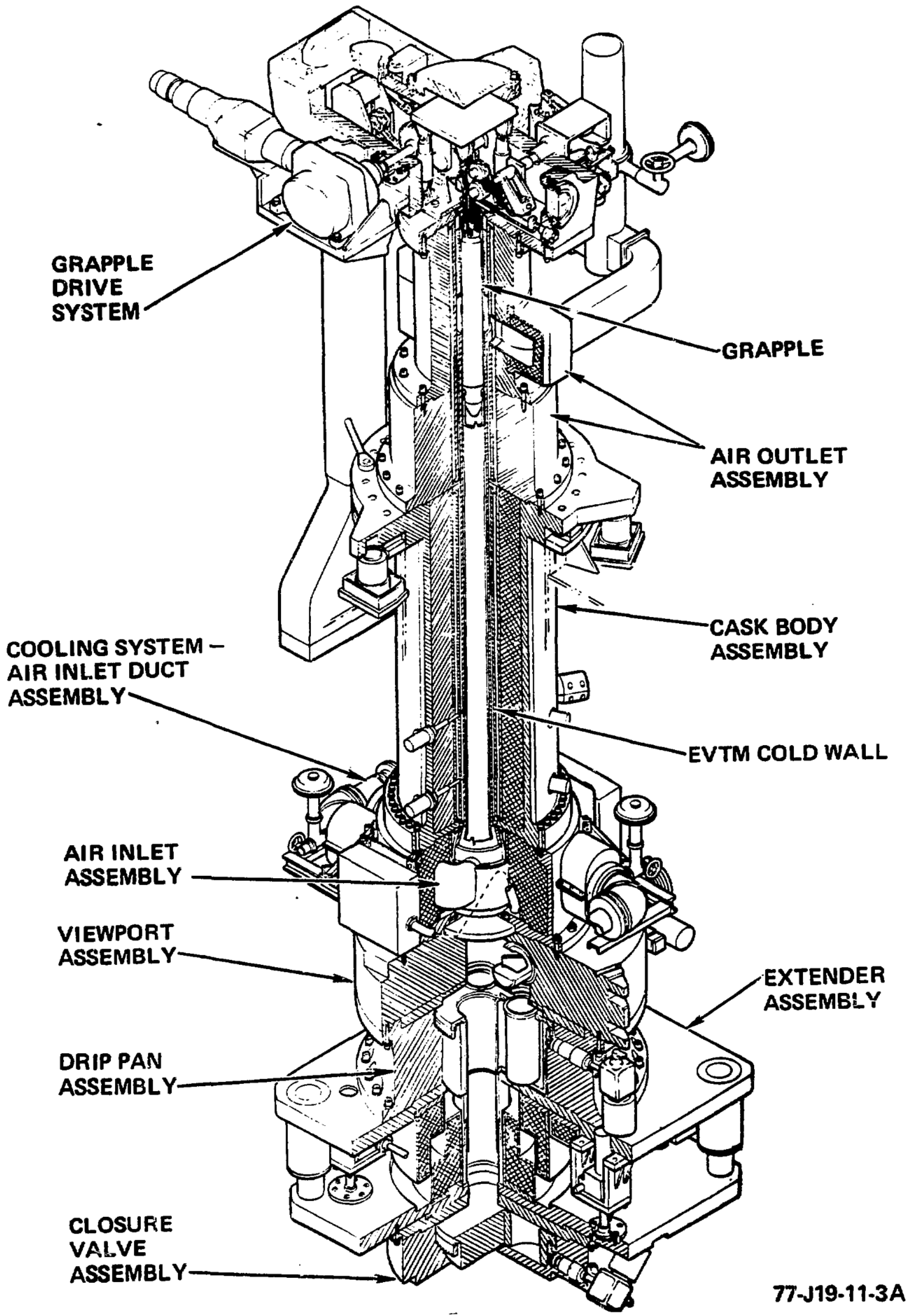


Figure 13 CRBRP Ex Vessel Transfer Machine

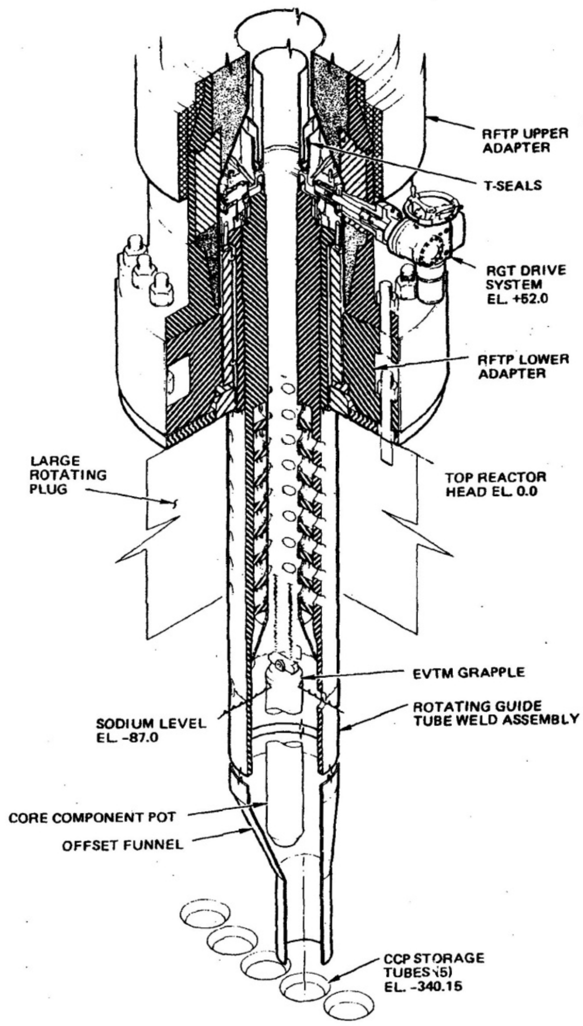


Figure 14 Rotating Guide Tube

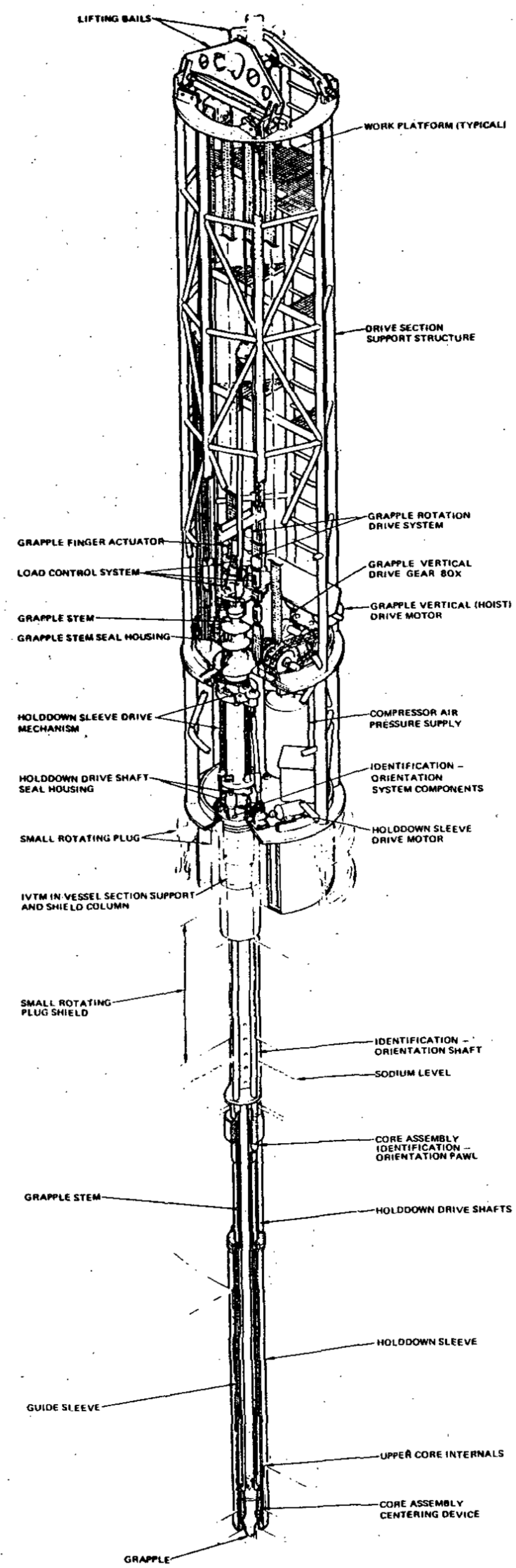


Figure 15 CRBRP In-Vessel Transfer Machine

The first idea, a single rotating plug, draws on the refueling design of the Sodium Reactor Experiment (SRE), a graphite moderated sodium cooled reactor. The SRE had a single rotating plug centered over the core with ports located in the head in such a fashion that by selection of a port and rotation of the head it would be possible to position a port over any individual core assembly. The SRE had a core design that was very different from the LMFBR design, but features of this approach could be adapted to the design approach being considered herein. The significance of the SRE system (as well as that of Hallam, for that matter) for this discussion is there was no in-vessel transfer machine so no need to provide the volume above the core to make such transfers. Core assemblies were removed directly from the core into a fuel handling cask where they were transferred to a cleaning cell and a storage cell. This approach has some of the key advantages of open vessel refueling in that it permits a shortened vessel and eliminates the transfer positions. If open vessel refueling (discussed later) proves to be too much to swallow, this scheme would be a good alternative. A photograph of the SRE reactor head and refueling machine appears below.[31]

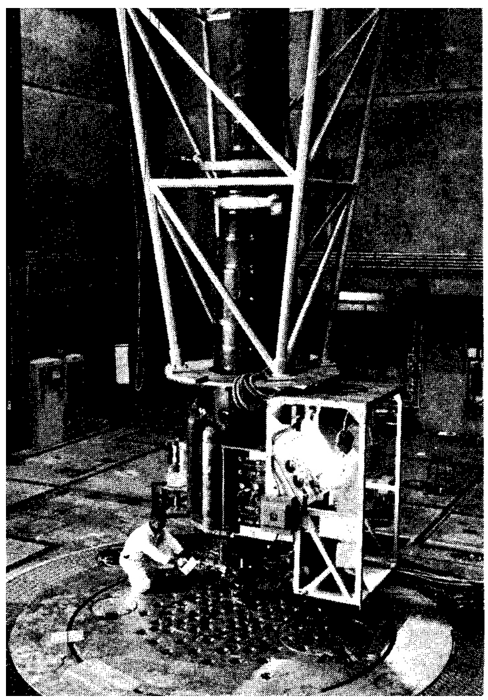


Figure 16 SRE fuel handling system

A large scale fast reactor has many more (and smaller) assemblies than SRE, but likely the same approach could be used with a centered rotating plug similar to the SRE plug described above. Because of the 12-fold symmetry of a hexagonally patterned core, it would not be necessary to have a port for each core assembly, assuming the rotating plug is centered over the core. For example, the core described in Appendix 2C has 847 fuel, blanket, and control assemblies would require at most 74 ports, many of which could be eliminated depending on the tolerance of the transfer machine for misalignment. In this regard, there is a tradeoff between the size of the ports and the number required. The larger the ports, the fewer will be required. There would be

31Starr, Chauncey; Dickenson, Robert W.; *Sodium Graphite Reactors*, Addison Wesley; 1958.

some details needing to be worked out, e.g. the access ports to the first and second rows are adjacent, so special ports would be needed there; the CRDM penetrations would need to be compatible (i.e. not excessively large) and the drivelines, which would move with the rotating plug, retractable so as to clear the core assemblies. It may prove necessary to modify the core design somewhat so that the control assemblies do not appear in symmetric positions. There is a more extensive treatment of this subject in Appendix 2G.

The fuel handling machine would be obliged to furnish the function served by the core component pots by having some sort of shroud, closable at the bottom that would be lowered into the sodium pool through one of the head ports. The shroud would be evacuated, presumably of argon or helium, drawing sodium up into the shroud, then a grapple would be lowered through the shroud, engaging the core assembly, the core assembly would be withdrawn into the sodium filled shroud, the lower valve would be closed, then the shroud containing the sodium covered core assembly would be fully withdrawn into the EVTM. Assemblies that are 7.19 in. across the flats will be 8.30 in. from corner to corner. Allowing 0.125 in. thickness for the shroud, the port diameter would be about 9 in. for a cross sectional area of 63.5 in.^{2}, which compares with the core assembly cross sectional area of 44.77 in.^{2}. There will be plenty of room on the head for these ports. The biggest complication with such a scheme is the UIS, which would interfere with access to the core assemblies. It would be necessary to design the UIS (as well as the suppressor plate) so that there would be access to the core assemblies below each port in the head. This would not have been a feasible approach on CRBRP since the UIS was obliged to provide backup hold-down for the core assemblies. There is no need for hold-down in the design approach herein proposed. The suppressor plate would also need to be supported from the rotating plug and have holes aligned with all the ports. Alternatively, the suppressor plate could consist of two parts, one connected to the fixed portion of the head and the other to the rotating plug.

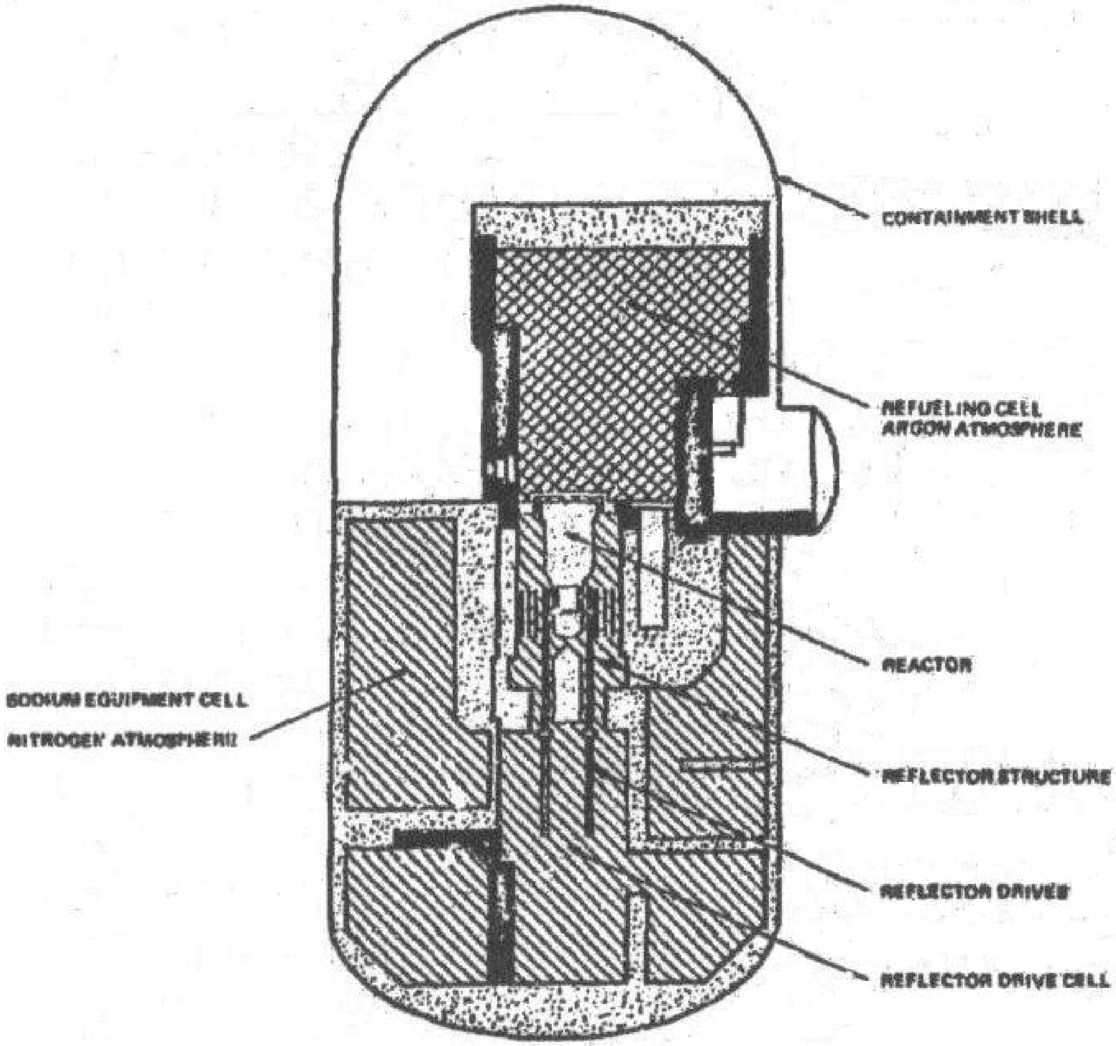
Since the rotating plug would probably be about 22 ft. or more in diameter to allow access to shield assemblies, space could probably be found for at least 50 of these ports. For the core design shown in figure 45, a minimum of 18 ports would be required. A number between 30 and 50 would probably prove adequate. Assuming a pattern could be found that would provide complete coverage of all the core assemblies, there would be no further need for the intermediate rotating plug, eliminating the IVTM and allowing the reactor vessel to be shortened by another 12 ft. Elimination of the intermediate rotating plug constitutes CRM 14. A conceptual port pattern has been devised and is described in more detail in Appendix 2G. The pattern provides 32 ports and requires a tolerance of plus or minus 0.8 in. corresponding to a port diameter of 10.1 in. Seven of the ports access a single assembly (within the 1/12 core sector, therefore accessing 12 assemblies in the whole core) and can accordingly be reduced in diameter. Other patterns undoubtedly could be devised.

The EVTM complexity is increased somewhat by the provision of the valved shroud and siphoning system. However, if the EVTM is operating within a refueling cell (described below), the shielding, which constituted a great deal of the weight of the CRBRP EVTM, could be significantly reduced or eliminated. The shortened fuel assemblies should reduce the length of the machine an equivalent amount. The machine would have a short travel distance to the EVST and would not pass through an equipment hatch.

It needs to be pointed out that for a removable assembly diameter of 22 ft, (the core diameter is 18 ft. and is surrounded by 2 ft. of removable shield assemblies) if two rotating plugs are used, the vessel diameter must grow to at least 44 ft. For two rotating plug refueling, the Upper Internals Structure (UIS) needs to be rotated out of the way of the assemblies to be refueled. One might design the UIS so it does not cover the shield assemblies, but the point is that a reactor vessel diameter in the 30 ft. range is not consistent with two rotating plug refueling unless the UIS is eliminated.

The SEFOR Refueling Approach

The ultimate in simplicity that eliminates the need for any rotating plugs was that selected for the Southwest Experimental Fast Oxide Reactor (SEFOR). In SEFOR, an argon inerted, shielded hot cell was located above the reactor provided with lead glass windows and periscopes, mirrors and television camera so that fuel elements could be removed from the reactor and examined. The cell was used for refueling and for transferring fuel from the reactor to an irradiated fuel storage tank located in the cell and from the fuel storage tank to a cask for shipment off-site. All operations within the cell were visible to the operators. An outline figure of the approach used at SEFOR follows:



- Figure 17 SEFOR Vessel and Refueling Cell

The following is drawn from a reference on SEFOR operation: "Extensive fuel transfer operations were accomplished, including approximately 2000 fuel transfers. This experience demonstrated the ease which fuel can be visually located and then grappled and transferred by remote mechanisms. Fuel rod transfers in a 15-min. time interval are limited only by the crane speed. Reflections from the silvery sodium pool contribute significantly to the ability to see and identify core locations. This favorable operation has confirmed the design basis for open-pool, hot-cell refueling, and gives increased confidence to LMFBR designs utilizing this form of refueling. Broad experience was also obtained with the refueling cell man-access suits. Many refueling cell maintenance and repairs have been accomplished, and a high degree of competence has been achieved with men working within the suits. More than 60 cell entries have been made and over 80 man-hours of in-cell operations logged."[32]

32Arterburn, J. O.; Billuris, G.; Kruger, G. B.; *SEFOR Operating Experience*; ASME Nuclear Engineering Conference; Mar.7-10, 1971

The idea of a hot cell with lead glass viewing windows, cranes, and manipulators, located directly above the reactor is foreign to anyone with commercial nuclear reactor experience and it creates the impression of an experimental facility. Manned entry through an air lock raises questions of ready accessibility and personal safety. Contamination levels in such a facility could be high and decontamination would be tedious and difficult. It is probably for these reasons that the SEFOR refueling approach was never pursued on any follow-on plant. However, for the design approach considered here, the cell need not be inerted except on rare refueling occasions or when new or spent fuel is being shuffled from row to row in the Ex Vessel Storage Tank (EVST). Following fuel handling operations, the cell would be deinerted and decontaminated.

It is important to point out that there is relatively little description of SEFOR in the open literature available on the internet. From the information available, there is no way to tell what the Upper Internals Structure looked like, how it was supported, and how it was dealt with when the head was being removed. Another area of uncertainty is the control rod mechanisms and how they were accommodated during head removal. It appears, from Figure 17, that the control system was operated from below the reactor and consisted of reflectors rather than absorbers. SEFOR was designed by General Electric, so somewhere in the General Electric document control system there must be answers to these types of questions. Since General Electric submitted a bid for the CRBRP design contract, their proposal may have included open vessel refueling, in which case there could exist another rendering of this type of approach at a larger scale than SEFOR.

If the single rotating plug option were selected, the UIS would be supported from the rotating plug, jacked up to a storage position prior to head rotation, and would rotate along with the inner plug, the suppression plate, and the control rod drivelines. Both the single rotating plug and the open vessel options accomplish a significant reduction in the reactor vessel height and lead to simpler refueling systems. Both capitalize on the loop-type design approach in a way unavailable to pool-type designs. Of the two, the open vessel approach is probably the more economic. Of the two, the open vessel approach is more likely to arouse virulent opposition. To avoid this, the single rotating plug is selected for the current design approach. More will be said about the open vessel approach and the two will be carried forward in parallel in the interest of understanding the criteria for selection between the two. Once a reliable cost estimate has been accomplished for the base plant design, and the cost impact of adopting the single rotating plug is known, the entity responsible for proceeding with plant construction will be armed with better information and thereby be in a better position to make a considered decision among these two alternatives.

The EVST arrangement proposed here is applicable to either refueling system arrangement. The tank itself is smaller (no CCPs), the large hatch between the containment and the Reactor Service Building is eliminated, the Reactor Service Building itself is reduced to little more than a Transfer Room (plus whatever space is required to house necessary auxiliaries, which would be considerable), a fuel handling cell, and a shipping room, and is represented here as cost reduction measures 15, 16, and 17. These ideas (without a single rotating plug) are captured in Figure 19.

The EVST should be cooled by natural circulation capable of operation under station blackout conditions. There is little advantage to providing natural circulation cooling to the reactor if the spent fuel in the EVST is exposed to potential failure on the occasion of a station blackout. A two tier tank as was designed for the CRBRP should be used to reduce tank diameter. All tank penetrations should be through the head in the interest of simplification. A guard vessel should be provided to eliminate leak issues. In the interest of promoting better natural circulation for the EVST, the fuel assemblies which have the highest decay heat load should be loaded into the lower tier so as to lower the thermal center of the contents in the tank. The maximum heat load for the EVST is about 3000 KW, which corresponds to the core decay heat five months after reactor shutdown. Two Na/NaK heat exchangers would be located above the tank with two natural draft air cooled heat exchangers provided to remove heat from the NaK. It will be necessary to provide the EVST with a downcomer to return the cooled sodium to the bottom of the tank. The downcomer could either be a set of pipes or the outer annulus of the tank itself. Since the EVST will be more tightly loaded than the CRBRP EVST, it may be required to provide built in neutron poison to achieve criticality control.

For the open vessel refueling case, the fuel assemblies' decay heat will be about 10 KW two weeks after shutdown and about 5 KW three months after shutdown. The decay heat from blanket assemblies would be about 55% lower. 5-10 KW is too much power to handle naked fuel assemblies inside the refueling cell without cooling of some form. Since there are no core component pots to load assemblies in to, the in-cell core component handling device must furnish the equivalent. One approach might be to lower a handling shroud (similar to the shroud described above in connection with the EVTM with a closed top) which contains the core component grapple to a foot or so above the assembly being withdrawn, siphon sodium into the length of the shroud, lift the core component into the shroud, valve off the bottom of the shroud, transfer the core component to the ex-vessel storage tank, and reverse the procedure. The shroud may require external fins to provide adequate radiative cooling to the fuel handling cell atmosphere.

EVST Placement and Refueling Design/Operation

The ex-vessel storage tank needs to be accessible from both the Refueling Cell and the Transfer Room. Since this is essentially what was done on the Superphénix design for the EVST, this would be nothing new.[33] The Superphénix fuel handling system is shown on Figure 18. Superphénix had rotating plugs for in-vessel transfer and had an A-frame arrangement for transfer from the vessel to the storage tank. In Superphénix, only the outer row of the EVST is accessible from the reactor service building handling room. A manipulator on the containment side moved spent and new assemblies between the outer row and interior rows within the EVST with the help of a rotating carousel on the EVST. The carousel drive for the EVST was located within containment.

33It would be similar to the Superphénix design with one important distinction. The tank would be cylindrical without the bulge for the A-frame. A cylindrical tank would be smaller in diameter (no core component pots), shop fabricated, and much less likely to experience the type of failure that befell the Superphénix EVST. The Superphénix EVST was fabricated in the field.

For the approaches proposed here, a refueling cell encloses the top of the reactor vessel and the side of the EVST nearest the RV including the carousel drive. During refueling operations, the refueling cell would be inerted. Either a Fuel Transfer Machine (FTM) or, for open vessel refueling, the fuel handling cell crane would access RV core components and each of the rows of the EVST during the refueling operation. The EVST would have been preloaded with new fuel & blanket assemblies as necessary to accomplish refueling. There would be no concurrent operations in the Transfer Room during the refueling operation.

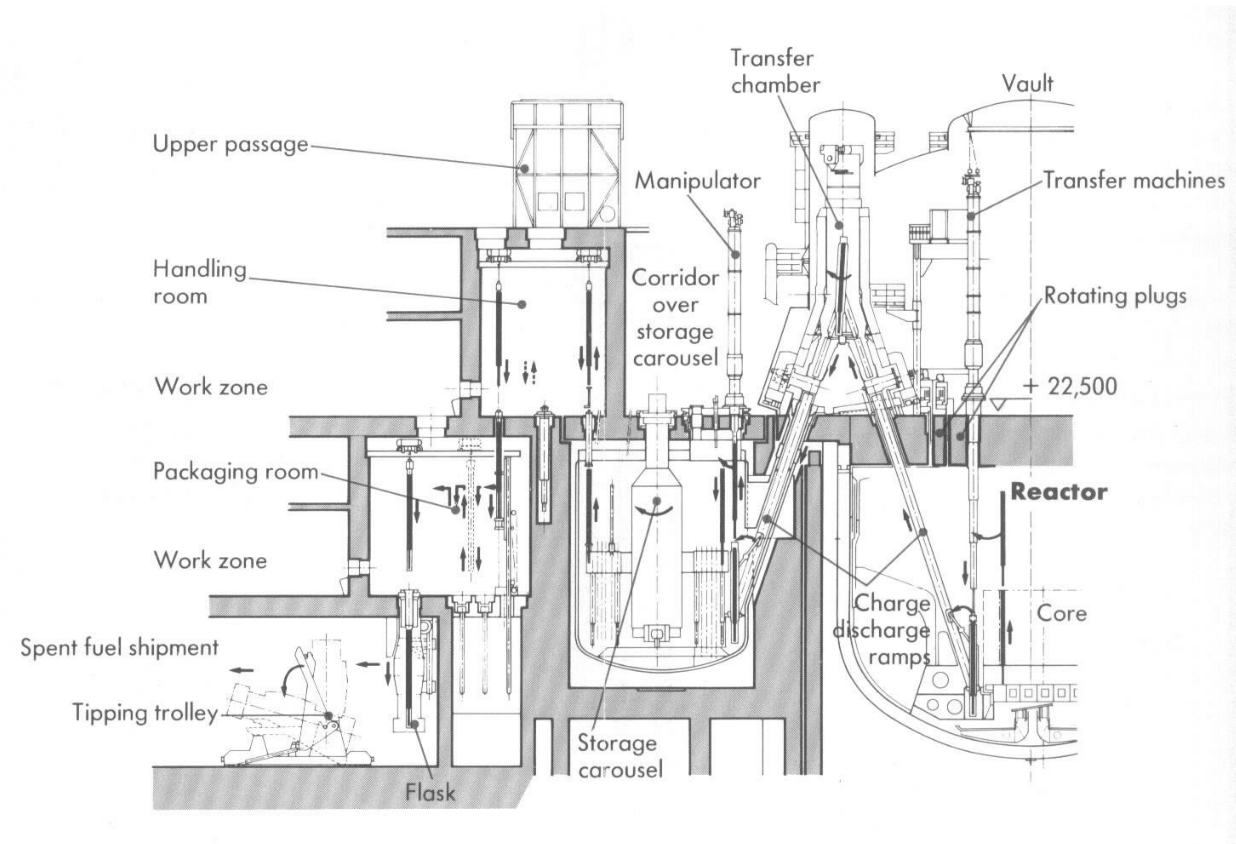


Figure 18 Superphénix Fuel Handling System

For the case shown in Figure 19a, the Fuel Transfer Machine (FTM) would be handled by the Refueling Cell crane, as was done on the SRE and Hallam. There would be room for a FTM that had an overall top to bottom length of up to 16 ft. Since the core assemblies length is 11.5 ft., 16 ft. may prove to be inadequate, requiring that the Refueling Cell roof be raised, a rather modest penalty. The FTM could be mounted on rails, but that would require that the RV be lowered about 8 ft. so the rails would clear the rotating plug riser and UIS jacking mechanisms (see fig. 21a). The FTM will require electric power to operate a cooling circuit, the shroud that is lowered into the RV and the EVST, the shroud door, a drip pan installed below the shroud, and instrumentation. This electric power would be furnished by the crane and if the FTM is mounted on rails, the rails would be fully within the inerted refueling cell and could be electrified. Either case would be a huge improvement over CRBRP, where a cable run to the EVTM was provided which had to move with the EVTM.

The CRDMs require two points of connection/disconnection. The absorber assemblies must be capable of being disconnected from their drivelines so that they can be replaced when needed. The CRDM driver mechanisms must be capable of being disconnected from their drivelines so that the mechanisms can be removed during refueling operations to permit access to the fuel and blanket assemblies. During refueling, the driver mechanisms would be stored on the refueling cell floor. Should any absorber assemblies require replacement during refueling, the FTM must be provided with the capability to remove their associated driveline.

The following is a listing of the major steps required to refuel:

- Shutdown reactor.
- Wait two weeks for Na^{24} to decay.
- Enter Refueling Cell and disconnect control assemblies.
- Jack up Upper Internals Structure (UIS).
- Connect Fuel Transfer Machine (FTM) to selected crane and electric power.
- Inert Refueling Cell.
- Deflate riser mechanical seal and melt the sodium seal.
- Test head rotation.
- Remove and store Control Rod mechanisms and drivelines and plug associated head head penetrations.
- Use Plug Handling Machine to remove selected head access port plug and install valve.
- Same as above for EVST.
- Position head over selected assembly, access assembly with FTM, withdraw assembly from Reactor Vessel (RV) and transfer assembly to EVST.
- Return new assembly to RV.
- Access other core assemblies accessible from selected RV head port (See Appendix 2G, Head Port Layout) and return new assemblies to the position vacated. When all planned core positions have been replaced with new assemblies, activate Plug Handling Machine to remove valve and insert plug in port.
- Select another RV head port and continue as above. There are 32 separate heads port for which this procedure is to be performed.
- At the completion of replacement of core assemblies, reverse procedure.

After suitable decay of the spent assemblies in the EVST, shipment of spent fuel could begin. The atmosphere in the Transfer Room would be inerted whenever the EVST port is opened or if handling sodium wetted spent fuel is occurring. The outer row of spent assemblies would be removed at the Transfer Room, cleaned with wet vapor nitrogen to remove sodium, and loaded into inerted spent fuel shipping casks. Five years after removal from the reactor, the fuel assembly decay heat will be down to about 1 KW. At such a low heat load, it probably will be possible to handle the assemblies in the Transfer Room without using the handling shroud. The FTM or the refueling cell crane will transfer spent assemblies from the inner rows to the outer row so they can be accessed in the Transfer Room. The operation will not be fast, but it need not be.

When the EVST has been emptied, new fuel and blanket assemblies can be brought in. This probably would not be scheduled until a year or so before the intended refueling outage to avoid unnecessary carrying costs of the new fuel. Since only the outer row of the EVST can be loaded from the Transfer Room, it will be necessary once again to use either the FTM or the fuel handling cell crane to load the inner rows. Sketches of the closed vessel and the open vessel refueling concept in elevation and plan views are shown below. The plan view is at the level of the floor of the refueling cell. Note that the support floor for the IHXs shown in the diagram will be 10-15 ft. above the refueling cell floor.

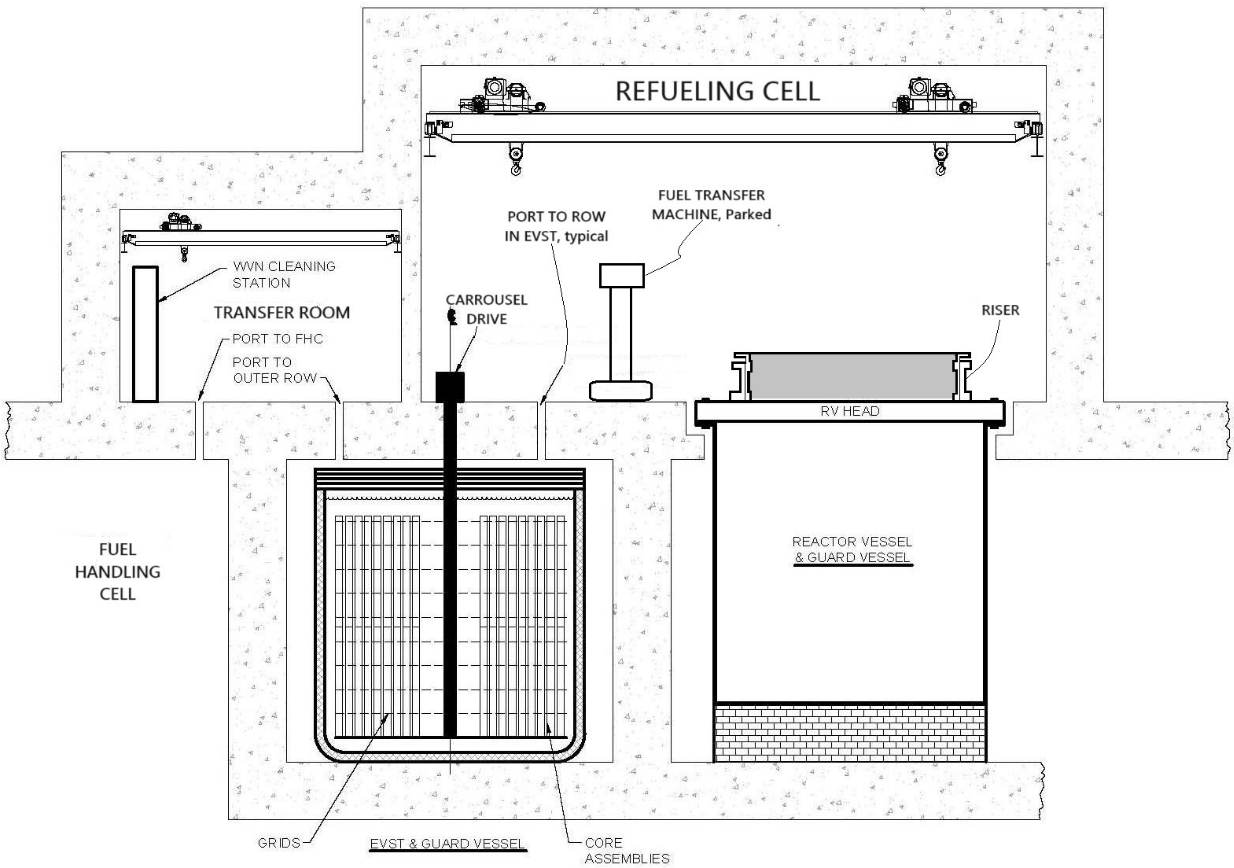


Figure 19a Elevation view of Fuel Transfer Machine refueling concept

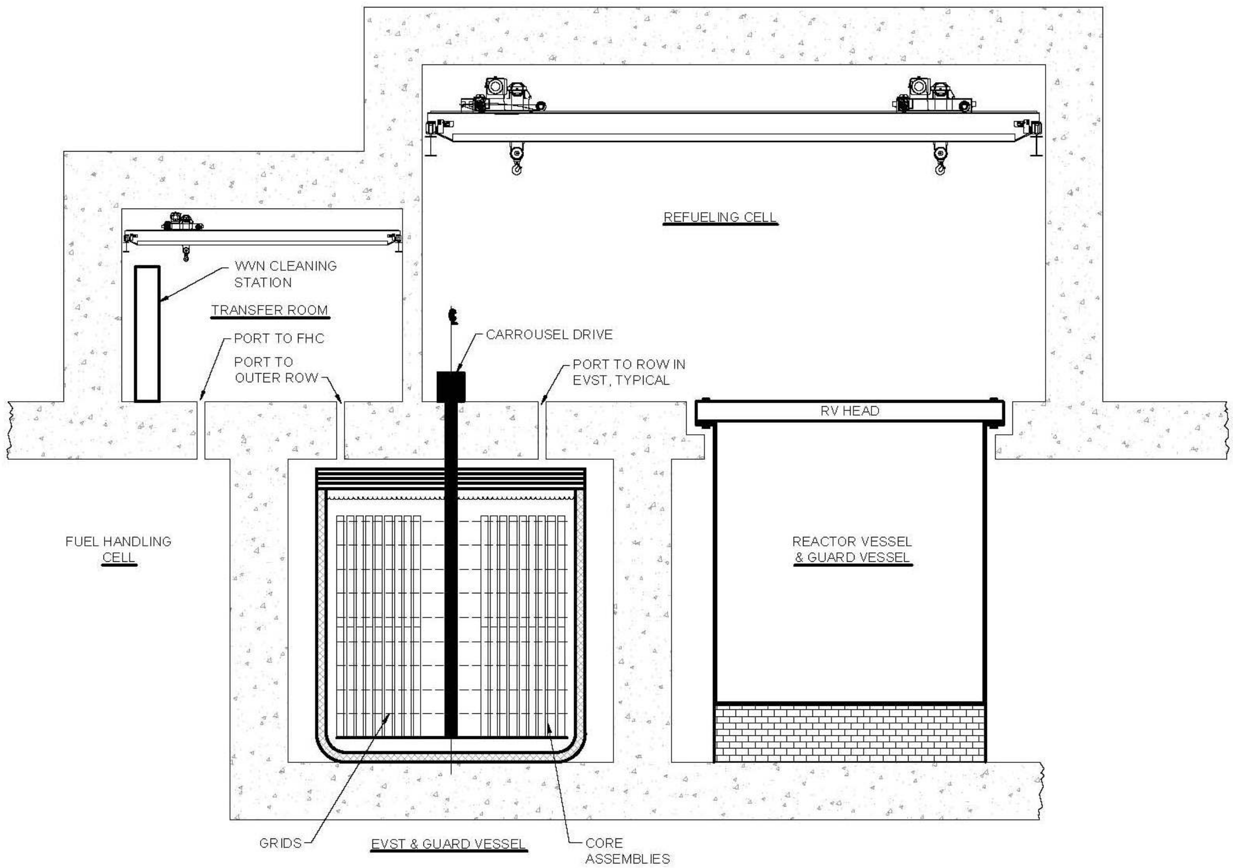


Figure 19b Elevation view of OVR refueling concept

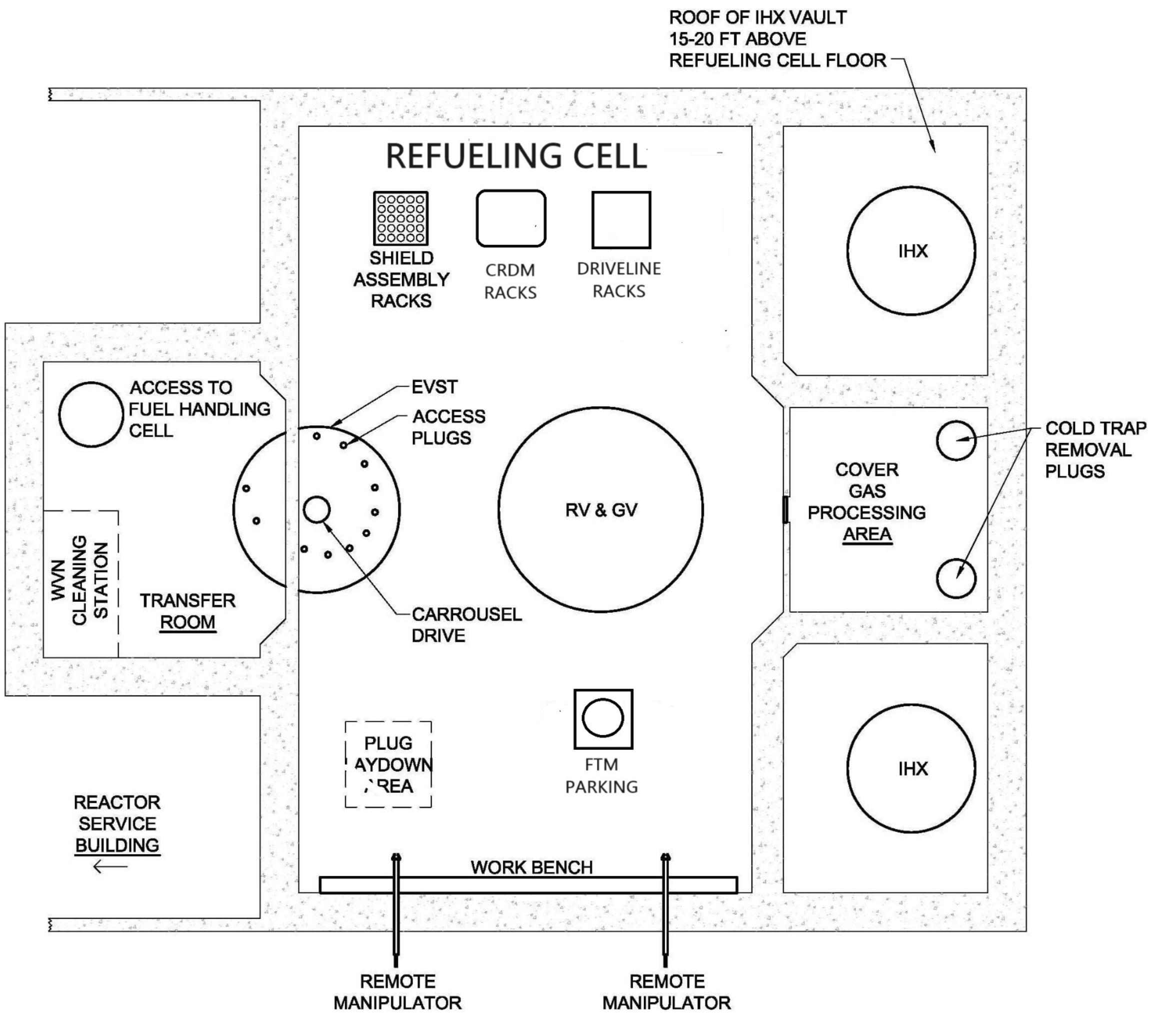


Figure 20a: Plan View of Fuel Transfer Machine refueling concept

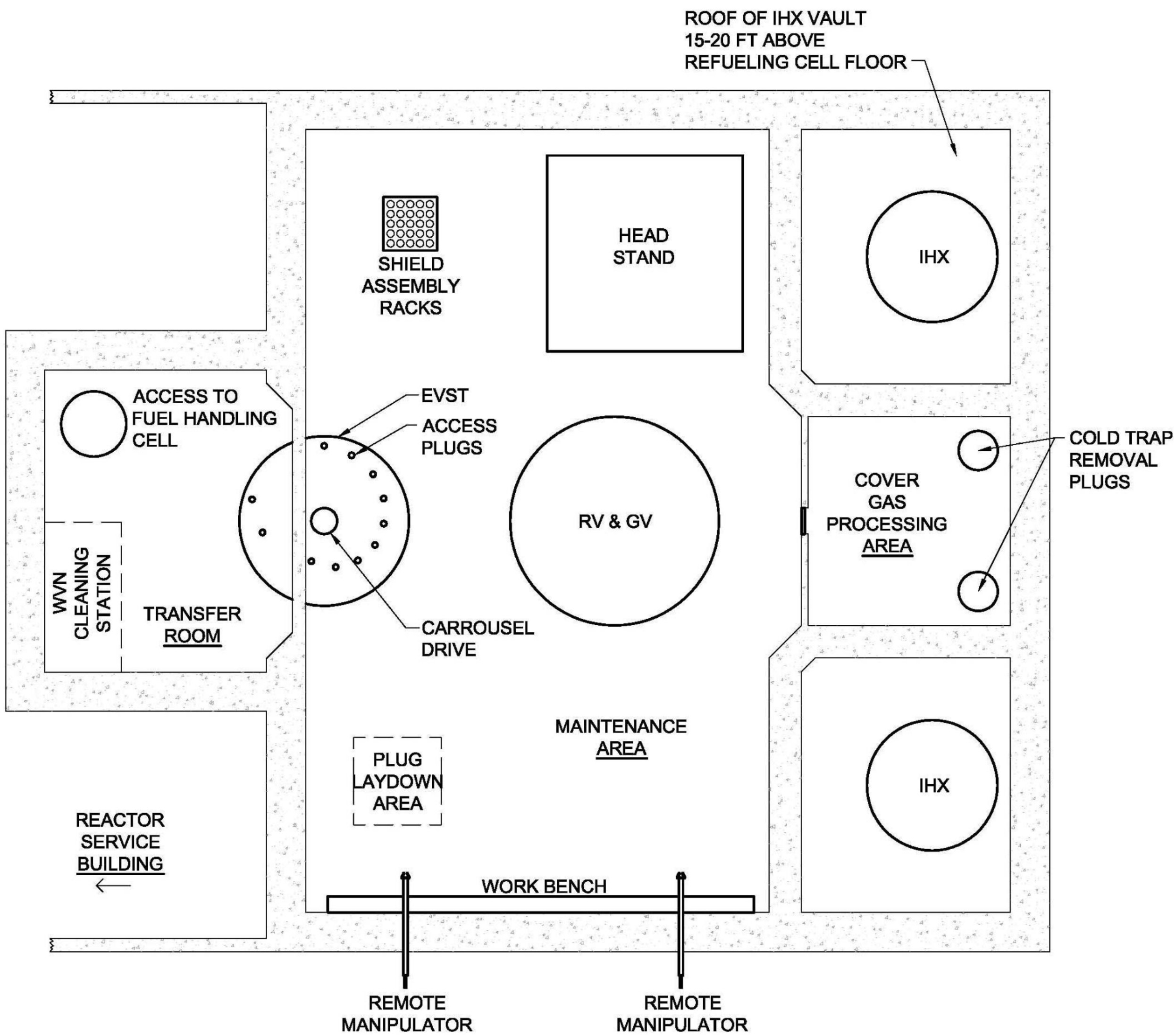


Figure 20b   Plan view of OVR refueling concept

It is worth considering the advantages of this refueling approach. The items below are made in comparison the CRBRP, but the list would not be much different if the comparison were made with Superphénix: Those options that are unique to open vessel refueling are identified with "OVR".

- The need for rotating plugs on the reactor vessel head is eliminated. This also eliminates the need for the systems required to seal these plugs during plug rotation or plant operation as well as the motors required to accomplish plug rotation. (CRM 18 OVR, but if a FTM is incorporated in the design, a single rotating plug is an improvement over two (or three) rotating plugs.)
- The plug risers, which house the seals and bearings for each of the rotating plugs, are eliminated greatly reducing head area congestion. The "design approach" requiring a single riser represents an improvement over two (or three) risers.

- The elimination of rotating plugs removes a source of misalignment of the control rod drive mechanisms improving confidence they will function as intended (see section 10). OVR
- The need for an upper internals structure jacking mechanism is eliminated. (CRM 19) OVR
- The need for shielding under the reactor head is reduced or eliminated, which reduces the length of the reactor vessel correspondingly (about 3-4 feet). (CRM 6)
- The need for space between the core barrel and the reactor vessel is eliminated, enabling the reactor vessel diameter to be reduced by at least 2 ft. (Eliminate transfer positions and required volume between the core barrel and the vessel wall – CRM 20)
- The need for a large inventory of core component pots (the inventory would be set by the number of positions in the EVST plus the transfer positions in the reactor vessel) is eliminated. (CRM 12)
- The need for valves to be mounted on the reactor head and the EVST is eliminated. (CRM 21) OVR
- The need for a machine to handle these valves is eliminated. (CRM 22) OVR
- The need for systems to purge the space between the valves is eliminated. (CRM 23) OVR
- The EVST diameter can be reduced since it will no longer be obliged to house all core assemblies contained within core component pots. Such reduction must be accomplished within criticality control constraints. (CRM 15)
- The EVST size would be set by the number of fuel, blanket, and control assemblies in a single core loading. For the example of the design being considered, that would be 847 assemblies – actually fewer storage positions than the CRBRP EVST was designed for.
- The reactor vessel can be shorter since there is no longer a need for clearance above the core for the horizontal translation of core assemblies. On CRBRP, this was a 14 ft. penalty in reactor vessel height. (CRM 11)
- The IVTM is eliminated. (CRM 13)
- The EVTM is eliminated and both it and the IVTM are replaced with a much simpler fuel handling cell crane. (CRM 24) OVR
- The auxiliary handling machine and plug handling machine are both eliminated. (CRM 23 & 25) A plug handling machine will be required for the single rotating plug option.
- The large hatch between the containment and the reactor service building is eliminated. (CRM 16) The rails through the containment and reactor service building and the bridge rails through the hatch are also eliminated. (CRM 26)
- The entire area under the head is accessible. There is an annular ring of area adjacent to the reactor vessel that is inaccessible with a straight pull IVTM. This may have implications for simplifying the reactor vessel design.
- The requirement for the reactor vessel to be top mounted is eliminated.

Many issues would require resolution of this proposed system for a large commercial plant.

- A provision needs to be made for uncoupling the control rod absorber assemblies. Presumably, the uncoupling operation would be performed two weeks after shutdown to allow for Na^{24} decay prior to inerting the refueling cell. This could be accomplished manually before the refueling cell is inerted.

- OVR -- Following uncoupling of the control absorber assemblies, the mechanisms and their drivelines would require removal. The mechanisms would probably have welded seals on the head, which would require cutting prior to mechanism removal. At the completion of refueling, these seals would require re-welding. During refueling, the uncoupled mechanisms and drivelines would require storing in the refueling cell.
- OVR -- The reactor vessel head, fabricated from stainless steel so as to have thermal expansion compatible with the RV, is 2 ft. thick and 30 ft. in diameter and is estimated to weigh about 400 tons. This is probably too much to lift with an overhead crane. Since the only reason for the head being 2 ft. thick is shielding, it could probably be fabricated out of four 6 in. slabs each of which is lifted individually. 100 tons is a more reasonable lift. Lifting fixtures would need to be attached to each piece.
- OVR -- Some primary coolant temperature needs to be selected for refueling operations. On CRBRP, refueling was accomplished at a primary temperature of 400°F, but with open head refueling, there will be an incentive to refuel at a lower temperature (probably 250-300°F[34]) to reduce the heat load to the refueling cell and to reduce the deposition of primary sodium vapor on the refueling cell walls. A lower refueling temperature will require a somewhat larger overflow vessel and greater reactivity control. It would also require highly effective cold traps. Another option may be to allow the refueling cell temperature to rise considerably above ambient.
- OVR -- Since plugs will be opened to the EVST, a similar argument may apply to EVST sodium. The opening of a plug does not represent as much heat load as an open reactor vessel, so the EVST temperature can probably be somewhat higher than the reactor vessel temperature.
- OVR -- Purity standards for the refueling cell argon (or helium if that is the cover gas being used) need to be established. On SEFOR, there is evidence from the available literature that more oxygen found its way into the coolant during refueling operations than would be acceptable in a commercial plant.
- OVR -- Provision must be made for removal of the head and its temporary storage within the refueling cell along with the control rod drive mechanisms. The lifting device used for head removal and its replacement must be single failure proof.
- OVR -- It must be demonstrable that there would be no fuel damage if the crane were to fail in transit to the EVST. The means for recovery from such a mishap needs to be devised.
- Fixtures for removal and replacement of the EVST plugs need to be devised.
- OVR -- The UIS and the suppression plate must be dealt with in some fashion. Figure 20 shows the UIS as being lifted with the bottom plate of the reactor head as a placeholder. With such a design approach, the lower head plate and UIS would be moved to a UIS parking position, the UIS disconnected, and the lower plate stored with the other head plates. The UIS parking position would be designed with a scale-like arrangement so as to prevent the UIS from being obliged to bear the weight of the lower head plate. Note that the UIS parking position is not shown on Figure 20.

There is another potentially large advantage of open vessel refueling (or the single rotating head scheme that could also embody the refueling cell concept) that deserves mentioning. All

34At low temperatures, the risk of plugging becomes significant. The solubility of oxygen at 250°F is about 1 ppm and about 3ppm at 300°F. Operation in this temperature range may not be feasible.

LMFBR designers need to consider how they would approach the situation where some large component needs to be removed from inside the reactor and either repaired or replaced. Such an operation very nearly became necessary on Fermi-1 following the fuel melt incident. If something like that were to happen in a plant similar to CRBRP, some kind of a leak-proof tent structure would need to be constructed above the reactor head and the area inerted before one or more of the rotating plugs could be removed. The interior of the tent would need to be outfitted with the handling equipment and fixtures required to accomplish whatever is intended. Very likely, personnel would be obliged to enter this area in personnel protection gear including breathing apparatus. Those who have given any careful thought to how this action would be performed know what a major task this would likely be. It is likely that all the fuel and blanket assemblies would need to be removed and replaced with dummy assemblies to remove that element of risk from the equation. The whole operation would be complicated by the fact that the details would have not been worked out by the original designers and would need to be solved *ad hoc* and on the spot. If the plant were designed for open vessel refueling, (or single rotating plug with a refueling cell) most of these issues vanish. There already would exist a cell capable of being inerted above the reactor outfitted with handling equipment including remotely operated manipulators, viewing windows, cranes, and fixtures.

The arrangement of the containment, Fuel Handling Cell, the Transfer Room, and the EVST could apply whether there were zero, one or two rotating plugs. The main difference between open vessel refueling and refueling using a FTM would be the EVST itself, the head design. It should be pointed out that various hybrid approaches could be contemplated which would be intermediate both in cost and complexity between the design approach recommended here and CRBRP/Superphénix. For example, one could envisage a single rotating plug on the reactor vessel with an offset arm or pantograph IVTM and transfer positions in the reactor vessel containing core component pots. This approach would require only a single port to be opened to the reactor similar to the ports on the EVTM. It would require increasing the diameter of the reactor vessel and an EVTM designed to accommodate core assemblies contained in core component pots. It also would lead to increasing the length of the reactor vessel and would require increasing RV diameter to permit handling of components under the head. It would eliminate much of the complexity in the head access area and simplify plant construction.

The Resulting Reactor Vessel Design

The main function of the 10 ft. high UIS on CRBRP was to provide a place where flow mixing occurs from core assemblies with widely different outlet temperatures. The outlet temperature from control and shield assemblies will be considerably lower than the temperature from fuel assemblies and these flows need to be mixed prior to their arrival in the outlet plenum so that any thermal striping is confined to the regions designed to accommodate it. The UIS also performs the functions of backup hold-down of core assemblies (not required in this "design approach"), a location for instrumentation to measure core assembly outlet temperatures, and alignment & protection from cross flow for the control rod drivelines. These functions could very likely be accomplished in less vertical space if there had been an incentive for doing so on the CRBRP project.

On the other hand, CRBRP was a 995 MWth plant design and the concept being advanced here is 3000 MWth, which will require a larger vessel diameter and a sturdier core plate to support a much heavier core. It also may prove necessary to provide some shielding in the core assemblies above the blanket with the gas plenum removed. In the final analysis, design analysis will be required, but there is reasonable basis for optimism that the reactor vessel height can be reduced to somewhere in the vicinity of 30 ft., which would be close to a 50% reduction from CRBRP for a plant with three times the thermal output. Importantly, none of the reactor vessel dimensions should be finalized until there emerges reasonable confidence in the design of the core, the head closure, and the vessel internals. It also needs to be acknowledged that there may be a price to be paid for a shortened bottom mounted reactor vessel in the way of its seismic response. The inertia of the heavy head may increase the buckling loads on the shortened vessel wall attendant with seismic ground motion. Increased buckling loads may require that the vessel wall be thickened somewhat.

In addition to the above items, a promising candidate for cost reduction would be the UIS itself. For the case of CRBRP, there wasn't much incentive to reduce the height of the UIS since the space above the core was needed to transfer core assemblies in-vessel with the IVTM. However, because of the potential severity of the thermal striping issue, the provision of a feature in the design to mitigate its effect may be necessary. In fact, there was never any guarantee that the CRBRP UIS, fabricated entirely of Inconel, would survive the environment to which it was to be exposed. Experimental efforts focused on the phenomenon usually suggested the problem was real and its effects were possibly underestimated. A desirable course of action is to eliminate the phenomenon altogether by regulating all the core assemblies so that their outlet temperatures are close to equal. While this may be achievable in part, it may prove to be difficult for the control and shield assemblies. To the extent that thermal striping can be reduced, the remaining purposes for the UIS would be to limit cross flow to the control rod drivelines, and have some place to house the core exit thermocouples which would likely not require a structure 10 feet high. There is no need for a requirement to provide backup hold-down since the pressure drop across the assemblies is insufficient to lift them against the force of gravity.

As the vessel height is reduced, at some point the outlet nozzles will begin to become a consideration. The CRBRP hot leg piping was 36 in. in diameter. The Japanese JSFR-1500 outlet piping is designed to be 50 in. in diameter, which is somewhat surprisingly small given that the loop flow in the JSFR-1500 design is nearly five times greater than the loop flow in the CRBRP. This may be a reflection of the shorter piping runs in JSFR-1500 and their corresponding ability to accept a greater head loss per linear foot of piping. In any case, the outlet nozzles of a 1200 MWe plant will certainly be larger than CRBRP, especially since, with the Heat Transport System contemplated (see section on HTS) there are just two of them. It is estimated the hot leg piping would have a diameter of at least 60 in. and the cold leg piping 48 in. Since the outlet nozzles must be above the core barrel, there will come a point where further shortening of the UIS will provide no additional return. There needs to be some margin above the top of the outlet nozzles so as to prevent cover gas from getting entrained into the outlet piping attendant with down transients.

The sketch below (without the CRDMs) is a composite of the ideas expressed above for the open vessel refueling case. A thermal liner is shown, which would be eliminated if analyses establishes that it is feasible to do so. The active core region is 18 ½ ft. in diameter and the control rods and their associated drivelines and mechanisms (not shown) are spread over a roughly circular array about 14 ft. in diameter. The reactor closure head is supported by the vessel, which represents a compressive loading on the 2 in. vessel wall of about 700 psi.

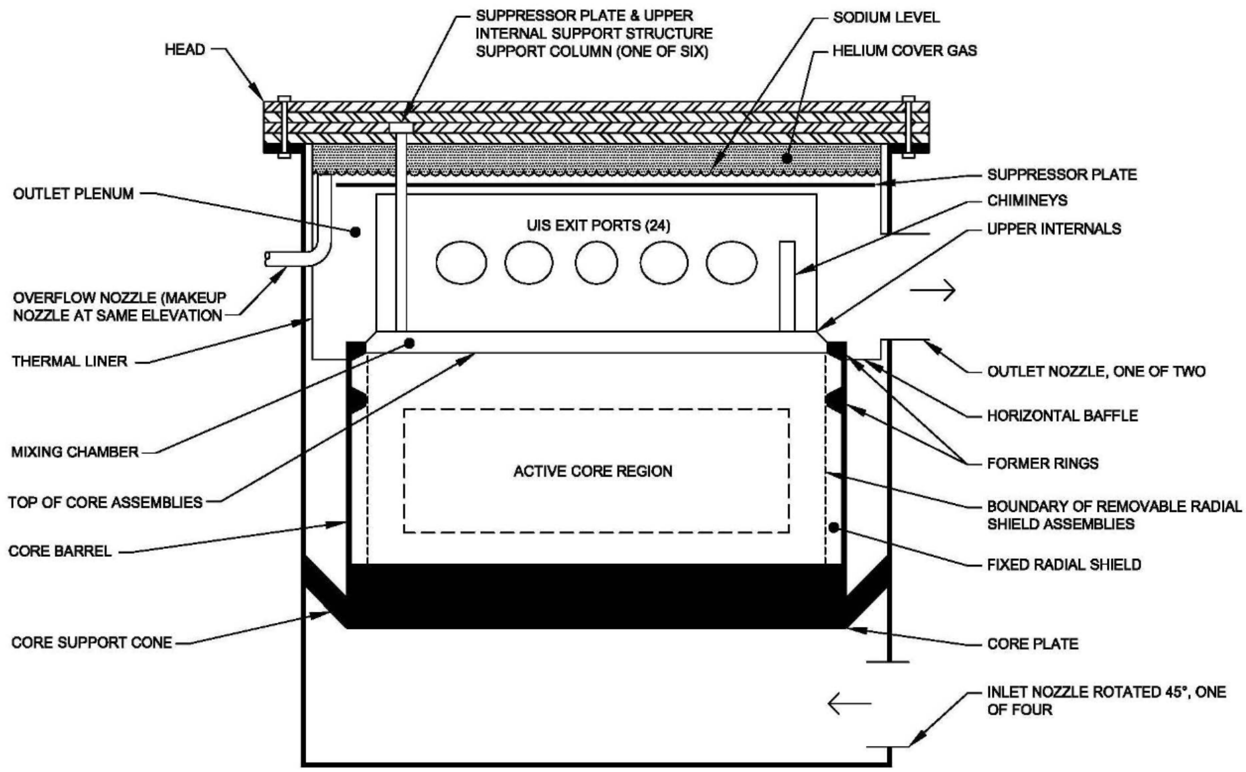


Figure 21 OVR reactor vessel design

As was stated earlier in this section, the single rotating plug option is the one being carried forward for the "design approach". The head would be different for the single rotating plug case. The rotating plug would be one piece and sized so as to give access to all the removable assemblies and, unlike CRBRP, there would be a fixed annulus outside the rotating plug. A port through this annulus could provide access to a DRACS if such a system were desired (see Section 8) but such a provision would probably require a small increase in vessel diameter. Also, it would be necessary to support the UIS and the suppressor plate from the head with a provision for jacking them upward similar to the CRBRP whenever the rotating plug is actuated. Core outlet thermocouple coverage would not be complete because of the need for refueling ports (discussed in Appendix 2G) but assuming the fuel assembly outlet temperatures are regulated at the assembly outlets, a representative thermocouple sampling should be sufficient.

Figure 21a, below, is a rendering of the "design approach" RV, showing outlines of representative primary and secondary CRDMs, the head riser (which supports the rotating plug), the UIS jacking mechanism, and a representative refueling port. The single rotating plug is inside the plug riser and the fixed annulus is outside. The loop seal in the head below the risers contains the sodium seal which is normally frozen but is melted during refueling to permit head rotation. Since there are 32 refueling ports, many of which are located in the CRDM area, it would be necessary to remove some or all of the CRDMs prior to refueling. Of course, the CRDMs and core thermocouples must be disconnected electrically prior to head rotation.

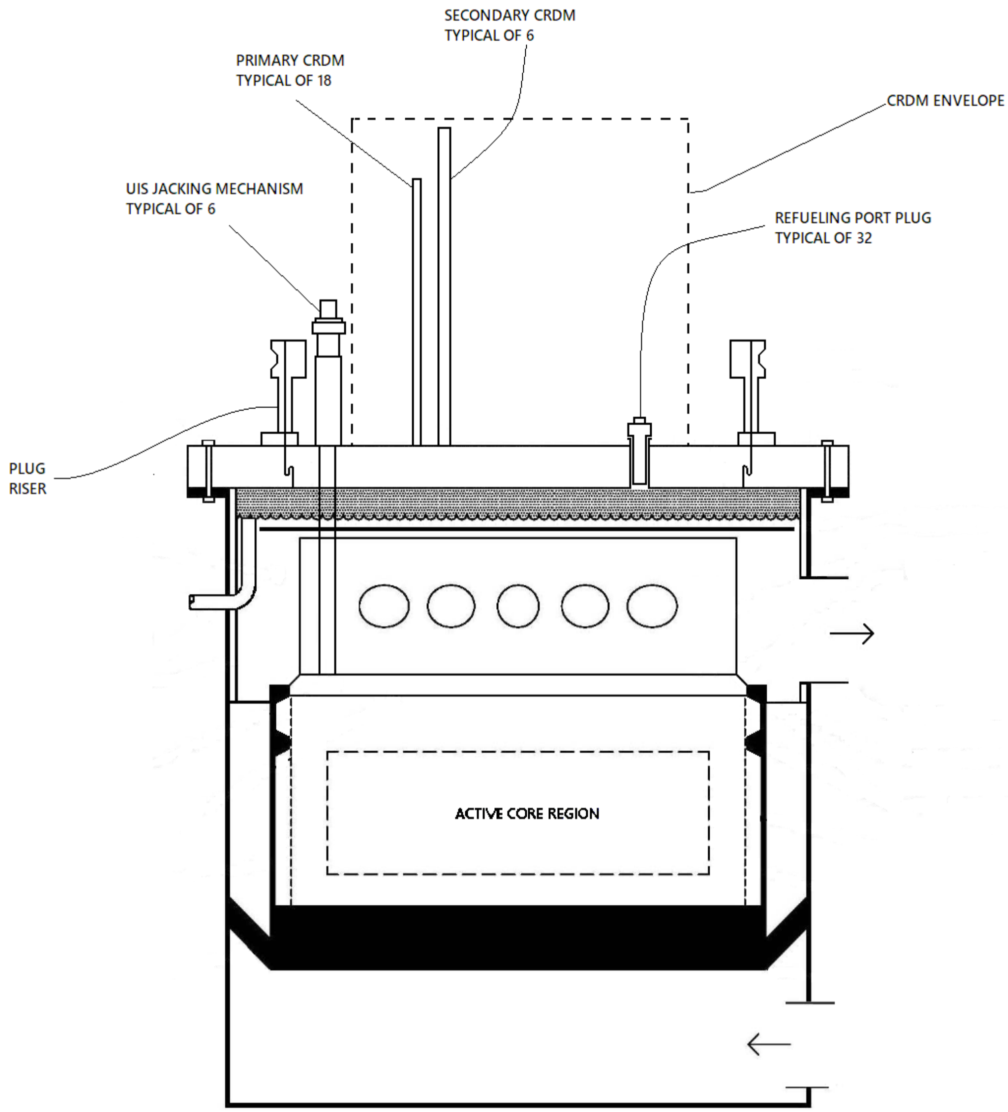


Figure 21a "Design approach" reactor vessel

There is another issue connected with the outlet nozzles. On CRBRP it was argued that a breach of the primary system pressure boundary would fill the reactor guard vessel before the outlet nozzles became uncovered. Filling the annular space between the reactor vessel and its guard vessel would lower the sodium level in the reactor by about 2.3 ft. If one adds the annular space between the inlet and outlet piping that is below the sodium level in the reactor, the result is another 1 ½ ft of sodium level. The outlet nozzles shown in the sketch above are 4 ft. below the normal operating sodium level, so this criterion is satisfied. This argument can be supplemented by pointing out that a primary system pressure boundary failure will leak before it breaks and the primary sodium makeup pumps (to be described later) will be able to keep up with any such leakage. Moreover, any cover gas entrained in the outlet nozzles will wind up in the IHXs which will probably have their own cover gas system connection or a vent since, in this concept, they are the high points of the system. If more space turns out to be needed because of pipe routing or some other reason, the horizontal baffle, which provides lateral support for the core barrel could be lowered along with the outlet nozzles or the vessel length could be increased up to four feet and still be less than half the length of the JSFR-1500 vessel.

The approach shown in the sketch above retains the basic configuration of CRBRP. The core barrel is shown as being 24 ft. in diameter – it could be increased to 26 ft. with little consequence making space available for a larger core or more shield assemblies if proven to be necessary. The UIS internals are shown having a 7.5 ft height. With bottom reactor vessel mounting, the load from the core is carried from the core plate through the core support cone to the vessel then downward to the vessel bottom. The core plate is shown as being 3 ft. in thickness, which is a foot thicker than CRBRP in order to accommodate the heavier core. It is likely it is thicker than it needs to be, but its required thickness needs to be determined by an analysis that considers all the penetrations. While there are more holes in the core plate than was the case with CRBRP, they are much smaller in diameter since the inlet modules have been removed from the design.

The tensile load on the vessel wall has been eliminated; replaced with a compressive load to carry the weight of the head. It is estimated that the vessel and its contents weigh somewhere in the vicinity of 5,000,000 lb. If the entire load were carried by the vessel wall (it isn't since the ~1,000,000 lb. weight of the sodium will mainly be borne by the vessel bottom) and the vessel were 2 in. thick, the total compressive load on the vessel wall, including the contribution from the head, would be less than 3200 psi at the bottom of the vessel. Since most of the weight is due to the core itself, one could design the vessel wall to be thinner above the attachment of the core support cone to the vessel wall, if desired.

It is possible that some kind if skirt arrangement will carry some of the load from the vessel wall down through the guard vessel or more likely, the vessel may rest on the guard vessel as was done at SRE or on some kind of Hallam-like standoffs. A thermal liner is shown above the horizontal baffle. It is presumed that some bypass flow will be directed into the annulus between the core barrel and the vessel wall then outside the thermal liner. The core assemblies are shown with a 10 ft. extension above the core plate with the 1 ½ ft inlet hardware inserted into the core plate.

Since the bottom mounted vessel design approach proposed represents a sharp departure from more recent precedent in LMFBR reactor vessel design, below is a summary of the most important advantages associated with each of the two approaches:

Top mounted vessel

- Vessel head is maintained at a fixed elevation
- Inspectability of vessel welds is simplified

Bottom mounted vessel

- Vessel wall in compression rather than tension – the greatest part of the load is limited to the bottom of the vessel below the core support cone

- Proven approach – little R&D required
- No issues with heat conduction to structural concrete other than at mounting flange

- Lower seismic excitation since vessel support occurs at a lower building elevation
- The static load on the vessel wall is lowest in the outlet plenum region where the temperature is highest possibly permitting elimination of the thermal liner and associated bypass flow
- Creates (stimulates) option for flat bottom of RV reducing vessel length
- Creates option for supporting core plate off bottom of vessel, reducing core plate thickness

It may turn out to be feasible to combine the reactor vessel and the core barrel. The volume between the reactor vessel and core barrel was used for transfer positions and in-vessel storage in previous designs, and such functions have been eliminated in this approach. Moreover, the under-the-head refueling approach led to an annular region adjacent to the reactor vessel that was inaccessible to the straight-pull IVTM used on the CRBRP design and therefore useless. Alternatively, the thermal shield could be thickened and serve as a core barrel as was suggested earlier in this section. Both of these approaches would result in a reduced diameter of the RV and probably eliminate the fixed portion of the reactor head. A consequence of doing so would be to eliminate the possibility of adopting the Direct Reactor Auxiliary Cooling System (DRACS) for decay heat removal. It will be shown in Section 8 that the preferred decay heat removal system does not involve a DRACS except as an alternative or fallback. However, the DRACS is sufficiently attractive as a fallback option to suggest not taking any steps in the vessel design that would eliminate it from further consideration.

In closing this section, it would be well to do some summing up. The principal incentive for all that has been proposed has been to shorten the RV. In addition to economy both in vessel fabrication cost and the expensive real estate needed to house it, a reduced length reactor vessel comes with additional benefits. The support columns for the UIS are reduced in length. A shorter vessel will have a lower seismic response which is further reduced by bottom mounting. The control rod drivelines are shorter and less likely to experience misalignment. There will be a lower pressure drop across the reactor vessel. The instrument lines to the core thermocouples will be shorter. The thermal shield, if it proves necessary, will be much shorter. It will probably be easier to engage fuel assemblies either with the EVTM for the single rotating plug concept or the refueling cell crane for the open vessel refueling case. The result is a reactor vessel that better performs its function and is more concentrated and focused on containing and getting the heat out of the reactor core.

7 Heat transport system

In the final analysis, the CRBRP Primary Heat Transport System (PHTS) was quite elegant, combining some of the best ideas from its American and German predecessors. Figure 22 is a schematic of one loop (of three).

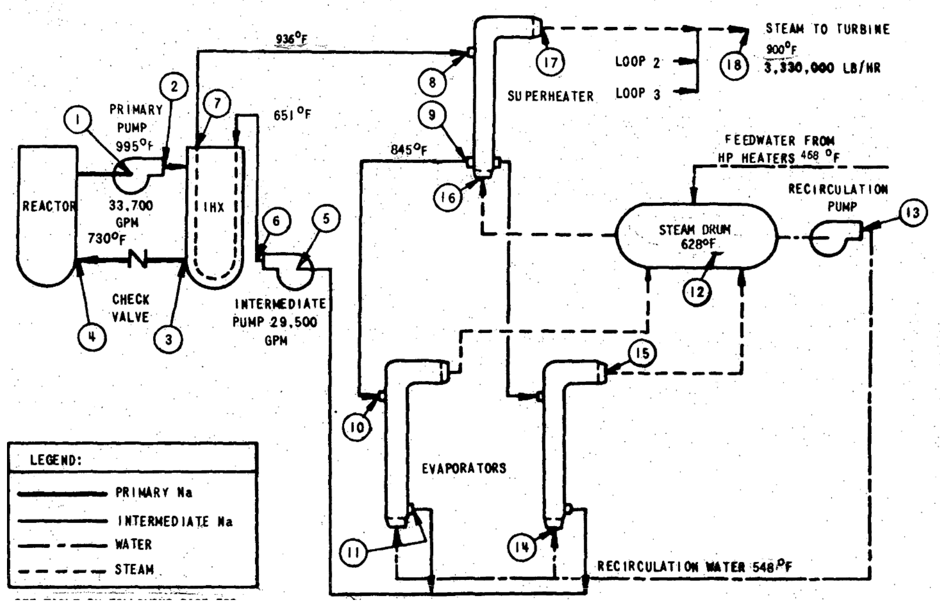


Figure 22 CRBRP HTS Schematic

The PHTS pumps were located in the hot leg as was done in FFTF and the German SNR-300. This was done so as to enable maintaining atmospheric pressure in the reactor vessel at all times. If the pump had been placed in the cold leg, it would have been necessary to pressurize the reactor above atmospheric to ensure the pumps would have adequate suction head during full flow operation because of the pressure drop across the IHX.

The Steam Generating System (SGS) consisted of two evaporators and one superheater per loop. The steam drum had a two for one recirculation ratio. Decay heat was removed from the steam drum via an air blast heat exchanger. There was an auxiliary feed water supply to the steam drums supplied with 1E power when the normal feedwater pumps were not available. One of the auxiliary feedwater pumps was driven by steam from the steam drums.

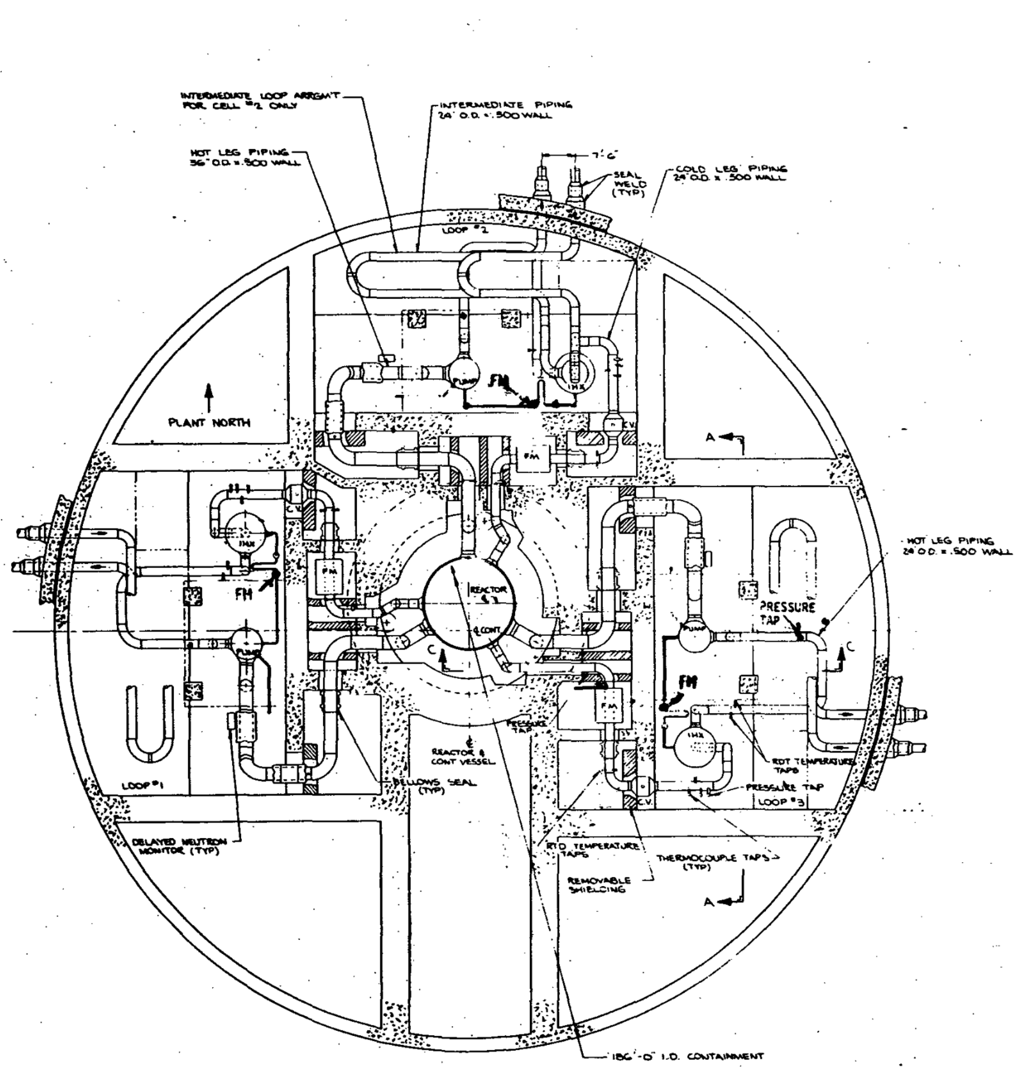


Figure 23   CRBRP HTS Containment Layout

The entire PHTS piping circuit including all expansion loops was maintained at a constant elevation. The expansion loop between the primary pumps and the IHXs protruded outward and established the containment diameter as is evident in the figure above, which is one of the primary defects of this design. The containment diameter of 186 ft. is considerably greater than commercially sized PWRs and resulted in much wasted space in an area that has very high real estate value.

A discussion of the PHTS would not be complete without a treatment of the overflow system, shown below as Figure 24. The RV continuously overflows to the Overflow Vessel while EM pumps take suction on the vessel and return sodium to the RV. The overflow vessel performs many of the same functions as the pressurizer in PWRs, maintaining inventory control through variations of the primary system temperature or power level changes. At operating temperature, the overflow vessel is nearly full, while it is at minimum level during refueling operations. Included in the overflow circuit are the cold traps, which remove oxygen and other impurities from the sodium. Figure 24 also shows the Overflow Heat Removal System (referred to on the CRBRP project as the "direct heat removal service") and the cooling system for the EVST.

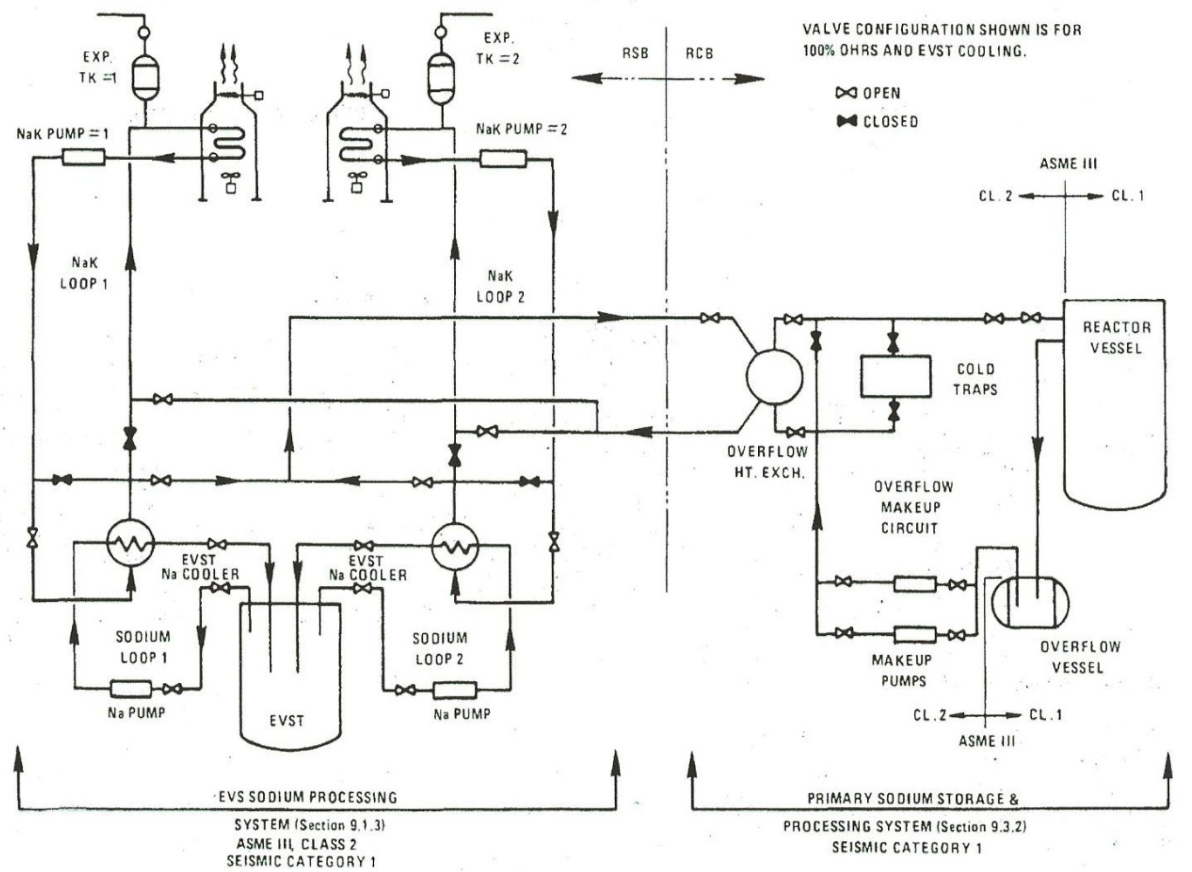


Figure 24 Overflow System, Overflow and EVST Heat Removal Systems

Since one of the very few places in the world that has performed any relatively recent work on loop-type designs is Japan, it would be well, when embarking on a treatment of the heat transport system (HTS) to start with an examination of the 1500 MWe Japan Sodium Fast Reactor (JSFR-1500). There have actually been two loop-type conceptual designs developed in Japan since Monju, the demonstration fast breeder reactor (DFBR) and more recently the JSFR-1500. The DFBR is a 660 MWe design embodying the so-called top-entry loop which was retained by the JSFR-1500. In fact, many of the JSFR features are a scale-up of the DFBR, so for the purposes of this discussion the following will be based on the more recent JSFR-1500. Most of the Japanese effort on this design that has been made available in the open literature seems to be focused on their safety approach and core design and there is woefully little on the heat transport system, but if one starts with what is available and uses a little imagination, it is possible to put some of the pieces together. A conceptual rendering of the JSFR-1500 HTS is shown below in figure 25[35]. By way of comment, representatives from Japan who describe their activities claim that in Japan future work will be based on the loop "to take advantage of Monju experience".

35 Figure drawn from *Progress on Fast Reactor Development in Japan*, H. Ohira, N. Uto, Meeting of the Technical Working Group on Fast Reactors, June 20-22, 2012

While that may be true, the Japanese vendors are fully capable of designing a pool-type reactor and have possibly chosen the loop because they have concluded it has more economic potential. There was considerable design activity at all four of Japan's reactor vendor design groups (Toshiba, Hitachi, Mitsubishi, and Kawasaki) funded by CRIEPI that was focused on the pool design in the 1980s.

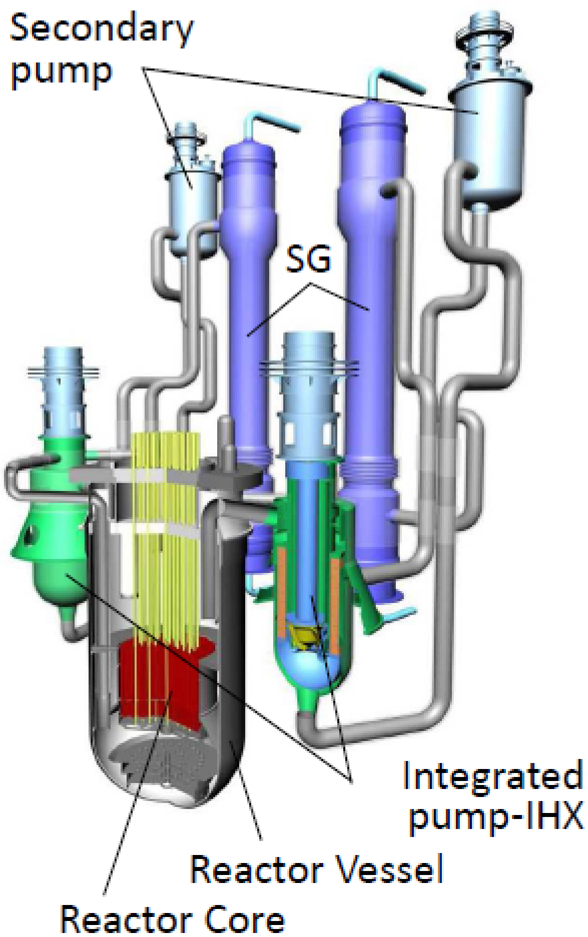


Figure 25 JSFR-1500 heat transport system

The JSFR-1500 has two primary loops and two secondary loops. The primary pumps appear to be centrifugal and have been integrated with the IHXs so as to eliminate the pump vessels, their associated guard vessels, and the interconnecting piping. The IHX tube bundle has been elevated above the reactor core to promote PHTS natural circulation. The designers refer to their concept as "through the head", which presumably means the reactor inlet and outlet piping penetrates the head rather than being routed through nozzles on the reactor vessel wall. There is precious little on the subject of refueling, but it was reported[36] that there is one pantograph machine and one rotating plug. A "pantograph" is an in-vessel handling machine with a scissors-like arrangement allowing variable extension in-vessel in the horizontal direction. Figure 26 below shows a "Plug

36IAEA-TECDOC-1531 2006, page 243

Rotation Mechanism" and a single rotating plug but no indication of the location of the pantograph. Presumably, the single rotating plug is offset to allow the UIS to be rotated out of the way of the pantograph but small enough so as not to interfere with the through the head penetrations. The pantograph machine is probably offset to one side of the rotating plug. In the same reference, it is also reported there is a mobile cask to storage outside containment, so a more or less conventional refueling system similar to CRBRP has presumably been adopted. Drawings of the reactor and combined primary pump/IHX are shown below along with a simplified general arrangement drawing depicting a two unit plant. Note that the containment is rectilinear and confined to the PHTS vaults – beyond that, there is no elevation drawing, but there may not be an operating floor in the conventional sense since the IHX appears to be elevated somewhat above the reactor vessel head.

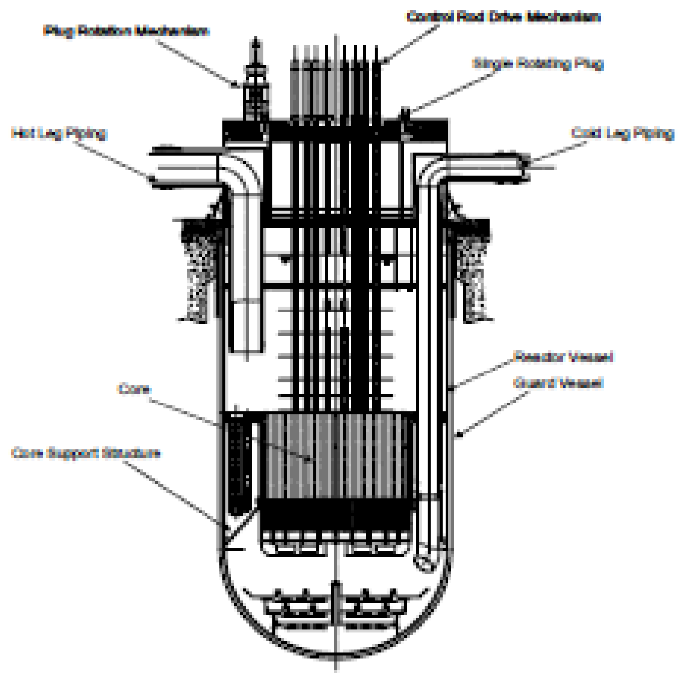


Figure 26   JSFR-1500 reactor vessel

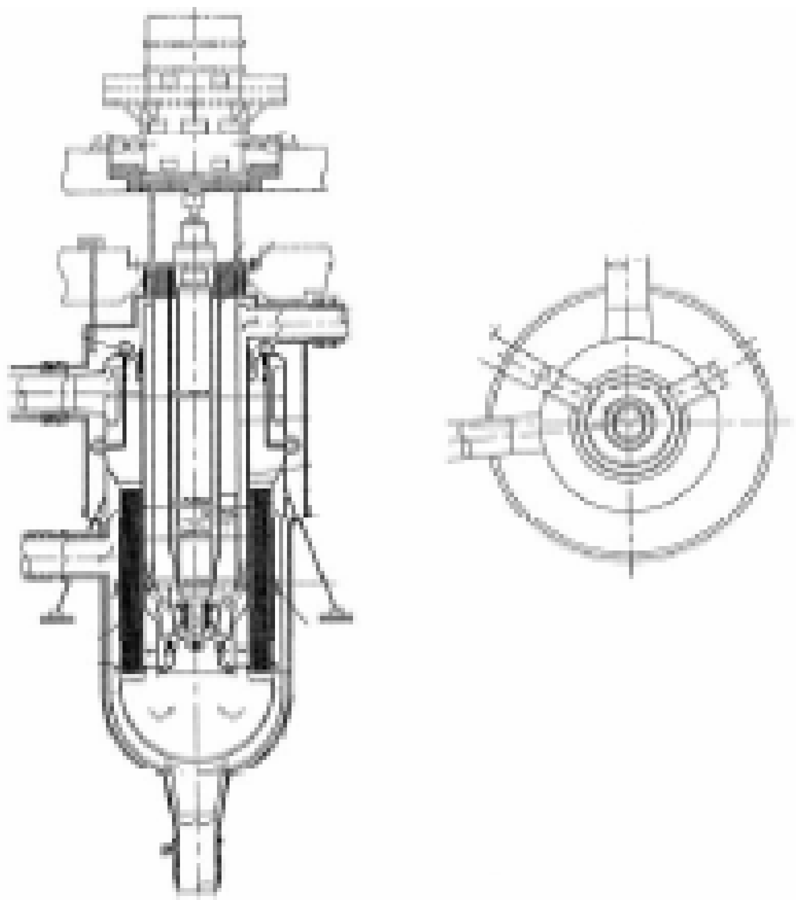


Figure 27 JSFR-1500 combined primary pump/IHX

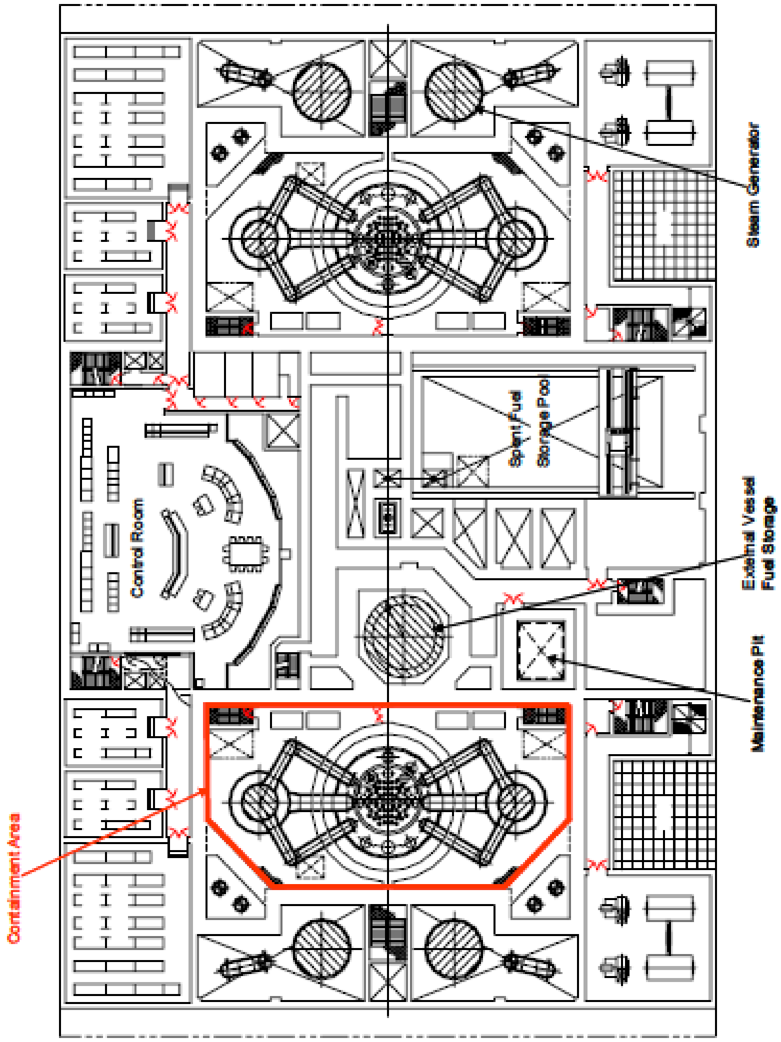


Figure 28   JSFR-1500 general arrangement

The reactor vessel is known to be about 35 feet in diameter. If one scales the above general arrangement drawing, it appears the containment space is about 115 feet long and 55 feet wide. Since the containment is reported to have a volume of 20,000 m^3, the average height of the containment must be about 95 feet. This compares with the volume of the CRBRP containment of 170,000 m^3, which was the second largest LMFBR containment designed worldwide behind only the German SNR-300. This concept clearly does achieve a significant reduction in containment volume. The reactor vessel height is about 69 feet. There are two cold leg lines for each loop, presumably to promote better flow distribution to the reactor core. The material of fabrication of the HTS is stated to be “12 Cr steel” because of its lower thermal expansion coefficient and higher creep strength in comparison to austenitic steels. This 12 Cr steel is ferritic and was developed in 1990s, including the technology of eliminating gaseous elements from the steel and injecting new knowledge about the effect of tungsten/molybdenum to improve high temperature strength.[37] The reduced expansion of the HTS piping is given as the reason the HTS can be made smaller. Another attribute of ferritic steels is that they are better conductors of heat than austenitic steels and therefore serve better as tube materials for the IHX. It is possible that the combined IHX/pump is mounted on some kind of movable foundation, but such cannot be readily confirmed. All primary system piping is protected with a "guard pipe", so the double walled piping concept of Fermi I returns with the JSFR-1500. One could question whether the double walled primary system piping is necessary in a two loop design, but in light of the elevation of the IHX/pump above the reactor, it may be necessary to maintain a different cover gas pressure in the reactor from that over the IHX/pump.

The decision to select two primary loops (as opposed to just one) may have been dictated by the decay heat removal concept, the subject of Section 8. The design shows one direct reactor auxiliary cooling system (DRACS) and two primary reactor auxiliary cooling systems (PRACSs). The PRACS includes a separate tube bundle in each of the IHXs. The DRACS has a separate tube bundle in the hot sodium side of the reactor vessel. Each system naturally circulates sodium (or more likely NaK) to an air cooled heat exchanger which is also cooled naturally. Each of the three systems is designed to remove 100% of the core decay heat independent of the other two. The design therefore embodies both redundancy and diversity. To achieve single failure resistance with a disabled primary loop when there is just one DRACS, two PRACSs are required, thus two loops. One could argue that the same reliability could be achieved with two DRACS and one PRACS, but such an approach would not yield diversity with a disabled primary loop. Since both DRACS and PRACS have their own tube bundle, the working fluid in both systems is probably NaK. The advantage in using NaK is that it significantly reduces the probability of freezing in the air cooler. DRACS is probably most suitable for a top entry system in order to provide access to the tube bundle. A concept level drawing of the JSFR-1500 decay heat removal system is included below in section 8.

There does not appear to be any reference to the fact that the double walled primary system piping comes as a direct result of the cold leg pump. While the pump has been nicely integrated into the IHX, it is still a centrifugal pump requiring cover gas to seal the shaft. If the pump were an EM pump, it would not require cover gas and potentially, the double-walled piping could be eliminated. It may prove to be desirable to retain double walled primary system piping regardless of whether an EM pump is chosen to eliminate the double ended primary system pipe break from having a significant effect on the containment design. If the pipe break is eliminated from the design bases, there would be no need for providing engineered safety feature (ESF) cooling to the concrete in the primary system vaults. The EM pump could be integrated into the IHX housing or installed in the cold legs. Since the primary system flow splits at the exit of each IHX, an EM pump could be installed on each cold leg for four pumps altogether. The use of EM pumps in the primary system brings additional advantages. The potential for sodium leakage is reduced since the sodium is circulated within a fully confined pump. The pumping flow can be matched to the decay heat profile, eliminating thermal transients from hot leg components. The pump mechanical seals are eliminated along with whatever system is provided to lubricate the pump motor and any reduction gearing. There are no moving parts to wear out that would require periodic maintenance. There is no cover gas to be supplied and processed. A near

37Ichimiya, M.; Mizuna, T.; Kotake, S.; *A Next Generation Sodium Cooled Fast Reactor Concept and its R&D Program*; Nuclear Engineering and Technology, Vol. 39, Number 3, June 2007 prototypic EM pump was tested by the Japan Atomic Power Company (JAPC) at ETEC in 2000-2001 with generally satisfactory results.[38 39] The JAPC EM pump had a flow rate of 42,250 GPM and a head of 37 psig. The flow rate would be fairly close to being correct for the design proposed for split cold leg piping, and the pump discharge pressure would be consistent with the head losses described in Section 6 above. There is an added advantage to splitting the cold leg piping. Doing so reduces the diameter of the piping and the nozzles into the reactor vessel. Every inch of reduction of these nozzles is most likely an inch off the length of the reactor vessel. So long as there is a third decay heat removal system, the two loop concept does not run afoul of the single failure criterion. It is therefore selected as the concept for the "design approach", with the third decay heat removal system identified in Section 8. Related to this is the decision to juxtapose the EVST and Reactor Vessel (Section 6), which eliminates a four loop option and allows three loops at the expense of increased containment volume. Moreover, the overflow vessel and cold traps can be conveniently located between the loops in a two loop design.

The top entry concept was probably chosen for JSFR-1500 to eliminate all nozzles from the wall of the reactor vessel. This is probably mainly an esoteric matter (or possibly an attempt to capture one of the stated advantages of pool-type plants) since the vessel nozzles do not add much to the cost of the vessel and as long as they are provided with guard piping, they introduce no new safety issues. Top entry does facilitate the use of a DRACS system for decay heat removal as will be described in the Decay Heat Removal System section, a perhaps debatable reliability improvement over a PRACS. However, top entry also may have the advantage of enabling the IHX/pump to be brought in closer to the reactor vessel. Although this is desirable, top entry would interfere with open vessel refueling. Besides, this possible advantage of the top entry system could be answered by locating the reactor vessel PHTS nozzles at or near the vertical plane that is perpendicular to the vertical plane that runs midway between the two IHXs. As will be discussed in the pool vs. loop discussion, the top entry vessel would have no overflow tank requiring the vessel to accommodate all the thermal expansion of the primary sodium as it heats up from refueling temperature to operating temperature. Assuming that reduction of the height of the reactor vessel is a worthy objective, an overflow tank must be included in the design. The inside height of the JSFR-1500 reactor vessel of 69 ft. is greater than the CRBRP vessel.

There are other features of the JSFR-1500 HTS design that may also open the door to controversy. For example, the design embodies two straight double-wall tube once-through steam generators. The designers may have chosen this steam generator design to create an ultimate pathway for a primary steam generator, however, from scaling the drawings the resulting units appear to be about 110 feet in length – a rather enormously long steam generator requiring a very high enclosure building. With an IHTS, it is questionable whether such a design would be more economic than say four helical coil steam generators and four IHTS pumps, both patterned after the Superphénix design and taking advantage of the Superphénix experience.

38Ota, H., Katsuki, M., Taguchi, J., Fanning, A. W., Doi, Y., Nibe, N., Ueta, M., Inagaki, T.; Development of a 160 M^3/min Large Capacity Sodium-Immersed Self-Cooled Electromagnetic Pump; Nuclear Science and Technology, Vol. 41, No. 4, pp. 511-523; April 2004.
39Fanning, A., Kliman, G., Kwant, W., Dahl, L., Inagaki, T., Ueta, M., Nibe, N., Ota, H., Katsuki, K., Doi, Y., Maekawa, I.; Giant Electromagnetic Pump for Sodium Cooled Reactor Applications; IEEE, 2003.

Helical coil units would be about 50 ft. in length, which would result in a lower cost of the building housing them.

Another consideration weighing on this matter relates to the tube area required in the steam generators in comparison to the IHXs. For the IHX, the temperature drop across the tubes is essentially constant assuming the PHTS and IHTS flows are equal. Moreover, since the IHX has sodium on both sides of the tubes, it can take double advantage of the excellent heat transfer properties of sodium. For the case of a once through steam generator, the water side heats initially, followed by boiling, then superheating. The point where boiling begins is a pinch point along with the waterside steam exit from the unit. This phenomenon is demonstrated graphically in the figure below where the temperatures in the figure correspond to Superphénix parameters in degrees Fahrenheit.[40] The figure also illustrates why there are limitations on how much the IHTS cold leg temperature can be reduced while maintaining high pressure steam conditions.

The steam generator tubes are obliged to contain the high pressure on the water side requiring them to be thicker than IHX tubes. The heat transfer profile combined with the greater tube wall thickness and poorer thermal conductivity of water/steam increases the tube surface area needed in the steam generator compared with the IHX. For example, in Superphénix which has four IHXs and four once through steam generators, each IHX has a heat transfer area of 1550 m^2 while each steam generator has a heat transfer area of 2570 m^2 even though poorer heat conducting austenitic steel was used for the IHX tubes and better heat conducting ferritic steel was used for the steam generator tubes. Using double walled tubing in the steam generators would make this comparison worse. The greater heat transfer area required in the steam generators is another argument for having four steam generators when there are just two IHXs.

40The figure is for illustrative purposes only. A more exact figure would show curved lines for all but the saturation line.

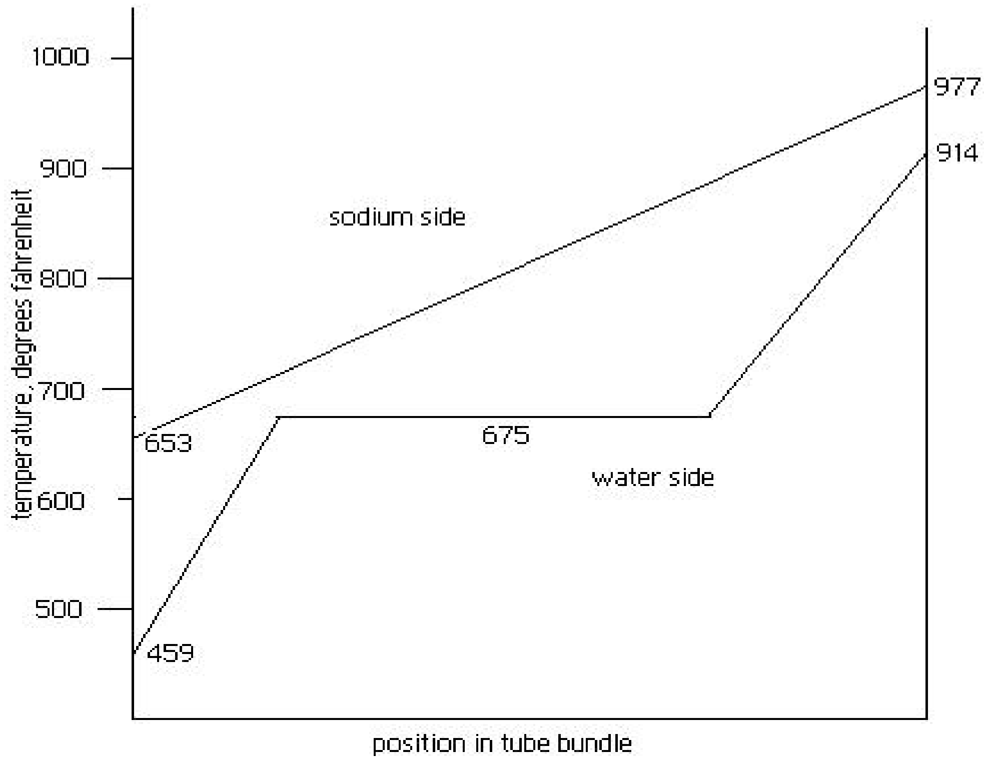


Figure 29 Temperature profile through once through steam generator

As was stated earlier, for the purposes of this design concept, Superphénix once-through helical coil steam generators will be assumed, representing CRM 27, each unit replacing two evaporators, one superheater, and one steam drum in CRBRP. The steam generators will be located in a building outside containment. In view of the relatively high probability that one or more of the steam generators will require replacement over the plant lifetime, the long wall of the steam generator building should be an exterior wall permitting access with heavy lifting machinery. It should also be the wall where the steam generators are located (on the interior).

For the case of two IHXs and four steam generators, each of the two intermediate loop flows would split into two paths to each steam generator then to the IHTS pumps where the discharge lines would merge to single lines back to the IHXs. Each steam generator should have its own IHTS pump to ensure balanced flow is achieved. An added benefit of this configuration is it should be possible to operate the plant at least at 50% power with one steam generator out of service. Since there are two reactor inlets from each IHX, the flow distribution to the reactor will not be affected. If there were a Sodium-Water Reaction Products System (SWRPS) activation, the plant would likely trip, however. In the interest of maintaining equal temperatures in the two primary loops, if the plant is operated with one steam generator out of service, a second steam generator should be brought out of service so that there is one steam generator in operation for each IHX.

Another feature of the JSFR-1500 HTS is the absence of cutoff valves in the intermediate loops and check valves in the primary loops. Absent valves in the IHTS loops, any sodium water reaction in a steam generator cannot be isolated from the balance of the loop and could contaminate the IHX. It is possible that the JSFR-1500 designers decided that with double wall tube steam generators, the possibility of a large scale sodium water reaction in the IHTS was sufficiently low so as to eliminate the need for these valves. The primary system check valves are provided to inhibit reverse flow on the occasion of a trip of a single primary system pump. If the check valves are removed, it may require all pumps to be tripped if any single pump trips. It is probably a good idea to remove the primary system check valves anyway to reduce system head losses. Depending on their design the head loss across each of the check valves could easily be 5 psi. Moreover, for the low flow rates accompanying natural circulation and the low end of the load following range, it may prove difficult to identify a satisfactory design that would both pass natural circulation flow and reliably accomplish its check valve function. For these reasons, removal of the check valves is probably a good idea. However, removal of the IHTS isolation valves would require evaluation if SWRPS activation is a credible event. Since the IHTS isolation valves were removed from the CRBRP design, there is good reason to believe they could be removed from the design approach.

Whether or not there are valves are relatively small points, however, the 69 ft. long reactor vessel in the JSFR-1500 design is not a small point and there has been little significant improvement in its design over predecessor loop type designs from the point of view of economics. If opportunity for significant improvement in the HTS is to be found over that achieved by JSFR-1500, it is probably in reducing the size of the reactor vessel.

Before leaving the steam generating system, since much of this paper is based on the CRBRP design, it should be pointed out that CRBRP used a recirculating steam generating system with two evaporators and one superheater per loop. The system was designed for a 2:1 recirculating ratio in the evaporators. This design was selected for two reasons. First, the steam generating unit size was kept low at about 110 MWth per unit and could be tested at a DOE facility at ETEC at nearly full size.[41] Second, as has been mentioned before, the original CRBRP decay heat removal system was dependent on there being a steam drum. It was a questionable decision to use one of the most vulnerable components in the plant as a vital part of the decay heat removal system and sole reliance on that system did not survive licensing. Also, the helical coil design, which is most adaptable for scale up to large sizes, tended to be the third choice among project personnel after the hockey stick and double wall straight tube designs. There were numerous trade-off studies among the project participants which usually came down in favor of the hockey stick design. What this demonstrates is the need to be suspicious of trade off studies. Such studies frequently have an unknown agenda and can be easily slanted even when supposedly objective decision criteria are being used.

From earlier discussions, there does not appear to be any conceptual reason for having more than one primary system loop once provisions have been made for decay heat removal that are sufficiently independent of the heat transport system. In fact, one of the advantages of the loop-type approach over the pool is that the loop affords the capability for single loop design which

41In fact, a 70 MW hockey stick unit, which was the planned CRBRP configuration, was tested at ETEC, but it was tested in the once through mode.

the pool does not. The principal problem with a single loop may be the size of the components – IHX, primary pump and piping. The reactor inlet would probably require splitting into at least three lines to give reasonable confidence of uniform flow distribution to the core. A two loop design modified as indicated above may represent a reasonable compromise and the best path forward.

The diameter of the PHTS piping requires discussion. In the thermal hydraulics treatment included in Appendix 2D, the need for making substantial reductions in the core pressure drop was discussed. There is an added incentive to minimize pressure drop in the PHTS since it is proposed that EM pumps be used for the PHTS which have relatively poor efficiency and have not yet been demonstrated at large sizes beyond about 40 psi. CRBRP used 36 in. diameter piping for the hot leg and 24 in. diameter piping for the crossover pipe and the cold leg. CRBRP was incentivized to minimize the hot leg pressure drop in the interest of minimizing the drawdown in the hot leg primary pump. For the case of the design approach being considered, minimizing pressure drop is desirable for both hot and cold legs.

Scaling the CRBRP hot leg from CRBRP to the proposed design flow parameters and two vs. three loops would lead to 66 in. for the hot leg piping diameter. If one maintains the same flow velocity in the cold leg, 48 in. is the result. This compares with 50 in. and 34 in. selected for the JSFR-1500 design and corresponds to a flow velocity of about 8.8 ft/sec. Such low flow velocities lead to extremely low head losses. The Darcy relationship indicates there would be about 0.5 psi head loss in 100 ft. of straight pipe at these conditions and about the same for each piping elbow. Since the head loss varies as the fluid velocity squared divided by the pipe diameter and the fluid velocity is inversely proportional to the pipe diameter squared, the head loss in a piping system for a given flow rate turns out to be inversely proportional to the 5th power of the piping diameter. Reducing the outlet piping diameter to 60 in. would therefore increase head losses in the hot leg by about 60%, which would probably be acceptable. Decreasing the hot leg piping diameter to the JSFR-1500 dimensions would increase head losses by a factor of four, and may not be acceptable.

In the above paragraph the subject of piping elbows is raised. The head loss in a piping elbow is sensitive to the radius of curvature of the elbow. To minimize these head losses, a radius of curvature of two times the piping diameter would be preferred. Since higher radius of curvature piping elbows reduces their flexibility to accommodate differential thermal expansion, it may become desirable to provide the IHXs with flexible mounts. Doing so simplifies the design of the PHTS and eliminates this issue from influencing either the distance between the IXH and the reactor vessel or the piping elbow design. However, mounting the IHXs in this fashion complicates the IHTS design since the penetrations of the IHTS piping through the containment building are fixed. Moreover, high radius of curvature elbows consume added space in the containment and will conflict with economic goals. For the purposes of this paper, it will be assumed the IHXs are fixed but the option of flexible mounting should be held in reserve pending further design. Radius of curvature PHTS piping elbows will be primarily set based on what fits in the containment.

With respect to the IHX, the JSFR-1500 designers provided for primary sodium to flow through the tube side and intermediate sodium through the shell side of the IHX. The reason for doing this may have been related to the fact that the pump and IHX are integrated into the same unit in JSFR-1500. Most LMFBRs designed to date have put the primary coolant on the shell side with the higher pressure intermediate sodium on the tube side. (In pool type reactors, there is no realistic alternative.) It is desired to minimize the PHTS pressure drop across the IHX and since it would probably be easier to accomplish this objective on the shell side, the design proposed here provides for the more conventional approach of putting the primary system flow on the shell side of the IHX.

The CRBRP IHX was originally designed to have a removable tube bundle as were many other LMFBR IHXs worldwide, including the FFTF and most pool-type plants. The primary sodium was on the shell side while the intermediate sodium was on the tube side. A characteristic feature of removable tube bundles is the central downcomer that carries intermediate sodium entering the top of the unit to below the bottom tubesheet. The removability feature was deleted on CRBRP midway through preliminary design, but the central downcomer was retained. Other features of removable tube bundle IHXs are the continuous primary side plenum outside the tube bundle with a bypass seal and the bottom primary exit. Both of these features were retained on CRBRP. The CRBRP IHX had 2850 7/8 in. diameter tubes 25.8 ft long with an overall length of 58 ft. and a shell diameter of 8.8 ft. A drawing of the CRBRP IHX is included below.[42]

42 R. W. Devlin, J. D. Bresnahan, WARD, *FFTF and CRBRP Intermediate Heat Exchanger Design, Testing and Fabrication*, Procedings of the US/USSR Seminar on Problems of Design, Development, Fabrication, and Test of Breeder Reactor Components, Feb. 6-9. 1978, Los Angeles, CA

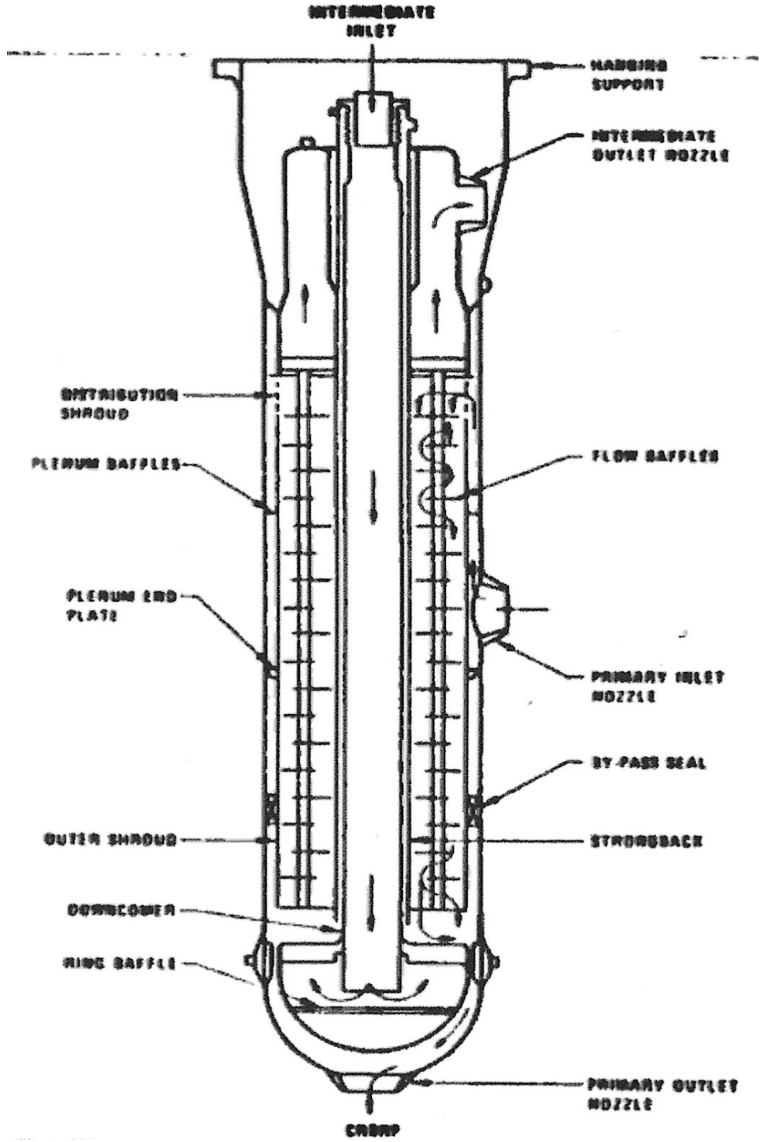


Fig. 30 CRBRP IHX

In view of the fact that sodium coolant presents a benign environment for the metals that contain it; the removable bundle feature presents a cost savings opportunity that should be capitalized upon, as it was on CRBRP. Since it is proposed (section 8) to put an auxiliary tube bundle in the IHX for the PRACS and this bundle may be of a helical coil design, any requirement for tube bundle removability faces another obstacle.

It happens that the designer of the CRBRP IHXs, Foster Wheeler, was commissioned to design an IHX with 1000 MWth thermal duty in the late 1970s as a part of exploratory studies for a follow-on plant to CRBRP. The design was required to have a primary side pressure drop of no greater than 10 psid. The resulting design shown in the figure below had 4284 1 1/4 in. diameter tubes on a 1 7/8 in. triangular pitch. The unit incorporated a 17 MWth helical coil Primary

Reactor Auxiliary Cooling System (PRACS) in the inlet plenum, had an overall length of 65.6 ft., a maximum diameter of 13 ft., and weighed 298 tons.[43]

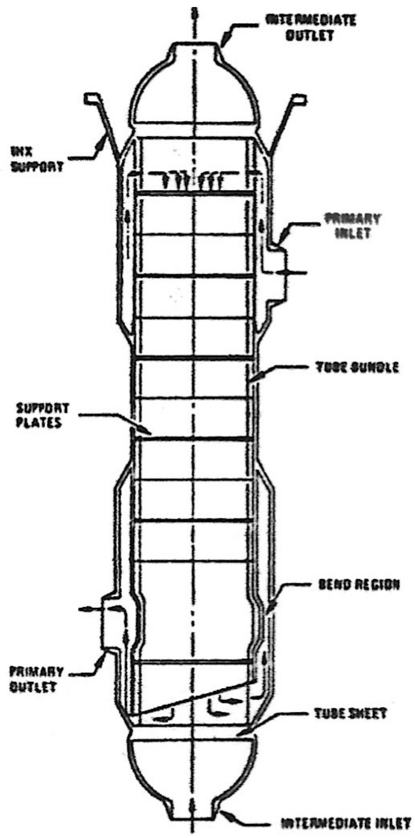


Figure 31 1000 MWth IXH

Note that this design looks like a more ordinary heat exchanger compared to its CRBRP equivalent with the shell side inlet and outlet on the side, tube side inlet and outlet on the bottom and top, the inlet plenum distinct from the outlet plenum, and no central downcomer. The PRACS coil would be in the upper part of the primary side inlet plenum, with a thermal center about 14.5 ft. below the top of the unit. The paper cited above reported on a trade off between three designs having different primary side pressure drops. Only the high pressure drop unit was deemed to be transportable over land -- the other two requiring barge shipment. A 1500 MWth unit would have a shell diameter approaching 16 ft., probably requiring barge shipment. This single fact could be decisive in the selection of the number of primary loops for the plant, particularly in view of the interest in keeping primary side pressure losses at a minimum. A four loop plant with 750 MWth IHXs may turn out to be preferable possibly dependent of the site selected. Whether the reactor vessel is barge transported or fabricated on site would influence this decision. Other factors influencing this decision are material selection (austinitic vs. ferritic

43 G. B. Brown, J. F. Cox, *IHX Portion of Paper on Advanced Work on Pumps and Heat Exchangers*, Proceedings of the US/USSR Seminar on Problems of Design, Development, Fabrication and Test of Breeder Reactor Components, Feb. 6-9, 1978, Los Angeles, California

steel), advances in heat exchanger technology since 1978, and the desired LMTD across the IHX tubes.

In section 8, it is proposed that a key system for decay heat removal in the "design approach" involves a committed separate coil at the top of the IHX. This system is called the Primary Reactor Auxiliary Cooling System or PRACS. The thermal center of the PRACS coil in the IHXs should be 30 ft. above the thermal center of the core to ensure abundant natural circulation. In Appendix 2E, it is shown that with a 30 ft. driving head, this natural circulation flow is about 14% of full flow at a reactor ΔT of 324°F and will vary as the square root of the ΔT.

The requirement that the PRACS coil be 30 ft. above the thermal center of the core leads to the IHX being above the Reactor Vessel (RV). The sodium level in the RV is about 15 ft. above the core thermal center (see figure 21) and the middle of the PRACS coil is about 15 ft. below the top of the IHX, so the IHX would extend about 30 ft. above the top of the RV. This constitutes a departure from previous sodium cooled reactor plant designs where the free sodium level in the RV was the highest point reached by the sodium in the PHTS. (It appears from the drawing above that the JSFR-1500 design may have located the IHXs slightly above the RV.) This comes as a consequence of both shortening the RV and incorporating PRACS into the design.

During normal operation, this arrangement is not a problem since the RV cover gas would be pressurized to about 15 psig. However, during refueling operations, the sodium freeze seal in the rotating plus must be melted to permit head rotation, so the RV pressure must be reduced to atmospheric. Thus, the top of the IHX PHTS sodium would be about 25 ft. above the RV sodium level, and a vacuum must be drawn at the top of the IHX PHTS sodium when the plant is shutdown for refueling.

Since the IHX will be the highest point of the PHTS, entrained gas will tend to collect there. With RV pressure of 15 psig, the IHX pressure would be slightly above atmospheric and the IHX could be vented to the Radioactive Argon Processing System (RAPS) (see Section 12) or it could be vented to a convenient place such as the overflow vessel. When the RV is depressurized during refueling, it will be necessary to pull a vacuum on the IHX to keep it filled. This could be accomplished using the RAPS compressor or more likely, by some other convenient special purpose device.

It is clear from the above that while it is possible to achieve a 30 ft. thermal driving head between the core and PRACS thermal centers, the IHX cannot be elevated much further without simultaneously lengthening then RV. Since the IHX length is 65.5 ft and there must be 10 ft. clearance below the IHX to accommodate the IHTS inlet piping, it will be necessary for the basemat elevation below the IHXs to be about 15 ft. below the RV basemat elevation. With the containment design proposed in this "design approach", this does not introduce any particular problem. The thermal center of the main IHX tube bundle would be 13.5 ft. above the thermal center of the reactor core, which would be sufficient for natural circulation it the PHTS loops, if such natural circulation were desired.

A final modification to the HTS that needs to be incorporated is to extend its load following capability down to 15% in order to provide maximum load following capacity for the plant. This kind of requirement would be applied to cause the plant to be compatible with a power grid having extensive renewable energy generating capacity. One challenge would be to develop a pump power supply and an electric motor capable of running at very low speeds. This should not be a problem for the EM pumps in the primary system and if it proves to be impractical for centrifugal pumps in the intermediate system, they could also be changed to EM pumps. In fact, EM pumps would likely be preferred for the IHTS so the plant transient response would be matched between the PHTS and the IHTS. Centrifugal pumps can have a long coastdown following trip that does not apply to EM pumps.

The capability of the steam plant, particularly the turbine generator, to operate at these low power levels needs to be assessed. Another issue that could crop up at very low flows could be competition from natural circulation. As is shown in Appendix 2E, the natural circulation flow with a 13.5 ft. thermal centers separation between the core and the IHXs would be 9.3% at full system ΔT. To operate the plant at 10% power may require the primary pumps to be shutdown and may lead to a reduced reactor ΔT if the natural circulation flow turns out to be greater than predicted – a likely result. While this is not impossible to design for, it likely would be desirable to accept 15% as the plant minimum for load following purposes.

To sum up the conclusions as they pertain to the JSFR-1500 HTS design:

- The top entry concept may be incompatible with open vessel refueling but would probably be okay for the single rotating plug concept where the entry would be through the fixed portion of the head.
- The centrifugal pump in the primary system should be replaced with cold leg EM pumps. The EM pump could be integrated into the IHX or installed in the cold legs. If the cold leg is split, it may be desirable to install a pump in each leg for a total of four primary system EM pumps.
- Since the elevated primary system piping concept has been abandoned in the design concept, all primary system piping will be guarded. It is noted that this guarded piping was considered to be the primary containment on the Hallam plant. This licensing approach is worthy of consideration.
- The hot leg piping diameter should be increased above JSFR-1500 dimensions to 60 in. and the cold leg to 48 in.
- Primary sodium should be on the shell side of the IHX.
- The JSFR-1500 RV design should be revisited.
- The IHTS & steam generator designs are suspect but probably not too important from an economic point of view. Replacing two double walled steam generators with four helical coil steam generators may be more economic and could provide more flexibility of operation.
- The DRACS may only be applicable to a top entry system, would not be applicable for open vessel refueling, and may require an increased reactor vessel diameter to adapt to the single rotating plug in the "design approach".

- An overflow vessel should be included in the design as a means to reduce reactor vessel height.
- The absence of cutoff valves in the IHTS should be revisited from the point of view of controlling possible sodium water reactions.
- The use of ferritic steel in the PHTS is desirable, particularly as tube material in the IHX, and if compatible with the selected parameters should be adopted, at least in part.

Otherwise, the two loop concept with two cold leg pipes per loop and close-in components with an elevated IHX with split cold leg piping appear attractive. A possible layout of the HTS major components and the PHTS piping is shown in the figure below. The inside plan dimensions of the containment building in this diagram are 80′ by 120′. There could be a down loop in each of the two RV outlet legs within the reactor vault before the piping enters the IHX vaults. (The RV vault wall would be moved outward.) Similarly, the IHX outlets could vertical with the EM pumps in the vertical leg. If it proves necessary to provide down loops in the cold leg piping, assuming the top of the diagram is north it would be necessary to move the IHXs east about 10 ft. and correspondingly increase the size of the containment building. The turbine building would adjoin the south side of the steam generator building to permit the Steam Generators to be on an outside wall.

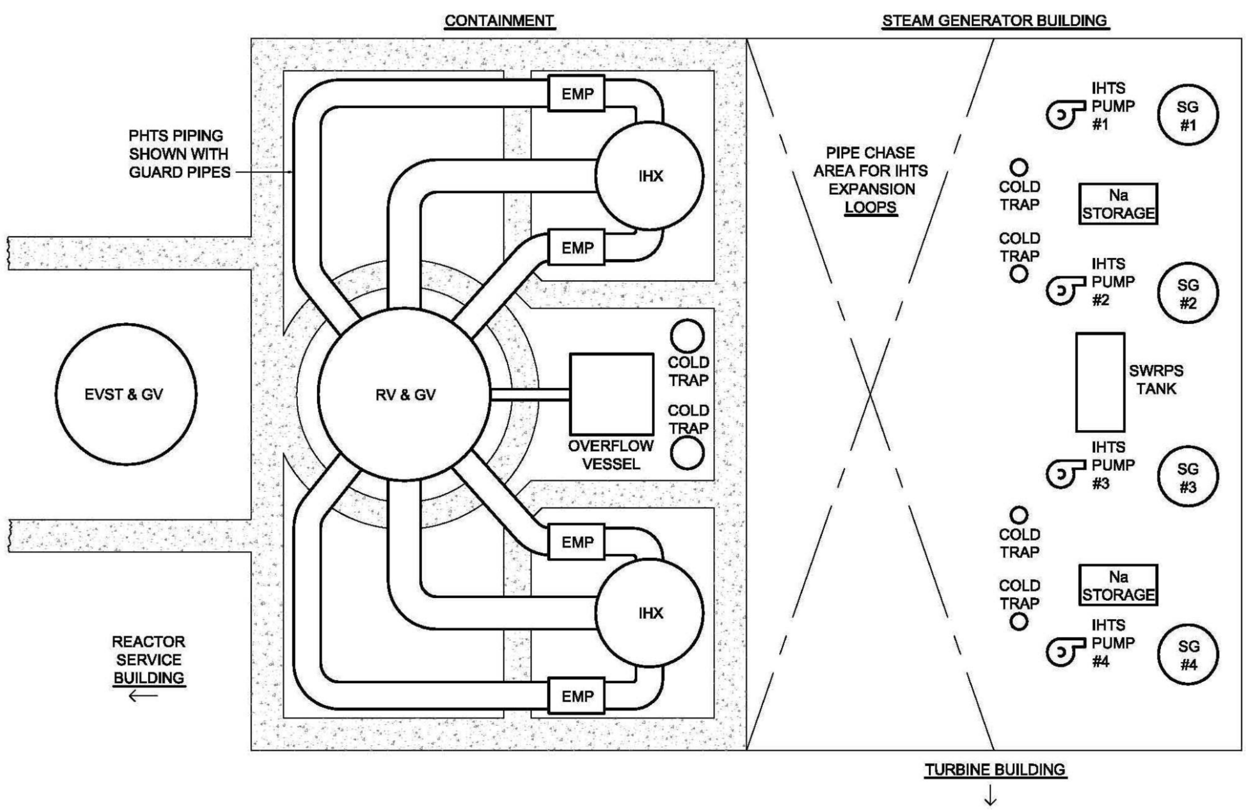


Figure 32 Proposed HTS layout

An alternative layout that would require evaluation would be to locate the IHXs north and south of the RV. Doing so would shorten the distance between the RV and the IHXs and would enable the four cold legs to be symmetric. The main problem with doing this is it would interfere with the refueling cell since the IHXs are much longer than the RV and would extend to the roof of the cell. The JSFR-1500 is a close coupled north-south layout with the RV outlet nozzles feeding directly to the IHX inlets (see Figure 25). How such a scheme accommodates thermal expansion would require input from the designer. Also, tube plugging would be challenging in a unit adjacent to and not shielded from the reactor.

It is necessary to enumerate the Cost Saving Measures of the proposed HTS system in comparison the CRBRP.

28. The elevated unguarded PHTS piping concept is abandoned, permitting greatly reduced containment volume.
29. Centrifugal pumps are replaced with EM pumps in the primary circuit.
30. Two primary loops down from three.
31. Elimination of requirement for pony motors on PHTS and IHTS pumps
32. Check valves are eliminated from PHTS.
33. Guarded PHTS piping eliminates need for liners on PHTS vaults.

Parameters selection

An important part of HTS design is the selection of primary and intermediate system hot and cold leg temperatures and steam conditions, including feedwater inlet temperature, steam pressure, and superheat. HTS parameters selection involves a compromise between the quest for high thermodynamic efficiency and the limitations imposed by the core materials, particularly the fuel and inner blanket cladding material. By and large this compromise has already been made on CRBRP with the resulting selected primary system hot leg temperature of 995°F. So long as the cladding material remains austenitic, the CRBRP result remains applicable modified upward somewhat by the proposed thermal-hydraulic improvements suggested in Appendix 2D and section 6.

Another consideration which pertains to this issue is the steam generator. For the "design approach", the Superphénix steam generating system has been identified as the reference design. The reasons for this are simple – Superphénix was one of the few LMFBRs incorporating a once through steam generator of a commercial design which actually accumulated some experience that was essentially trouble free. There are three good reasons for being especially careful with the steam generator system in a LMFBR. First, the consequences of a water to sodium leak are significant. Second, historical experience with LMFBR steam generators has not been good. Third, the duty a once through steam generator is exposed to is challenging.

Consider figure 29 above on the temperature profile of a once through steam generator. This profile actually uses Superphénix parameters. At the feedwater inlet, the temperature across the tubes is nearly 200°F, placing the inner wall of the tubes (the water side) in tension and the outer wall in compression. A similar high ΔT occurs when boiling is complete at the onset of superheating. In the boiling region the water side tubes are constantly being exposed to wetting and drying as bubbles form and break away from the tubes. This wetting and drying will cause large swings in the water side surface temperature of the tubes exposing them to fatigue and possible failure. It was concern over these issues that led the CRBRP designers to select a recirculating system, which didn't eliminate the problems but did ameliorate them somewhat.

Although the Superphénix system was trouble free for about five years of operation, there is no guarantee that the Superphénix steam generating system will survive a 40 or 60 year plant lifetime but unfortunately, aside from the recent Russian experience at BN-600, Superphénix is the best experience available for a commercial-grade steam generator.[44] It is for this reason that the Superphénix steam parameters are selected for the purposes of this study. Using the Superphénix steam side parameters essentially locks in the IHTS parameters but not the PHTS parameters. Superphénix PHTS, IHTS, and water/steam side parameters are presented below:

	Hot leg	Cold leg
PHTS	1013°F	743°F
IHTS	977°F	653°F
Water/steam side	914°F	459°F

Table 1 Superphénix HTS parameters

The BN-600 experience with their steam generators is too compelling to be ignored. As of the end of 2018 they had not experienced a steam generator leak since 1991. The BN-600 steam pressure is about 2015 psig and temperature is about 944°F. BN-800 is about 1970 psig with steam temperature of 914°F. At a minimum, some sort of collaboration should be attempted and wholesale procurement might be considered. Such a path could lead to changes in the parameters and reconsideration of the decay heat removal system since the BN-600 steam generating system involves separate evaporators and superheaters (as well as reheaters) and a steam drum.

Back to Superphénix, the ΔT on the hot side of the IHX is 36°F while the ΔT on the cold side is 90°F giving a LMTD of 59°F across the IHX. These rather strange numbers are partly the result of efforts to minimize the size of the IHX tube bundle which is located within the pool while taking advantage of the highest reasonable temperature that the fuel system could deliver. For a loop type plant, the IHX tube bundle can be allowed to grow without excessive economic penalty. A constant ΔT across a tube bundle yields the most efficient utilization of tube surface area. If a constant 40°F temperature drop were to be adopted across the IHX, the PHTS hot and cold leg temperatures would become 1017°F and 693°F respectively for a PHTS temperature rise across the reactor of 324°F. This compares with a 265°F increase across the CRBRP reactor, a 22% increase. For a scaled up core, the 22% greater ΔT translates into about 22% less primary system flow and 22% lower flow velocity in the PHTS reducing pressure drop by 39%. For all the reasons detailed in Appendix 2D, this is a highly desirable outcome. Total primary system flow for these parameters would be about 225,000 GPM requiring each pump to be capable of delivering about 56,250 GPM. The primary and intermediate flow rates would be the same. The proposed parameters are repeated in the table below.

44There is more experience with the EBR-II steam generators but the design is not practically scalable.

	Hot leg	Cold leg
PHTS	1017°F	693°F
IHTS	977°F	653°F
Water/steam side	914°F	459°F

Table 2 Proposed HTS parameters

Before leaving this subject there is another matter in need of discussion. Superphénix was not the only plant where the PHTS ΔT was greater than the IHTS ΔT. In fact, on CRBRP, the ΔT across the hot side of the IHX was designed to be 59°F while on the cold side it was 79°F. This raises the question of why the PHTS and IHTS flow rates weren't equalized on CRBRP. The answer is transients. On the occasion of a reactor trip the reactor power immediately drops to the neighborhood of 6% and continues down to below 2% after one minute. Meanwhile, even if the primary pumps are tripped immediately when the reactor is tripped, primary flow will be greater than decay power because of pump coast-down. The PHTS pumps, being centrifugal, have a significant moment of inertia and it required 20-40 seconds for them to coast down to the 10% flow which would correspond to the pony motor flow rate. During this period, the primary system flow rate considerably exceeds the decay power from the reactor core with a resulting sharp decrease in hot leg temperature. This sharp temperature decrease exposes hot leg components to stress. It was primarily for this reason that the CRBRP PHTS temperature rise across the reactor was limited to 265°F even though IHTS ΔT was 285°F. The smaller the PHTS ΔT, the less severe will be the transients.

For the design approach being advocated herein, this consideration is not germane since EM pumps are being proposed for the PHTS system. Indeed, the ability to reduce transient response is one of the reasons for adopting EM pumps for the PHTS. When a reactor trip occurs, the EM pumps can be programmed so that PHTS flow rate matches reactor decay power. Fairly soon after trip (probably immediately), natural circulation will assume the responsibility for core cooling and the EM pumps can be shut down.

8 Decay heat removal system

The decay heat removal system is a key component in the description of any nuclear plant since it addresses the very feature about nuclear power that differentiates it most from other sources of energy, viz. it is a heat source that cannot be fully shut down. When the control rods are inserted and the nuclear reaction is halted, radioactive fission products continue to generate heat that must be removed from the reactor. It is important to recognize that it was not the nuclear reaction but the decay heat that did the damage at Chernobyl, Three Mile Island, and Fukushima Daiichi. At the instant the nuclear chain reaction is halted, the decay heat energy is about 6.6% of full power, declining to about 1% after an hour.

The CRBRP steam generator auxiliary heat removal system has been mentioned twice before in this paper and this section will begin with a more complete description of that system in the interest of thoroughness on this important topic. The operative portion of this system is shown in the figure below. There are protected air cooled condensers (PACCs) that are supplied with steam directly off each of the plant's three steam drums. Steam condensing in the protected air cooled condensers is returned to the respective steam drum by natural circulation. The PACC itself was designed to be cooled by blowers, two of which were powered by emergency diesel generators and a third by the plant battery. Each PACC had a design heat removal capacity of 15 MW or 1.5% of the plant's full power. The SGAHRS was designed to operate with the pony motors in the PHTS and IHTS operating. The auxiliary feedwater system took suction from a protected water storage tank and one of the three auxiliary feedwater pumps was steam driven with the steam coming from any of the three steam drums. The system was designed so that a single PACC could remove 100% of the reactor decay heat so long as one auxiliary feed pump was operable. One hour after shutdown, the auxiliary feedwater can be secured.

Despite the fact that the system was designed for operation with pony motors and blowers, the designers were mindful of the desirability of preserving a natural circulation capability, and designed accordingly, maintaining the IHX thermal center above the core and the evaporator thermal center above the IHX. Six years after design activities had been launched; the capability of the SGAHRS to operate in the natural circulation mode was evaluated, and found to be more than adequate. A re-evaluation was performed in 1982 after the core design had been modified. The result was that following station blackout conditions with the turbine driven auxiliary feed pump operating, the primary flow leveled off at 3%, intermediate flow at 4% and recirculation flow at 11% in about three minutes after shutdown. Reactor Vessel outlet temperature never exceeds 1025°F, well within allowable margins.

This remarkable result was buried in section 5.7.5 of the PSAR under 5.0 "Heat Transport and Connected Systems" and 5.7 "Overall HTS Evaluation". It did not appear in the Table of Contents anywhere. Its total length was slightly more than three pages and included just two figures. It cited two references[45][46] that would have provided a more thorough description but included only a perfunctory summary in the PSAR. So it turns out CRBRP had station blackout capability long before Fukashima and the project made just about no effort to capitalize on the accomplishment. The only way to explain this is that the CRBRP was designed before Fukashima when the importance of decay heat removal during station blackout was not fully appreciated.

45R.R. Lowrie, W.J. Severson, "A Preliminary Evaluation of the CRBRP Natural Circulation Decay Heat Capability", WARD-D-0132, 1978

46W.J. Severson et al, "Summary Report on the Current Assessment of the Natural Circulation Capability with the Heterogeneous Core", WARD-D-0308, Feb. 1982

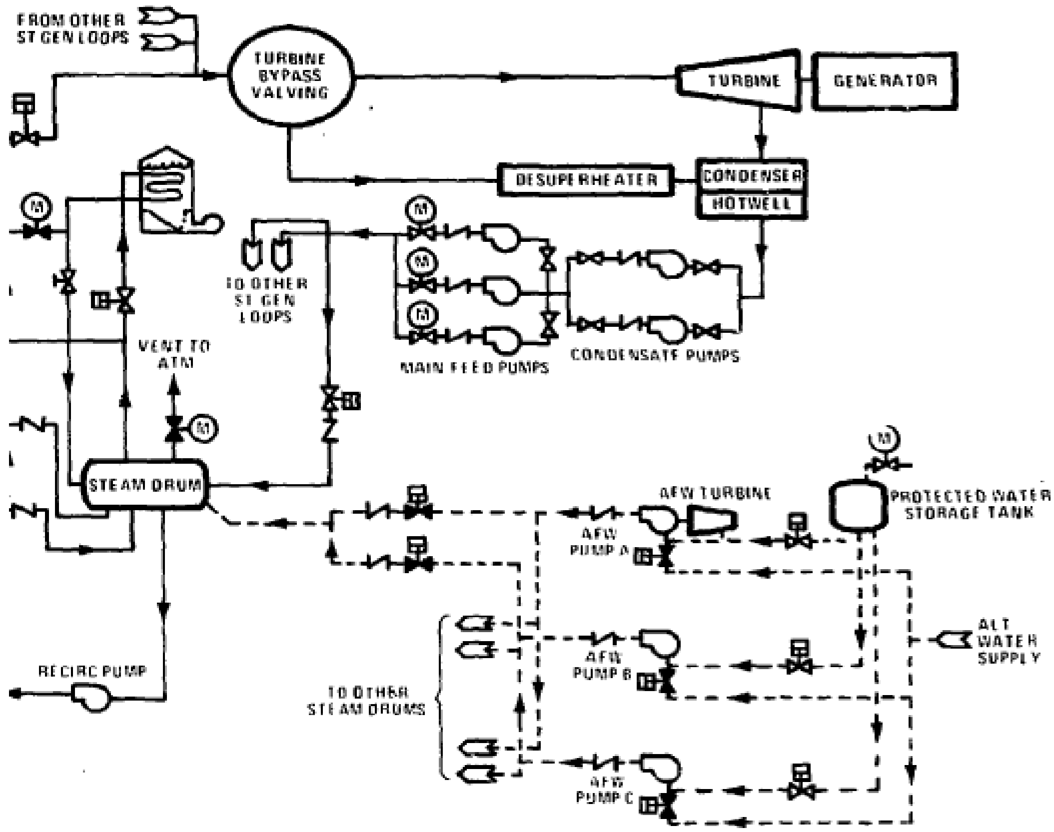


Figure 33 CRBRP steam generator auxiliary heat removal system and auxiliary feedwater system

In retrospect, there is no question the SGAHRS would have been more attractive had it been designed specifically for operation without electric power from the beginning, which it obviously could have been. The PACCs would have been natural draft with no fans and the pony motors, if they were still provided, would have had no safety significance. It seems likely that if the plant's decay heat removal system was known to be operable during station blackout, other parts of the plant design would probably been effected, making it less reliant on 1E power. Eliminating the blower in the PACCs would probably have led to a larger PACC unit but little else. Since the system was totally closed, the only occasion it had for requiring any make up water was during the first hour of operation, primarily due to the sensible heat of the sodium inventory and the fact that the superheater would no longer be removing any heat. During this one hour period, relief/control valves would open on the steam drum so makeup water would be needed. A second auxiliary feedwater pump would probably have been steam driven. If the SGAHRS had been designed from the beginning for lights out operation, the CRBRP would have been one of the first commercial reactors to have had such a capability and it could have been a good advertisement for one of the particular merits of the LMFBR concept. This SGAHRS as it was designed did not satisfy the NRC, probably because it was dependent on the steam generating system, which was known from Fermi-1, the British PFR, and the Russian plants to have a record of questionable reliability. The project therefore proposed a second decay heat removal system, which the project called the Direct Heat Removal Service (which unfortunately shares the acronym, DHRS, with Decay Heat Removal System) shown in the figure below.

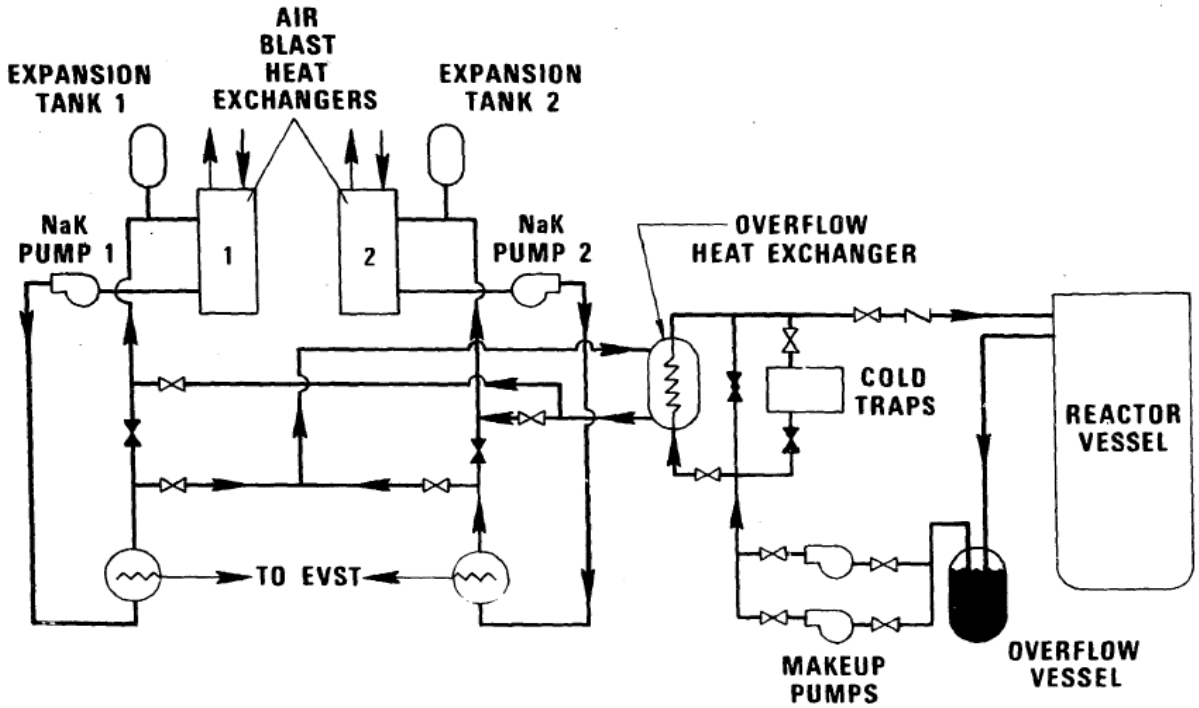


Figure 34 CRBRP Direct Heat Removal Service (DHRS)

This system was a rather clever way to bring preexisting capability to bear to strengthen the reactor decay heat removal system. Since the EVST may be loaded with heat producing spent fuel, it is necessary to provide it with cooling that is reliably powered. The CRBRP DHRS borrowed upon this capability when needed. The only item added to the plant was an overflow heat exchanger and some connecting piping that enabled the plant operator to shift the reactor decay heat load to the EVST air blast heat exchangers that were already in the design. There may also have been a capacity upgrade of the NaK pumps and the air blast heat exchangers. Each air blast heat exchanger was capable of removing 5.5 MW so the two operating together would handle the core heat load after about one hour following shutdown.

Numerous other means for decay heat removal have been used or proposed, particularly in the years since CRBRP was cancelled. The proposed JSFR-1500 design, shown in the figure below,[47] is very typical. There are two systems, a Direct Reactor Auxiliary Cooling System (DRACS) and a Primary Reactor Auxiliary Cooling System (PRACS). The DRACS involves a single cooling coil which is immersed through the head inside the reactor vessel. NaK flows through the coil and naturally circulates to an air cooled heat exchanger located high above the reactor. The PRACS involves independent coils in each of the two IHXs again containing NaK and again naturally circulating to an air cooled heat exchanger. Each of these three systems is capable of removing all the decay heat from the reactor independent of any electric power.

47 *Progress on Fast Reactor Development in Japan*, H. Ohira, N. Uto, Meeting of the Technical Working Group on Fast Reactors, June 20-22, 2012

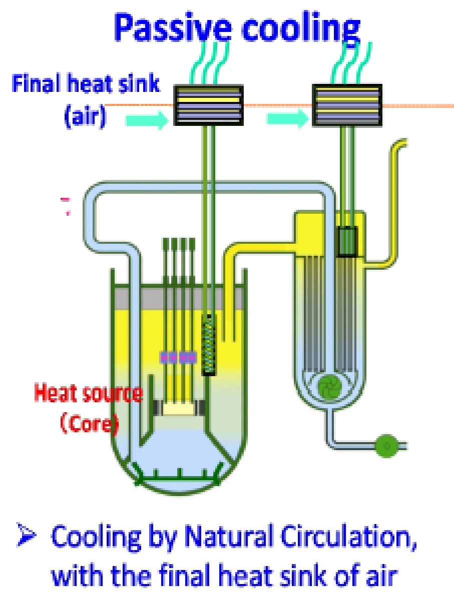
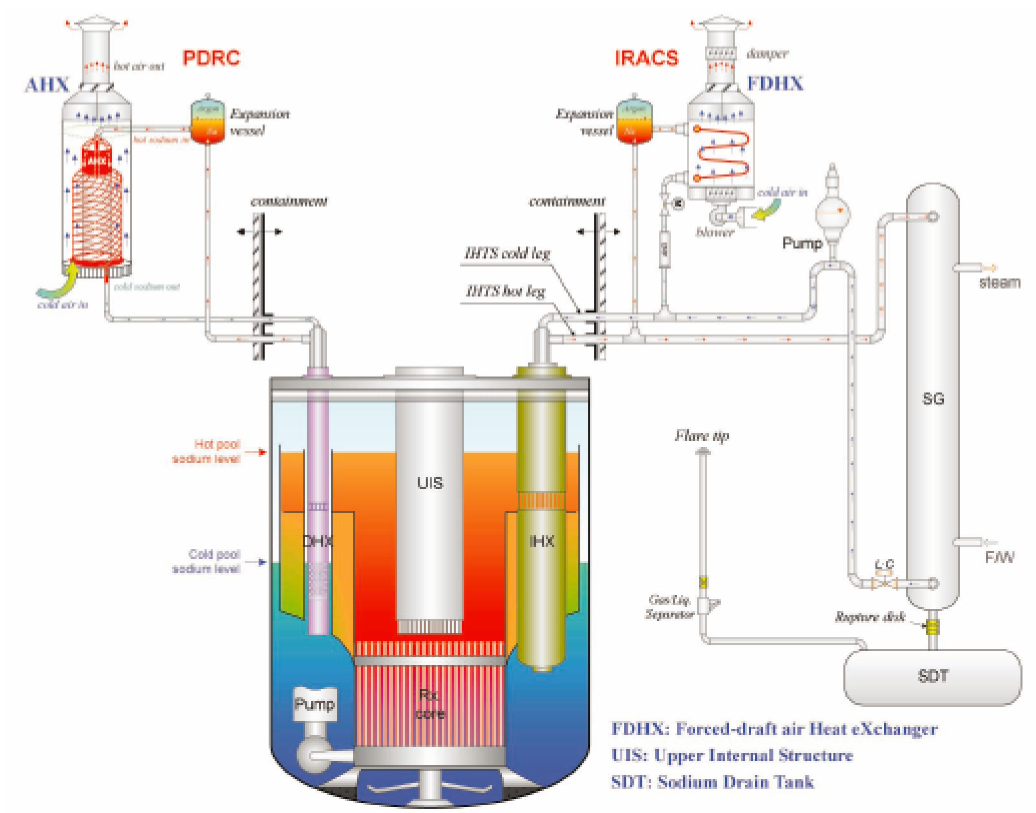


Figure 35 JSFR-1500 decay heat removal concept

A variant of the PRACS can be found on the Korean plant design KALIMER-600 shown in the figure below.[48]



48 Kwi-Seok Ha, Hae-Yong Jeong, *Comparison of the Decay Heat Removal Systems in the Kalimer-600 and DSFR*, Nuclear Engineering and Technology, Vol. 44, No. 5, June 2012

Figure 36 KALIMER decay heat removal concept

KALMIR-600 uses a DRACS which they call a passive decay heat removal circuit and an Intermediate Reactor Auxiliary Cooling System (IRACS) that taps off the intermediate loops and is cooled by a forced draft heat exchanger. The Korean designers probably felt that having one circuit, the passive decay heat removal circuit that is totally cooled by natural circulation would be sufficient and it would be easier to control a forced draft system. There is no reason why the IRACS system couldn't be designed with air heat exchangers that are cooled naturally. IRACS with EM pumps is the decay heat removal system installed in the Russian BN-800.

Among the other systems that have been proposed is one that naturally circulates air by the guard vessel of pool type plants. Variations of this concept have been used on PFR in the UK and Phénix and Superphénix in France. They are generally somewhat marginal in their performance and depend on the large inventory of sodium within the pool to slow the temperature rise before the guard vessel cooling can keep up with the decay heat. Since the design approach that is being proposed purposefully avoids a large sodium inventory, this sort of system would likely not be attractive.

Just about any system that is 100% redundant, totally passive, can operate without electric power (including instrumentation), and is independent of the steam generating system would be better than the world's fleet of LWRs to date. Of the systems shown above, the DRACS would probably not be compatible with open vessel refueling, but it would be compatible with the single rotating head concept, which is another advantage of the single rotating head system over the open vessel system. For the concept presented here, any DRACS would be located as high in the RV as possible (below the minimum safe level) and its primary side outlet would be connected to a downcomer that would penetrate the core support cone. The maximum thermal driving head would be on the order of 10 ft. Another problem with DRACS is there doesn't seem to be any way to avoid a considerable bypass flow in the RV from outlet to inlet without some sort of valve. Such a valve could be pressure activated by the PHTS pumps, but such a solution could interfere with load following at low power levels when PHTS pump discharge pressure is at a minimum. A creative solution to this problem is an attractive candidate for an R&D effort, and is included in Appendix 9.

From a reliability point of view, the difference between the PRACS and the IRACS is small assuming the IRACS is cooled by a naturally circulating air heat exchanger. The IRACS is more straightforward, not requiring a separate coil in the IHX and using sodium to the air cooled heat exchangers in place of NaK. However, with IRACS the IHX tube bundle is part of the decay heat removal circuit requiring that the thermal center of the IHX tube bundle be elevated well above the reactor core – possibly in the range of 30 ft. As was discussed in Section 7, this could lead to a requirement to lengthen the RV so as to enable circulation in the PHTS during refueling operations. Moreover, it would be necessary to ensure that the sodium in the air heat exchanger is kept molten during normal plant operation when the IRACS is not in use. This would probably be best accomplished by maintaining a small flow through the air heat exchanger with some kind of a damper on the air side that is kept shut and limits air flow. IRACS also results in the entire IHTS becoming important to safety, at least up to and including the IHTS isolation valves, if such valves are present. These undesirable features have the result of making the PRACS the more attractive option. If necessary to achieve belt and suspenders diversity and redundancy, both PRACS and IRACS or PRACS and DRACS or DRACS and IRACS could be deployed together on the same plant, accepting the economic penalties described above. Another option would be to have two 50% PRACS coils within each IHX.

It would be desirable to have a diverse system available. For this purpose, the Overflow Heat Removal Service (OHRS)[49] of CRBRP certainly would be diverse and would bring with it all the benefits previously stated. Because of its ready controlability and ease of operation, the OHRS would probably be the preferred means for routine decay heat removal during refueling or other routine shutdowns. Because of the need to prevent freezing in the ultimate heat sink, it would be necessary to use the NaK cooled OHRS to reduce the sodium temperature to the 250-300°F range that would probably be preferred for refueling operations. (This temperature range would apply to open vessel refueling but not necessarily to the single rotating plug option.) It would also be preferable for the EVST to be capable of natural circulation to its Na/NaK heat exchanger and the associated NaK system to naturally circulate. In addition, the NaK to air heat exchangers should be capable of functioning without air blast fans, i.e. natural draft. When used for this service, the EM pumps providing the overflow should be designed for a minimum flow rate of about 1% of full PHTS flow or about 2250 GPM.

A brief treatment of the thermal capability of these two systems needs to be included in this discussion. For the case of CRBRP, the thermal capability of each of the SGAHRSs was 15 MW. Since the SGAHRS was designed for one system to be out of service followed by a single failure, each of the three SGAHRSs could handle the entire decay heat load of the plant. This 15 MW turns out to be an overestimate of the heat load for three reasons. First, the SGAHRS was designed for CRBRP stretch conditions, which were 1121 MWth as opposed to design conditions of 995 MWth. Second, the designers always were holding station blackout as an option, which tended to lead to additional capability than required for the reference case. Third, SGAHRS had to deal with the sensible heat of the HTS that was above the temperature of the steam drum. This imposed a considerable additional load on the system during the first hour after shutdown. A better indicator of the amount of decay heat removal capability would be the CRBRP OHRS, which was designed for 11 MW.

Another benchmark can be drawn from the Foster Wheeler 1000 MW IHX discussed in Section 7. That unit was required to have a PRACS coil capable of removing 17 MW. That was probably based on a 3000 MW three loop plant design, later called the Large Scale Prototype Breeder. Assuming the single failure criterion was applied would lead to a 34 MW decay heat load for the 3000 MW design, which would be consistent with scaling up the CRBRP OHRS to 3000 MW. Since the single failure criterion would apply to the "design approach", and there are just two IHXs, each PRACS coil as well as the OHRS would be obliged to have a 33-34 MW capability.

Summarizing, the proposed decay hear removal system consists of two totally passive PRACS, each capable of removing 100% of the reactor decay heat and an OHRS dumping its heat to two

49The term “Overflow Heat Removal Service” is more descriptive than “Direct Heat Removal Service”, and will be used in the balance of this paper.

naturally circulating EVST heat exchangers. Additionally, since the PHTS and IHTS will naturally circulate, heat can be dumped to atmosphere from the steam generators with feedwater being supplied by steam or motor driven auxiliary feedwater pumps, although this circuit would not be safety grade. A de-superheater would probably be required for the steam driven feedwater pumps and some form of preheat would be needed for the auxiliary feedwater, since extraction steam would not be available. If this configuration fails to provide adequate reliability, since there are four IHTS loops, the two PRACS could be exchanged for four 50% IRACS at an associated capital cost penalty or, alternatively, a DRACS could be added, for the single rotating plug concept.

The elimination of the requirement for the IHTS and SGS to be important to safety is considered CRM 44. Whether PRACS is less expensive than the SGAHRS is not assured, but since it does not require separate evaporators, superheaters, and steam drums, and does not require the SGS and IHTS to perform a safety function, it would certainly seem to be.

9 Containment

The CRBRP containment was cylindrical with an ellipsoidal head, a flat bottom basemat, and an operating floor. The operating floor was 86 ft. above the basemat and the cylindrical portion extended 83 ft. above the operating floor. There was a 400 ton polar crane mounted near the top of the cylinder. The area above the operating floor was accessible during plant operation while the area below the operating floor was not due to the high radiation level of the Na^{24}. The flat basemat was a consequence of the decision to adopt a cylindrical structure and the 83 ft. portion above the operating floor was probably a consequence of the removable IHX tube bundle, which had been part of the design in the early stages when the containment design approach was selected. (To remove the tube bundle, it would be necessary to have a totally enclosed device, handled by the polar crane that would adapt to the IHX, permit evacuation and inerting, lift the bundle out of the IHX shell, and button up the IHX. How the tube bundle, once removed, exited containment was something left for detailed design.) There were large unoccupied volumes below the PHTS loops. The huge volume above the operating floor was mostly unused. The polar crane was used to lift the Auxiliary Handling Machine, the ex-vessel portion of the IVTM, and various floor valves but would probably have been tested to near its limit only if an IHX tube bundle replacement were necessary.

The concept advocated herein abandons the cylinder in favor of a rectilinear arrangement, so there is no need for a common elevation basemat. Other than the refueling cell, there are no operations going on, so there is no need for an operating floor. If it was alright on CRBRP for the volume below the operating floor to be inaccessible during power operations, why not put the entire containment off-limits during operations? What purpose is served by a massive polar crane if the IHX tube bundle is not removable and there is no refueling equipment to handle? There is no longer a need to store an IVTM or Auxiliary Handling Machine. A Plug Handling Machine and necessary floor valves can be stored in the refueling cell and the EVTM can be parked there. It would probably be a good idea to design the containment so there is a "soft patch" over each of the IHXs in the event that there is ever a need to replace one. Otherwise, the containment volume should be whatever is needed to house the refueling cell, reactor vessel, PHTS, overflow vessel, in-containment storage vessel, cold traps, overflow heat exchanger, RAPS (possibly), and needed auxiliaries (see Section 12). This approach reduces containment volume by at least 80% without meaningfully impairing operability, and this for a plant three times the power output of CRBRP.

Once the geometry is settled, it is now necessary to turn to the design basis. In LWRs, the containment is typically designed to accommodate the double ended primary system hot leg pipe failure. For PWRs, this leads to 580°F water exiting the break and immediately flashing to steam and pressurizing the containment structure to up to 60 psig. To contain such pressures in a relatively large structure, it is necessary to adopt the cylindrical or spherical geometry commonly observed for containment buildings at nuclear plants worldwide. It is also necessary that the containment be large. If it is desired to reduce the profile of the containment in a PWR as was the case at the Donald C. Cook plants in Michigan it becomes necessary to incorporate some sort of pressure suppression system inside containment such as ice condensers.

In the case of sodium cooled reactors, the double ended pipe break in the primary system is also considered (perhaps ill-advisedly if the leak before break postulate can be positively confirmed for low pressure ductile piping systems), but the consequences are much more benign. The sodium flowing from the postulated break is well below its boiling point, so provided the space into which it flows is inerted it does not pressurize the space other than by adding sensible heat to the surrounding atmosphere. For the design concept presented here, the entire primary heat transport system is surrounded by guard piping located in cells that are inerted with nitrogen or argon, so there is little or no reaction of the hot sodium with the atmosphere. A similar arrangement was adopted on the Hallam reactor and was referred to as the "containment" for that plant, which was licensed.[50]

The refueling cell has a single barrier between sodium and the cell interior. That cell is designed with a liner so any escaped sodium will not cause damage to structural concrete. While such an event may cause modest pressurization of the cell and may require the provision for some sort of concrete cooling or catch treys as were used on CRBRP, it does not require any robust containment other than shielding, at least in the conventional sense. For the design approach being advanced, there isn't need for any protection against major primary sodium leaks since the entire primary system is double walled. The primary sodium storage vessel and the overflow vessel are unguarded inside inerted cells. The cold traps, overflow heat exchanger, and connecting piping are unguarded inside inerted cells. Short runs of the intermediate sodium will be inside containment. For these runs, there would be a choice of either enclosing the applicable runs in a double wall or lining the effected cell, with the cell liner likely to be the preferred choice.

Other events that have been proposed as potentially challenging the containment are refueling accidents where a spent fuel assembly is dropped and its fission products released. This event would require a leak tight fuel handling cell, but the cell is already obliged to be leak tight in

50 Beeley, R.J., Mahlmeister, J.E., *Operating Experience with the Sodium Reactor Experiment and its Application to the Hallam Facility*, Atomics International, 1961.

order to keep air out and argon or helium in. A control rod ejection accident is another that has been postulated, but for oxide fueled systems it turns out to have acceptable consequences provided the reactor shutdown system does its job. Moreover, for the low pressure drop system envisioned in the concept being considered, it is difficult to postulate an event that could result in a control rod ejection. Local subassembly blockage is a very reasonable postulated accident, particularly since it has already happened on two sodium cooled reactors, SRE and Fermi-1. However, in both of those cases the consequences were contained by the primary system and it is difficult to conceive of any sequence following from a subassembly blockage that would breach the primary system and impose special requirements on a containment building.

Illustrative of this subject is a press information booklet on the Sodium Reactor Experiment (SRE) that was prepared sometime around 1957.[51] "The SRE Building is not designed for containment in the sense that it can withstand an internal pressure and still be leak tight. An important aspect of the sodium-graphite concept is that it does not require a specially constructed containment shell. No foreseeable nuclear accident could so increase the pressure that an external containment shell would be necessary." While the SRE was not a fast reactor, it was one of the sodium cooled reactors that experienced a partial meltdown from which there was no significant release.

Some operations will be undertaken in the containment that would require cell liners in places other than the refueling cell. For example, the cold traps will need to be replaced periodically and the cover gas processing system will contain components that require periodic replacement. Every place there are cell liners, it will be necessary to make provisions for them to be tested for leak tightness, which will probably involve pressurizing the cell to 5 or possibly 10 psi. Depending on the design details, it may turn out that it is most efficient to line the entire containment.

Based upon the forgoing, one could reasonably conclude that the primary need for containment is to provide adequate shielding during operation and accommodate postulated refueling events. There is one bug in the ointment. Fairly early in the development of breeder reactors, there emerged the famous Bethe-Tate model.[52] This model postulates an event that would normally trip the reactor but fails to do so for some reason, overheating of the sodium coolant in the core to the point of boiling, sodium voiding which injects positive reactivity, and then a massive core meltdown followed by the unsupported upper part of the core falling into the debris at the bottom. The result is the hypothetical core disruptive accident or HCDA as it is generally known.

The Fermi-1 designers were caught up with the HCDA during licensing and what amounted to a "core catcher" was installed in the bottom of the reactor vessel. Of course, the unintended consequence was that a piece from the core catcher broke loose during plant operation blocking flow to several fuel assemblies which led to the famous partial core melt at Fermi-1 and contributed significantly to the demise of that plant. Maybe a core catcher isn't such a good idea

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51*Technical Information, The Sodium Reactor Experiment*; published by Atomics International; undated
52H. A. Bethe, J. H. Tait; *An estimate order of magnitude of the explosion when the core of a fast reactor collapses*; UKAEA-RHM (56)/113, U. K. Atomic Energy Authority, Risley, Warrington, Lancashire, England, 1956

after all. What this proves is that it is not constructive to go overboard with safety provisions and installed complexity for hypothetical events when designing a nuclear power plant.

Since the FFTF was unlicensed[53], the next time a sodium cooled reactor was obliged to confront the regulatory process was the CRBRP project. The initial approach of that project was to install a diverse and redundant reactor shutdown system and provide a containment vessel for "defense in depth", even though there was no postulated event that required it. There was no core catcher either inside the vessel or external to it.[54] The CRBRP containment was a 186 ft. diameter steel shell designed to a pressure of 10 psig. From Probabilistic Risk Analyses (PRA) it was estimated that the probability of failure of the shutdown system was less than 10^{-6} per year.

When the licensing process was engaged, all kinds of NRC questions surfaced dealing with the HCDA. The project wound up enlisting HCDA experts from Argonne who performed a series of analyses that were furnished to the NRC for evaluation. Numerous meetings occurred focused on the topic. The end result was that even though the containment wasn't necessary for any realistic postulated event, the project was required to add a concrete confinement building outside the containment along with a containment cooling system, an air filtration system on the exhaust between the containment and the confinement and other systems deemed necessary by the NRC. In order to present an event that could challenge containment, the project postulated (probably with considerable "encouragement" by the NRC) a major breech of the 35,000 gallon sodium storage tank (which is normally empty) simultaneous with failure to maintain the inert environment in the cell in which that tank is normally contained, causing a sodium fire lasting for 550 hours. This event is at odds with the single failure criterion unless the tank if filled with sodium and the cell deinerted, in which case the reactor would have been shut down for long enough that the Na^{24} would have decayed to near ambient, i.e. two weeks or more. This sodium was also obliged to contain a radiological source term for assessment of site suitability that was somehow based on a specified (300 MW-sec) HCDA, which of course is an impossible combination. So it appears the NRC was going to buy into a non-mechanistic (i.e. non-deterministic) approach with a specified fire and source term as the basis for containment. It also appears that the ASLB had not bought into the idea yet[55] and of course, the interveners were demanding that the HCDA be a DBA.[56] The big problem with the HCDA being a DBA is that there are all kinds of HCDAs, depending on a plethora of assumptions. It is highly likely that HCDA=DBA means there would be no LMFBRs licensed by the NRC, which is likely exactly what the interveners had in mind.

Since then, the situation has not improved. Some entities began to endorse metal fuels as the solution to the loss of flow (LOF) and transient overpower (TOP) events without scram. For this case, it is the absence of a strong Doppler coefficient in metal fuels that comes to the rescue in ameliorating these hypothetical scenarios. So a perfectly acceptable fuel form that was selected

53There was regulatory review of the FFTF design but at that time the regulator and the sponsoring agency, the Division of Reactor Development and Technology (RDT), were both part of the AEC. The breakup of the AEC did not occur until 1975, over three years after the CRBRP project had been initiated and utility industry participation for that project secured.
54The precedent for the external core catcher was probably the PFR in the UK, which had installed an elaborate external core catcher under the reactor vessel.
55Partial Initial Decision, ASLB No. 75-291-12, Feb 26, 1983
56Sholly, Steven, UCS Comments on Supplement to FES on CRBRP, Sept. 13. 1982

in the first place mainly because of its strong Doppler coefficient and is supported with extensive and worldwide characterization data is proposed to be abandoned in favor of an inferior fuel form for the sake of a theoretical event. Keep in mind that it was the EBR-1 meltdown that at least partially supported the need for an effective Doppler coefficient in LMFBRs. In their JSFR-1500 design the Japanese have gone even further. They have taken to referring to beyond design basis events as "design extension conditions"[57] for which they plan to use deterministic means for evaluation. They have designed a core catcher in the bottom of their reactor vessel and have modified their fuel assemblies by providing them with an inner duct which is intended to carry molten fuel away from the core. With the core catcher, they are taking the same path as Fermi-1 and inviting the same result. With their inner duct, they are compromising their fuel performance both in breeding ratio and heat generation per assembly. Economics are certain to be compromised. The Russians have not excluded themselves from this picture. They have incorporated into BN-800, 1) a passive emergency shut-down system with hydraulically suspended rods (which is possibly not a half bad idea); 2) a special cavity over the core to reduce sodium void reactivity effect; and 3) a core catcher in the lower part of the reactor vessel to collect and retain core debris under the conditions of "heavy accidents".[58]

If one considers the serious accidents that have befallen the nuclear industry, they were the result of the unforeseen and the design bases provided little in the way of prevention or protection that was not incidental. In the case of Three Mile Island, the potential for an operator to misread his instrumentation and as a result take a course of action exactly contrary to that which would have been in the best interest of the plant had not been evaluated. (Of course, it would be impossible to evaluate every possible error that could be made in the operation of a plant. However, the complexity of commercial PWRs partly arising from their regulation certainly contributed to cause this operator error.[59] This contribution to the TMI accident has been recognized implicitly with the requirement for increased technical support for all nuclear power plant senior operators.) Although the containment building mitigated the consequences of that event, a containment building designed for a much lower pressure would have been equally effective. Even a confinement building alone probably would have been sufficient to keep site boundary doses well below allowable limits.

The basic problem at Fukushima was the plants had not been designed for station blackout. Why is it that the regulator required the plant to be designed for a double ended pipe break in the primary system but not for the case where all the lights go out? The design bases for Fukushima did little to mitigate the consequences of that event. Then there is Chernobyl where the operator was seemingly out of his mind. Is there a design basis for that? Whatever design bases there was for that plant seemingly didn't do much good. There will undoubtedly be more accidents and the regulators will realize they goofed again and add more requirements without eliminating the ones that may have caused the problem in the first place. The Three Mile Island operator; who did what he believed to be correct and had been trained for; simply did not realize he had been duped by his instrumentation. When plants become overwhelmingly complicated, that is

57Ichimiya, M.; Mizuna, T.; Kotake, S.; *A Next Generation Sodium Cooled Fast Reactor Concept and its R&D Program*; Nuclear Engineering and Technology, Vol. 39, Number 3, June 2007
58Pakhomov, Ilia; "BN-600 and BN-800 Operating Experience" Gen. IV International Forum; Dec. 19, 2018
59Flashing of the reference leg on the pressurizer level indicator caused erroneous high level indication during the TMI event, which led the operator to shut down the injection pumps. This was a phenomenon that was often postulated and well known on plants long before the TMI accident.

bound to happen. The point of all this is that if it has been dreamed of and made a regulatory requirement, it probably won't happen. We are still waiting for the first double ended primary system pipe break to occur on a LWR anywhere in the world. If nobody has ever considered it before, it may happen. For the case of the LMFBR there has been way too much dreaming about the HCDA.

The SRE designers had the right idea. With the exception of the refueling cell and the reactor vault, the only purpose served by the building surrounding the primary system is shielding. The refueling cell should be lined and leak tested since the reactor vessel may be open (OVR) and fuel will be transferred either in an EVTM or inside sodium containing shrouds within that cell. Since the reactor vault includes the cold traps, cover gas treatment systems, and the overflow system, it too should be lined, leak tested, and inerted. There is no reason for lining and inerting the IHX vaults. Approximately five feet of concrete is required to provide access to adjacent structures and provide a generally benign environment around the plant. Interior walls inside containment are likely to be ~4 ft. in thickness in the interest of support for upper floors and to provide shielding from the reactor and PHTS. A shield of about 5 ft. concrete thickness should surround the reactor vessel.

The experience of the CRBRP project reveals that it does little good to install features in the design for vague and arbitrarily defined "defense in depth". Under the circumstances, one should propose a design that is reasonable, is based on events that have some kind of decent probability of actually happening and let the regulatory process take its course. The containment should be rectilinear, include the primary heat transport system, the overflow vessel, the primary sodium storage tank, the primary system cold traps, the cover gas processing system, the refueling cell, possibly a PHTS drain tank that would permit draining one loop, necessary support systems, and as little else as possible. IHTS expansion loops inside containment should be avoided by placing the IHXs near the containment boundary and using bellows seals for the IHTS penetrations. It is much more economic to accommodate IHTS piping expansion within the steam generator building than inside containment. As was mentioned in section 16, the proposed JSFR-1500 containment was expected to have a volume of 20,000 m^3. With the reactor vessel shortened it should be possible to keep the containment volume in the vicinity of 20,000 m^3. This 20,000 m^3 estimate is easily consistent with the dimensions shown on figure 20 in Section 6 and figure 32 in Section 7.

There may be some reasonable things that could be done to further enhance the safety of the design. Some of the suggestions in this paper, such as shortening the reactor vessel will improve control rod drive reliability. Simplification of the plant makes it more comprehensible for those who will be charged with operating it. Certainly, the plant should be designed for station blackout with strong natural circulation capability in the PHTS that would become operative of the occasion of any LOF event. To that end it may prove possible to eliminate safety-grade electric power including the emergency diesels, preserving only battery power supplies for instrumentation. Such a step would be a big safety improvement. Specific suggestions in this area are the topic of Section 11. Other steps to make the reactor shutdown system more reliable or innovations such as self actuating shutdown systems may be reasonable approaches that could be taken to deal with this matter. This topic is treated in Section 10. Adding abstractions such as "defense in depth" has been proven to be an unproductive course of action.

Regulatory mandates to add features to the containment to deal with the HCDA are probably unavoidable, but the project leadership must insist that any such features are beyond the design basis and not advocated by the plant owner. There is likely to be a way to design the reactor vault so that it serves as an inner containment while the containment building itself acts as the outer containment. This inner containment approach seems to have been proposed for the JSFR-1500, where a containment is shown surrounding the reactor vessel.[60] Other features may be added if they do not interfere with the plant's operability and don't add excessive costs. The extremes CRBRP went to are clearly beyond the pale and contributed to the Project's termination for excessive cost. Ultimately, the plant owner can walk away from the project if licensing requirements become unreasonably costly and make the plant uneconomic. With this in mind, it would be judicious to minimize plant investment until there is a clear path through licensing. Certainly, no early component fabrication should be undertaken as was done on the CRBRP.

At this point, it is necessary to enumerate the cost reduction measures inherent in the design being advanced as compared with CRBRP as they pertain to the containment:

34. Adopt rectilinear containment structure vs. cylindrical and eliminate the requirement for a single elevation basemat.
35. Significantly reduce containment volume.
36. Eliminate requirement for single elevation operating floor. The requirement to have a floor that is accessible during operation inside containment is unnecessary.
37. Reduce design pressure from 10 psi to 5 psi or any pressure that will permit reliable leak testing
38. Eliminate confinement building
39. Eliminate containment cooling system
40. Eliminate air filtration processes that extend beyond CAPS
41. Eliminate all cell liners not required for containment leak testing

10 Reactor control and shutdown systems

As has been stated before, the CRBRP design included two redundant and diverse control rod drive systems. The primary system design was similar to that used on both LMFBRs and LWRs on many previous plants. The secondary system was of a more recent vintage. Plant control was accomplished using the primary system while the secondary system was fully withdrawn during operation. Both systems would unlatch and the control rods would fall into the core under the influence of gravity with a scram spring assist on reactor trips for the primary system and a hydraulic assist on the secondary system.

The CRBRP project evaluated three different design approaches for the primary system, the Collapsible Rotor-Roller Nut Mechanism (CRRNM), the Magnetic Jack, and the Ball Screw.[61]

60 H. Ohira, N. Uto, *Progress on Fast Reactor Development in Japan*, Meeting of the Technical Working Group on Fast Reactors, June 20-22, 2012

61Pitterly, T. A., Lagally, H. O.; Review of FFTF and CRBRP Control Rod System Designs; October 4, 1977.

All three systems had a track record of extensive accumulated experience. The CRRNM design was selected primarily because of the extensive experience that had been accumulated on both commercial and naval reactors exceeded that of the other two concepts combined with its ability to accommodate the requirements for very fine control and a minimum 1000 lb. drive-in capability to free a stuck rod. The CRRNM was the only shutdown system installed on the FFTF. The system was extensively tested as a part of both the FFTF and the CRBRP projects to ensure it would perform as intended even though it was essentially the same system that had been used on many earlier reactors adapted to operate within a LMFBR environment.

Early in the conceptual design stage of the CRBRP project, it was decided that a second totally diverse system would be incorporated into the design. The thinking was that two independent diverse shutdown systems would reduce the failure to scram probability to below 10^{-6} per reactor year making design and analysis for ATWS events unnecessary. That thinking was based on ATWS probabilistic analyses which suggested failure to scram probability was less than 10^{-3} per reactor year for a single system. Thus, two diverse and redundant systems should reduce the failure to scram probability to the range of 10^{-6}. It is worth mentioning that the so-called "line in the sand" on the CRBRP project was more directed at resistance to install a core catcher or anything resembling one. That mindset appears to have been primarily driven by the Fermi-1 experience rather than a refusal to consider and analyze the consequences of HCDAs.

As one of the three reactor vendors involved in the CRBRP project, General Electric was assigned the task of developing the secondary shutdown system. The system would be required to shut down the reactor from full power conditions to hot standby temperature (~600°F) in the unlikely event of a stuck rod following the maximum anticipated reactivity addition fault in the reactor.[62] The original design with a homogeneous core called for just four of the control rod drive mechanisms (CRDMs) to be of the secondary design. Thus, the reactivity necessary to meet the above requirements were to be accomplished by just three rods. In the heterogeneous configuration, the Westinghouse designers decided they could reduce the number of primary control rods allowing the number of secondary control rods to be increased to six. The two control rod drive system designs, primary and secondary are shown in the figure below.[63]

62The same stuck rod criterion applied to the primary system as well.
63The figure and much of the subsequent description is drawn from McKeehan, E. R., Sim, R. G.; Clinch River Breeder Reactor Secondary Control Rod System; US-ERDA/Japanese-PNC Working Group Seminar on LMFBR Components 12/5-8/1977; September 14, 1977

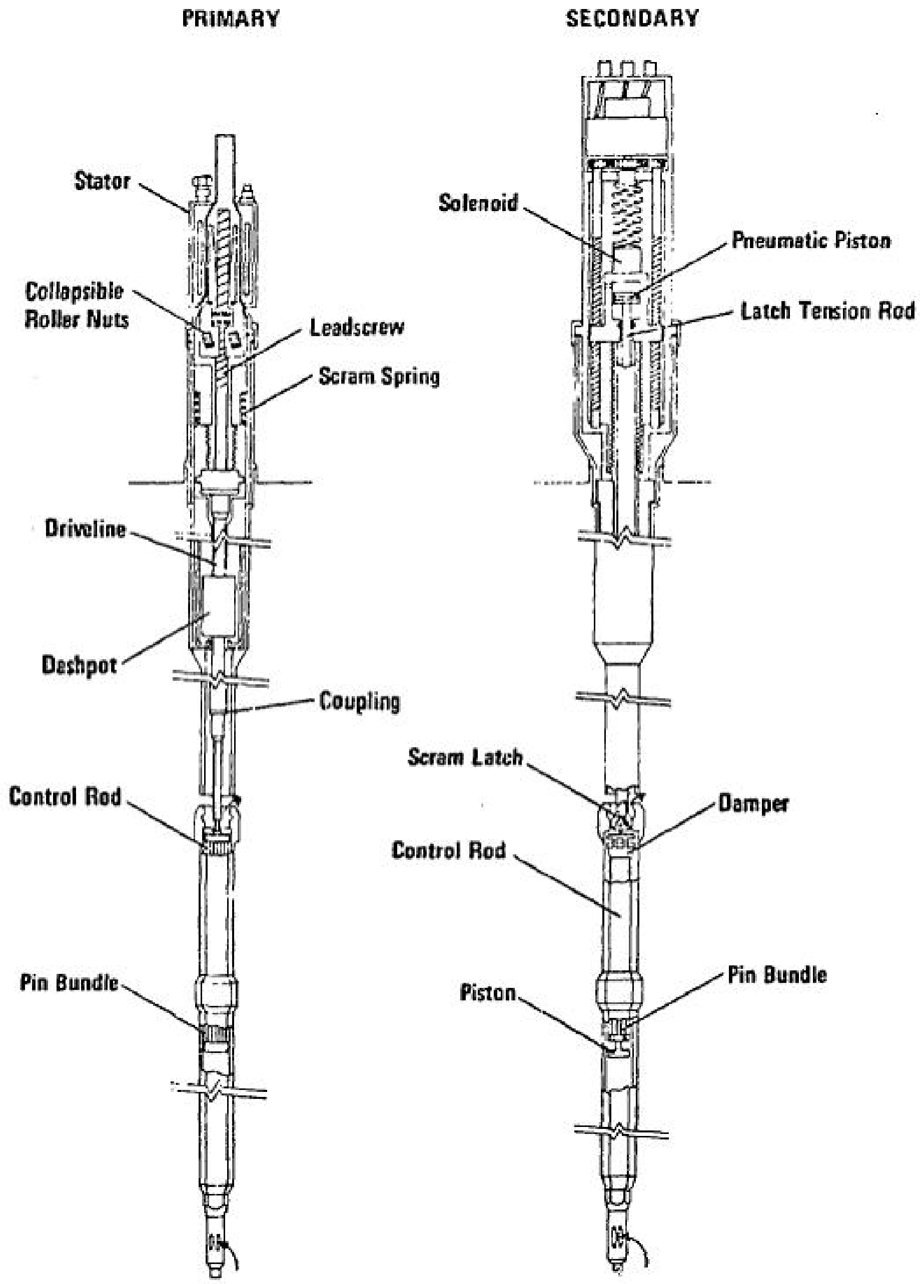

Figure 37 CRBRP primary and secondary control systems

In the primary design, the scram latch is located within the mechanism outside the reactor. For the secondary system, the scram latch is located just above the control assembly. In the primary system, the control assembly when unlatched is driven by gravity and a spring. In the secondary system, a piston is placed below the assembly with high pressure primary coolant above the piston and low pressure coolant below the piston, thus driving the control rod in by both gravity and hydraulics. The number of absorber pins in the primary assembly was 37 arranged in a hexagonal array. In the secondary assembly, 31 pins are arranged in a circular array. The bundle for the primary system is hexagonal and operates within the hexagonal duct while the bundle for the secondary system is circular and the assembly operates within a cylindrical guide tube that is installed within the duct. Tripping of the primary system is accomplished by de-energizing the latch magnets which then release the roller nuts from the lead screw with a spring assist. Tripping the secondary system is accomplished by spring load to open pneumatic valves which vent the pneumatic piston located within the mechanism leading to ¼ inch of motion of the tension rod that activates the scram latch. Each system used its own sensors and logic circuits, which are diverse from one another. Even the systems for breaking the control rod motion following a trip were diverse. The secondary system designers did a fairly credible job demonstrating that the two systems were quite immune to common mode failure problems between each other.

The two systems together should have been enough to eliminate HCDAs from further consideration on the CRBRP project but they weren't because of an ingrained Bethe Tait mindset that prevailed at the NRC and certain national laboratories and likely remains intact today. At the time CRBRP licensing was being pursued there existed numerous groups in the country that were committed to the analyses of HCDAs. There was a conviction held by some that the LMFBR will never win public acceptance unless it can be shown to be able to survive an HCDA. The fault in this reasoning is it misses the point. The issue is the reliability of the reactor shutdown system. Why is it assumed that it is not possible to provide a shutdown system that is demonstrably reliable? There is a contrast between the shutdown system and the decay heat removal system. While there has been a total failure of a decay heat removal system there has never been a total failure of a reactor shutdown system when it was called upon. Fukushima Daiichi is the obvious example of a failure of the decay heat removal system but less obvious was the Browns Ferry unit #1 fire in 1975. The Browns Ferry fire rendered all the safety grade systems that were provided for decay heat removal inoperative but action on the part of an informed operating crew brought systems to bear for ultimate core cooling that had never been intended for that purpose. So there have been at least two total failures of the decay heat removal system worldwide vs. no failures of the reactor protective system.

If sufficient reliability in the reactor shutdown system cannot be achieved with two independent diverse systems such as was provided on CRBRP, what can be done to increase that confidence? The shutdown system must perform its intended function without any margin for mistakes. The counter to the public acceptance argument in the preceding paragraph is that public acceptance will not likely be won by a technology that is uneconomic or so unreliable that it cannot be demonstrated that the system intended to shut down the reactor is reliable and therefore mitigation systems are required. There are additional measures that could be taken to improve the shutdown system reliability even beyond that achieved on CRBRP.

Along these lines, one of the key issues that required resolution for the adaptation of the CRRNM on the CRBRP was the misalignment introduced by the clearances required by the rotating plug supports, both risers and bearings. Up to one inch of misalignment was expected from this source requiring numerous design features combined with extensive testing for its accommodation. The CRBRP project demonstrated through a test program that it could accommodate this misalignment however; with either of the refueling approaches herein recommended this misalignment source is greatly reduced. Not only does this simplify the design but it improves confidence that the shutdown system will function as intended when called upon. It is noted that the secondary system in CRBRP was less affected by this misalignment issue since it used just a thin tension rod to the scram latch. Another confidence building improvement flowing from the design approach proposed herein is the much shorter driveline arising from the reduced length reactor vessel, reducing uncertainties in the control rod location arising from differential thermal growth. Reducing the length of the reactor vessel also improves its seismic response reducing yet another source of misalignment in the control rod drives. Yet another shutdown system enhancement inherent to the design approach has been alluded to elsewhere; the reduced core pressure drop, which removes the most credible motive force for rod ejection accidents.

A concept that appears to have been nearly forgotten in the collective consciousness of the nuclear industry is the idea of using a "partial insertion" as a part of the reactor protective system. The partial insertion involves powering the control rods a fixed amount (such as five inches) into the core using the control rod drive mechanisms operating at a much higher speed than is normally used for in and out motion. The amount by which the control rods are partially inserted could be established as the amount necessary to reduce the reactor power to zero plus some suitable margin. The advantage of the partial insertion is it makes use of the force available from the control rod drive mechanisms to power the rods inward should there be any obstruction in the path of the control assemblies. Another advantage is it takes the scram breakers out of the picture. It was a scram breaker failure at the Salem plant that started the ATWS discussions with LWRs. For a system that includes diverse and redundant shutdown systems patterned after the CRBRP designs, one of the two systems, most likely the primary system could use the partial insertion approach for reactor protection while the secondary system would drop the control rods by gravity. (Of course, if power is lost to the primary CRDMs, they would unlatch and the rods would fully insert.) The reactor plant that used the partial insertion concept had collapsible rotor roller nut mechanisms.

The automatic reactor cutback is a variation of this principle. Cutbacks involve set points slightly below the reactor trip set points and are generally intended to prevent trips from occurring in the first place. They have the added advantage of injecting diversity into the total system that is intended to protect the reactor. Cutbacks would involve mechanically driving the primary control assemblies into the core and be less susceptible to issues which could interfere with the free fall of the control assemblies.

Another concept from the past (SRE) is to operate the control assemblies fully within thimbles that are inserted into the core. The thimbles would be evacuated of sodium and filled with cover gas. Using thimbles would eliminate flowing sodium from interfering with control assembly operation and would permit continuous monitoring of control assembly alignment. Such an arrangement eliminates much of the discussion about the shutdown system reliability. If the mechanisms unlatch, the assemblies will fall. The shortened reactor vessel and refueling scheme proposed are the avenues which make such an approach more attractive. This is the sort of thing that is feasible in a low pressure sodium system but unthinkable in a water cooled reactor.

The BN-800 has adopted an approach specific to this issue[64]. During normal operation, the reactor's control rods are hydraulically held in place by the flowing sodium at the top of the reactor core. If sodium flow decreases, the rods fall into vertical control rod channels in the reactor core and stop the chain reaction. Details are unknown from information available in the open literature, but from this description alone, there would not appear to be protection for TOP-ATWS events. Also, such a scheme would not be consistent with a plant designed for load following.

The unfortunate licensing fate of CRBRP prompted interest in the development of self actuated shutdown systems, especially at EPRI where two concepts had been patented by Larry Minnick, formerly with the Yankee Atomic Power Company and then an EPRI employee.[65] The above Combustion Engineering referenced report describes three systems, two of which had been earlier patented by Mr. Minnick and compares them with a more technically modest concept based on levitated balls.[66] All of these proposed design approaches were intended to protect against loss of flow events without scram. A more recent concept coming out of Japan is shown in the figure below.[67]

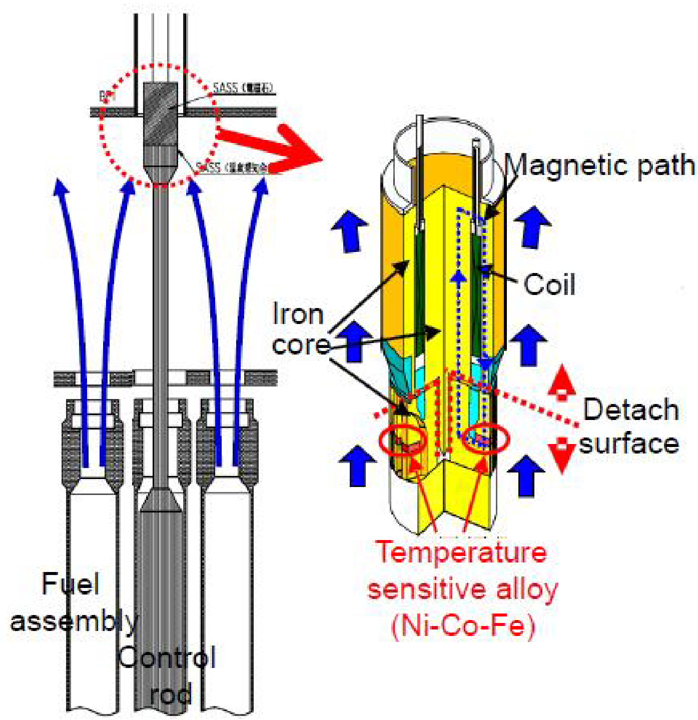

Figure 38 Curie point latch

64Nuclear Engineering International; *Fast Reactor Progress at Beloyarsk*; 14 Jan. 2011
65Dupen, C. F. G., Combustion Engineering Inc.; *Self Actuated Shutdown System for a Commercial Sized LMFBR*; Prepared for EPRI; August 1978
66The levitated balls contained a neutron absorbing material, most likely boron. The idea behind the levitated balls concept was that they would be kept levitated by primary system flow. When flow dropped below some set level, they would fall into the active core region stopping the chain reaction.
67Kubo, Shigenobu, Japan Atomic Energy Agency; *A safety design approach for sodium cooled fast reactor core toward commercialization in Japan*; IAEA Technical Working Group, Vienna, February 27-29, 2012

This concept involves using a latch that is held in place magnetically using an alloy with a curie point that is sufficiently high so the latch remains engaged during normal operation but unlatches when the surrounding temperature reaches the curie point of the selected material and is significantly higher than that corresponding to normal operation. Since this concept is actuated by temperature rather than flow, it would presumably protect against both TOP- and LOF-ATWS events. This sort of system could potentially be offered as a remedy for failure to scram scenarios. It might, for example, be installed as an additional latch on the secondary control rod system.

This section makes no specific recommendation other than the retention of the CRBRP system, which is considered adequate. A decision to incorporate any additional system or systems would be founded on further analyses or be part of a settlement with the regulator.

11 Eliminating 1E electric power

In section 3 the many advantages of sodium cooled reactors were enumerated in comparison with light water reactors. Mainly, these advantages follow from the relatively low energy content of the primary coolant. LWRs are required to deal with the prospect of a primary piping system failure which pressurizes and heats the containment building and requires containment cooling as well as both high and low pressure water makeup sources to ensure the core is kept covered with water. These various LWR safety systems require that there be reliable electric power in large amounts to power very large motors driving the vital safety system pumps.

The governing rule is Criterion 17 from 10CFR50 Appendix A and is stated below:

> *Criterion 17—Electric power systems*. An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.
>
> The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.
>
> Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all

onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

This criterion or something very close to it governed the design of the electric power system at Fukushima. Obviously, the criterion is not fool proof. It would be much better for nuclear power plants to be able to accommodate total loss of electric power from all sources without experiencing serious consequences. The design approach being presented has that capability. It is useful to consider the 1E loads on the CRBRP and evaluate what has been eliminated using the approach of this paper.

The CRBRP design provided for three separate diesel generators. The underlying reason for three was related to the choice of the SGAHRS for decay heat removal. The thinking was that if one loop were disabled, the single failure criterion required three separate diesel generators supplying three class 1E "divisions". Divisions 1 and 2 supplied essentially identical loads in their respective divisions while division 3 was provided primarily to supply loop 3 SGAHRS. The CRBRP project had not selected a diesel generator at the time of the project's termination but the peak loads on diesels 1 and 2 including a 15% allowance for expansion were expected to be around 3400 KW. This is equivalent to about 4600 horsepower – very stout diesels indeed. At these large sizes, diesels become difficult to start and require extensive maintenance and frequent testing to assure reasonable reliability. If a diesel is taken out of service during plant operation, operational constraints are likely to be applied by technical specifications because of the single failure criterion. If the engines could be much smaller some of these issues would vanish.

In order to evaluate possibilities, again CRBRP will be used as a convenient point of departure in view of the abundant information available in the PSAR. As stated above the 1E loads were estimated to be 3400 KW for each of divisions 1 & 2, which were based on an identified load of 2967 KW. These loads can be conveniently divided into eleven groups which will be separately treated below.

1. *Steam Generator Auxiliary Heat Removal (SGAHRS) 1414.7KW* By far the largest emergency load is that associated with the SGAHRS and their associated auxiliary feed water pumps which is replaced with naturally circulating systems in the proposed design that require no power other than instrumentation.
2. *Ex-Vessel Storage Tank (EVST) 253 KW* The proposed design approach would replace the active system used in CRBRP with a passively cooled system. As above, the only load remaining would be for monitoring instrumentation.
3. *Annulus cooling 1268.2 KW* The annulus cooling system was part of the CRBRP system for dealing with the HCDA, which is considered inappropriate as a design basis. Of this total load, 1103 KW occurs when there is no SGAHRS load – presumably because the

core is assumed to be on the floor of the reactor vault so SGAHRS wouldn't do much good. The remainder, 165.2 KW occurs coincident with SGAHRS operation and will be considered here as a deduction.

4. *Diesel support 29.6 KW* This number includes such things as fuel supply and engine cooling. It probably cannot be eliminated altogether but likely can be reduced with smaller engines.
5. *Control Room support 81.2 KW* This includes supply, return, and filter fans necessary to maintain an habitable environment in the control room and can probably not be eliminated. However, modern control rooms will inevitably be much smaller with much less electric power loads than was the case for CRBRP, which was patterned after nuclear plant control rooms that were extant at the time.
6. *Containment isolation 4.8 KW* This is a small load provided to supply power to valves necessary to isolate the containment that probably cannot be avoided.
7. *Primary sodium make-up pump 18KW* This pump was required to be supplied with 1E power because of the DHRS employed on CRBRP. Since the proposed design includes this feature, it is retained as an emergency load.
8. *Lighting 50 KW* This is lighting for areas of the plant necessary to be illuminated to enable operators to accomplish and maintain safe shutdown. A number similar to this will likely apply to any nuclear plant although it can probably be reduced by using the more efficient systems available today combined with the reduced size control room.
9. *Battery charger 115.5 KW* It is not clear why the CRBRP designers used the full load of the battery chargers to prepare their load list. The battery would be fully charged whenever there was a loss of off-site power so all that would be required would be enough power to supply the DC load. On CRBRP, the DC load was about 50 KW per division, which will be used for this analysis. There is no reason to expect it to scale with plant size.
10. *Emergency chilled water (including chiller) 583.5 KW* Slightly more than half of the CRBRP emergency chilled water load is for systems such as SGAHRS and EVST cooling that is not applicable to the concept proposed here. The control room load is very high and can certainly be reduced for a more modern control room design with fewer heat sources.
11. *Emergency service water 176.4 KW* This system cools the diesels, provides the heat sink for the emergency chilled water system and supplies the fire protection system. The system is cooled by a tower that requires power for its fans. With smaller diesels and a smaller chilled water system, this load can probably be reduced somewhat.

The table below summarizes and compares CRBRP requirements with the design approach proposed and constitutes CRM 42:

	CRBRP[68]	Proposed design
1. SGAHRS	1414.7	0
2. EVST	253	0
3. Containment annulus	165.2[69]	0
4. Diesel support	29.6	15

68Figures are drawn from Table 8.3-1A in the CRBRP PSAR.
69The annulus loads following a postulated HCDA are predicted to be 1268.2 KW.

5. Control room support	81.2	50
6. Containment isolation	4.8	5
7. Primary sodium makeup pump	18	40
8. Lighting	50	25
9. Battery charger	115.5	50
10. Emergency chilled water	583.5	200
11. Emergency service water	176.4	90
Totals	2967[70]	475

Table 3 Emergency loads per division, KW

Allowing 15% growth in loads as was done in the CRBRP PSAR leads to the requirement for 700 horsepower diesels and there would be just two of them. At this much smaller size, many options present themselves for diesels that are much more reliable, easier to start, and cheaper than the three 4600 horsepower machines that would have been required on CRBRP. This is also much smaller than the emergency diesels required on LWR plants offering another competitive advantage for the LMFBR.

However, there is an important distinction in the above table between the emergency loads for CRBRP and those for the proposed design. For the case of CRBRP, the loads are those necessary to maintain the plant in a safe state, e.g. decay heat removal from both the reactor and the EVST. (This is somewhat fictitious since it was shown in Section 8 that CRBRP actually had station blackout capability, at least for the decay heat removal system if not for the EVST. Despite this capability, as was stated in Section 8, the project never sought nor realized any licensing benefit from the capability.) For the proposed design, electric power necessary to maintain safe shutdown has been reduced to zero and the loads defined are those that would be desirable to be supplied in the event of loss of all offsite power. This constitutes CRM 43. It would be desirable to maintain control room habitability and lighting but not essential for safe shutdown. In the highly unusual event of a station blackout it may be necessary to monitor instrumentation locally with some sort of battery operated portable power supply and there likely would be the occasion for checking the status of valves by viewing manual indicators, or throttling back on naturally circulating air cooling of decay heat removal systems but a committed 1E power source is not needed. Thus, the emergency diesels provided to accommodate loss of offsite power and the entire emergency power supply may not need to be 1E.

A careful definition of how this would be done would be a worthy project for further study. Operating a plant that is designed for station blackout and knowledge of exactly what is to be done by the operators of the plant should one occur is a far better alternative than being reliant on large 1E diesel generators and experiencing core damage when a station blackout occurs.

There is one load on the list above that warrants special attention – the primary sodium makeup pumps. As the reactor cools down, the sodium level in the reactor vessel will drop. It is necessary to maintain the outlet nozzles covered to ensure primary system flow to the PRACS

70Note that the numbers in this column do not add to this figure despite the fact that they were all drawn from the same table in the PSAR. The number shown is the figure cited in the PSAR table as the total.

heat exchangers in the IHXs. There are four possible approaches for dealing with this issue. 1) A DRACS could be incorporated into the reactor vessel. With open vessel refueling, a DRACS would require its own nozzles to carry the NaK to the DRACS cooler and could complicate refueling. With the single rotating plug concept, DRACS would be entirely feasible, penetrating the horizontal baffle and head outside the rotating plug. The DRACS heat exchanger(s) would need to be located adjacent to the vessel wall in the outlet plenum so as not to interfere with refueling and there would need to be one or two down-comers within the vessel penetrating the core support cone to return the cooled sodium to the inlet plenum. 2) The reactor vessel height could be increased so there would be no possibility of uncovering the outlet nozzles. Using a volumetric coefficient of expansion of sodium of $1.6 \times 10^{-4}/°F$, accommodation of 10% volume change from 400°F to 1000°F could be accomplished by adding two ft. to the reactor length. 3) A 1E power supply expressly committed to the makeup pumps could be adopted. Since the makeup pumps are small loads, the power supply (about 40 KW) could be furnished by batteries, but the pumps must be capable of continuous operation if they are necessary for OHRS function. 4) Since the overflow tank is provided with a cover gas, overflow tank inventory could be transferred to the reactor vessel by pressurizing the overflow tank above reactor cover gas pressure.

12 The devil is in the details: ALM, SWRPS, IGRP, Rad. Waste and others

This section deals primarily with six remaining systems, five of which are unique to the LMFBR and the sixth, Liquid Radioactive Waste (LRW), is shared with LWRs. The five unique systems are the Auxiliary Liquid Metal (ALM), Sodium Water Reaction Products System (SWRPS), the Radioactive Argon Processing System (RAPS), the Cell Area Processing System (CAPS), and the Inert Gas Receiving and Processing (IGRP) system.

Auxiliary Liquid Metal

The Auxiliary Liquid Metal (ALM) system subsystems are 1) Sodium and NaK Receiving, 2) Primary Sodium Storage and Processing, 3) EVS Processing, 4) Primary Cold Trap NaK Cooling, and 5) Intermediate Sodium Processing. This subsection will focus on those subsystems which offer the greatest opportunity for cost reduction.

The Sodium and NaK Receiving subsystem was designed to receive sodium in 80,000 gal. railcars and NaK in steel drums. The system on CRBRP was designed for the capability to unload a railcar in 16 hours, a requirement that might be questioned. Otherwise, this system is straightforward and there isn't much that can be said about it.

The Primary Sodium Storage and Processing subsystem is quite another matter. The system was designed to have the capability to totally drain the entire primary system into an in-containment storage tank, two tanks located in the Reactor Service Building (RSB), and the balance into the

Overflow Vessel. The probability of being required to drain the entire primary system including the RV does not justify this requirement. The table below reproduced from the CRBRP PSAR is illuminating.

	Ft^3	Gallons
Reactor Vessel Drain		
Reactor Vessel	13,328	99,960
PHTS piping which drains to the Reactor Vessel	1,950	14,625
Reduction in PHTS Na level	2,328	17,460
EVST reserve	1,000	7,500
Overflow/makeup piping	625	4,688
In-Leakage from a Single IHX Leak	1,350	10,125
Total usable storage required	20,581	154,358

Table 4 Total Primary Storage Requirements (Sodium Volumes at 400°F.)

It is presumed that "Reduction in PHTS Na Level" means sodium in the PHTS loops that does not drain into the RV. "EVST Reserve" is also not clear (and not explained). The "In-leakage from a Single IHX Leak" probably is a calculation of the amount of IHTS sodium that would wind up in the primary system should there be an IHX tube failure. Elsewhere in the PSAR text it is stated that the in-containment primary sodium storage tank and the overflow vessel each have 35,000 gal capacities. Therefore, since the PSAR states that the total primary system storage capacity is 190,000 gallons, the two tanks located in the RSB must be sized at 60,000 gallons each. It is proposed that both of these tanks be deleted as CRM 45. The 35,000 gallon in-containment primary storage tank would have been sufficient to drain a single loop. Also, the 35,000 gallon Overflow Vessel (OV) was sized in part to meet the requirement for draining the total primary system and does not need to have a 35,000 gallons capacity to meet its requirements as an OV. The PSAR does state that the net sodium overflow volume of the PHTS from 400°F to THDV conditions is 1439 ft^3, which is the principal requirement on the OV. Allowing for TOP transients at power, an OV size of about 2000 ft^3 should be adequate and is captured as CRM 46. These numbers may need to be changed for the commercial-sized plant, but the underlying requirement would carry over.

The primary system cold traps provide yet another opportunity for significant cost reduction. Sodium proceeding to these traps is cooled first by a regenerative heat exchanger then a NaK cooler. The NaK in turn is cooled by a Dowtherm J system then chilled water. In contrast, the IHTS sodium cold traps are cooled also by a regenerative heat exchanger but in place of NaK, HVAC air is the ultimate heat sink. Use of the IHTS approach raises safety issues with the highly radioactive primary sodium should there be a sodium leak from the cold trap. However, the Dowtherm J has a temperature use range from -100°F to 575°F so it could be used in place of the NaK. This measure is captured as CRM 47.

There is a drain vessel associated with the primary sodium makeup pumps. There is no obvious need for a drain vessel for these pumps that is separate from the in-containment primary system storage tank and it should be considered for elimination as CRM 48. Elimination of components in this system eliminates ALM piping connections, inert gas services, associated instrumentation and operational monitoring, simplifying plant operation.

The EVS Processing system consists of two loops with EVST sodium leading to a sodium to NaK heat exchanger then to an EM pump and back to the EVST. On the NaK side, coolant flows to an air blast heat exchanger, an EM pump, and back to the sodium to NaK heat exchanger. In addition, there is a third loop with naturally circulating sodium from the EVST leading to a sodium to NaK heat exchanger. The NaK side also naturally circulates to a natural draft heat exchanger. The third system was made necessary to meet the single failure criterion with one loop out of service.

The CRBRP system may well be the optimum system for this service. A change that might be considered would be to borrow the idea that led to the Overflow Heat Removal System; namely, make the two normal heat removal systems naturally circulating and have the third loop circulated by an EM pump to a sodium to NaK heat exchanger that ties into the PRACS on its NaK side.

The Intermediate Sodium Processing subsystem may also be close to the optimum system. However, since the design concept has just two intermediate loops, there would be just two Intermediate Sodium Processing subsystems as opposed to three in CRBRP.

SWRPS

Beginning with the SWRPS, since a drawing of the system is not included in the CRBRP PSAR for some reason; the figure below captures the essential features of a typical system. If there were to occur a significant sodium water reaction such as caused by a tube rupture inside the steam generator, large quantities of hydrogen would be produced ($2Na + 2H_20 \rightarrow 2NaOH + H_2$) causing a pressure pulse on the sodium side, causing a surge of sodium into the expansion tank, failing rupture disks to the separation tanks, and shutting fast acting feedwater and IHTS sodium valves. The inlet to the separation tanks enters the tanks tangentially causing a swirling action that separates the reaction product (NaOH) from un-reacted sodium and hydrogen. The un-reacted sodium drains into the sodium dump tank while the hydrogen is directed up a flare stack, failing a rupture disk there. If the argon float on the expansion tank cannot keep up with the increasing pressure there, a rupture disk could also fail on the argon side of the expansion tank, venting argon to the dump tank.

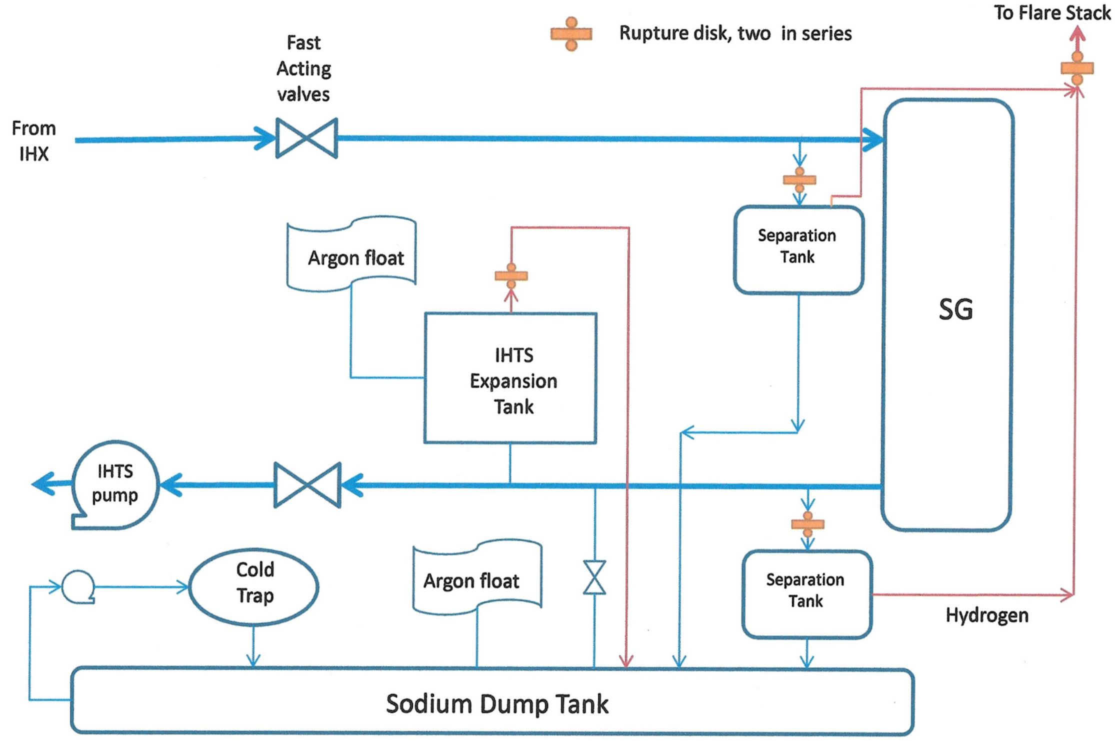

Figure 39 Typical Sodium Water Reaction Products System

Hydrogen passing up the flare stack fails rupture disks located there. In the interest of preventing an explosive mixture of hydrogen and oxygen in the atmosphere above the flare stack, the hydrogen is ignited as it passes a non-return valve at the stack exit. The stack itself is normally purged with nitrogen between the rupture disks and the non-return valve to prevent an explosive situation from occurring in the stack should there be a SWRPS actuation. Nitrogen is also supplied to the sodium system to dilute the hydrogen and replace the sodium that is rapidly being drained.

In the design of this system, the lines to the separation tanks and the expansion tank should be as close to the steam generator as possible in the interest of providing relief to the steam generator and minimizing contamination of the IHTS. Some systems may provide a cover gas in the sodium side of the steam generator expressly for the purpose of absorbing some of the pressure pulse attendant to a sodium water reaction. Following any significant sodium water reaction, the effected steam generator may require replacement and the separation tanks will be cut out and replaced. Reaction products in the dump tank will be removed by the associated cold trap and any remaining IHTS sodium will likely be drained into the dump tank for processing by the cold trap.

This is probably a system that is ripe with opportunities for improvement, but none are obvious, so the following discussion will be directed to areas where opportunities are more transparent.

The Inert Gas Receiving and Processing (IGRP) system on the CRBRP was comprised of four subsystems, the Argon Distribution System, the Nitrogen Distribution System, the Radioactive Argon Processing System (RAPS), and the Cell Area Processing System (CAPS). The better understanding of the system will be achieved by beginning with RAPS, proceeding to CAPS, then the IGRP and Nitrogen Distribution systems.

RAPS

The figure below shows a process flow diagram for RAPS as it was designed for CRBRP.

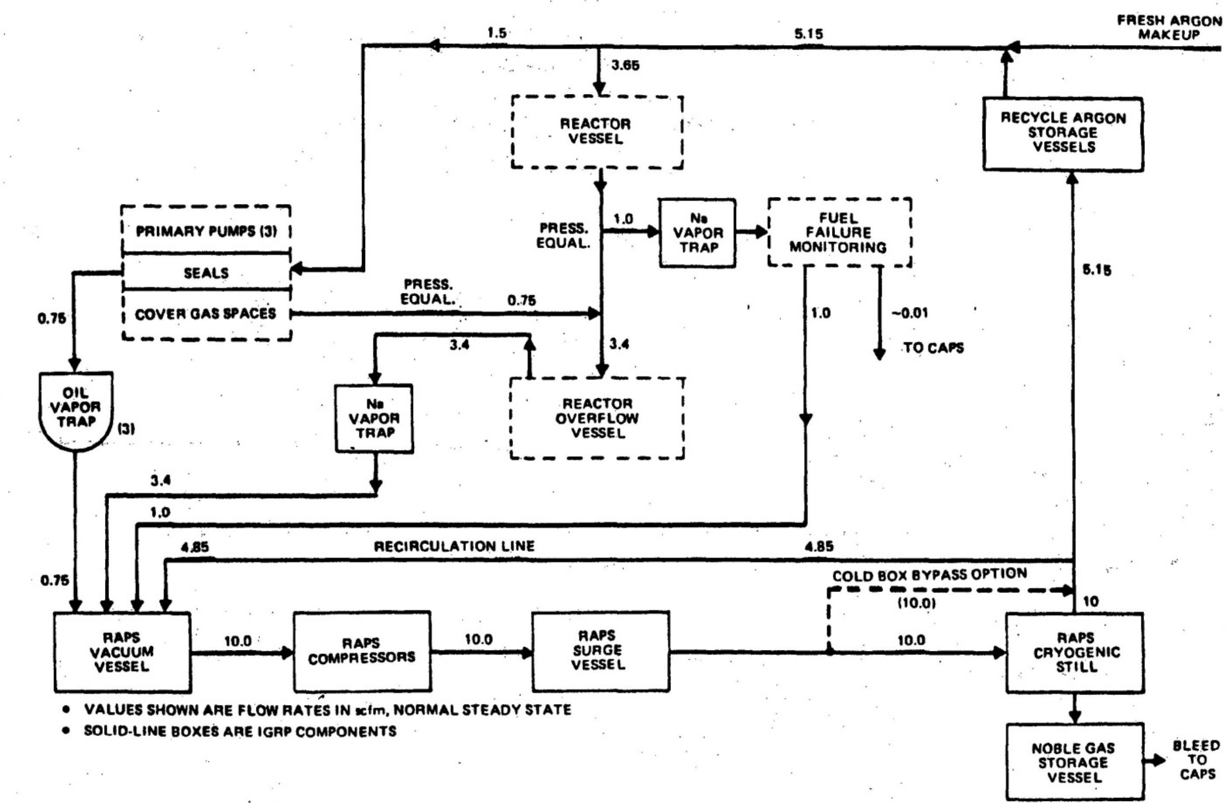

Figure 40 CRBRP Radioactive Argon Processing System (RAPS)

The numbers shown in the figure are the flow rates in SCF/min. In the center of the diagram, a pressure equalizing line for the reactor vessel, overflow vessel, and PHTS pumps is evident. RAPS is intended to serve those components plus the fuel failure monitoring system. The vessels and pumps exhaust through vapor traps to a vacuum vessel, then to a compressor and surge vessel on their way to a cryogenic still that separates the xenon and krypton from the argon before proceeding to the recycle argon storage vessel. The sodium vapor traps are provided to condense any sodium iodide that might be entrained in the cover gas preventing it from flowing into the processing system.

The fuel failure monitoring system is an obvious candidate for elimination. It was planned for installation on CRBRP partly because of the government's involvement and an associated intent to derive experimental data from that project. Nonetheless, the CRBRP PSAR acknowledged "the failed fuel monitoring system is not required to operate when there are more than 60 failed fuel pins", which corresponds to 0.17% failed fuel. The plant was designed for operation with 1% failed fuel. This amounts basically to an acknowledgment that the system is not required. The elimination of this system was previously identified as CRM 5, which eliminated gas tagging of fuel assemblies.

Since the proposed design concept replaces the centrifugal PHTS pumps with EM pumps which do not require cover gas, that part of the RAPS system can be dispensed with. It is worthy of pointing out that the RAPS included an oil vapor trap downstream of the PHTS pumps. The PHTS pump seals were gas seals that were lubricated with oil. Although most of the oil passing the seals would be carried away by the RAPS, inevitably some of the seal oil would find its way into the PHTS sodium, where, after reacting with the sodium, it would need to be removed by the cold traps, shortening their life. Thus, another advantage of EM pumps manifests itself. The pump seal purge was the single supply to RAPS than could not be halted. With the pump seal purge out of the picture, the opportunity of halting RAPS processing flow becomes an option – one that will be exercised later in this discussion.

A third simplification would be to eliminate the recirculation line. The CRBRP PSAR states "The recirculation-loop feature in RAPS permits maintaining a steady throughput under conditions of changing output demand requirements." This recirculation line is a relatively small matter, but it was made necessary on CRBRP since the compressor was operated at constant flow rate of 10 CFM. The compressor for the current design will be variable speed, eliminating the need for recirculation. Once the pump seal purge and failed fuel detection purge have been eliminated, the only real reason for having RAPS is to remove Kr^{85}. Among the fission product inert gasses, Kr^{85} is the only one with a relatively long half life of 10.76 years. The next longest lived inert gas fission products are Xe^{131m} with a half life of 11.96 days and Xe^{133} with a half life of 5.27 days. Of these two, Xe^{133} is the more challenging because of its greater fission yield.

In the CRBRP design, the cryostill was intended to be operated at liquid argon temperature of -302.5°F. This compares with liquidus temperatures for krypton and xenon of -244.1°F and -162.6°F respectively. The designer's intention was that krypton and xenon (mainly krypton) would collect in the cryostill and once a year they would be bled off into the noble gas storage vessel. After a few weeks, the only gas left in the storage vessel would be Kr^{85}, which would subsequently be discharged to atmosphere in a controlled fashion. Since there would be only about 700 Ci of Kr^{85} produced in a year, such a procedure would be acceptable.

The xenon isotopes do complicate the design of RAPS, however, since so much xenon is produced in fission. The table below reproduces data from the CRBRP PSAR (except for the last column) defining the production rate of the inert gas isotopes assuming operation with 1% failed fuel. The fifth column shows the radioactive load in Curies on the cryostill after a year of operation. With the exception of Kr^{85}, these are all equilibrium loads which are reached after a month of operation – only the Kr^{85} continues to build. The inventory in the cryostill gives a fair approximation of what the inventory of radioactive fission gases in the cover gas would have been had there been no RAPS. This observation will become important in the discussion that follows. The final column shows the energy load associated with the named isotope on the cryostill.

Isotope	Half life	Decay constant (per day)	Input rate (Ci/day)	RAPS cryostill load (Ci)	Energy load (watts)
Xe^{131m}	11.96 day	0.058	112	1900	0.5
Xe^{133m}	2.23 day	0.306	3760	1.1E4	2.9
Xe^{133}	5.27 day	0.131	65,100	4.7E5	319
Xe^{135m}	15.7 min.	63.6	95,600	2.0	0
Xe^{135}	9.16 hr.	1.81	334,000	8.8E4	165
Xe^{138}	14.2 min.	70.2	170,000	2.5	0
Kr^{83m}	1 86 hr.	8.98	16,400	160	0.4
Kr^{85m}	4.4 hr.	3.78	30,000	2100	0.1
Kr^{85}	10.76 years	1.77E-4	2.05	720	0.8
Kr^{87}	76 min.	13.1	52,000	180	0
Kr^{88}	2.79 hr.	5.96	64,400	1700	13.3

Table 5 CRBRP Radioactive Nuclide Input Rates to Reactor Cover Gas & Radioisotope Load in Cryostill

The total energy load on the cryostill turns out to be 502 watts, of which 14.5 is attributable to krypton isotopes (mainly Kr^{88}) and the balance, 487.5 watts to xenon isotopes. This 502 watt load on the CRBRP cryostill would need to be taken into account in the engineering design of RAPS, but was not considered important enough to deserve mention in the PSAR.

With the design contemplated, the picture changes. The plant's thermal power is three times that of CRBRP and it is proposed to use vented fuel. Thus the radionuclide input rates and associated energy loads theoretically increase by a factor of 300. Discharging 210,000 Ci/yr. of Kr^{85} to the atmosphere annually would probably not be well received, and assuming RAPS flow rates increase by a factor of 3 to account for the higher thermal power, the thermal load on the cryostill could be the 150 KW range. Loss of power to such a system or any of the other RAPS components has the potential of producing a large source term that would challenge containment.

First, consider the Kr^{85} problem. 210,000 Ci seems a formidable number, yet its volume turns out to be just 140 liters or about 5 ft^3 at STP. It would generate about 230 watts of thermal energy. While there are stable isotopes of krypton that are fission products and will be present in the cover gas along with Kr^{85}, even if the total krypton volume is an order of magnitude greater, these are numbers that would suggest that long term storage would be manageable over the expected lifetime of the plant.

Second, consider changes that could be made to the RAPS flow rate. On CRBRP, the RAPS flow rate was 5.15 $ft.^3$/min., which included the flows from the overflow vessel, the pumps, and the fuel failure monitoring system. The design basis of the CRBRP RAPS is to "maintain the cover gas at an acceptable level of radioactivity" ("acceptable" not defined), to provide a source of low radioactivity gas for the head seals, and for cover gas pressure control. The head seal leakage is identified in the RAPS section of the PSAR as being 7 SCC/min., which works out to be 0.00025 ft^3/min. Therefore, of these three criteria the most likely to govern RAPS flow rate is the cover gas pressure control. When the plant starts up from shutdown, the RV sodium expansion causes the RV to overflow to the Overflow Vessel thus reducing the total volume of the cover gas between the RV, Overflow Vessel, and PHTS pumps, which winds up in the RAPS storage vessels. Since the storage vessels are maintained at a higher pressure than the cover gas, the only way to get it there is through the RAPS. The calculated rate of expansion for the conceptual design selected for this study turns out to be approximately 3 ft^3 per minute for a power level change of 1% per minute, which is reasonably consistent with the 5.15 $ft.^3$/min. given the higher sodium inventory of CRBRP.

There is no reason why cover gas pressure control needs to be accomplished using the same system that is used for cover gas processing. A separate compressor could tap into the equalization line through a sodium vapor trap, discharging to a surge vessel, a flow control valve, then back to the equalization line. A flow rate of 3.0 $ft.^3$/min. for this system should be adequate. Once having provided this separate system for cover gas pressure control, the RAPS processing flow rate can be reduced by at least a factor of 100. Reducing the RAPS flow rate reduces the energy burden on the processing system components and allows some decay to occur (particularly Xe^{135} and Kr^{88}), although this benefit is reduced because of the 5.27 day half life of Xe^{133}. The figure below shows a revised potential system, where the flow numbers are on $ft.^3$/min.

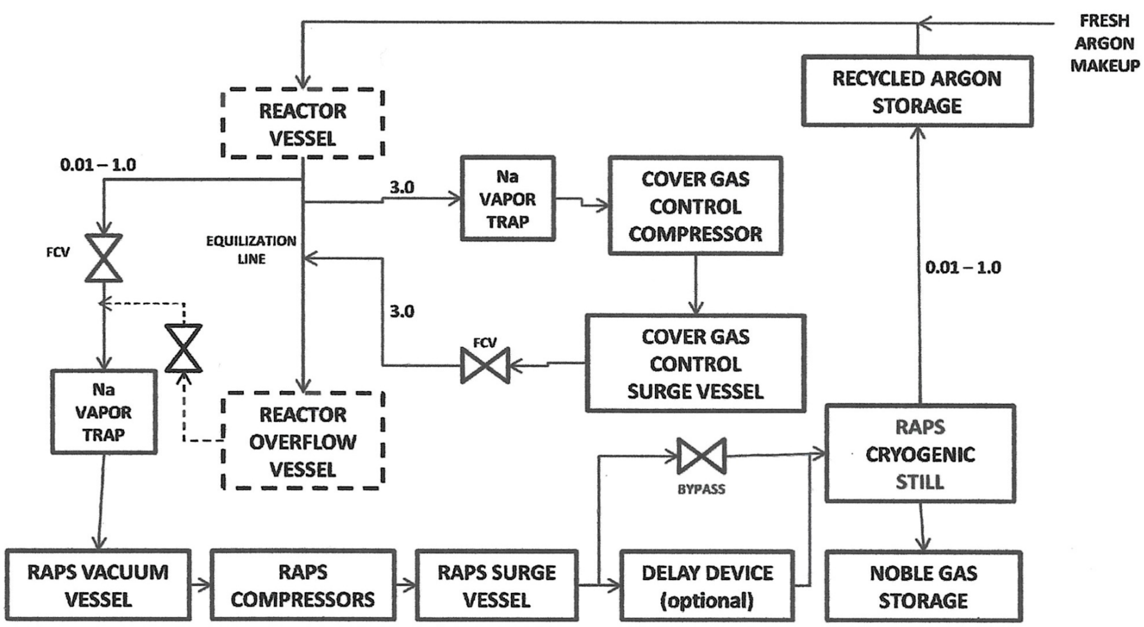

Figure 41 Proposed Cover Gas Control & Processing System

It is necessary to give consideration to the factor of 300 for the energy difference of the cover gas activity between CRBRP and the design concept. In the CRBRP, it was assumed that all the fission gas associated with 1% failed fuel immediately found its way into the cover gas. Since a fuel element failure could occur in the region of highest heat production in the fuel element, such an assumption is semi-plausible and it certainly is conservative. If the plant systems can handle the consequences associated with such conservatism, making the assumption avoids justifying a lower number. Since the assumption is troublesome given the design approach being contemplated, it needs to be reconsidered.

The fission gases are born in the fuel pellets and it is necessary for them to migrate to the plenum above the fueled area in order to make their way to the reactor coolant and then to the cover gas. The amount of time that is required for such migration to occur under the circumstances envisioned in the design approach being advocated is unknown, but it cannot be instantaneous. After a relatively short period of plant operation, the pellets are in intimate contact with the clad, so gas must migrate upwards through the pellet column including passing through the upper axial blanket. If this process requires as much as two weeks on average, the cryostill problem is solved – most of the xenon will have decayed in the fuel.

The technical literature is rich in articles on fission gas migration in oxide fuels. It appears to be widely accepted that there is little migration of fission gas in fresh fuel up to a burnup of about 5%, when the fission gas bubbles begin to agglomerate and migrate along grain boundaries. Other areas of common ground in the technical literature are that migration occurs at a higher rate as temperature increases, and migration can be accelerated by transients. At higher burnup, the gas migrates along grain boundaries, but these boundaries won't be lined up when the gas moves up from one pellet to the one above it, which further retards the process. The effect of the upper axial blanket on retarding flow is something that does not appear to have been addressed, but it should present another rather formidable barrier. Many of the upper axial blanket pellets will have swollen making intimate contact with the clad, but will not have agglomerated fission gas bubbles expanding grain boundaries in the same fashion as the fueled regions. A fair amount of pressure buildup will be needed to pass this barrier. Once the fission gases finally pass the upper axial blanket and arrive in the coolant, some will dissolve or be entrained in the coolant stream and remain there until they finally make it to the cover gas. The time required for this to happen is something that should be amenable to experimental determination.

Part of the answer to all this complexity and uncertainty is to make the RAPS processing rate variable in the range from 0.01 to 1 ft.^{3}/min. The flow rate would be determined by the activity of the cover gas – when it exceeds 1 watt/ft.^{3}, its flow would be reduced in accordance with a predetermined formula down to as low as 0.01 ft.^{3}/min. There is no need to consider rates as high as to those planned for CRBRP, since at 1 ft.^{3}/min., the cover gas will be turned over once daily and the DF of the processing system is so high that 5-7 turnovers will suffice for most purposes. Refueling occurs so infrequently, it can await satisfactory cover gas activity levels. There may be some cases where the processing system would be shut down completely to await xenon decay, i.e. following a transient. In any case, flexibility is necessary for a satisfactory RAPS design.

As was stated earlier the Overflow Vessel on CRBRP has a capacity of 35,000 gallons or about 4700 ft^3. The PSAR states in a table that the net sodium overflow volume from 400°F. to THDV conditions is 1439 ft^3. The total primary system volume at THDV was 23802 ft^3. Presumably, there is some allowance in the Overflow Vessel for overpower transients – perhaps 300 ft.^{3}.

The CRBRP RV cover gas volume was 410 ft.^{3}, which compares with the design concept cover gas volume of about 750 ft.^{3}. The Overflow Vessel needs to make an allowance for overpower transients occurring at full power, so another 250 ft.^{3} should be adequate for that purpose. Between the reactor and the overflow vessels, the total amount of cover gas is 1000 ft^3. At a rate of 0.01 ft.^{3}/min., the entire cover gas volume would be processed in about 70 days. The dominant reason for RAPS processing is to provide a means for removal of Kr^{85}, which is insensitive to processing flow rate. Slowing the processing allows more xenon to decay in the reactor vessel where it contributes a little to plant power level. In fact, it could be argued that the approach of having a separate system for cover gas pressure control could have been taken on CRBRP so as to reduce the burden on the processing system.

It is necessary to consider the effect of the higher cover gas energy level on the RAPS processing components. Assuming that the reactor cover gas has a volume of about 750 ft.^{3} at full power, the cover gas could have an energy level as high as 200 watts/ft^3 if CRBRP PSAR assumptions are used. The cover gas control surge vessel would not experience such a high level since it would mainly be filled during plant startup when the plant would be operating at low power after it had been shutdown. Assuming power level at startup to be 5%, cover gas energy level would be at most 10 watts/ft^3 (more likely, close to zero). If the surge vessel were sized for 1500 SCF, it would have a heat load of at most 15 KW, dropping to under 10 KW once the Xe^{135} has decayed. This should be a fairly manageable engineering problem.

Another occasion when the cover gas control surge tank would be filled would be during the up transient associated with load-following. From 15% to 100% power there would be 3 SCF per minute flowing into the surge tank for 85 minutes for a total of 255 SCF. So long as the cover gas energy is less than 50 watts/ft.^{3}, the surge tank would be okay, but would not be able to handle a similar load-following transient the following day. A decision on load following capability would need to be made based on the activity in the cover gas.

In order to design the RAPS processing system components, it is necessary to decide the level of cover gas energy they will be obliged to process – the idea being that on the rare occasions of higher energy levels in the cover gas, the RAPS processing system will be shut down until the levels decline. The key parameter is the time required for the fission gases to escape from the fuel pin, enter the coolant, and then enter the cover gas space. This is something that can be supported by experimental data when the time comes to embark on preliminary design but for the purposes the current work, one week will be assumed to be reasonably conservative. After a week, the fission gas isotopes have decayed such that their contribution to the cover gas energy is down by 76%. This corresponds to about 50 watts/ft^3 in the cover gas.

If the components in the RAPS processing system are obliged to handle cover gas at energy levels of 50 watts/ft^3, they must be resized from their CRBRP version. There appears to have been about 1400 ft.^{3} of cover gas in the CRBRP design (400 in the RV, 400 on the Overflow Vessel, and 200 in each of the PHTS pumps). From the PSAR data, the RAPS surge vessel contained about 3600 SCF, the vacuum vessel between 125 and 206 SCF (pressure varying between -7 and -2 psig), the Recycle Argon Storage Vessel 2200 SCF, and the cryostill 58 SCF. There is no need for such a large surge tank when the compressor speed is variable, especially since RAPS system surge can be accommodated in the Recycle Argon Storage Vessel, which stores gas after it has been processed. The only purpose for the Surge Vessel would be to absorb any pressure pulses from the compressor. If the vacuum vessel and surge tank were reduced in size to about 40 SCF, the attendant heat loads would be a manageable 2 KW. Since isotopes are being concentrated in the cryostill, its size is not as important. An approach might be to put some sort of delay device in the circuit ahead of the cryostill to allow isotope decay. Because of the 5.27 day half life of Xe^{133}, this approach would only marginally improve things unless the delay device was sized to be about 100 ft^3. At 0.01 $ft.^3$/min. processing rate, a 100 ft^3 delay device would reduce the Xe^{133} concentration another 63%. Such a device might consist of a 2 ½ in. diameter coiled tube with a total tube length of 3000 ft. It is shown as optional with a bypass, which would be mandatory at higher flow rates, in Figure 37.

It remains necessary to consider the overflow vessel. On CRBRP, the RAPS took suction on the overflow vessel, so the cover gas in the overflow vessel was the same as the reactor. At full power, the overflow vessel would be nearly full of sodium, but there will need to be some cover gas left to accommodate overpower transients – say 250 $ft.^3$. There is no reason to burden the overflow tank with high energy cover gas if it can be avoided. Since there is little if any flow in the equalization line during steady state conditions, a simple solution would be to take the RAPS processing line off the equalization line as close to the reactor as possible. Another fix would be to design the overflow connection to the reactor so as to avoid entrained cover gas in the reactor overflow. This could be accomplished by simply providing a small downward pointing pipe connecting to the overflow penetration interior to the reactor vessel. Heat generated in the overflow tank would eventually be returned to the reactor via the overflow pumps, but it would be desirable not to be obliged to rely upon this mechanism. In the event that the Overflow Vessel cover gas becomes contaminated to levels approaching unacceptability, Figure 37 shows the capability to switch RAPS processing from the reactor to the overflow tank.

The discussion in this subsection has assumed the cover gas is argon while earlier helium was given as the preferred cover gas for the design concept. The design of the cryostill would certainly be different with helium cover gas and it is unlikely that one would use liquid helium as the carrier medium. Both krypton and xenon are solids at liquid argon temperatures, so one could use liquid argon or nitrogen to condense (and freeze) the krypton and xenon, with a barrier separating the argon/nitrogen from the helium cover gas and the chunks of frozen xenon and krypton. The design of a cryostill for this application shouldn't pose much of a problem. The remainder of the components would work for either cover gas.

CAPS

A flow diagram of the Cell Area Processing System or CAPS as it was designed for CRBRP is shown below. The figure is a reproduction of a similar figure appearing in the PSAR except in the PSAR, the HEPA filter was omitted. The solid boxes are CAPS components.

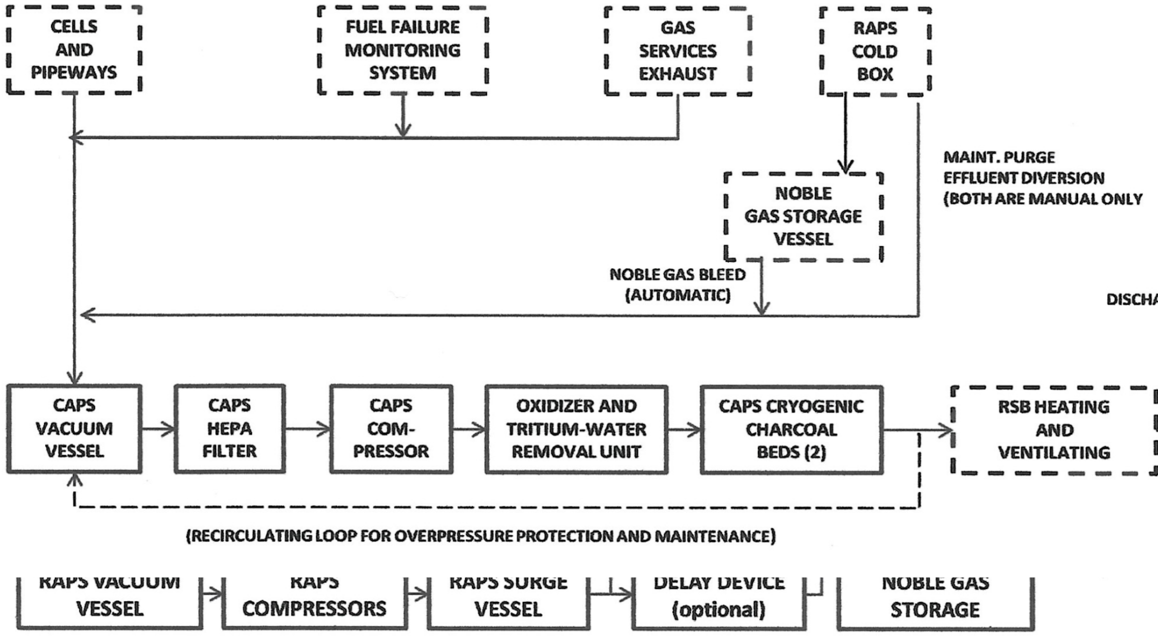

Figure 42 CRBRP Cell Area Processing System (CAPS)

Regarding the CAPS inputs, the Gas Services Exhausts are intermittent and are located at various stations around the plant – CAPS would receive exhausted nitrogen, argon, or air from vessel cover gases, cooling gases, cleaning, bagging, and fuel handling operations, and other services. On CRBRP, it was the intent to purge the cryostill annually, with the argon from the still directed to CAPS and the noble gases to gas storage, where it would be bled to CAPS over a fixed period of time. The fuel failure monitoring system continuously discharged about 1 ft^3/min to CAPS, as shown on Figure 38. The only input to CAPS that was continuous was the cells and pipe ways where a purge stream of about 38 SCF/min was maintained. The idea was that cover gas leaking through various seals and tritium leaking through pipe walls would be captured in CAPS and treated prior to release to the atmosphere.

A recirculation loop, shown by a broken line in Figure 38, returns the CAPS output to the vacuum vessel if radioactivity above an acceptable level is detected by the radiation monitoring system. Also, if the effluent radioactivity is high, the CAPS compressors will be shut down. The tritium-water and alcohol removal process uses an oxidizer and a freeze-out, dryer; it oxidizes tritium, collects tritiated water and alcohol and passes them to the Radioactive Liquid Waste System, where they are prepared for off-site disposal. The decontamination of radwaste gas is performed in two cryogenically-cooled charcoal delay beds. In the beds, the short-lived gaseous radioactive species are adsorbed and then decay; they are thus removed from the process gas stream. RAPS and CAPS have different process methods, i.e., the distillation-process removal of noble gases in RAPS rather than the delay beds and the oxidation process removal of tritium in CAPS. In each subsystem, however, the input is collected in a vacuum vessel, from which it is transferred to a surge vessel. It is then treated in the respective cold box.

The oxidizer for tritium is obvious overkill. In an LMFBR, there is no tritium production from neutron absorption by deuterons and the only source of tritium is from fission, where the expected production rate is less than 1 per 10^4 fissions. The PSAR states that at least 99.8% of the tritium will combine with sodium and be removed by the cold traps. The main source of tritium is expected to be that which diffuses trough the PHTS and IHTS pipe walls. The total annual release rare for tritium is given as 0.069 Ci/yr and the DF of the tritium removal unit is given as 100. Thus, if there had been no tritium removal unit, the tritium release would be 6.9 Ci/yr, (given the conservative assumptions used) which compares with the releases from other sources of 1700 Ci/yr, 1200 of which is Kr^{85}. The elimination of the tritium removal unit is CRM 49.

Although it is not explicitly stated in the CRBRP PSAR, CAPS is a backup for RAPS. The CAPS cryogenic charcoal beds therefore play an important role when they are viewed from this perspective. The charcoal beds delay the xenon isotopes until they decay before release. This is probably the most important function performed by CAPS. The CAPS for the design concept would be much the same as its CRBRP counterpart with the fuel failure monitoring system and the tritium removal unit deleted and the HEPA filters placed at the end, rather than the beginning, of the process.

IGRP System

Although technically, RAPS and CAPS are part of the IGRP system, for the purposes of the discussion, the IGRP system minus RAPS and CAPS will be treated separately. IGRP minus RAPS and CAPS is basically a system for the distribution of argon and nitrogen throughout the plant. This turns out to be important for the main purpose of proposing a design approach that meets the basic requirements without unnecessary provisions that add cost and complexity to the plant without commensurate benefit. One of the first things that greets the reader of the IGRP sections of the CRBRP PSAR is that there are 12 P&ID drawings for the argon distribution system and 12 P&ID drawings for the nitrogen distribution system. This system turns out to sprawled out all over the plant with miles of piping and hundreds of valves. It is ripe with opportunity for cost saving.

Of the 12 P&IDs for the argon distribution system, 5 are for the RSB, 5 are for the RCB and 2 are for the SGB. Starting with the SGB, close examination of the P&IDs reveals that there is much that is unnecessary. It appears that the designer's approach was to provide permanent systems for applications which may only occur seldom in the life of the plant. For example, there is argon piping to four locations for intermediate sodium cold trap line venting. The piping leads to a shut off valve and a spool piece. The idea being that when one needs this feature, the spool piece is installed. The only time this would ever be used is when the intermediate system cold trap is being replaced, which occurs perhaps once every 20 years. In addition to the pump cover gas, there is a pump seal purge to the lower seal and its associated oil collection tank and oil trap; there is a pressure equalization line between the pump and the IHTS expansion tank; and there is a supply line to the pump seal oil supply tank. Here are three more good reasons for using an EM pumps for the IHTS. The SGB has its own bank of liquid nitrogen storage vessels, associated vaporizers, filters and numerous associated valves. It also has an auxiliary argon supply. Once the pumps and once in a blue moon connections are deleted, the only things needing argon in the SGB are the Sodium Dump Tank, the IHTS Expansion Tank, and the spaces between the rupture discs in the SWRPS system. There is no reason why any needed argon for these services can't be supplied from the RSB.

In the RCB there are 17 spool piece connections, all of which could be eliminated. When there occurs a rare need for argon to be supplied to one of these points, it could be accomplished with hose. There is no need for permanent installations and spool pieces that are likely to disappear long before they are needed. As was the case with the SGB, replacing the PHTS centrifugal pumps with EM pumps eliminates nine connections. The pressure equalization lines connecting the OV to the PHTS pumps are in this system. A careful review of the five P&IDs for the RCB would undoubtedly lead to many other options for cost reduction. Systems such as this receive relatively little oversight from the customer organization and as a result, they grow in such a fashion so as to satisfy every possible perceived need. If the system designer receives little in the way of feedback there is a tendency to add to the system, usually unnecessarily. This system is a good example of the devil in the details and a good opportunity for cost reduction. It is captured here as CRM 50.

Dowtherm J

Dowtherm J is an organic heat transfer fluid that does not react either with water or the alkali metals that was used in two places on the CRBRP plant, 1) as the cooling fluid for the NaK that was used for the primary system cold trap. It in turn was cooled by chilled water. 2) the space coolers in the fuel handling cell were cooled by Dowtherm J, which were in turn cooled by chilled water.

Dowtherm J would be needed as the cooling fluid for the space coolers of the refueling cell in the concept being advanced. It should be noted that Dowtherm J was planned for use on the CRBRP plant nearly 50 years ago. It was an improvement over the cooling fluid used by FFTF for similar applications called Mobiltherm. In the intervening years, it is likely a more advanced fluid exists that could replace the Dowtherm J, although Dowtherm J continues to be marketed by Dow.

Once a suitable cooling fluid has been identified, a careful assessment of the possibility of using that fluid in place of NaK, everywhere NaK is used should be performed. NaK is more reactive than sodium and since it is liquid at room temperatures, a leak of NaK is considerably more difficult to contain and control than a leak of sodium, which usually freezes at the leak site. Eliminating NaK everywhere it exists in the plant would be a big positive for the LMFBR. This item is captured as CRM 51.

An important qualification to this cost reduction measure is the experience of the SRE with tetralin, which was used to cool the seals on the primary sodium pumps. The tetralin made its way into the primary system and caused plugging of several fuel assemblies leading to their overheating and meltdown. Armed with that experience, the tetralin was replaced with NaK everywhere it existed on the primary system and was not used on Hallam.[71] A NaK leak into the primary system is not without consequences, but NaK won't interfere with core cooling.

Nitrogen

In the CRBRP design, nitrogen supplied 1) a low pressure header feeding all of the normally inerted cells and pipe-ways within containment and the RSB, 2) a high pressure line for actuation of valves in cells that are normally inerted, 3) a line to the CRDM assembly recirculation cooling system, 4) a line to provide sparging gas to the sodium component cleaning operation in the RSB, and 5) a line to purge the RAPS cold box, 6) the CAPS cold box is inerted with nitrogen at a continuous low flow rate during operation in order to provide a dry, non-frosting, non-condensing atmosphere, 7) for service maintenance operations available at service stations located within the RSB, 8) a controlled pressure N_2 supply is provided separately to the autoclave located in the RSB, 9) as a cover gas for the Dowtherm tanks used in the chilled water system, and 10) as a cover gas for the SWRPS in the SGB.

There are obvious places where economies can be made in this system. As was the case with argon, the both the RSB and the SGB had their own supply of nitrogen in the CRBRP design, which was unnecessary. For nitrogen, there is a greater need for a large supply to be available in short notice to support the SWRPS than anywhere else in the plant. If SWRPS is activated, nitrogen must fill the volume previously occupied by sodium in the effected steam generator and to purge the hydrogen generated by the sodium water reaction. Therefore, it might make sense to eliminate the separate RSB supply and have the entire plant supply originating in the SGB. Also as was the case with argon, there are numerous connections requiring spool pieces, most of which can be done away with.

Liquid Radwaste

There are two sources of liquid radwaste in the CRBRP design; the sodium removal and decontamination system and plant drains. Plant drains include the personnel shower, and all the cell drains throughout the plant. Each of these two sources has its own subsystem consisting of a filter, collection tanks, another filter, an evaporator, a demineralizer storage or monitoring tanks, and either discharge or reuse paths. Six complicated P&IDs are required to describe this system. All this complexity is provided so as to reduce the dose to the most exposed member of the public who obtains all his or her drinking water from the nearest point adjacent to the plant offsite to 0.13 mrem/year. This is a sad tale of woe common to every nuclear plant in the country.

There is nothing that can be done about floor drains or showers, but the system intended to prepare sodium wetted spent fuel for shipment, which is by far the largest radioactive source for

71 R.J.Beeley, J.E.Malmeister, *Operating Experience on the SRE and its Application to the Hallam Nuclear Power Facility*, Atomics International, 1961

the liquid radwaste system, does not need to be available until 15 years after startup of the plant, at the earliest. If it were possible to ship sodium wetted spent fuel to the reprocessing facility, a large cost driver could be eliminated. Since the systems for treating floor drains and showers is separate from the system for cleaning sodium wetted fuel, the latter system can be added to the plant 15 years downstream. Section 14 describes the fuel cycle facilities necessary to implement a power supply system that includes LMFBRs and describes the need for committed reprocessing facilities for LMFBRs. If the first such facility were co-located with the LMFBR, the need for a radwaste system for treating effluent from a sodium cleaning facility would be obviated. Deferral or elimination of this system is captured as CRM 52. Space should be provided in the RSB for its eventual installation.

13 Summary and conclusions

The essence of this monograph is to make a convincing argument that there are abundant opportunities for reducing the capital cost of the LMFBR by designing for and realizing the inherent attributes of the metal coolant and the breeding principle. It should be possible to design a plant that is capable of continuous operation without refueling for periods of about ten years and operating base loaded or load following at the option of the owner. If the reactor is capable of operating continuously for ten year periods between refuelings, a simpler (but slower) refueling system can be adopted, which creates numerous opportunities for capital cost reduction.

As a result of the foregoing sections, the key reference design parameters turn out to be:

Electrical output (nominal)	1200 MWe
Thermal output (nominal)	3000 MW
Thermodynamic efficiency	~41%
Reactor outlet temperature	1017°F
Reactor inlet temperature	693°F
Number of PHTS loops	2
Number of PHTS pumps	4
PHTS pump concept	electromagnetic
PHTS pump location	cold leg
Primary pump head	30-40 psig
PHTS total flow rate	225,000 GPM
PHTS pump flow rate	56,250 GPM
Number of IHTS loops	2
Number of IHTS pumps	4
IHTS pump location	cold leg
IHTS pump concept	EM
IHTS flow rate	225,000 GPM
IHTS head	36 psig
IHTS hot leg temperature	977°F

IHTS cold leg temperature	653°F
Number of IHXs	2
Number of steam generators	4
Steam temperature	914°F
Steam pressure	2400 psig
Feedwater temperature	490°F
Cycle length	10 years
Load following capability	15-100% of full power
Refueling down time	six months
Capacity factor between refuelings	95%
Maximum peak fuel burnup	300,000 MWD/MTU
Average fuel burnup	150,000 MWD/MTU
Refueling concept	Single rotating plug
Decay heat removal concept	PRACS + OHRS with IRACS or DRACS option

The heat output, thermal, and steam pressure conditions are the same as Superphénix since the same steam generator configuration is being proposed. Note that the above steam conditions are an improvement over CRBRP primarily because a once through steam generating system is selected rather than the recirculating system chosen for the CRBRP plant design. Within the context of the proposed design approach identified here, improvements can probably be made by either increasing steam pressure moderately and/or increasing PHTS and IHTS temperatures by about 50°F while maintaining 2400 psig steam conditions.

It is imperative that the LMFBR take advantage of its natural attributes and be designed to be economic. Approaches to design and licensing must be rethought with the goal of developing a concept that can compete on the marketplace with all alternatives, including renewables, natural gas, and especially light water reactors.

- The plant should be designed to be capable of load following from full power down to 15% power so as to be compatible with a power grid that is supplied with a substantial fraction of renewables. Base-loaded operation could be economically preferable, but the extent to which capital cost reduction is successful correlates with the facilitation and desirability of load following operation. I.e., a less expensive plant is a more attractive candidate for operation in a load following mode.
- The loop type design is more flexible and is most likely to be more economic, easier to construct, and more reliable that the pool type design. (See Appendix 5)
- PHTS expansion loops should mostly be in the vertical direction to minimize containment volume.
- There should be two primary system loops with one IHX in each loop. The IXHs should be located as closely to the reactor vessel as can be reasonably achieved so as to minimize containment volume and the demands on the expansion loops.
- The primary pumps should be electromagnetic (EM) and located in the cold leg. The EM pumps should be capable of providing flow over all ranges continuously from 15% to 100%. The IHTS pumps may be centrifugal if that is the more economic option but EM pumps would be preferred so as to match the PHTS pump coastdown characteristics. The IHTS pumps should also be capable of operation down to 15% flow.

- There is no need for and there should be no pony motors, either on the PHTS or the IHTS.
- The reactor cover gas should be helium so as to enable better separation and removal of fission gases, but argon would be acceptable.
- There should be four helical-coil once through steam generators and four IHTS pumps located in the cold leg. The steam generators should be isolable on the water side. The capability of sodium side isolation is optional, but probably not necessary.
- The core should be designed for a ten year refueling interval with the whole core, internal blankets and the most if not all of the radial blankets replaced at each refueling. The refueling shutdown may require up to six months.
- The fuel should be vented to the primary coolant. The fuel should be capable of a 15 a/o average burnup and peak burnup of 30 a/o.
- Orificing of blanket assemblies should be either controllable from outside the reactor or self actuating. It would be highly desirable to be able to control the flow to both fuel and blanket assemblies as well to reduce the effects of thermal striping.
- The pressure drop across the reactor should be less than 20 psid.
- Hydraulic hold down of core assemblies may be eliminated as unnecessary with the reduced core pressure drop.
- The refueling system should be single rotating plug. The EVST should be sized to accommodate a full core load. The fuel handling system should be designed to permit loading the spent fuel cask five years after the fuel has been removed from the reactor through a wash station.
- The reactor head should have one centrally located rotating plug. The reactor vessel should have no in-vessel transfer position. Core component pots should be eliminated. The vessel should be about 30-35 feet in height and 28-30 feet in diameter.
- The containment should be confined to the refueling cell and the primary HTS vaults. Any provisions for HCDAs mandated by the regulator must be beyond design basis. Safety emphasis should be on plant simplification, highly reliable natural circulation decay heat removal, and enhancement of reactor shutdown system reliability.
- Decay heat removal should be through a PRACS with an OHRS for backup and normal shutdown operation. An IRACS or DRACS would be a reasonable alternative to a PRACS or supplement for PRACS if necessary.

Although the simplicity of the "design approach" is promising, it is expected that it will not be easy to obtain a reliably accurate cost estimate. Much of the cost of the plant will be in the major components – particularly the reactor vessel, and to a lesser extent the IHXs, EM pumps, and steam generators. Many of the specialty items e.g. cold traps will need to be procured “build to print”. Identifying potential vendors for LMFBR components who can make reliable cost estimates in the U.S. will not be a trivial undertaking. The primary reason the CRBRP reactor vessel was procured at such an early stage was the perception that there was only one credible vendor for such a task remaining in the U.S. at that time, and that vendor would shut down its capability for vessel manufacture if the CRBRP vessel was not procured early. The estimate for the structures, piping, and installation should be more straightforward given that the preliminary design is available.

Another area that is likely to be problematic is the state of current knowledge in sodium technology. There was a good deal of work done in this area at the Liquid Metal Engineering Center in the 1960s and 1970s, which continues to be somewhat available. Technical development activities with liquid metals were also performed at the Hanford Engineering Development Laboratory, Argonne National Laboratory, and the Knolls Atomic Power Laboratory. A five volume set on Sodium and NaK Engineering Technology was prepared during that period and the associated technical literature continues to be available, although not necessarily on the internet. It will likely turn out to be necessary to spend much time in technical libraries.

Once a believable cost estimate exists for this configuration, and the estimate shows total costs to be considerably less than an equivalently sized LWR as expected, consideration could be given to options which potentially improve the plant's performance and operability. Although it may be tempting to do so, the cost of licensing should be known first. So long as the investment has been kept under control, there is no reason for seeking a limited work authorization or beginning construction immediately after a construction permit has been secured. The preliminary design can be modified if desired so long as the PSAR is kept up to date. For example, it may also be desirable to expand the containment volume to better accommodate growth items unforeseen during early design, alternative primary sodium treatment systems, or even the adoption of four IHXs rather than two. Any such expansions should be made only after assessing their cost impact and ensuring the resulting capital cost remains significantly below LWRs.

14 A Path Forward

The forgoing has been directed at identifying some of the engineering challenges that need to be addressed to arrive at a commercially attractive LMFBR design. However, the path forward must deal with more than plant engineering. An initial plant which serves as a demonstration that embodies the basic principles in this work must be built to establish widespread utility industry confidence in the concept. This section addresses some of the institutional issues and facility requirements that have been alluded to previously but not systematically considered.

In the United States, the government seems to perform construction projects well when it is faced with an emergency. The Manhattan Project is an excellent example. If the country were to defer deployment of the breeder reactor until natural gas supplies were exhausted, it would be faced with an emergency, and government performance might be equal to the task. Unfortunately, if this were to happen, it could be too late and would be much more difficult. There is not enough plutonium available to fuel a large fleet of breeders from a dead start. The plutonium needed must either come from the reprocessing of LWR spent fuel or be bred by the breeders themselves, and the breeders are more effective at the task of producing plutonium. With no reprocessing the only separated plutonium currently available in the U.S. is unused weapons plutonium which is of insufficient quantity to fuel a large fleet of LMFBRs. Reprocessing, breeder reactor development and deployment must begin well before natural gas supplies are depleted.

As was stated in Section 1, the breeder reactor is dependent on reprocessing. It would be possible to fuel a small number of LMFBRs with plutonium that has been recovered from weapons but there would be no point in doing so. The LMFBR spent fuel must be reprocessed to recover the bred plutonium if the plants are to be sustainable and increase in number. Either an LMFBR demonstration plant together with a reprocessing plant is built at the same time or a reprocessing plant is constructed first. Clearly, it would be more tractable to sequence the construction of these two building blocks if possible.

If one contemplates constructing a reprocessing plant first, there is the question of who would be the customer. A reprocessing plant produces three streams, uranium, plutonium, and fission products/actinides. The market for the plutonium stream would be the breeder reactors which would come later so there would no near term market. The uranium stream, which has by far the greatest mass rate of flow, would have enrichment somewhat higher than natural uranium and therefore should have value greater than natural uranium. It could be re-enriched or blended with higher enrichment uranium and reused by LWRs. This was the intent when the Barnwell plant was being constructed. At a market price of \$35/lb. for U_3O_8 the uranium stream of a 1000 MTU/yr. reprocessing plant would fetch perhaps \$50 Million per year which, by itself, is not a compelling return on the capital investment of the plant. Complicating the economics further, the uranium stream contains about 0.02% each of U_{234} and U_{236}, and about 0.0001% U_{233} which together cause the uranium stream to be about 20 times more radioactive than non-reprocessed uranium. This increased radioactivity would need to be considered in the designs of both the enrichment plant and the fuel fabrication plant that uses reprocessed uranium.

It is somewhat of an irony that the entity that stands to gain the most from the availability of commercial reprocessing is the federal government – the same entity that was responsible for stopping Barnwell plant construction in 1976-7. After Barnwell was halted, the government assumed responsibility for the disposal of nuclear waste generated in the United States with the Nuclear Waste Policy Act. Its attempts to fulfill that responsibility so far have resulted in the expenditure of about \$20 Billion and have yielded only a construction permit application and some exploratory shafts inside Yucca Mountain in Nevada. In view of the enormous size and complexity of the facilities contemplated for installation at Yucca combined with the remoteness of the site, it is no stretch of imagination to expect at least another \$50 Billion would be required to complete the repository project – that is assuming it were even possible to do so, given the obvious political issues involved. Assuming that the repository is designed for disposal of 100,000 MTHM (metric tonnes heavy metal), if the cost estimate suggested above is approximately correct, the cost of waste disposal is at least \$500 Million per 1000 MTHM, which was the planned annual throughput of the Barnwell plant.

There are two major cost drivers for a repository of the type contemplated for construction at Yucca Mountain – the substantial volume of the waste and the actinides contained in the waste. A reprocessing plant would reduce the waste volume by a factor of over 100 and the waste stream would be amenable for removal of the actinides. Once the actinides have been removed, the remaining fission product waste stream could be vitrified and stored in a convenient surface facility. After about 300 years, the total radioactivity of the remaining vitrified fission products is comparable to the uranium from which it was originally derived.

The Barnwell plant was designed to reprocess about 1000 MTHM per year. The original cost estimate of the Barnwell plant was in the $300-400 Million range. Construction of the same plant today would likely be in the $1-2 Billion range. If the federal government were to pay ~$250M per year for volume reduction of the waste for which they are responsible, a private entity may be incentivized to proceed with construction of such a plant if given reasonable assurances that it would be licensable. Development of a reliable process for removal of the actinides from the waste stream is a task that could be assigned to the national labs or it could be contracted. It is unlikely that the cost of implementing such a process on the waste stream would be anywhere close to $250M/yr, which would result in a savings to the government as compared to a repository, which has proven to be politically impossible to build. A Barnwell-type plant would probably not be able to reprocess LMFBR fuel, but there would be no spent LMFBR fuel to reprocess until 15 or more years after the first plant begins operating. About three years of Barnwell-type plant operation reprocessing LWR spent fuel would be sufficient to provide the needed plutonium for the initial loading of a single LMFBR.

The CRBRP project provided a model of government industry cooperation from which it is possible to learn lessons and draw conclusions. The government (DOE) was in charge of that project and it was the government's executive involvement which turned out to have been a mistake. The fates of both CRBRP and the repository program are object lessons of the difficulty experienced by DOE with the completion of a major nuclear project. The annual appropriations process combined with changing administrations makes the government particularly ill suited to carry out and complete any project that requires more than four years to execute. The presence of the Defense Nuclear Facilities Safety Board (DNFSB) that has oversight on DOE adds another layer of unquantifiable risk into the DOE equation.

For CRBRP, the utility industry formed the Breeder Reactor Corporation (BRC) which provided $750 million from the contributions of over 400 electric utility companies in exchange for rights to the intellectual property flowing from the project. The BRC in turn created the Project Management Corporation (PMC) which was the operational arm of the project and was responsible for communicating project progress to the BRC. PMC together with the DOE formed a joint project office for the purpose of providing overall management of the project participants. Although the PMC personnel had meaningful roles in the project, all the key positions were occupied by DOE employees, so the utility industry ability to participate in the key decision making process was limited. It was always clear that the DOE had ultimate responsibility. In many cases, particularly early in the project before the joint project office was formed, utility industry leadership was not even consulted on key decisions.

For any future LMFBR project, it will be necessary to once again enlist the participation of the utility industry. This will not be an easy sell – the legacy of the CRBRP project is unpleasant and the key decision makers in the utility industry are likely to have long memories. Nonetheless, there is no alternative. The ultimate users of the technology must be in the driver's seat from the beginning if it is to have any prospect of being accepted by them. It would not be necessary to begin with a major financial solicitation as was done with the BRC, but an "interested party" could be formed drawn from utility industry personnel with modest expenditure. In all likelihood, the "interested party" could wind up as a group within EPRI, but other special purpose institutional arrangements made by the electric utility personnel may be preferable.

Utility industry leadership in an LMFBR project is not unprecedented -- the Fermi I project was executed under utility industry leadership, first by Detroit Edison, then by Atomic Power Development Associates (APDA) acting in conjunction with the Power Reactor Development Company (PRDC), both of which were creatures of a consortium of interested electric utility companies. (The utility consortium were joined by other interested companies such as Allis-Chalmers and Westinghouse. There was also international participation from Japan and other countries.) There was some financial contribution made by the AEC, but most of the financial support came from the participating private utility companies. The AEC acted in an advisory and regulatory role, but the decision making was entirely the responsibility of APDA and PRDC. This institutional arrangement resulted in a project that came much closer to success than CRBRP.

For there to be any prospect of enticing the electric utility industry into a new breeder project, the position of the government must be transparent. So what should be the federal government's role? Foremost, it is essential that the government withdraw its objection to commercial reprocessing and recognize the uncoupling of nuclear power from nuclear weapons proliferation. This needs to be done at a policy level in the President's office. It is long past time to disavow the legacy left behind by Jimmy Carter. The government may also need to furnish some fraction of the plutonium necessary for the initial core load should the reprocessing plant experience difficulties. Beyond that, there is not much that should be expected from the federal government. The politics of a democracy will always be focused on the short term. Taking highly visible steps that are politically risky to address a future need that may not materialize for fifty or more years is something that will not happen within the U.S. government.

The federal government could make a meaningful contribution with limited R&D assistance. However, any such development activity carried out by the National Labs should have involvement and oversight of the private sector participants that are moving forward with actual projects. The labs could provide considerable assistance with core design, materials identification, fuels performance data, and development assistance with items such as the flow control devices that are needed on the blanket assemblies. The extent to which the labs can be made partners and supporters of the project can potentially have enormous impact on a favorable outcome.

The "interested party" identified above would represent the private sector participants and may include some industrial participation such as the potential owner(s) of the first reprocessing plant. It probably should limit participation by engineering firms that might be later contracted to develop detailed design to limit their influence at the conceptual stage. There is adequate engineering talent available in the electric utility industry. An early purpose for the "interested party" would be to develop a conceptual design for the LMFBR that has a firm grounding in analysis and is consistent with the needs of the electric utility industry.

Once commercial reprocessing is in place, one worthy role for the national laboratories would be to develop processes for removal of the actinides from the waste stream. With the actinides removed from the Purex waste stream, the remaining fission products could be vitrified and stored in some above ground facility. Geologic disposal should turn out to be unnecessary given sufficient effectiveness of actinide removal.

After sufficient confidence has been developed in the conceptual design and believable cost estimates are in place, the preliminary design for the LMFBR plant could be developed and submitted to the NRC. The CRBRP project made the mistake of setting a schedule which required major component procurement before there was a license. The project had no alternative but to capitulate to all the NRC's demands. The "interested party" must retain the option to walk away should the licensing process lead to a design that is inconsistent with utility industry objectives. It is unlikely that the NRC would agree to a one-step licensing procedure for the first LMFBR to be licensed since Fermi-1, but a major retrofit at the FSAR stage requiring significant demolition and reconstruction would seem unlikely. Once a preliminary design certification (or PSER if the site has been selected) is in hand, the project may proceed at its own selected pace.

When a reprocessing plant is operational and a plutonium stream becomes available, the LMFBR becomes closer to realization. It would be necessary for some provision to be made for fabrication of the fuel to be used in the plant since existing LWR fuel fabrication facilities are incapable of fabricating plutonium bearing fuel. At the time of the CRBRP project, there was a plan to fabricate fuel for the CRBRP in an unused facility near the FFTF. The remains of that effort could be recovered or alternatively the fabrication facility could be capitalized into and collocated with the demonstration plant. Since the power plant requires refueling only once every ten years, the throughput of the needed fabrication plant is modest. Nonetheless, the fabrication plant is likely to be expensive (~$300 million) so its financing needs to be thought through. Since the fuel in the internal and radial blankets is depleted uranium, their fabrication could be accomplished at an existing commercial facility.

The demonstration plant would most probably be undertaken by some sort of PMC-like or APDA-like entity created and funded by the nation's electric utility industry. The entity would be an outgrowth of the "interested party" above and would develop the detailed design, select the site, procure components, and initiate plant construction. The host utility would derive the benefit of the power produced and would be expected to furnish the funding equal to a comparably sized LWR minus some percentage, perhaps 30%, to account for the greater risk associated with the deployment of a new concept. Any additional funding would need to be made up by a consortium of electric utility companies. Needless to say, if the total cost of the plant comes anywhere close to the cost of an equivalently sized LWR, the project is unlikely to go forward; so the contributions from the consortium should be small.

After the power plant has operated ten years and the spent fuel has cooled sufficiently to permit reprocessing, a reprocessing plant committed to LMFBR spent fuel would become necessary at some point. LWR reprocessing plants would probably be able to keep up with the plutonium requirements of new LMFBRs for a while, but eventually the spent LMFBR fuel must be reprocessed. Once there are several LMFBRs in operation, a market for plutonium will develop which would justify the construction of such a facility. The figure below illustrates the process described above with the essential nearer term components shown in heavy lines.

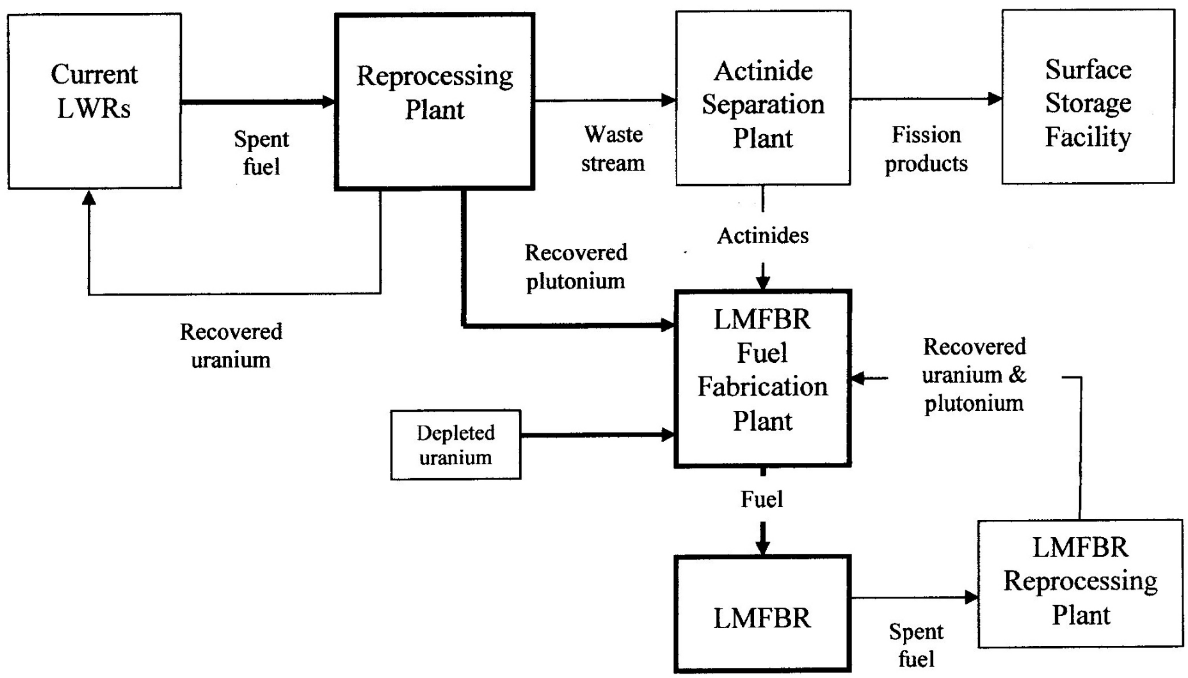

Figure 43 LMFBR fuel cycle

There is one final point to be made before closing. Once a market for plutonium develops, the cost of plutonium will inevitably rise and affect the core design and likely the refueling design. The design concept being advanced here is intended to get the ball rolling and get some early plants built so experience can be acquired with the concept well before the "energy crunch" develops. It may be 30, 50, or hopefully 100 years before the United States is faced with the "energy crunch". However, if one were to begin the "path forward" proposed in this section in the relatively near term, in view of all that needs to be done, it could easily be thirty or forty years later before the first plant shuts down for its initial refueling. A second plant would likely not begin until after some operating experience had accumulated with the first plant. Bringing a new technology such as is envisioned here to fruition takes a very long time and when one considers the time required to build plutonium inventory sufficient to support a large fleet of LMFBRs, it would be a mistake to delay starting the process. It can fairly easily be shown that even with a 100 year time horizon, the sooner the first steps are taken in this plan, the better.

Appendix 1 Fast spectrum reactor neutronic considerations

Fast spectrum reactors can be used with either the U^{238}/Pu^{239} cycle or the Th^{232}/U^{233} cycle.[72] The breeding ratio with the Th^{232}/U^{233} cycle is somewhat lower than with the U^{238}/Pu^{239} cycle. Nonetheless, as was mentioned earlier, there are some situations where the Th^{232}/U^{233} cycle would be preferred such as in India, where there is relatively little naturally occurring uranium but abundant thorium. As a result, the Indians plan to initiate their LMFBR program using enriched uranium as the fissile material, but switch once they have accumulated sufficient U^{233} to support the Th^{232}/U^{233} cycle. Their initial reactors will be fueled with fairly highly enriched uranium mixed with thorium to breed U^{233}. The Th^{232}/U^{233} cycle is also considered to have a lower non-proliferation potential since the U^{233} is highly radioactive and must be handled remotely. Although there has not been much analytic work on the subject, the Th^{232}/U^{233} cycle may also have lower sodium void reactivity and therefore be attractive from that perspective.

There is a price that must be paid if fast neutrons are to be used in a fission reactor: the concentration of the fissile material must be much greater. The cross section for fission in a fast reactor is much lower than in a thermal reactor. While the cross section for fission of the U^{235} in a LWR is about 550 barns, the fissile component in a LMFBR will typically have a fission cross section in the 1.5-2.0 barn range, depending on the spectrum. As a result, the neutron flux tends to be much higher and it is necessary to raise the enrichment of fissile isotopes to a higher level than is required in LWRs. Whereas LWRs are typically fueled with uranium enriched to 2½-4% in U^{235}, a large fast breeder may have Pu^{239} in the 15-30% range. Even with the higher concentration of fissile material in the core, the neutron flux will be at 20-100 times greater in a LMFBR than in a LWR. As a consequence, radiation damage of structural materials becomes a greater concern and must be dealt with in the design.

Because of the excellent heat transfer properties of sodium, the core tends to be much smaller and is somewhat pancake shaped. A large breeder reactor core may be just 3 ft. in height but 15 ft. or more in diameter.

Since core neutron energy is maintained high, there is no xenon or samarium poisoning of the reactor as is the case with thermal spectrum reactors. The accumulation of fission products in the core does insert negative reactivity but most of the neutron spectrum is well above the resonance region avoiding resonance captures of the type typified by xenon and samarium. While the average cross section of fission products in a thermal reactor is about 75 barns, that number is orders of magnitude lower in a fast reactor.

Burnable poisons cannot be used to extend core life as is sometimes done with LWRs. It is also not practical to poison the coolant with a material that can be readily removed as is routinely done on PWRs. Because of the spectrum, there is no convenient burnable poison material. Moreover, the use of a poison for this purpose would be a sink for neutrons and would conflict with the objective of breeding.

72As will be shown in the section on actinide burning, other cycles involving artificial isotopes can also be used.

In addition to there being essentially no resonance region, there are no materials that exhibit high neutron capture cross sections such as hafnium that would be suitable for use as a control material. The best candidate for control material is boron, which exists in nature in two stable isotopes, B^{10} and B^{11}. B^{10} is the better fast neutron absorber and represents about 20% of naturally occurring boron. If natural boron turns out to be inadequately absorptive, it will be necessary to enrich the boron in the B^{10} isotope. The operative reaction is B^{10} (n,α) Li^7.

For those readers who are familiar with the four factor formula used with thermal spectrum reactors, $K_\infty = \eta\varepsilon pf$, the terms ε and p, fast fission factor and resonance escape probability have no meaning in fast reactors. The term f, thermal utilization, would need to be redefined as the quotient of captures by the fuel and captures by all core materials. We can call it "neutron utilization" since there is no term for this factor that has wide spread usage in the nuclear community. If we were to retain the term "f" for this "neutron utilization", then for fast reactors, $K_\infty = \eta f$.

The delayed neutron fraction, β, in fast reactors is comparable to thermal reactors, however since plutonium is the preferred fissile material in fast reactors and since β is lower for plutonium than for U^{235}, the delayed neutron fraction is lower. Prompt neutron lifetimes are two to four orders of magnitude shorter in fast reactors than in thermal reactors. The smaller β and shorter prompt neutron lifetimes increase the burden on the control system. More is said on this subject in Appendix 2A where the fuel form is discussed and Section 10 on control systems.

The concept of spectral hardening and softening is somewhat unique with fast reactors.[73] The term applies to the energy spectrum of the neutrons in the core. In any fast reactor, the spectrum will be softer, i.e. have lower energy than a fission energy spectrum since there will inevitably be neutron collisions with structural materials and the coolant. Even more important are the collisions with U^{238}. About 80% of the non-absorptive neutron collisions with U^{238} are inelastic, meaning the neutron is briefly absorbed then re-emitted at a significantly lower energy followed by a gamma emission from the excited U^{238} nucleus. Most of the remaining 20% of the non-absorptive U^{238} collisions result in fission. Small metal fueled reactors with high Pu^{239} enrichment will tend to have the hardest, i.e. highest energy spectrum while large oxide fueled reactors with lower Pu^{239} concentrations will have a softer spectrum. All other things being equal, it is preferable to have a hard spectrum since the neutron reproduction is greater at higher incident neutron energy. Thus, a harder spectrum leads to a higher breeding ratio. However, all other things are not equal as will be discussed in Appendix 2A on the fuel form appearing next.

Spectral hardening plays an important role in safety analyses of fast sodium cooled reactors. All other things being equal, increasing the coolant temperature decreases collisions with the coolant resulting in spectral hardening. Since this would lead to reactor instability, this phenomenon must be compensated for with some other effect such as Doppler. Importantly, sodium voiding accompanying boiling also hardens the spectrum thereby inserting positive reactivity.

73The concept of over- and under-moderation in thermal spectrum reactors is related. LWRs typically are designed with a small amount of under moderation so that a coolant temperature rise with resulting coolant density decrease causes further under moderation, thus inserting negative reactivity and enhancing reactor stability.

Appendix 2 Reactor core and Head Port design

A *Fuel form*

The precedent for the use of metal fuel in the early reactors was established by the production reactors, which used metal fuel clad in aluminum. Aluminum was chosen as a clad because of its availability, thermal conductivity, and low thermal neutron absorption cross section. Aluminum is not suitable for use above about 400°F, so zirconium with its tolerance for high temperatures and its low thermal neutron absorption cross section was selected for the early thermal reactors. For the case of fast reactors where thermal neutron absorption is not an issue, stainless steel with its excellent strength at high temperatures, its compatibility with sodium, and its relatively low cost proved to be the preferred cladding.

As early as the late 1950s, the desirability of changing the metal fuel form to the oxide began to emerge. First, the metal form did not perform well with burnup. With burnup as low as 10,000 MWD/MTU or about one atom percent, the fuel swelled and became spongy with fission gas buildup, partly as a result of its poor creep strength. Second, the metallic form is chemically reactive and difficult to handle. Some cladding materials considered as alloying agents to improve its creep strength were found to chemically react with uranium. Third, the metal exists in several different phases, which change with temperature, limiting its application to temperatures that are lower than would be desired in commercial sodium cooled fast reactors. Fourth, the metal is more difficult to reprocess than the oxide, adding steps that increase reprocessing costs. Fifth, the LWR industry was moving to oxide fueled systems and if synergies were to be achieved between the two classes of reactors in reprocessing and fabrication, it would be necessary for the breeders to use the same fuel form. Sixth, and perhaps of greatest significance is the Doppler Effect.

In the early 1950s, it was thought that Doppler wasn't too significant in fast reactors since the spectrum was well above the resonance region and the neutron flux in that region is relatively low particularly in small reactors. Illustrative of this mindset, a text book on fast reactors initially published in 1961 contains the following statement: "The Doppler Effect is small and of minor importance in fast reactors".[74] As core calculations were beginning on large, oxide fueled fast reactors, it was found that there was a small but not insignificant neutron population in the resonance region that could result in a significant contribution from the Doppler Effect.[75] This is due to the softening of the spectrum caused by oxygen in the fuel pellets, the higher flux levels associated with larger sized, lower enrichment cores using a less dense fuel, and the poor thermal conductivity and higher operating temperature of the oxide. The melting point of UO_2 is about 5150°F and PuO_2 is about 4400°F. The melting point of the mixed oxide, intermediate between

74R. G. Palmer & A. Platt, *Fast Reactors*, Temple Press, 1961

75Greebler, P., and Hutchins, B. A., "The Doppler Effect in a Large Fast Oxide Reactor – Its Calculation and Significance for Reactor Safety," Proc. Symp. Phys. Fast Intermediate Reactors, Vienna, Vol. 3, 1961, p. 121, International Atomic Energy Agency, Vienna, 1962.

these two extremes, declines up to 150°F with burnup, probably due to the buildup of fission products. At peak power levels, the oxide temperatures are not far below these limits, particularly when uncertainties are applied.

About the same time that interest in the use of oxide fuels was emerging, in 1955 the EBR-1 reactor experienced a partial meltdown caused by inward bowing of the fuel assemblies. The inner surfaces of the fuel ducts ran hotter than the outer surfaces, causing them to bow inward with increasing power. This inward bowing effect was a positive power coefficient which led to an excursion that terminated in a meltdown of approximately 40% of the core. This accident[76] led reactor designers to focus on the importance of the prompt reactivity coefficients, and to recognize that as many of the reactivity coefficients as possible needed to be negative.

Contributing to this situation is the delayed neutron fraction. The delayed neutron fraction, β, for the isotopes of interest in reactors, both LWRs and LMFBRs, are given below:

isotope	β
U^{235}	0.0073
Pu^{239}	0.0023
U^{238}	0.0157

Table 6 Delayed neutron fractions

Assuming a fast fission factor of 1.03 to account for U^{238} fissions gives a β of 0.0078 for LWRs. With plutonium fuel and U^{238} accounting for about 20% of all reactor fissions, the LMFBR β would be calculated to be 0.0050, considerably lower than for LWRs. It turns out that the β for LMFBRs is actually lower still. The determination of β for the reactor needs to account for the importance of the delayed neutrons produced. The "importance" of a neutron is a measure of the likelihood of that neutron participating in a subsequent fission. If it more likely to be absorbed than to cause a subsequent fission, its "importance" will be low. This is particularly germane in a fast reactor where core heterogeneity is enhanced by the presence of internal, radial, and axial blankets. Once the importance has been accounted for, the result is known as beta effective or β_{eff}. Since a good fraction of the U^{238} fissions occur in the radial and axial blankets where their importance is low, β_{eff} for LMFBRs is better approximated by 0.003 - 0.004. For the case of CRBRP with a heterogeneous core design, β_{eff} was calculated to be 0.0034. This smaller margin to prompt criticality further supports the argument of the need to provide the reactor with stability enhancing negative prompt reactivity coefficients.

Caught in the middle of this transition was the Fermi-1 reactor. In the mid 1950s when the plant was being designed there was early indication that the oxide fuel form could be superior to metal fuels but the magnitude of the Doppler Effect was unknown and the designers chose metal for the initial core to benefit from the negative reactivity coefficient resulting from thermal expansion of the fuel. For a small core such as Fermi-1, thermal expansion is a somewhat more important factor than for larger cores. It was also known that a higher breeding ratio and higher power density could be achieved using metal fuel. Sufficient quantities of plutonium were not available at the time for the initial core loading of the Fermi reactor so 25.6 wt. % enriched

76D. Okrent, *Meltdown and Analysis*, Fast Reactor Information Meeting, Paper II-B, p. 77, Chicago, Nov. 1957 uranium was therefore selected for the initial fuel load. The metal fuel was in the form of pins alloyed with 10% molybdenum which tended to stabilize the uranium phase to the γ form which remained stable over the range of temperatures of interest to the Fermi-1 designers. Moreover, the alloying with molybdenum improved the creep strength of the uranium over that of the pure metal. Although alloying penalized the breeding ratio somewhat, its advantages were considered more important than its disadvantages.

Nonetheless, concurrent testing that was being performed in the Materials Test Reactor (MTR) showed that even with alloying; the maximum allowable burnup in Fermi-1 with this fuel would be about 10,000 MWD/MTU, which corresponds to about 1 a/o. MTR data predicted that in the hotter regions of the core near its center, burnup limitations as low as 0.6 a/o would need to be imposed. Because of all these impracticalities, it was planned to replace the Fermi-1 core eventually with a Pu/U mixed oxide core with a predicted breeding ratio of 1.28.[77] Unfortunately, because of the problems the plant experienced with its steam generators and the partial melting incident, core replacement never occurred.

By the early 1960s, it became clear that if the LMFBR concept were to be successfully commercialized, it would be necessary to utilize oxide fuels and achieve burn-ups of at least 100,000 MWD/MTU. Several questions presented themselves. The fissioning process would generate fission gas within the fuel pin. This fission gas needed to be accommodated with some sort of gas plenum, expected to be about 3-4 ft. long. This plenum could be located either above or below the axial blankets or a combination of both above and below. Would the gas successfully transport out of the fuel pellet, through the fuel and axial blanket regions into the plenum? Would the fuel pin be sufficiently strong to accommodate the fission gas pressure? Would it be better to vent the fission gas into the reactor cover gas and remove it there?

In furtherance of the objective to explore the benefit to be obtained from the Doppler Effect, the Southwest Experimental Fast Oxide Reactor (SEFOR) project was initiated early in 1964. The 20 MWth SEFOR project was a joint effort funded by the AEC, a group of interested domestic electric utility companies, and included international participation through the Karlsruhe Laboratory in Germany. Completed in May, 1969, its primary objective was to demonstrate and measure the Doppler Effect in a LMFBR. It turns out that the SEFOR project demonstrated several additional concepts important to LMFBR engineering, some of which are treated in this monograph.[78] The SEFOR reactor was provided with a control scheme that allowed super-prompt critical reactivity excursions in order to demonstrate the reactor behavior under extreme transients and the effect of Doppler in mitigating these transients.

While SEFOR construction was underway, a new and unexpected fuel problem arose. The phenomenon of stainless steel swelling at doses beyond 10^{22} n/cm^2 was first reported in 1967.[79] The phenomenon is apparently caused by a super-saturation of point defects, leading to voids collapsing into agglomerations producing internal strain in the material. When it was first observed, there was some evidence that it could lead to as much as 10% strain at fluences of 10^{23}

77*Fermi 1, New Age for Nuclear Power*, E. Pauline Alexanderson, ed., American Nuclear Society, 1979, pages 132-3

78J. O. Arterburn, G. Billuris, G. B. Kruger, "*SEFOR Operating Experience*", ASME, July 19, 1971

79C. Cawthorne, E. J. Fulton, *Voids in Irradiated Stainless Steel*, Nature, Vol. 216 (Nov. 1967), p. 575

n/cm^2. Since large LMFBRs will have flux levels as high as 10^{16} n/cm^2-sec, the onset of swelling could occur at two months of operation. In addition to producing elongation of the fuel pins, since the fluence tends to be greatest at the core centerline, the interior sides of the fuel ducts receive a greater fluence than the exterior causing the assembly to tend to bow outward at the ends. If the fuel assembly is restrained horizontally, as it certainly must be the bowing results in a stress, which is relaxed by creep, which may be caused by temperature alone or be enhanced by radiation.

In 1963 the phenomenon of neutron embrittlement on stainless steels from neutron irradiation was first reported.[80] The phenomena associated with embrittlement are closely related to swelling. As the swelling approaches 20%, the strength of stainless steel drops precipitously and the material becomes brittle. This embrittlement is believed to be due to micro-fractures between voids and other dislocations in the crystal structure aligning perpendicular to any applied stress once the swelling reaches a critical level. Embrittlement may limit the ultimate exposure of fast reactor fuel to something in the 200,000-300,000 MWD/MTU range corresponding to about ten years of residence in the core.

By the mid 1960s, the unexpected fuel phenomenology produced a somewhat predictable effect. Throughout the LMFBR design community there emerged great activity focused on testing of fuel in order to get the best possible characterization data. By the time the CRBRP project was underway, a consensus had emerged that 20% cold worked 316 stainless steel was sufficiently resistant to swelling that it could be used with reasonable confidence for exposures up to a peak of 150,000 MWD/MTU with average burnup of 80,000 MWD/MTU. This was sufficient for CRBRP requirements so it was selected as the reference material for both the clad and the ducts.

As has been described earlier, there has not been much meaningful activity in breeder reactor development since the cancellation of CRBRP. However, two alloys have emerged that have promising applications as core assembly materials – D9 and HT9. D9 is an austenitic steel that is a variant of SS316 having slightly higher nickel and lower chromium with a small amount of titanium added. HT9 represents a more radical departure from the SS316 precedent – it is a ferritic alloy containing 12% Chromium and 1% Molybdenum. Both of these materials have enhanced swelling resistance as compared with 316SS and provide reasonable confidence that there are avenues available for performance improvement of steels that will enable the use of long lived cores.

A record high burnup level of about 35 a/o has been successfully reached with a six pin test assembly at BOR-60 while 260 pins have achieved 25-30 a/o burnup.[81] All of this high burnup Russian experience has been accomplished with advanced alloy materials that are akin to but not identical with D9 and HT9.

Following the termination of the CRBRP project, technical representatives from ANL began resurrecting interest in the use of metal fuels in LMFBRs. The claim was that the low burnup issues had been resolved through the use of improved alloying materials and that metal fueled

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80A. C. Roberts, D. R. Harris, *Elevated Temperature Embrittlement Induced in a 20 %Cr-25 %Ni Nb Stabilized Austenitic Steel by Irradiation with Thermal Neutrons*, Nature, Vol. 200 (Nov. 1963), p. 772
81IAEA-TECDOC-1569

reactors offered more acceptable methods for dealing with ATWS events. However, even if it were possible to design a metal fuel that could reach the same burnup as an oxide, a very dubious possibility, the other five disadvantages of metal fuels stated at the beginning of this section would still apply, in particular the lack of any strong Doppler feedback. Moreover, there appears to be limited interest in metal fuels abroad – those countries that continue to develop the technology are doing so based on the use of oxide fuels. For these reasons, the design approach presented herein will be based on the use of oxide fuels with whatever cladding material offers the best performance at high burnup and is consistent with the identified operating parameters.

B *Vented fuel*

From the above discussion, it is clear that the fuel represents perhaps the most challenging single aspect of LMFBR design. The challenge is made no easier by the requirement that the fuel pin contain the fission product gases. The inert gases, krypton and xenon represent over a quarter of the fission product inventory so the pressure inside the fuel pin rises quickly with burnup. For the case of CRBRP, at a peak burnup of 10 atom percent, the pressure in the fuel pin was expected to be about 1000 psi. From the foregoing discussion it is clear that the damage being sustained by the cladding during reactor operation makes it ill equipped to deal effectively with the even higher pressures attendant with burnup greater than 10 atom %, but higher burnup is almost mandatory if the LMFBR is to have much of an opportunity of being economic. This raises the question of whether it is feasible to vent the fuel eliminating or greatly reducing the fuel pin pressure.[82]

Another incentive for venting the fuel pins is to reduce the length of the fuel assembly. Despite the fact that just 5 ft. 4 in. of the CRBRP fuel assembly is fueled, it is nearly the same length as a LWR fuel assembly and the pressure drop across the core is greater than is the case for a LWR. For CRBRP, core pressure drop was approximately 110 psi, about double that of a PWR. Partly this is attributable to the LMFBR core lattice being tighter and fuel assembly ducting with orifices to regulate flow to each assembly, but reducing the fuel assembly length by four feet would have reduced the core pressure drop on CRBRP by about 10-15 psi. Reducing fuel assembly height pays off in a corresponding reduction of both the reactor vessel and containment height. If EM pumps are used for the PHTS, there is an added incentive for reducing core pressure drop to compensate for the lower pumping efficiency of EM pumps.[83]

82Another approach for dealing with this issue has been to move the gas plenum to the bottom of the assembly where it is exposed to cold leg sodium and would therefore be at a lower pressure. This approach was taken on the Superphénix plant. There are two objections to this approach. First, a pin failure will result in fission gas passing upward through the fuel assembly injecting positive void reactivity. For a single assembly this would not be enough to create much of an excursion, but it may cause a reactor trip. The second objection is that placing the gas plenum at the bottom raises the thermal center of the core requiring the thermal centers of the IHXs and steam generators be raised an equal amount to preserve natural circulation capability. This second objection could introduce a fairly large adverse capital cost impact.

83If under-the-head refueling is maintained, reducing the length of the fuel assembly has a double pay-off since the requirement for transferring fuel assemblies above the core but under the head will also be reduced by four feet.

A third incentive for venting is to increase the fuel fraction of the fuel & blanket assemblies. If high burnup is attempted with unvented fuel, the cladding thickness must be increased to contain the higher fission gas pressure decreasing the volume available for fuel & blanket material. This is counterproductive to attempts to achieving long cycle time which require good breeding ratio thus a high fuel fraction in the core.

There are two options for fuel pin venting. One would be to collect the fission product gases within some chamber in the fuel assembly then carry away the gases using plumbing connected to each assembly. A second option would be to vent the gases directly to the coolant. Since the second option presents the easier solution, is less of a problem with refueling, and is potentially more reliable, it is worth considering first.

If fission product gases are vented to the coolant, it would be necessary to prevent sodium from flowing into the fuel pins during transients or reactor shutdown. Sodium chemically reacts with oxide fuel causing it to swell considerably. Prevention of such intrusion might be accomplished with some form of check valve or relief device that limits flow to one direction only. The fission product gases, if vented to the coolant would quickly accumulate in the cover gas where they could be removed by cryogenic distillation. Cryogenic distillation depends of the differing liquidus temperatures among the inert gasses. For the purpose of this discussion, the liquidus temperatures for inert gases of interest are presented below;

He	-268°C
Ne	-246°C
A	-185°C
Kr	-152°C
Xe	-107°C

Table 7 Boiling point of inert gasses

From the above, it is seen that cryogenic distillation should work for the fission gasses with either argon or helium as a cover gas, but it would certainly work better with helium. Neon is included in the list because it is the decay product of Na^{22} and will gradually build up in the cover gas if not removed. This probably doesn't matter since Ne^{22} is stable. The xenon fission products do not pose much of a problem since the longest lived among them is just 5.27 days[84], however Kr^{85} has a half life of 10.44 years. Kr^{81} has a half life of 2×10^5 years, but little Kr^{81} is produced in fission. The means for dealing with krypton is one of the topics discussed in Section 12. Alternatively, the gas could be stored, at least allowing for Kr^{85} decay. About $25 \cdot 10^6$ cm^3 of fission gas at STP would be produced yearly by a 3000 MWth reactor, a tenth of which would be krypton with the balance xenon. Most of the xenon produced is stable or has a short half life and could be released. Assuming the krypton can be separated from the xenon, less than 100 ft^3 of krypton at standard temperature and pressure would be generated yearly, which would be a manageable storage problem.

Other fission products that might find their way into the coolant would include iodine, which would rapidly combine with the sodium and be removed by the cold traps. One potential

84Except for Xe^{131m}, which has a half life of 11.96d but has a very low fission yield.

problem with venting the fuel assemblies into the coolant is likely not to be from the fission product gases or iodine but from Cs^{137}. Cesium is an alkali metal with a melting point of 160°F and a boiling point of 1240°F, well below the temperature of the fuel when it is operating at full power. Although much of the cesium in spent fuel will combine with oxygen in the fuel, some will remain in its elemental state and be vented with the fission gases. It will then intimately mix with the sodium. It cannot be directly removed by chemical means, and will not be removed by the cold traps.[85] With its 37 year half life, it will build up in the coolant and complicate plant maintenance. It was stated earlier in the paper that the Cs^{137} activation in the BN-350 reactor at end of life was 6-7 $\mu Ci/cm^3$. Venting the fuel to the coolant could lead to levels a hundred-fold higher, which would be equivalent to ~1000 $\mu Ci/cm^3$ and would lead to unacceptably high radiation fields in any areas carrying primary coolant. BN-350 ultimately dealt with their Cs issue by installing reticulated vitreous carbon filters, which reduced Cs^{137} concentration by a factor of 800.[86] Carbon filters were first developed by ANL for use on EBR-II and require the sodium to be cooled to the 300-400°F. range. Such a device could be installed as a parallel stream on the line to the cold traps. Alternatively, some form of distillation could be used.

Venting the fuel to the coolant could create a licensing issue. Since the very beginnings of nuclear power development, a basic principle of design has been there are three barriers between the fission products and the environment: the fuel cladding, the primary system, and the containment. While not eliminating it altogether, venting the fuel to the coolant diminishes one of these barriers. One could argue that since the vented fission products are being continuously removed, the barrier is still, in effect, present, but the outcome of such an argument cannot be predicted with certainty.

A corollary fringe benefit of vented fuel is to eliminate the need for gas tagging of core assemblies. This, of course, raises another licensing issue. With vented fuel, it would become more difficult to locate a failed fuel assembly. One might question whether it is really necessary to locate a failed fuel assembly. Operating with failed oxide fuel has been demonstrated by testing at both EBR-2 and FFTF and operationally at BOR-60 without serious consequences but not over the long cycle lengths being advocated herein. The problem is that when sodium combines with UO_2, the resulting compound has a lower density than the UO_2. Although most if not all fuel material exposed as a result of a cladding breach is likely to be swept away by the coolant and wind up in the cold traps, it does not require much of a stretch of imagination to envisage sodium entrained fuel material blocking flow in the failed fuel assembly which could only be detected by the core exit temperature detectors.

An alternative to venting to the coolant might be to vent the fuel to a header that is collected and processed. One could envision the fuel pin vents being collected in some kind of a chamber at the top of each assembly then routed through channels in the upper internals structure, through the head, then to a processing station. This would not solve the failed fuel location issue since tagging would serve no purpose if the assemblies are continuously vented to a header, but it would eliminate the problem of cesium in the coolant. From a licensing perspective, this

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85It is possible that advantage might be taken of the higher chemical activity of cesium in comparison to sodium. Any oxygen in the coolant will preferentially combine with the cesium, for example.

86O.G. Romenenko, K.J. Allen, D.M. Wachs, H.P. Planchony, P.B. Wells, J.A. Michelbacher, *Cleaning Cesium Radioactivity from BN-350 Primary Sodium*, Nuclear Technology, 2005 approach could well be considered worse than venting to the coolant since it would bring the fission products outside both the clad and the primary system boundary.

Regardless of the venting method used, if peak burnup in the vicinity of 30 atom percent is contemplated, fuel venting may be a necessity. Moreover, it improves neutronics performance and carries with it the prospect of shortening the reactor vessel and the containment by 4 ft., a worthy benefit.

C Nuclear design and core layout

The core design is, without question, the most important component of the total plant design. It drives the reactor vessel design and indirectly the refueling system design, the primary heat transport system design, and the containment design. When designing the plant, it should be the first design to be undertaken to a sufficient level of detail that confidence in its design is established. On the CRBRP project, that did not turn out to be the case as was described in Section 6. The point is that before any significant activity is undertaken on the plant design, there needs to be high confidence that the core design will not change radically.

If a reactor were to be designed with uniform enrichment, the resulting radial flux profile would be the fundamental mode of the wave equation in cylindrical geometry, which is actually the $J_0(r)$ Bessel function, but is reasonably closely approximated by the cosign function, which is the correct function describing the axial flux distribution. Although there is some neutron reflection at the core boundaries, it is not as pronounced as is the case for a LWR. Most of the power would be produced in the center of the core with relatively little production at the peripheries. As the core burned down, the flux would gradually move outward toward the regions where less burnup had occurred, but most of the burnup would be in the center of the core. Designers have combated this problem in LMFBRs by using higher enrichment assemblies in the outer core region, providing axial blankets above and below the core and a radial blanket with reflector/shield assemblies outside the radial blanket. The blanket assemblies are primarily intended to improve neutron economy and breeding and the shield assemblies are primarily provided to protect permanent structural components from excessive neutron fluence, but both also tend to reflect some of the neutrons back into the core. The result is a flatter flux profile across the core. The early CRBRP design was executed exactly this way. There were 109 inner fuel assemblies with fissile enrichment of about 16% and 90 outer fuel assemblies with a fissile enrichment of about 23.2%. The outer enrichment zone was surrounded by 150 radial blanket assemblies which occupied ~2 ½ rows. The fuel assemblies consisted of 217 0.23 in. diameter pins while the blanket assemblies consisted of 61 0.51 in. diameter pins. It was planned to refuel one third of the core annually. This core design is shown in the figure below where the shaded assemblies comprise the radial blanket and the outer enrichment zone comprises the outer two rows of the fueled core..

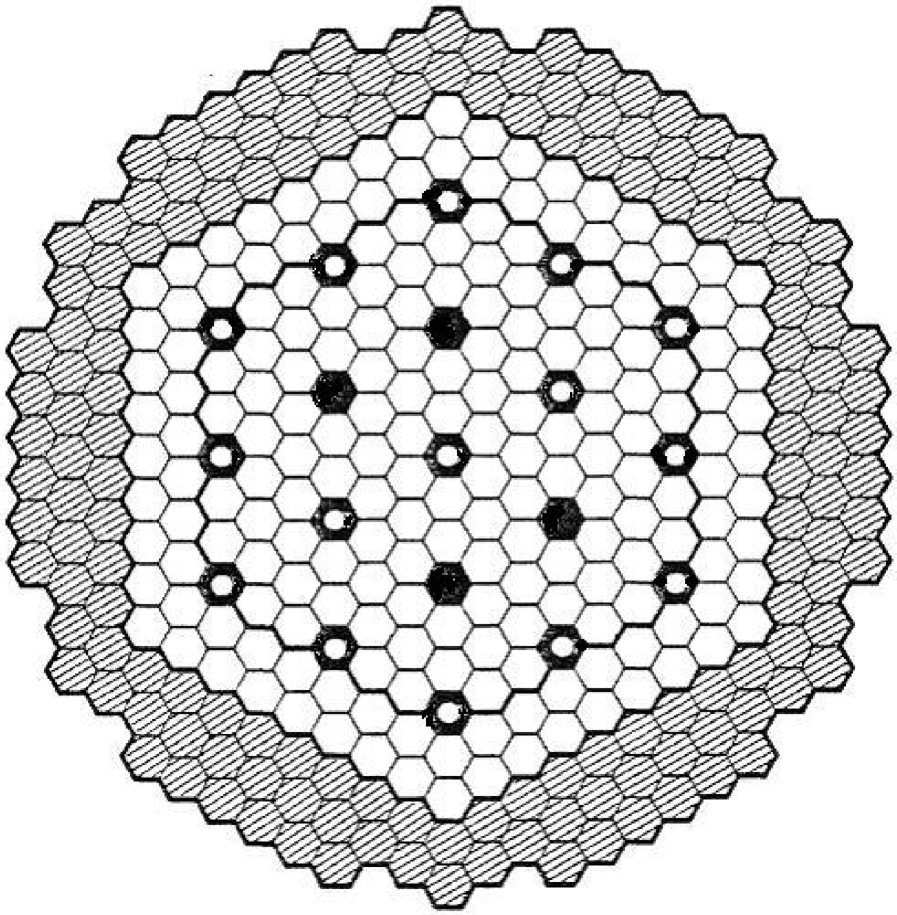

Figure 44   Early CRBRP two region homogeneous core design

When the CRBRP project was initiated in 1972, there was prepared a set of "Demonstration Plant Design Guidelines", which were agreements between the government and utility company entities that were funding of the project on how the design would go forward. One of these design guidelines was a requirement that the plant be designed to operate with a minimum breeding ratio of 1.2. By the time the conceptual plant design had been completed in 1975, it was obvious that the breeding ratio guideline would not be met. The project office requested that some action be taken to address the issue.

Initially the prime contractor argued that the demonstration plant design guidelines would be met in the initial cycle if LWR recycled plutonium were deployed as the fuel rather than the low Pu^{240} plutonium that was planned to be used.[87] The case wasn't argued long since even with recycled LWR plutonium (which was not then available); the guideline could not be met beyond the initial cycle. To address the issue, the nuclear design personnel proposed a heterogeneous core design, selectively locating radial blanket assemblies in the high flux region of the core. At that time, they called this approach the "alternative fuel management scheme" or AFMS, since they wished to retain the homogeneous design as the plant's reference design pending further

87The CRBRP planned plutonium isotopic concentrations were 0.1% Pu^{238}, 86% Pu^{239}, 11.7% Pu^{240}, 2% Pu^{241}, and 0.2% Pu^{242}.

evaluation of the AFMS, particularly its thermal-hydraulic performance. Although the resulting design had a somewhat higher fissile inventory, it satisfied the applicable demonstration plant design guidelines and offered some additional advantages. It reduced the number of fuel assemblies required and increased the amount of power produced by the blanket assemblies. Since fuel assemblies were expected to be considerably more expensive to fabricate than blanket assemblies, this was seen as having a direct economic benefit. AFMS also eliminated the need for two separate fuel enrichment zones. It was more effective at flattening the neutron flux profile than the two region concept and led to lower radial power peaking. The better flux flattening could be attributed to the better fine tuning available with the inner blanket assemblies. The neutron flux was somewhat lower due to the higher fissile inventory.

Fewer control rods were required primarily because simultaneous with the design of the heterogeneous core, it was decided to enrich the primary control assemblies. This was a sensible thing to do since it is much less expensive to enrich the boron for the control assemblies than to fabricate additional control assemblies plus their drive mechanisms. In addition, fewer control assemblies means more core positions that can participate in the breeding process, either as a fuel assembly or an inner blanket assembly. The decision to enrich the control assemblies could have been made independent of the core design selected, and would have improved the performance of the homogeneous design somewhat. The secondary control assemblies had been designed to be enriched from the outset. With the heterogeneous design, the number of secondary control assemblies was increased from four to six mainly because the heterogeneous design resulted in each secondary control assembly being adjacent to two inner blanket assemblies, which decreased the secondary control assemblies' individual effectiveness.

The effectiveness of the axial blankets improves with the heterogeneous core design because the heavy metal volume fraction of blanket assemblies is higher than it is for fuel assemblies. Since the inner blanket assemblies are mixed in with fuel assemblies, the average heavy metal volume fraction of the axial blankets above and below the core increases. For the case of CRBRP, fuel assembly heavy metal volume fraction was 0.325 while blanket assembly heavy metal volume fraction was 0.539. Averaged over the region containing fuel and inner blanket assemblies, this yields a heavy metal volume fraction of 0.395 in the axial blanket, a 21.5% increase. This could have been capitalized upon by decreasing the thickness of each of the axial blankets by three inches, which would have decreased overall assembly length by 6 inches, but there was no point in taking the indicated action as the reactor vessel dimensions were frozen at that point.

On a related subject, there does not appear to be any good reason for why the upper and lower axial blankets should have the same thickness, particularly in view of the fact that the control rods extend down into the core from the top, shielding the upper axial blanket. As an example, at end of cycle 4 on CRBRP when the control rods would be expected to be near their uppermost position, the lower axial blanket produced 1.85% of the average fuel assembly heat while the upper axial blanket produced 1.32%. Some trimming of the upper axial blanket perhaps in favor of extending the lower axial blanket should be evaluated in any future LMFBR design.

An unexpected benefit of the heterogeneous configuration was a lower sodium void worth in the fuel assemblies probably attributable to the fact that adjacent internal blanket assemblies acted as neutron sinks. In addition, the heterogeneity introduced a greater degree of incoherence in the voiding sequence. The heterogeneous design was adopted as the reference design prior to the project's termination. The CRBRP heterogeneous design as it appears in the CRBRP PSAR is shown in the figure below.[88] In the figure, the blanket assemblies are cross hatched, the fuel assemblies are white, the nine primary control assemblies are black with an open circle and the six secondary control assemblies are solid black.

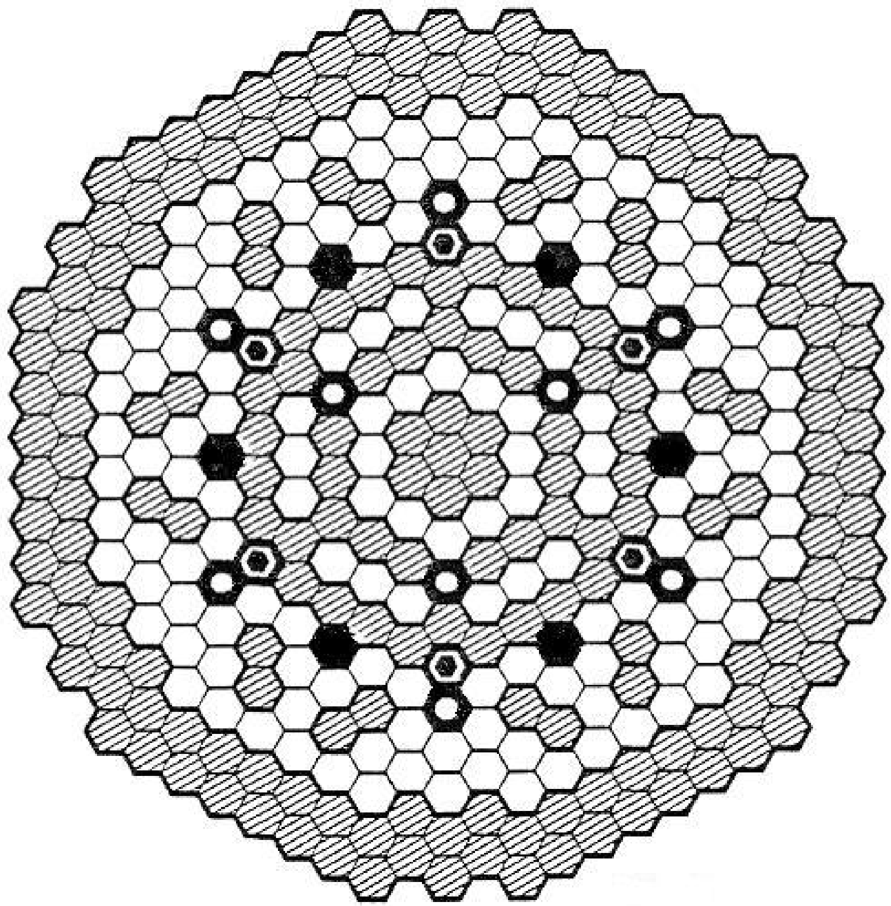

Figure 45 CRBRP heterogeneous core design

When describing LMFBR core designs, it is conventional to refer to the center assembly as occupying "row 1", the six adjacent assemblies as occupying "row 2" and so on. Note that the seven center assemblies are all blanket assemblies. Row 3 consists of fuel assemblies; row 4 has 15 blanket assemblies and three primary control assemblies, and so on. The corners of row six are occupied by positions that are blanket assemblies for the first half of a two year cycle and fuel assemblies for the second half. Row seven has six primary and six secondary control assemblies along with 24 fuel assemblies. Note that in comparison with the homogeneous

88The initial AFMS design proposed by Westinghouse personnel was somewhat different than the design shown in the accompanying figure. Many improvements were made to the core design as more analyses were performed. For example, the blanket assemblies in rows 8 and 9 were reconfigured to improve the worth of the control rods in row 7. Additionally, the alternate blanket/fuel positions did not exist in the original rendering. Approximately four years were required for Westinghouse nuclear and thermal-hydraulics personnel to settle on the core design shown.

design, the heterogeneous design has just two rows of radial blanket assemblies and has twelve additional positions occupied by fuel, blanket, or control assemblies.

The heterogeneous core design proposed by the prime contractor had 156 fuel assemblies with about 30.5% fissile enrichment, 77 internal blanket assemblies and 132 radial blanket assemblies. The six positions that were intended to be loaded with internal blanket assemblies for one year then changed out for fuel assemblies were provided to recover reactivity lost during the first year of operation.[89] 12 of the 324 removable shield assemblies in the homogeneous core design were replaced with radial blanket assemblies. Some of the shield assemblies were changed to Inconel to improve their shielding effectiveness. At the end of two years, the entire central core region would be replaced including all the internal blanket assemblies. There were two rows of radial blanket assemblies and the inner row blanket would be renewed after four years of operation while the outer row would be renewed after five years.

CRBRP was led to the heterogeneous design primarily because of the artifact that required it to use the same pin diameter as the FFTF. This was mainly motivated by the large fuels data base that was expected to be collected by FFTF operation and the need to make use of that data base on CRBRP. Another incentive may have been the concern that if CRBRP were to use a different fuel from FFTF, it would remove one of the incentives for completing the FFTF project. After the design team had made their proposal and it had largely been acceptable to the Project Office, representatives from Argonne National Laboratory pointed out that the same breeding ratio could have been achieved in a conventional two region core by using fuel with a pin diameter of 0.24 in. rather than the 0.23 in. used in CRBRP. The pressure drop across the pin bundle would have been greater but it probably could have been accommodated by the oversized primary system pumps. If the constraint of using the FFTF fuel design had not been applied to the CRBRP project, it is doubtful the heterogeneous core design would ever have been developed.

It turns out that heterogeneous designs have another important hidden advantage. Much of the plutonium production occurs in a region of high neutron importance (the word “importance” as used in this context should be assumed to have the meaning associated with its use in perturbation theory) which opens the door to very long lived cores. With homogeneous designs, the neutron flux gradually moves outward toward the radial blanket as increasing amounts of fissionable material are produced there. Plutonium has a lower importance in the outer reaches of the core than it does in the center, and the core gradually loses reactivity. Heterogeneous cores can be designed so that the plutonium production is greater near the center of the core where it has the greatest importance to reactivity. This is of key significance for designing a reactor where the reactivity swing is to be minimized so that it can operate for long periods between refueling.

89These alternating assemblies turned out to have the highest radial power peaking factors in the reactor core during the even years when they were occupied with fuel assemblies. At end of cycle 4, these assemblies had a peaking factor of 1.231 compared with 1.109 for the next highest assembly. At the beginning of cycle 3, the highest radial power peaking factor anywhere for a fuel assembly pin on the core was just 1.176, which is remarkably low and demonstrates the effectiveness of heterogeneous designs in flattening the radial flux profile. Had the project not been cancelled, it seems likely that this alternating concept would later have been abandoned in favor of a design that did not require this change out.

A secondary effect that also contributes to the designer's ability to provide for long core lifetime is the shielding afforded by the blanket pins closest to the fuel assemblies. The power level and plutonium production provided by the interior blanket pins increases as the fissionable content in the outer layers of pins increases. It is this line of thinking that suggests that thickening the layers of the parfait, particularly the blanket layers, may contribute to the long life objective.

It is instructive to consider the dimensions of a large heterogeneous core with parameters set to the assumptions of this paper. A 271 pin bundle will be assumed for the fuel assemblies. This is one row of pins greater than CRBRP and equal to the number of pins per fuel assembly used on Superphénix. There is nothing particularly magic about a 271 pin fuel assembly bundle. The larger the number of pins per bundle, the lower the fuel fabrication cost, the better the breeding ratio, and to a small extent, the lower the pressure drop across the assembly. It is difficult to argue that a 331 pin bundle wouldn't be preferable to a 271 pin bundle. A larger fuel assembly would reduce the flexibility in the core design but this is probably not a strong argument. Other considerations do come into play such as refueling and shipping, both of which are more challenged by a larger fuel assembly. The 271 pin bundle is selected primarily on the basis of Superphénix precedent and the objective of avoidance of unnecessary surprises more than any other reason. A 271 pin fuel bundle with 0.33 in. pins, a slightly higher wire diameter, and duct thickness about the same as CRBRP would measure 7.19 in. across the flats. The wire diameter was increased from a CRBRP value of 0.056 to 0.060 so as to improve flow area. There will be more on this subject in the thermal dynamics section.

The number of pins per blanket assembly is another matter. If it is planned to leave the blanket assemblies in place for ten years, the pins must be small enough so their pin power limits would not be exceeded at the end of the cycle. The only way to make this determination with certainty would be with analysis of the selected core design, but a reasonable estimate can be made by extrapolation from CRBRP analyses. For CRBRP, the inner blankets produced about 7% of full reactor power at the beginning of an equilibrium cycle and about 16% power at the end of an equilibrium cycle. At this point, many of the CRBRP blanket pins were approaching limiting pin power. The CRBRP blanket pins each contained 6.1 times the volume of the respective fuel pins. Given the greater fuel pin diameter in the proposed design (there is 2.3 times the heavy metal per inch in 0.33 in. diameter pins vs. 0.23 in. diameter pins), 2 years of residence in the CRBRP core is about equivalent to 3.6 years in the proposed core allowing for the different assumed capacity factor. Therefore, 10 years in the proposed core is about equivalent to 5.5 years for the CRBRP or 2.7 cycles. If the proposed blanket pins are no more than 6.1/2.7 = 2.26 times greater in volume than the fuel pins, the system should work. It turns out that this corresponds to a blanket pin diameter of about 0.47 in. with the same pin wall thickness of 0.015 in. For the duct size selected, this corresponds to a 127 pin bundle.

Blanket assemblies with 127 pins per bundle for the duct size selected would have a pin diameter of 0.47 in. The inner blanket assemblies would be producing about 25% of total core power at the end of the 10 year cycle.[90] If one scales up to 3000 MWth from CRBRP accounting for the

90The proposed core design has 107,587 fuel pins, 28,956 internal blanket pins, and 25,146 radial blanket pins or 66.5%, 17.9% and 15.6% respectively. By comparison, the CRBRP core had an average of 34,503 fuel pins, 4,880 inner blanket pins, and 8052 radial blanket pins or 72.7%, 10.3% and 17.0% respectively. This comparison suggests it may prove necessary to add inner blanket assemblies to the core of the proposed design. At 25% power from the inner blanket, some of the inner blanket assemblies may be approaching or exceeding pin power limits.

difference in the number of pins per fuel assembly, there would be approximately 375 fuel assemblies, 228 internal blanket assemblies, and 396 radial blanket assemblies. The number of control assemblies would increase, but not necessarily in proportion to size. For the purposes of this discussion, it is assumed there would be 30 control assemblies. Fortuitously, there was a heterogeneous oxide core design study performed by ANL at the 3500 MWth size and is shown below.

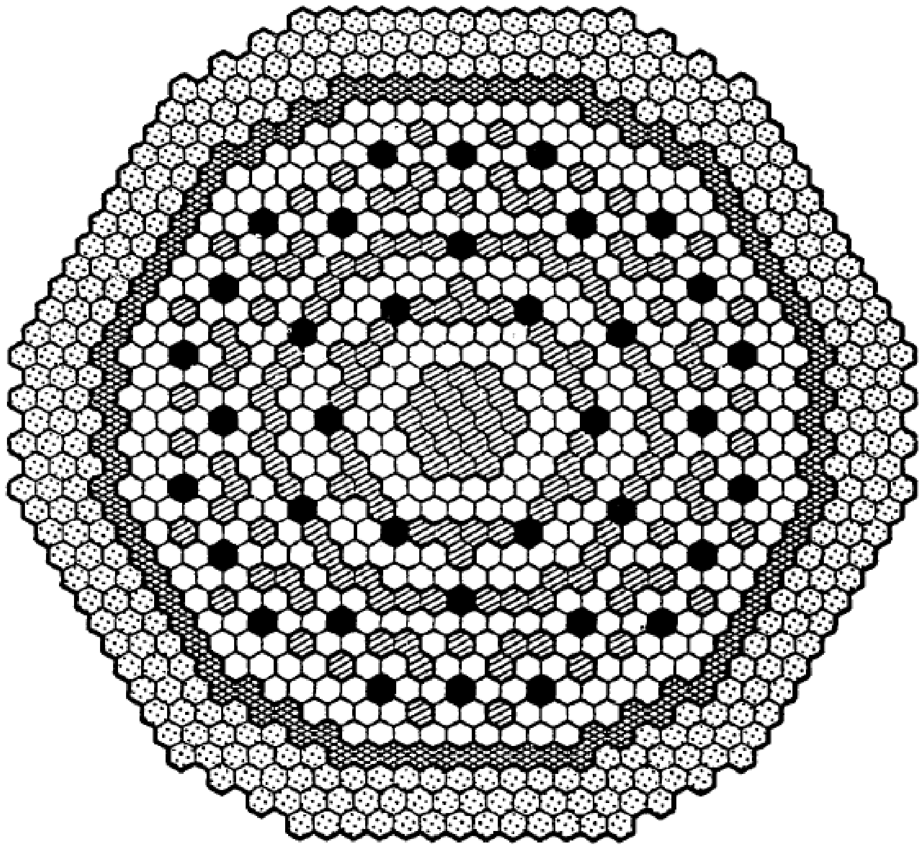

Figure 46 ANL 3500 MWth core design study

The ANL study[91] appears to have been conducted primarily to explore the advantages of metal fuels, but it nonetheless provides some information that is useful. It had smaller diameter fuel pins (0.285 in.) and thicker clad (0.022 in.) than is assumed herein. There are 396 fuel assemblies, 163 internal blanket assemblies, and 90 radial blanket assemblies. The active core region was 40 in high. This core was designed for annual refueling of 1/3 of the core. Note that there is just one row of radial blanket assemblies and three rows of shield assemblies. The ANL study enabled reduction of the number of shield assemblies by retaining just one row of metal shield assemblies and replacing the outer rows with boron assemblies. The inner row of metal shield assemblies helps reflect some neutrons back into the core, while the boron assemblies

91Fujita, E. K.., and Wade, D. C.; *The Neutronic and Fuel Cycle Performance of Interchangeable 3500 MWth Metal and Oxide Fueled LMRs;* ANL-FRA-163; March 1989.

make a better shield material for protection of the core barrel. The single row of radial blanket assemblies is an acknowledgment of the low value of the second row of radial blanket assemblies and the usefulness of making more room available for core assemblies. Note also that of the 36 control assemblies, 24 are surrounded on six sides by fuel assemblies. The remaining twelve have blanket assemblies on two sides only. This action was taken to improve the worth of the control assemblies. All of these ideas have merit.

The ANL core design could be expanded by two additional rows and still result in a core diameter of less than 21 feet using the pin diameter and duct size assumed in this paper. Adding some room for fixed radial shielding would bring the core barrel diameter to less than 23 feet. Assuming the space between the core barrel and the reactor vessel is minimized, a topic treated in the refueling section of this paper, there should result a reactor vessel diameter no greater than 28 ft. This 28 ft. diameter reactor vessel is the target diameter for the purposes of this paper. It is possible it could be reduced further pending detailed core design calculations. 28 feet probably represents a reasonably good upper bound.

One final topic that requires treatment is the height of the active core region. On CRBRP and FFTF, the active core was 36 in high which is a fairly typical number. Core designs tend to be short in LMFBRs since the sodium coolant is effective at removing heat. Three feet of active core height are usually all that is required to raise coolant temperature the required amount while remaining within linear power limits and additional length would add to the fissile inventory and was seen to serve little purpose. It is noteworthy that the Superphénix plant had an active core height of 1000 mm (39.37 in) and the planned Superphénix II was being designed for an active core of 1200 mm (47.24 in). It appears that the French saw an advantage to lengthening the active region of the core. There are two good reasons for doing so. First, a longer core leads to lower power density per foot of pin length and/or permits a higher ΔT across the reactor. The lower power density per foot allows for higher hot channel factors, should they develop in the design. Second and of greater importance, a greater core height increases the total heavy metal inventory in the core and allows for more total energy production per refueling cycle within limits of allowable burnup.

If one uses the assumed plant capacity factor (defined later in this section), assumes a ten year interval between refuelings, assumes 65% of the energy developed by the reactor comes from the fuel assemblies in the active core region, assumes an average of active fuel assembly 15 a/o burnup would be consistent with a peak burnup of 30 a/o, uses the number of fuel assemblies in the proposed design (identified later in this section) and uses the thumb rule that 1 gram uranium = 0.95 MWth-day, it is straightforward to calculate (0.65% X 365days per year X 3000 MWth X 10 years X 0.95 MWth per gram HM ÷ 0.15 a/o X 0.95) that 47.480 MTU must be in the active core region of the fuel assemblies. For this to work consistent with the parameters selected for the fuel assemblies, the active core region must be at least 47 in high. Rounding to the nearest whole foot, a 48 in high active core region will be used. Adding a foot to the active core region has the added advantage of holding down the linear pin power of the inner blanket assemblies, which could prove to be problematic toward the end of the cycle after substantial plutonium has bred into them. An alternative would be to add fuel assemblies, but doing so would add to the reactor vessel diameter and would probably be much more expensive than simply lengthening the fuel assemblies when the only purpose is to increase the heavy metal inventory.

The core design that is being proposed here is not radical. By way of comparison, the Superphénix reactor, which also had a nominal 1200 MWe output, had 0.33 in. diameter pins with 271 pins per bundle. There were a total of 364 fuel assemblies, 233 blanket assemblies (in 91 pin bundles) and 21 control assemblies. It was planned to replace 1/5 of the core each year for a 5-year residence time of the average fuel assembly. Even with a 5 year residence time, burnup was intended to be limited to 9 a/o. Superphénix clad thickness at 0.022 in. is slightly thicker than the 0.015 in. proposed here. The core proposed here would run at a slightly lower pin power than Superphénix and rather than annual refueling, the whole core would be refueled on ten year intervals. The big changes from Superphénix are the heterogeneous core design, vented fuel, and controllable flow to the blanket assemblies as will be discussed in the thermal-hydraulics subsection which follows. It should be noted that once thermal-hydraulic analysis of this proposed core approach is undertaken, it is expected that it will prove unnecessary to provide variable flow to the fuel assemblies however the internal blanket assemblies would most likely require it.

The tricky part of making this long lived core work will be to achieve the desired reactivity swing with burnup. It would be desirable for the reactivity to slowly increase for the first five to six years of operation and decrease for the last four to five years. The control rods would be almost fully withdrawn at both the beginning and the end of the ten year cycle and inserted to their maximum allowable at some point near the middle of the cycle (except for the secondary rods, which would be always fully withdrawn when the reactor is operating). Although the CRBRP project may have been the first to incorporate a heterogeneous core design as the reference design for a plant that was intended to be built, that project did not invent the concept. Prior to the mid 1970s, heterogeneous core designs were referred to as “parfait cores”.[92] The trick to controlling the reactivity swing of these “parfait cores” may be to thicken the layers of the parfait. Adopting a core that is larger than necessary to remain within peak linear power limits may well be what is needed to thicken the layers and achieve the cycle length objective.

Another degree of flexibility that could be called upon would be to extend the fuel assemblies by a foot or so and insert a layer of blanket material between two 24 in. fueled zones. This is the so-called “axial parfait core”. In addition to its potential for improving breeding, such an approach would also tend to reduce the hot channel temperature by providing a greater degree of intra-assembly mixing. Doing this would add to the length of the reactor vessel and would introduce irregular reactivity behavior from the control rods over their travel through the inner blanket. It could theoretically lead to oscillations between the upper and lower core zones if the nuclear coupling between them is weak; however, there is no obvious mechanism (such as Xenon poisoning in a LWR) that would initiate such oscillations. This axial parfait approach might be worthy of consideration if the proposed core proves incapable of meeting the operational goal of a ten year refueling interval. A core design that incorporates all the features being advocated (except for the horizontal inner blanket layer) is shown in the figure below.

92It is of some interest to note that the 10,000 MWth feasibility study performed by Argonne National Laboratory mentioned earlier in this paper used an annular core.

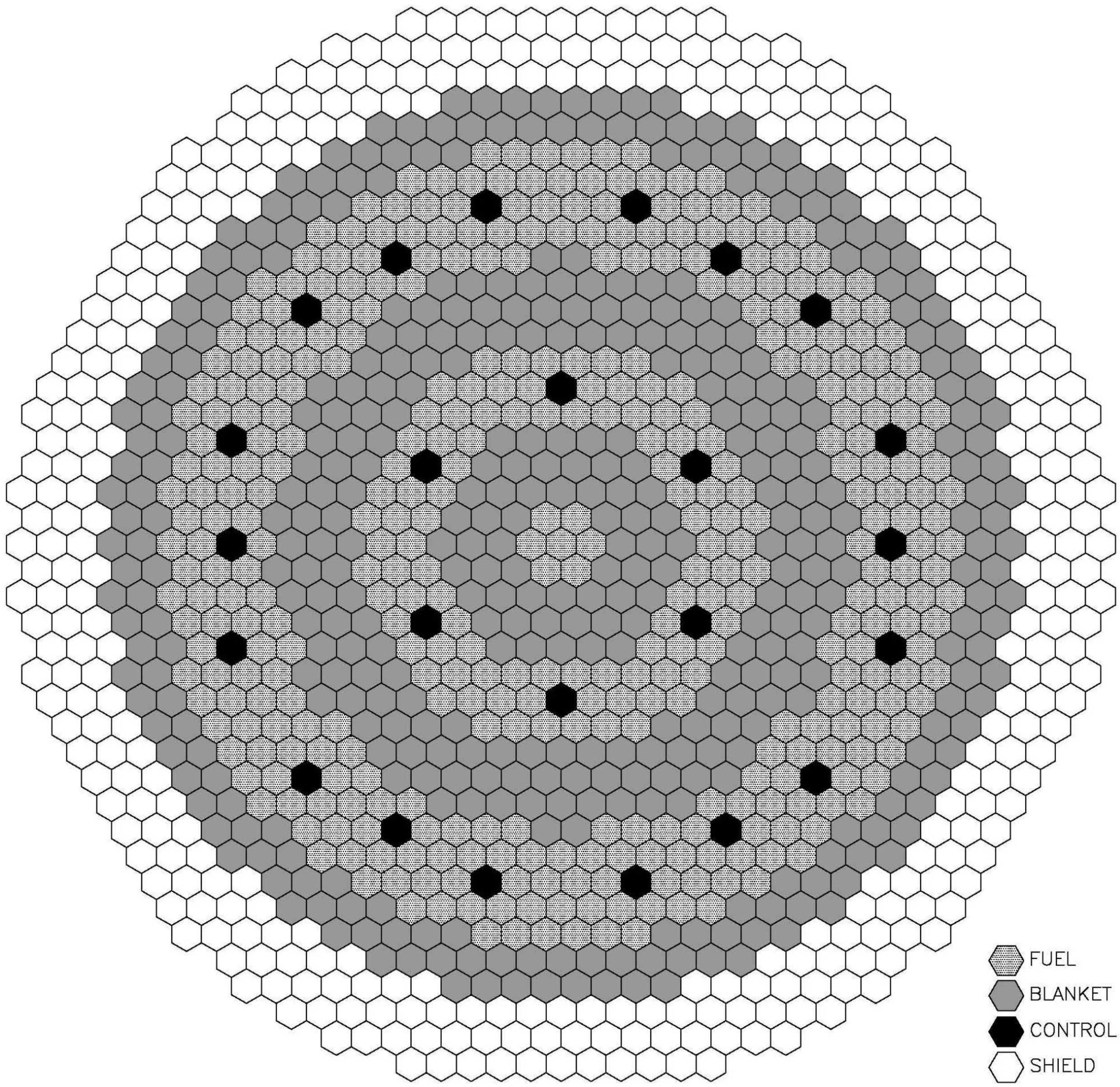

Figure 47 Proposed core design

There are 397 fuel assemblies, 228 internal blanket assemblies, 198 radial blanket assemblies, 24 control assemblies, and 342 shield assemblies. This is comparable to the scale-up of the CRBRP core but with 22 more fuel assemblies and 198 fewer radial blanket assemblies. Fewer radial blanket assemblies make sense from two points of view. First, the CRBRP core design had two rows of outer radial blanket assemblies as does this design. Second, the inner row of CRBRP radial blanket was intended to run for two equilibrium cycles and the outer row for 2 ½ equilibrium cycles. The proposed radial blanket is intended to run for just one equilibrium cycle. It is reasonable therefore to expect that the proposed design would have a relatively thinner radial blanket. The assembly count comparison between the CRBRP scale-up, Superphénix, the ANL core and the one proposed in this paper is shown in the table below.

	CRBRP scale-up	Superphénix	ANL (3500 MWth)	Proposed design
Power level, MWth	3000	3000	3500	3000
Fuel assemblies	375	364	396	397
Inner blanket assemblies	228	0	163	228
Outer blanket assemblies	396	233	90	198
Control assemblies	30	21	30	24
Height of core, in.	36	39.37	40	48
Fuel pin diam., in.	0.23	0.33	0.285	0.33
Clad thickness, in.	0.15	0.22	0.22	0.15
Fuel pins per bundle	217	271	271	271
Blanket pins per bundle	61	91		127
Blanket pin diam., in.	0.51	0.55		0.47
Refueling interval	Biannual w/ mid cycle adj.	1/5 of core, annually	1/3 of core annually	Once every 10 years

Table 8 Core assembly comparison of various designs

In the proposed design, all of the control assemblies are surrounded on six sides by fuel assemblies, so as to enhance their worth. Since it is planned to run the core for ten years between refuelings, the radial blanket is two layers thick in contrast to the ANL approach of having a single layer of radial blanket. The neutron flux profile will probably tend to move outward as a result of the thicker blanket however some counteracting inward motion of the profile would also be expected because of the heavy concentration of blanket assemblies toward the middle of the core. As shown, this core is about 18.5 feet in diameter – an additional three rows of shield assemblies would bring the diameter to just above 22 feet, somewhat above the 21 ft. goal. The inner blanket layers are purposefully thick in the interest of reducing reactivity swing with burnup, particularly early in the life of the core. There are somewhat more inner blanket assemblies than the ANL design (which had 16% greater thermal power) and almost the same number of fuel and radial blanket assemblies. There are six fewer control assemblies which could readily be added to the outer fuel annulus at the expense of either six inner blanket assemblies or six fuel assemblies. As many as 18 fuel assemblies can be traded for inner blanket assemblies and still remain within the CRBRP scale up parameters if it proves necessary to do so in order to prevent exceeding inner blanket pin power limits.

There is little doubt that if analysis were to be pursued on this core design, it would change. As was stated earlier, four years of design activity occurred on the CRBRP heterogeneous core before the design was settled. Had the project not been terminated, it is likely that further changes would have been made. The purpose of advancing the design shown is to suggest a starting point and to propose some ideas that may have merit towards the objective of developing a core design that can be operated for ten years between refueling outages.

It is necessary to identify the core assembly length for the following subsection as well as determining its impact on the reactor vessel, the EVST, the refueling cell and the other components of the refueling system. The CRBRP core assemblies were designed to be 14 ft. in length, which included 3 ft. for inlet and outlet hardware. (The split between 8″ for the upper axial blanket and 14″ for the lower axial blanket is almost certain to change. BN-600 has lower and upper axial blanket thicknesses of 13.8″ and 11.8″ respectively, which appears to be a result of optimization. Earlier designs had different splits. The choice must await analyses.) The inlet and outlet hardware will be treated in the following paragraph. As for the remaining 11 ft, 4 ft. of fission gas plenum has been removed, 12″ of fueled region has been added, and 6″ of upper axial blanket has been removed. With the fission gas plenum removed, it will probably be necessary to provide some shielding above the upper axial blanket to protect the UIS from excessive fluence. Without performing analyses, it is not possible to determine how much shielding will be required, so a placeholder of 12″ will be assumed. This 12″ figure probably isn't too far off the mark since the lower shield is 20″ in length – a value which has been retained – and the effectiveness of any upper shielding would get an assist from the control assemblies. This leaves the core assembly length at 9 ½ ft. plus inlet and outlet hardware.

The outlet hardware consists of a load pad for the core restraint system and a fixture for a grapple. This occupies about 8 in. and there is little opportunity for improvements. The inlet hardware, however, appears to offer opportunities for shortening. There is a discrimination fixture at the bottom of the assembly and windows for admitting coolant. Other than that, it is little more than a 2 ½ ft. hollow tube. The assembly orificing is accomplished in the shield block above the inlet nozzle. There does not appear to be anything in the PSAR or the available open literature that accounts for this long inlet nozzle. It should be reduced by at least one foot, which would leave the overall assembly length at 11 ½ ft. It is likely that another half foot could be removed from this inlet nozzle without compromising function.

There is a piece of information that is missing that is required by the core designer, viz. what capacity factor should be assumed. When CRBRP was designed, nuclear plants in the US were lucky if they were able to achieve a 75% capacity factor. It is possible that there was no conscious decision that led to the 75% assumption for the CRBRP design, and that is what was typically used by NSSS vendors.[93] However, things have changed since then. Nuclear plants in the US routinely achieve 90% capacity factors year after year including provisions for their refueling outages. This is a reflection of their value as base loaded plants with high capital costs

93The design capacity factor for the first year of operation was assumed to be 35%, 55% in the second year and 75% thereafter.

and low fuel cycle costs as well as intelligent capitalization on experience by their utility industry owners. The same capital argument applies to LMFBRs, and little good would be served by advertising them as being designed for a lower capacity factor than that which has been routinely achieved by LWRs. They must be designed assuming they will be run at nearly full power between refuelings even though, because of load following, this may turn out not to be the case. Since the refueling is expected to require up to six months, the plant should be designed for 95% capacity factor between refuelings. This would work out to be slightly better than 90% overall capacity factor assuming six month shutdowns for refueling.

The table below compares the heavy metal loading of the fuel, axial blankets, inner blanket and radial blanket between CRBRP and the selected concept.

	CRBRP	Proposed design	Proposed design, modified definitions
Fuel	5.189	50.570	50.570
Upper axial blanket	2.112	8.820	14.030
Lower axial blanket	2.112	15.440	24.560
Inner blanket	8.270	45.600	31.270
Radial blanket	12.707	39.600	39.600
Total	30.390	160.030	160.030

Table 9 Heavy metal loadings, CRBRP vs. proposed design (MTU)

The CRBRP figures as well as the first column for the proposed design include the upper and lower axial blanket extensions of the inner blanket into the inner blanket numbers. If those extensions are incorporated into the numbers for the axial blankets for the proposed design, the result is the final column, which gives a better picture of the actual heavy metal content of the axial blankets. The radial blanket includes the extensions in all cases. The proposed design has just over five times the heavy metal loading of CRBRP with proportionately less in the radial blanket and upper axial blanket and proportionately more in the fuel.

It also should be noted that the JSFR-1500 design, which is dealt with in this paper when treating the heat transport system, has a proposed fuel pin diameter of 0.43 in. Achievement of the 10 year cycle could alternatively be accomplished with larger fuel and internal blanket pin diameters. Possibly, some reduction of the number of core assemblies could be effected. A larger pin diameter would be likely to lead to a larger core diameter and a larger reactor vessel diameter.

D Thermal-hydraulic design

The main problem with heterogeneous designs has already been alluded to – viz. large swings in power generated by the inner blanket assemblies during operation. The CRBRP blanket assemblies had just 61 pins per bundle. At beginning of cycle 3 the fuel assemblies were predicted to produce 84.24% of full core power while the inner blankets produced 7.72%, entirely from fission of U^{238}. At the end of cycle four, the split was predicted to be 72.02% for the fuel and 17.2% for the inner blankets. Once the plutonium builds up in these pins to the neighborhood of 6%, their individual pin power was predicted to become somewhat greater than that of the fuel assemblies, and it would become necessary to remove them from the core to prevent them from overheating.[94] One could ameliorate this problem by using blanket assemblies with 91, 127, or even 169 pins at a small penalty to breeding ratio and a somewhat greater cost for blanket assembly fabrication which would delay the onset of the problem.

But then another problem emerges. Both the fuel and the blanket assemblies are orificed so as to optimize coolant use by the entire core. On CRBRP, the flow had to be optimized for the inner blanket assemblies assuming the plutonium concentration that would exist after they had resided in the core for two years. The blanket assemblies were overcooled at the beginning of their cycle and a little bit under-cooled at the end of the cycle. If one's objective were to design a core capable of operating for ten years between refueling, optimum flow for an internal blanket assembly at beginning of life would be woefully inadequate for the same assembly after it had resided in the high flux region of the core for ten years and had built up a substantial plutonium concentration. Failure to achieve optimal flow in all the core assemblies means that more flow must be provided to assure adequate flow in the assemblies that are producing the most heat. Optimum flow for the blankets at the end of cycle would be excessive at the beginning of the cycle and would lead to excessive thermal striping of the upper internals structure. Non-optimum flow also means that more flow must be supplied to the core increasing pressure drop. For the case of CRBRP, the flow split to the fuel, inner blanket and radial blanket assemblies was set at 65%, 17%, and 12% respectively. This makes reasonable sense when the power split is 72%, 17% and 11% respectively at the end of an equilibrium cycle. But at the beginning of the cycle, the power split is 84%, 7% and 9%. It is easy to see that for this case, the inner blankets were overcooled by about 140% at the beginning of the cycle.

Overcooling the inner blankets means the fuel assemblies must run hotter. Using the flow and power splits above and assuming a 265°F mixed mean temperature rise across the reactor, if the assemblies receiving 65% of the flow are generating 84% of the power, the average temperature rise across these assemblies must be 342°F while the temperature rise across the inner blankets that are receiving 17% of the flow but generating just 7% of the power is 109°F. For the case of CRBRP, 6% of the flow was provided for control & shield assemblies, reactor vessel cooling and leakage. If the power to flow ratio of the remaining 94% could be maintained equal,[95] the temperature rise would be 282°F, 70°F lower than the 342°F cited earlier. Achieving greater balance in the temperatures between the fuel and the inner blanket assemblies could provide a pathway for increasing the mixed mean reactor outlet temperature by 50°F or more. A 50°F increase in reactor mixed mean outlet temperature translates into a thermodynamic efficiency improvement of about 3%, which would improve the plant's electric power output by about 10%.

In the paragraph before last, the subject of thermal striping was brought up. This would be a good time to describe it, since it seems to be a problem unique with liquid metal cooled reactors,

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94At the end of cycle 4, the CRBRP inner blankets were predicted to have a Pu content of 215 Kg out of a total 7662 Kg heavy metal. This would correspond to a Pu content of 2.8% averaged over the 64 in. length of the fueled region of the assembly.

95The control assemblies require 1.26% flow, the shield assemblies 1.34% flow the thermal liner 2% flow with the balance being leakage. All of these flows require revisiting with an objective of reducing them substantially.

and it is aggravated by heterogeneous cores. When the sodium exits the core assemblies, it is mixed in the upper internals structure. There is a period before complete mixing occurs when the regions of the upper internals structure that are closest to the core are exposed to widely varying temperatures attendant with turbulent flow and the mixing from the coolest and the hottest assemblies. These temperature swings can occur with a short period – on the order of a second. Temperature variations of 200°F or more produce internal stress in the structures and expose them to fatigue and potentially, ultimate failure.

Heterogeneous core designs have the obvious appeal of effectively capitalizing on the most important attribute of the LMFBR – its breeding capability – inside the high flux region of the core itself, by making relatively inexpensive blanket assemblies contribute in a meaningful way to the job of producing power along with the much more expensive plutonium-bearing fuel assemblies. Except for the change-out of the six assemblies after odd years of operation, the CRBRP core came reasonably close to the objective being advocated, i.e. whole core refueling on infrequent intervals. With a higher fuel pin diameter and a greater number of blanket pins per bundle, the heterogeneous core design offers a promising possible approach to achieving the ten years between refueling that is advocated, except for the thermal-hydraulic problem caused by the heat production growth in the internal blankets.

There is an obvious solution that both optimizes flow and prevents thermal striping. The fixed orificing could be abandoned in favor of variable flow control devices at the inlet (or outlet) of each core assembly that is controlled by the assembly outlet temperature. It may turn out to be possible to remove the orifices from many of the fuel assemblies while providing less restrictive orifices for the remainder, orifice the shield and control assemblies, and provide variable flow only to the blanket assemblies. As more flow is directed to the inner blanket assemblies, the flow to the fuel assemblies (which would be producing less power) would decline while total core flow remains approximately constant.

Variable flow control to core assemblies was actually provided on the Hallam reactor plant by mechanically varying the core assembly orifices remotely through a flexible drive shaft.[96] This Hallam feature had problems with galling and binding of the control cable which may have discouraged further development of the concept. Nonetheless, the company that designed the Hallam reactor (North American Aviation) participated in the CRBRP project as Rockwell International. After it became clear that the heterogeneous design would probably be adopted on the CRBRP project, Rockwell was assigned to perform a study of variable flow control concepts which addressed and resolved the problems that occurred at Hallam and offered a totally diverse design approach.[97] One of the proposed concepts, similar to the Hallam approach is shown in the figure below.

96C. J. Baroczy et al., *Development of a Variable Orifice for HNPF Fuel Channels*, NAA-SR-5369 (May 1, 1961)
97Ostermier, B. J., Vitti, J. A.; *Development of Blanket Assembly Flow Control Devices*; ESG-DOE-13269; Rockwell International; 1978

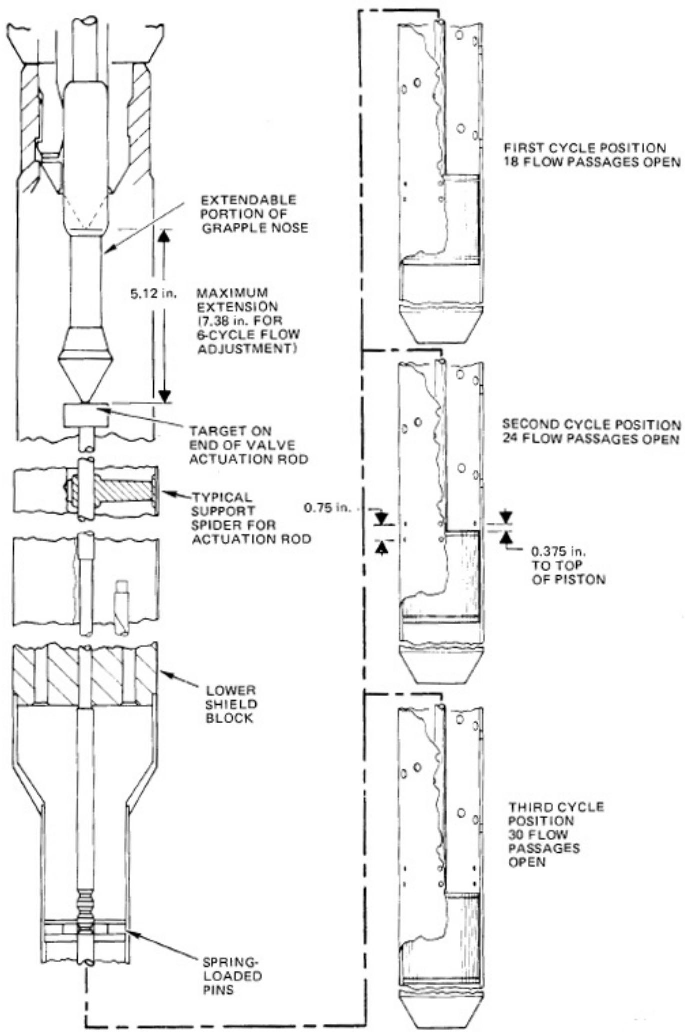


Figure 48 Variable orifice concept

In this concept, a valve actuation rod under compression is inserted down through the center of the blanket assembly to a piston located in the inlet orifice region. In the concept shown, there are three possible positions for the piston. As the piston is moved from its uppermost location downwards, an increasing number of flow holes are exposed increasing flow through the assembly. The valve actuation rod is operated by an extendable portion on the grapple assembly at the end of the in-vessel transfer machine (IVTM). The alternative design proposed in the above cited reference involved rotating each of the inner blanket assemblies to expose greater flow area in the orifice region, which would be provided for by design of the core support structure region. These two manual concepts were developed for the CRBRP project and are not consistent with the overall design approach being advocated. A remotely actuated device would be preferable.

The above reference did not propose any automated flow control devices other than to identify four possible approaches for doing so:

1. A bimetallic device

2. Devices relying on differential expansion between an assembly bundle element and the assembly duct
3. Devices relying on the thermal expansion of a fluid
4. A device using an externally monitored temperature signal to drive an external system to adjust an assembly variable orifice (manual adjustment of the orifice position during operation is also a possible variation on this concept).

Of the above four ideas, no mechanistic implementation was proposed. The fourth (without the manual adjustment) is the most straightforward and is proposed for the purposes of this design approach. Development of such a device is an excellent choice for further R&D that would not necessarily involve operation of expensive facilities and could be performed on a limited budget. The following discussion is mainly offered as guidance.

Considerable simplification of the variable flow concept could be achieved by controlling the flow at the top of the assembly rather than the bottom. This approach was avoided in the aforementioned reference to avoid pressurizing the assembly duct and to allow for debris sweep-out during accident conditions. It is an approach that would be worthy of reconsideration if the core pressure drop could be significantly reduced lowering the pressure on the assembly ducts and if consideration of accidents requiring fuel sweep-out for mitigation were eliminated.

As for the means for actuation of the variable flow control devices, electromagnetic pumps have been developed that are capable of operating in this environment for up to 100 years.[98] Assuming the control valve is electrically actuated, the EM pump development cited above provides a straightforward pathway to development of valve actuation motors. For orifices located at the bottom of the core assemblies, the power supply for these devices and their control would require some conduit probably through the fixed shield. (There is already instrumentation at the bottom of the upper internals structure for core assembly temperature measurement.) Locating the flow control at the top of the assemblies would greatly simplify this approach eliminating the need for running a conduit through the fixed radial shield. It would also be much more accessible if access becomes necessary. As an aside, controlling core assembly flow would capitalize on the ducted fuel in a way that PWRs will never be able to duplicate with un-ducted fuel.

The development of a core assembly flow control device has not been actively pursued to date likely out of concern for possible malfunctions that could shut off flow to a core assembly. Interruption of flow to selected assemblies was the cause of the partial meltdowns at the SRE and Fermi-1 reactors. There is a natural aversion to taking steps in the design that raise the prospect of such an event. This concern would need to be dealt with before such a device could be deployed. If this problem cannot be solved, the only alternative may be to abandon the heterogeneous design however, even in conventional homogeneous designs the radial blanket assemblies closest to the core will eventually build up plutonium and generate substantial amounts of power. Removal of the radial blanket may be a solution provided the fuel pins are sufficiently large in diameter to supply the necessary breeding. Such a design would replace the radial blanket assemblies with additional removable shield assemblies. It should be recognized

98Ota, H. et al; *Development of 160 M3/min Large Capacity Sodium-Immersed Self-Cooled Electromagnetic Pump*; Journal of Nuclear Science & Technology, Vol. 41, No. 4, pp. 511-523, April, 2004.

that even in such a core, the power generation by individual fuel assemblies will not be constant throughout life. The neutron flux will tend to move outward as burnup builds in central assemblies at a rate greater than the peripheral assemblies. Moreover, there is a question whether a homogeneous core, regardless of the fuel pin diameter, can be designed to operate to high burnup levels with whole core refueling intervals of ten years that is advocated herein.

In the interest of reducing pumping power, another objective of the core design should be to reduce the head loss across the reactor as much as possible and significantly less than 50 psig. The larger fuel pin diameters, shorter fuel assemblies, and the elimination of inlet modules (see section 5) will help as will the removal of the orifices from the highest flow fuel assemblies and the provision for individual control of blanket assembly flow. As mentioned above, a larger fueled region of the core also helps. 50 psig should be easily within reach and even lower pressure drops on the order of 20 psig should be achievable. A reduced pressure drop across the core offers the potential to eliminate core assembly hydraulic hold down.

On the subject of hydraulic hold down, it is instructive to examine the details of the pressure drops across plant and core components on the CRBRP. CRBRP was designed for a fuel assembly pressure drop of about 110 psig. The assemblies with the highest flow had a 40 psig pressure drop across the assembly inlet orifices. It is not known how to justify this feature in the CRBRP design. Perhaps at some point, there had been some thought to increasing the fuel assembly pin diameter, (which would have reduced the pin bundle flow area increasing core pressure drop) but if so it was not advertised by the contractor. More likely, the primary pump capability had been selected before the core requirements had been determined. Eliminating those orifices on the highest flow assemblies would immediately reduce the pressure drop across the core to 70 psig. Eliminating the four foot plenum gas space and reducing the thickness of the axial blankets would probably yield another 10-15 psig. The more open lattice of the fuel design being proposed would be good for further reductions.[99]

The CRBRP designers did not have much of an incentive for streamlining the core assembly hardware since the primary pump head had been established early in the project and its head was more than sufficient. In fact, it was more than was needed. The pressure drop across the fuel assembly orifice with the highest flow is ample testimony to that. On CRBRP assemblies, the inlet nozzle, shield, rod bundle inlet, rod bundle outlet, and outlet nozzle together represented 18.4 psig of head loss for the fuel assembly with the highest flow rate. If there had been an incentive to reduce this number, it could undoubtedly have been be lower.

The figure below shows how hydraulic hold down of core assemblies was accomplished on the CRBRP design.

99The pressure drop through a core assembly pin bundle is proportional to the flow velocity squared multiplied by the wetted perimeter of all the pins and duct surfaces in the pin bundle divided by the available flow area. For the concept being proposed, the wire diameter was selected by evaluating the assembly flow area divided by the product of the wetted perimeter and the number of pins per assembly and comparing the result with CRBRP. The wire diameter selected resulted in a 4.9% improvement in this parameter. The flow area per pin is 0.0619 in^2 for the design being proposed in contrast to 0.0334 in^2 for CRBRP. This results in a 46% drop in the flow velocity in the pin bundle which would reduce the pin bundle pressure drop by 71%.

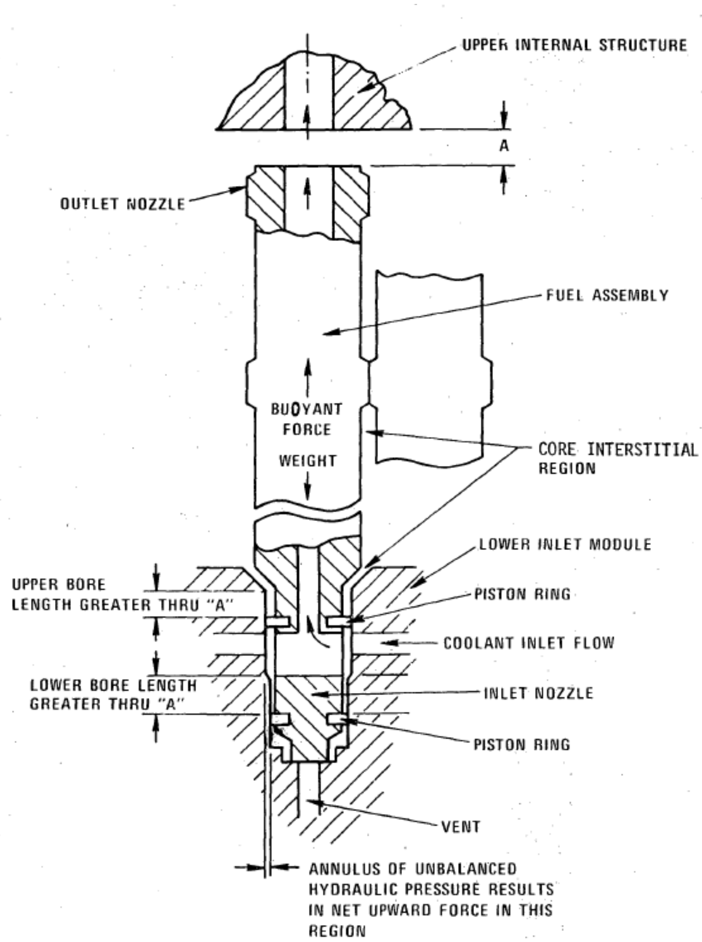


Figure 49 CRBRP hydraulic hold down feature

Coolant flow enters a region just above the fuel assembly inlet nozzle. The region below the inlet nozzle is vented to a low pressure region through ports in the core plate that lead to the interstices between the core assembly ducts. There will be some leakage flow past the piston ring in the fuel assembly inlet nozzle. This through core leakage flow joined with other leakages from the lower inlet modules amounted to about 1.05% of total core flow.[100] This is flow that would otherwise be available for cooling the core. These relatively small leakage flows can turn out to be quite consequential. The entire plant is designed for thermal hydraulic conditions which include significant uncertainties. After the plant has operated and actual conditions are measured, the plant output can be increased to "stretch conditions" which remove many of the thermal hydraulic uncertainties. Any actual leakages are lost for good and cannot be recovered when moving to stretch conditions. 1.05% total leakage flow could correspond to an additional 12 MWe of generated electric power.

With pressure drop across the fuel assemblies controlled to be in the range of 20 psig, it should be possible to eliminate the hydraulic hold down feature from the fuel assemblies. The upward force from a 20 psig pressure differential across the fuel assemblies would be about 390 lb. The

100On the FFTF reactor, this bleed flow was directed to the annulus between the core barrel and the reactor vessel. The FFTF did not have lower inlet modules. This subject is discussed in more detail in the reactor vessel and internals section.

fuel assemblies with the dimensions that have been selected are expected to weigh about 1100 lb. Given approximately 100 lb. of buoyancy from the sodium, the weight of the assembly would be more than sufficient to overcome the hydraulic lifting force. Reducing the pressure drop across the core assemblies further would add margin. Elimination of this hydraulic hold down would be one more simplification to the core design and it should be pursued. It would be desirable if the core pressure drop could be reduced even further was evident from the discussion of the heat transport system. There is precedent for a significantly lower pressure drop across the core than the CRBRP was designed for. The pressure drop across the Monju core was designed to be just 36 psig and the JSFR-1500 design core pressure drop is 43 psig.[101]

Reducing the core pressure drop also enables the realization of another advantage of sodium cooled reactors vs. water cooled reactors as discussed in Section 3. In water reactors, the control rod mechanisms are pressurized necessitating treatment of control rod ejection accidents. On CRBRP, control rod ejection accidents required treatment due to the potentially high hydraulic lifting force on the control assemblies should hydraulic holddown fail. With the lifting force reduced significantly below the weight of the control assemblies, there is no credible mechanism for control rod ejection accidents, and they can be eliminated from the design basis. This is identified as CRM 53.

A lower pressure drop across the core should not result in any flow maldistribution problems. The CRBRP PSAR states (section 4.4.2.6), "At the10% pony motor flow level after shutdown; insignificant flow redistribution occurs between the parallel flow core assemblies. However, for the core natural convection cooling mode, the effect of dynamically approaching low flow with worst case decay heat loads results in a-power-to-flow ratio greater than one. Consequently, core temperatures increase and natural convection phenomena such as inter- and intra-assembly flow redistribution due to different thermal heads and hydraulic characteristics of the core assemblies become important. In general, the core thermal head becomes significant relative to the form and friction loss across the core below 5% full flow." The message here is reasonably clear. As the primary system flow is decreased and natural circulation becomes more important, the flow distribution actually improves. There does not appear to be any incentive to design the core for any more pressure drop than is absolutely necessary to remove the heat produced by the fuel and blanket assemblies and the lower the pressure drop, the better. In fact, in the extreme, the prospect of designing the reactor for natural circulation at power is worthy of consideration and is treated in the next subsection.

The following summarizes the steps that have been (or will be) proposed for reducing the core pressure drop in comparison to CRBRP values:

1. Eliminating the orifices on the highest flow fuel assemblies reduces core pressure drop from 110 psi to 70 psi.
2. Increasing the number of fuel assemblies beyond that required for scale up to the higher thermal power of the proposed design: 5.6% reduction in average fuel assembly flow translates into an 11% reduction in assembly pressure drop.[102]

101IAEA TECDOC-1531, *Fast Reactor Database*, 2006.
102This assumes the flow maldistribution between the fuel assemblies in the proposed design is the same as the flow maldistribution of the CRBRP fuel assemblies.

3. Increasing reactor ΔT (see section 16 under parameters selection): 39% improvement. Items 1, 2, and 3 alone would reduce the pressure drop across the fuel assemblies to 38 psi. Assuming the split between the pressure drop across the pin lattice and assembly hardware remains the same as CRBRP (2/7 across the hardware and 5/7 across the pin lattice) this 38 psi would be divided into 11 psi across the hardware and 27 psi across the pin lattice. The following items would reduce the pressure drop across the pin lattice:
4. Increasing relative flow area in the pin lattice (flow area divided by number of pins) from 0.0334 in^2 to 0.0619 in^2: 46% velocity reduction and 71% pressure drop reduction.
5. Reducing the pin length from 114.4 in. to 72.4 in.: 37% reduction.
6. Greater wetted perimeter of the proposed design per pin: 40% increase.

Items 1, 2, and 3 alone would reduce the pressure drop across the fuel assemblies to 38 psi. Assuming the split between the pressure drop across the pin lattice and assembly hardware remains the same as CRBRP (2/7 across the hardware and 5/7 across the pin lattice) this 38 psi would be divided into 11 psi across the hardware and 27 psi across the pin lattice. The following items would reduce the pressure drop across the pin lattice:

Items 4, 5, and 6 would reduce the pressure drop across the pin lattice from 27 psi to about 8 psi. The result is a pressure drop of 19 psi across the core even without taking any credit for streamlining efforts on assembly hardware. It is apparent from this result that the objective of reducing core pressure drop may have been pressed too far. For example, if the CRBRP wire diameter were retained there would be a small penalty in pressure drop but the breeding would be better. Tradeoffs such as these are made as a part of the design process. The point to be made is there is ample opportunity to make significant reductions in the core pressure drop and there is a big payoff for doing so.

The table below compares the key CRBRP fuel assembly design features with those being proposed

	CRBRP	Proposed design
Number of fuel assemblies	156	397
Pins per assembly	217	271
Total number of pins	33,852	107,587
Pin outside diameter (in.)	0.23	0.33
Wire wrap diameter (in.)	0.056	0.060
Pin spiral wire pitch (in.)	11.9	17.1
Pin triangular pitch (in.)	0.2877	0.4017
Clearance between pins at wires (in.)	0.0017	0.0017
Clad thickness (in.)	0.015	0.015
Pin length (in.)	114.4	72.4
Pellet column length (in.)	64	70
Fission gas plenum length (in.)	48	0
Lower axial blanket length (in.)	14	14
Upper axial blanket length (in.)	14	8
Active fueled core region length (in.)	36	48
Total fueled region (in.)	64	70

Core pellet diameter (in.)	0.1935	0.2935
Shield block total length (in.)	20	20
Load pad OD (in.)	4.745	7.19
Duct across flat ID (in.)	4.335	6.79
Load pad thickness (in.)	0.205	0.2

Table 10   Key fuel assembly design features comparison

Before leaving the subject of thermal hydraulic design, it is the practice of the designer to incorporate margin in the design so as to account for uncertainties. Uncertainties can include instrumentation accuracy, manufacturing tolerances, analysis uncertainties and physical properties and correlation uncertainties. These uncertainties must be combined with attention to the degree of independence of each source of uncertainty from the others using some algebraic formulation. When combined, these uncertainties produce a deviation around the expected condition that can be represented as a distribution curve with some characteristic standard deviation, usually represented as σ. Design values are then obtained by applying some multiple of σ to the nominal conditions. The results are design margins.

These margins appear in all the analyses and generally are beneficial in ensuring that hard limits are not exceeded in the plant. However, they do penalize the plant when it comes to output. Again, using CRBRP as an example, the thermal hydraulic primary hot leg design temperature was set at 995°F. As this temperature was used to design components that would be exposed to hot leg temperatures, there is little doubt that the plant would operate reliably at that temperature. However, the expected temperature in the hot leg was much lower. Even after 30 years of operation with expected fouling of heat exchange components and expected tube plugging, the nominal hot leg temperature was expected to be 960°F, 35°F below thermal hydraulic design value (THDV) conditions.

If it were possible to run the hot leg temperature up to its THDV condition, one would expect to improve thermal efficiency by about 2% which would yield about 7% more generated electric power. However, that is generally not possible because the steam plant is also designed for THDV conditions including the temperature and pressure of the steam inlet to the turbines. The solution to this dilemma is to design the steam plant for nominal conditions at the steam generator steam outlet. It probably would be prudent to provide some margin for error but not to the extent used to arrive at THDV conditions, which are generally predicated on a 3σ level of uncertainty. A similar approach was taken on the CRBRP project. If economic performance in the plant is to be achieved, every opportunity, no matter how small, must be capitalized upon.

E   Natural circulation reactors

Natural circulation LWRs are known to have been designed, built, and operated at power successfully as a part of the Naval Reactors program. It is clear from the preceding subsection that if it were possible to design a LMFBR for natural circulation at power, a large number of problems would be solved. It would not be necessary to orifice any of the core assemblies as each would regulate its own flow. There would be no need to provide variable flow control to any core assemblies or raise any of the issues associated therewith. The ducts could be removed from all core assemblies, reducing parasitic neutron absorption, further reducing core sodium head loss, and improving breeding. Core assembly hold-down would become a non-issue. The issues of the location of the primary system pump and its design would be eliminated since there would be no pump. Thermal striping in the upper internals structure would no longer be of any concern. The primary system would be simplified. The old bogeyman of the loss of flow ATWS would be eliminated once and for all.

Elimination of the core assembly ducts is particularly appealing. Part of the core design process for LMFBRs involves determination of the ΔT across the ducts since this ΔT is responsible for duct bowing. Duct bowing is dealt with in core design with the location of the load pads on the ducts and the core restraint system. It was this duct bowing, combined with the use of a metal form fuel that was directly responsible for the meltdown at EBR-I. Duct bowing also complicates refueling since it inevitably leads to creep opposing the ΔT-induced bowing. Adding to the complexity is radiation induced swelling of the ducts. Elimination of the ducts would also probably result in the elimination of the wire wrap in favor of grids as are used in PWRs. The grids would be designed both to create the spacing of the pins and provide for assembly to assembly contact. The load pads on the ducts would be eliminated in favor of more modest grid contact plates that would permit either greater heavy metal density in the core, lower core pressure drop, or some combination of the two. Another benefit of removal of the ducts would be to greatly reduce any issues associated with assembly blockage. If there had been no ducts on Fermi-1, it is doubtful there would have been any assembly overheating.

Evaluation of natural circulation capability is a fairly straightforward process. The density of sodium at the selected primary cold and hot leg parameters (see section 7) of 693°F and 1017°F is 54.10 and 51.33 lb/ft^3 respectively. Assuming the thermal centers of the core and the IHXs are 30 ft apart in elevation,[103] the natural circulation driving head is about 0.58 psi. From the previous subsection, the core pressure drop at 100% flow is 19 psi and assuming the remainder of the primary system circuit pressure drop is about 11 psi yields a total primary system pressure drop at full forced circulation flow of 30 psi. Given that the pressure drop across any fluid system varies as the square of the flow[104], the available natural circulation driving head would yield a flow of about 13.9% of full system flow with forced convection for these temperatures.

While this result provides confidence that natural circulation is more than adequate to remove decay heat, it is difficult to be optimistic in the face of this result if one is contemplating the use of natural circulation for heat removal at power. For the case of the naturally circulating Naval reactors, not much is available in the open literature since the designs are classified, but since all current U.S. Naval reactors are water cooled, it seems reasonable to assume that the naturally circulating Naval reactors are probably assisted by allowing some boiling to occur in the core which would substantially raise the driving head, an option not available in a sodium cooled reactor.

103This 30 ft. elevation differential between the core and the IHXs would require that the IHXs be elevated above their location described in this paper.

104This is actually a first order approximation since it assumes the flow regime is turbulent throughout the range but is a reasonable approximation for the purposes of this discussion.

Perhaps the situation is not quite so grim. It is obvious that the greatest benefit can be achieved by increasing the ΔT across the reactor. If it were possible to double the ΔT across the reactor, the heat transferred out of the reactor for a given flow would be doubled and, to a first order of approximation, the thermal driving head would also be doubled. The first question needing an answer is how low can the cold leg temperature be decreased? The answer probably is associated with the minimum trapping temperature. Assuming that the trapping temperature cannot be reduced much below 250°F, the lower limit on the IHTS cold leg temperature is probably 350°F which would suggest that the lowest possible primary system cold leg temperature would be around 400°F. As far as the hot leg is concerned, it would probably be counterproductive to assume anything much above 1050°F because of materials issues that are raised at these high temperatures, so 1050°F will be assumed. These temperatures correspond to a 650°F ΔT, which is two times greater than the reference values assumed above. Taking into account the increased thermal driving head associated with this ΔT and using the same pressure drop figures assumed above, the maximum allowable power from this reactor would become 13.9% X $2.0^{1.5}$ ≈ 40%. At this juncture, things begin to look interesting. With the elimination of the primary system pump, simplification of the PHTS design, and the elimination of core assembly ducts[105], a reduction of system pressure drop by another 50% should clearly be achievable for another factor of $2^{1/2}$ which brings the 40% figure up to 56%. With these assumptions, increasing the core size by about 75% should just about do the trick.

How would such a PHTS be configured and what would be the implications on the design of the balance of the heat transport system? The reactor vessel of course would necessarily be larger. The design approach assumes a core diameter of 22 ft. and a reactor vessel diameter of 28 ft. The core diameter would need to be increased by $1.78^{1/2}$ or 33% to 29 ft and the vessel diameter to about 36 ft. There would be just over 350 metric tonnes of heavy metal (MTHM) in the core. Because of the greater core size, refueling would be required less frequently – on the order of once every 18 years. Assuming the fuel is a ceramic with a sizable Doppler coefficient, reactor control would be accomplished by a combination of control rod motion and steam plant demand. There would probably be four elevated IHXs surrounding and in close proximity to the reactor vessel with primary system flow on the lower pressure drop shell side and intermediate flow on the tube side. Return primary flow from the IHXs might be directed back into the reactor vessel at an elevated location in order to cool the region between the core barrel and the vessel wall. An overflow vessel would be incorporated into the design with an EM pump returning overflow sodium to the reactor vessel. There would be a connection to a cold trapping system, probably as a part of the overflow system, with an EM pump returning sodium back to the primary system.

105For the design concept being considered, the across the flats dimension of the fuel assemblies at the load pad is 7.19 in and the across the flat dimension at the inner duct wall is 6.79 in. For this design, removal of the ducts would make 10.8% additional flow area available. This would probably be accomplished by adopting a somewhat larger wire increasing the pin to pin pitch slightly. Since pin bundle pressure drop is proportional to the product of the wetted perimeter and the flow velocity squared divided by the flow area and the flow velocity is inversely proportional to the flow area, pin bundle pressure drop is inversely proportional to the cube of flow area. In addition, removal of the ducts reduces the wetted perimeter by 14.2% which proportionately reduces pressure drop. The combination of these two effects would result in a 39% reduction of pin bundle pressure drop. A greater reduction would be expected from assembly hardware. On CRBRP, the high flow assemblies were calculated to experience 8.7 psi across the inlet nozzle, 6.4 psi across the shield, 1.1 psi across the rod bundle inlet, 0.6 psi across the rod bundle outlet, and 1.6 psi across the outlet nozzle. All of these pressure losses would be expected to either vanish or decrease significantly with ductless fuel. The resulting full flow pressure drop across the core with this ductless fuel would therefore wind up somewhere in the vicinity of 5 psi.

(There would be no incentive for either of these auxiliary functions to be naturally circulating.) A primary reactor auxiliary cooling system would be incorporated (see section 8) with separate tube bundles built into the IHXs and probably NaK on the tube side. A cover gas system would be provided with its associated treatment system circulating cover gas from and to the reactor vessel. IHTS flow rate would be matched to primary system flow with forced convection variable speed pumps, probably electromagnetic. There would be a greater number of control rods because of the larger core, probably in proportion to the core size.

Design of the steam generating system with 350°F IHTS cold leg temperature is not straightforward. It would not be possible to design for 2400 psig steam conditions. To understand why consider the case where feed water inlet temperature is 300°F. Saturation temperature for 2400 psig steam is 663°F. Assume $\mathcal{W}_{Na}$ X $C_{PNa} = \mathcal{W}_{H2O}$ X C_{PH20} where C_P is the specific heat and $\mathcal{W}$ is the weight rate of flow in the steam generator of sodium and water respectively. Thus at the onset of boiling in the steam generator, the sodium side temperature would be 713°F. 55% of the intermediate sodium energy would be consumed in adding approximately 325 BTU/lb of energy to the water side. Since an additional 383 BTU/lb would be required to convert the water to steam, the conversion to 100% quality steam would be incomplete at the steam generator outlet. Neither lowering nor raising the feed water inlet temperature or increasing the steam generator heat transfer area can help this problem.

There are two possible solutions to this dilemma. One would be to accept a much lower steam pressure on the order of 350-400 psig with a significant amount of superheat. Associated with this idea would be about a 6% loss in thermodynamic efficiency between 400 psig and 2400 psig steam at 950°F which represents a loss in plant electric power output of about 180 MWe. Another more aggressive solution would be to have two separate steam plants. In this scheme, IHTS sodium would pass through two separate steam generators. The first would lower IHTS temperature from 1000°F to 675°F and the second would lower IHTS temperature to cold leg temperature, 350°F. The hot plant would operate at a steam pressure of 2400 psig while the cooler plant would operate at 300-350 psig. It would be possible to design such a plant with the same number (four) of steam generators if two IHTS loops were combined to flow through each of the hot steam generators and then through each of the cooler steam generators. This scheme would recover about half of the lost thermodynamic efficiency.

However, there is yet another problem. Primary system flow would not vary linearly with core power level but, to a first approximation, it would increase with the square root of the power level. If one takes the power to flow ratio to be unity at 100% power, at 50% power, the power to flow ratio would be about 0.7. At this lower power to flow ratio, the ΔT across the reactor would decrease from 650°F to 460°F. Perhaps some control of the primary system inlet temperature could be achieved by regulating IHTS flow and the steam plant(s) but at a minimum, steam conditions would certainly change. It is unlikely that such a plant would be useful in a load following mode. More likely it would ascend to full power very slowly and remain there.

Startup also presents some interesting phenomena. If one were to assume a shutdown reactor at 400°F cold leg temperature with the decay heat level is 0.5%, flow rate would be about 7% and the ΔT across the reactor would be about 50°F. Presumably removal of this decay heat would be accomplished by the decay heat removal system. Prior to reactor startup heat removal would be shifted to the steam plant and the decay heat removal system would be removed from service. Once the reactor is taken critical, power level would be increased raising the reactor ΔT. At 5% power reactor ΔT would be about 145°F and at 20% power, reactor ΔT would be about 290°F. If there were two separate steam plants, at this temperature the second steam plant would be started. It may be easier to control this reactor by maintaining the cold leg temperature constant although maintaining a constant hot leg temperature, though more difficult, is not impossible.

The above suggests that it should be feasible to design a reasonably practical naturally circulating LMFBR if one were motivated to do so for any of the reasons suggested above, but it would require a fair amount of thought. Extensive testing of natural circulating phenomena in the core and the remainder of the PHTS would inevitably be part of the price to be paid for any such endeavor. Ductless fuel, while being highly beneficial, would be a new participant in the technology that would require much study and irradiation testing. Would bowing still be an issue? How should core restraint be approached? Considerable design and testing would be needed to optimize a system that minimizes head loss. Design of a steam plant that could operate under significantly changing steam conditions could be a challenge. This idea appears to have sufficient merit to warrant further development, perhaps in a future paper or better with a carefully thought out test. For the current purposes and in consideration of the many potential difficulties associated with such a reactor including the almost total lack of experience with even related concepts, it is not considered necessary for consideration for an initial large-scale LMFBR design. However, the design approach being contemplated with its emphasis on reducing core pressure drop may very well provide a pathway to such a reactor.

F Core design summary

Given an objective of developing a core design approach that potentially could operate for ten years at a high capacity factor between refueling, the following summarizes the main points of this section.

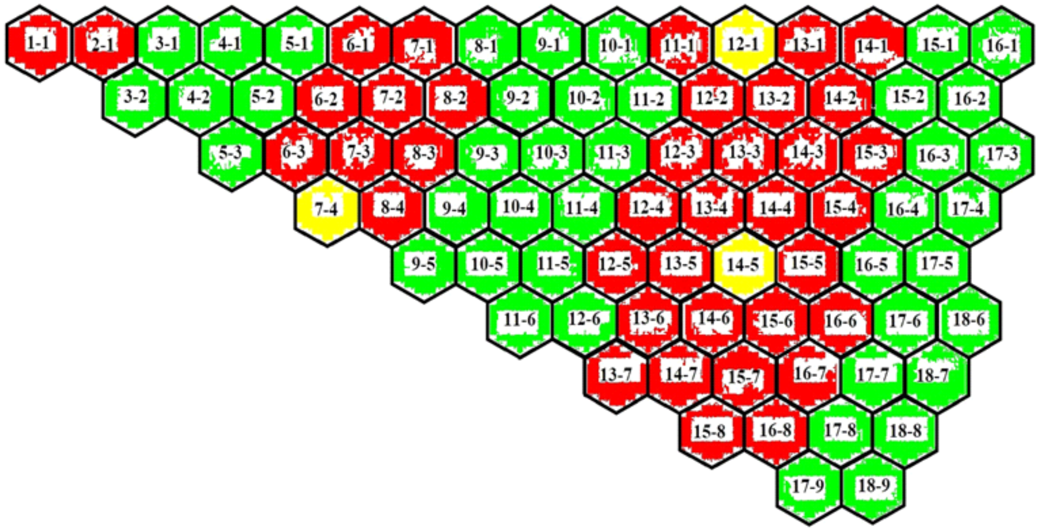
- The boron in the control assemblies should be fully enriched. It is much cheaper to enrich boron than add control assemblies.
- The thickness of the upper axial blanket should be reduced by about six inches. This is in response to the presence of blanket assemblies in the active section of the core which increases the heavy metal fraction of the axial blankets and the presence of the control assemblies, which decrease the effectiveness of the upper axial blanket.
- The fuel assemblies have 271 pins per bundle with 0.33 in. diameter pins. The blanket assemblies are in 127 pin bundles.
- The heterogeneous core design is adopted in the interest of minimizing reactivity swing with burnup. For the same reason, the layers of the “parfait” probably need to be relatively thick.
- The active core region is four feet high.
- The radial blanket thickness should generally be limited to no more than two rows in the interest of fuel cycle economy and minimizing reactor vessel diameter.

- Use a high density material such as Inconel for the inner row of radial shield/reflector assemblies. Use boron assemblies for the outer two rows. These steps are to be taken in the interest of minimizing reactor vessel diameter.
- In view of actual experience with operating LWRs, a 90% capacity factor should be assumed for the life of the plant.
- Adopt remote flow control for the blanket assemblies. The flow control device for the blanket assemblies should be on the assembly outlet rather than the inlet in the interest of simplification and improved access.
- Significantly reduce core assembly pressure drop to no greater than 20 psig.
- Eliminate orifices for the highest flow fuel assemblies.
- Eliminate core assembly hydraulic hold down in the interest of reducing core assembly length, simplifying the design, and reducing leakage flow.
- Reduce bypass flow to the extent possible in the interest of improving flow to heat producing assemblies.
- Design the components of the steam plant for nominal conditions rather than THDV conditions.

G Head Port Layout

The Reactor Vessel design described in Section 6 is one of the key cost reduction measures of the "design approach" and is enabled by a reactor core that requires refueling infrequently. Refueling that occurs only once in ten years permits longer refueling outages, which in turn allows for elimination of much of the head shielding and elimination of the need for in-vessel handling of core assemblies. The purpose of this subsection is to establish the feasibility of this refueling approach which was adopted by SRE and Hallam but has not been attempted for an LMFBR core. For the case of SRE and Hallam, there were far fewer fuel assemblies and they were not adjacent (there were moderator cans surrounding each fuel assembly), which facilitated access by a single rotating plug. The greater number and adjacency of the fuel and blanket assemblies complicates the problem for an LMFBR.

To begin, the figure below is a section of the core and blanket for the "design approach".



.Figure 50 1/12 core section

Since the core has 12-fold symmetry, a 1/12 section totally describes the entire core. The fuel assemblies are shown in red, the radial blankets in green and the control assemblies in yellow. It is a convention to number the core rows from the center outward, the middle assembly being the sole occupant of row 1, the adjacent 6 assemblies make up row 2, the next 12 assemblies compose row 3, and so on. Starting at the center, the across flats assemblies are position 1, their next neighbor position 2, and so on as shown in the figure above. There are 83 unique positions, each having its own distance from the center of the core. The same section from the figure above is represented below with the distances from the center assembly shown below the assembly position number. The distances have been normalized to the assembly dimensions with the unitary being the across flats dimension, which for the "design approach" is 7.19 in.

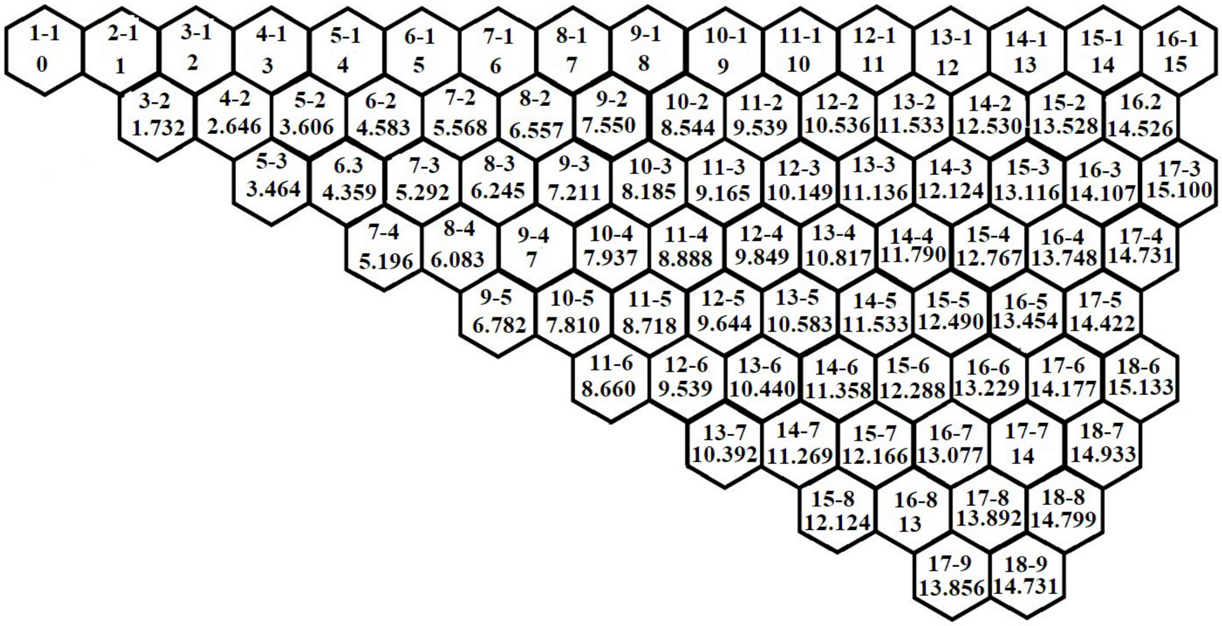


Figure 51 Normalized distances for assembly positions from the core center

The reason these distances from the center of the core is important results from there being a single rotating plug requiring multiple head ports to access the assemblies. It turns out that of the 83 unique positions, there are seven pairs that are equidistant from the center, 17-4 and 18-9 at 14.731 units, 15-1 and 17-7 at 14 units, 14-1 and 16-8 at 13 units, 14-3 and 15-8 at 12.124 units, 13-2 and 14-5 at 11.533 units, 11-2 and 12-6 at 9.539 units, and 8-1 and 9-4 at 7 units. Thus, the 83 unique positions have 76 unique distances from the center of the core, requiring at most 76 head ports.

It is instructive to consider the situation at the center of the core where the layout of the head ports is the most challenging. The figure below shows the inner five rows of the core with 9 in. head ports superimposed on the drawing.

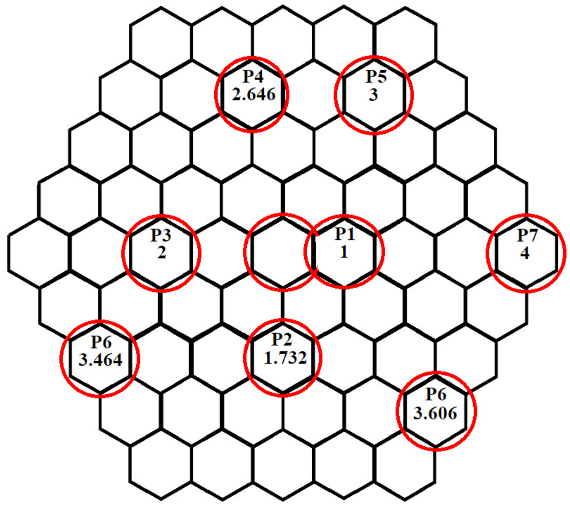


Figure 52 Inner Port Locations

The center port is the most challenging. The plug would be figure 8 shaped with the circular plug over row 2. When it comes time to access row one, the plug must be turned around. The plug handling machine must be capable of this maneuver. The closest that two plugs approach is between core positions 1-1 and 3-2. For 9 in. plugs, the web of metal between these two plugs is about 4.45 in. in minimum thickness. This contrasts with the space between plugs 1 and 3 of 5.38 in. The space between 4 and 5 is the same. Note that there is just one P6 plug -- it is shown twice in the above figure and its actual distance from the core center would be midway between the distances shown. It would probably be located near the lower right position in the figure above, and its diameter would be 11.5 in.

As one proceeds out from the center, more space becomes available for these plugs and one could readily arrive at a design for 75 port plugs (one less than the 76 mentioned above since plug 1 serves two locations).. However, the distances from the center are close in many cases, which creates the option for larger diameter plugs covering a group of core positions. The table below shows the results of the grouping that is possible when using 11.5 in. diameter plugs, which allow a diametric tolerance of 2 1/2 in. or 0.347 times across the flats. The equivalent figure is shown below the table. The "color code" relates to the figure. The positions in the figure that are not colored correspond to positions for which there is a unique port. These unique ports would be 9 in. in diameter, while all others would be 11.5 in. in diameter. Plug 19 requires the greatest alignment tolerance at 0.3 times across the flats.

Port No.	Lattice position & distance from center					Total assemblies accessed	Color code

1	1-1	2-1,1				7	gray
2	3-2, 1.732					12	
3	3-1, 2					6	
4	4-2, 2.646					6	
5	4-1, 3					6	
6	5-3, 3.464	5-2, 3.606				18	lt. green
7	5-1, 4					6	
8	6-3, 4.359	6-2, 4.583				24	yellow
9	6-1, 5	7-4, 5.196	7-3, 5.292			24	turquoise
10	7-2, 5.568					12	
11	7-1, 6	8-4, 6.083	8-3, 6.245			30	blue
12	8-2, 6.557	9-5, 6.782				18	maroon
13	8-1, 7	9-4, 7	9-3, 7.211			30	orange
14	9-2, 7.550	10-5, 7.810				24	green
15	10-4, 7.937	9-1, 8	10-3, 8.185			30	violet
16	10-2, 8.544	11-6, 8.660	11-5, 8.718			30	tan
17	11-4, 8.888	10-1, 9	11-3, 9.165			30	lt. blue
18	12-6, 9.539	11-2, 9.539	12-5, 9.644			36	purple
19	12-4, 9.849	11-1, 10	12-3, 10.149			30	dk. green
20	13-7, 10.392	13-6, 10.440	12-2, 10.536	13-5, 10.583		42	hot pink
21	12-1, 11	13-3, 11.163				18	brown
22	13-4, 10.817					12	
23	14-7, 11.269	14-6, 11.358	14-5, 11.533	13-2, 11.533		48	violet
24	14-4, 11.790	13-1, 12				18	lt. gray
25	15-8, 12.124	14-3, 12.124	15-7, 12.166	15-6, 12.288		42	blue
26	15-5, 12.490	14-2, 12.530	15-4, 12.767			36	yellow
27	14-1, 13	16-8, 13	16-7, 13.077	15-3, 13.116	16-6, 13.229	54	lt. blue
28	16-5, 13.454	15-2, 13.528	16-4, 13.748			36	gray
29	17-9, 13.856	17-8, 13.892	15-1, 14	17-7, 14	16-3, 14.107	48	pink
30	17-6, 14.177	17-5, 14.422				25	red
31	16-2, 14.526	17-4, 14.731	18-9, 14.731	18-8, 14.799		48	lt. green
32	18-7, 14.933	16-1, 15	17-3, 15.100	18-6, 15.133		42	lavender

Table 11 Refueling Ports

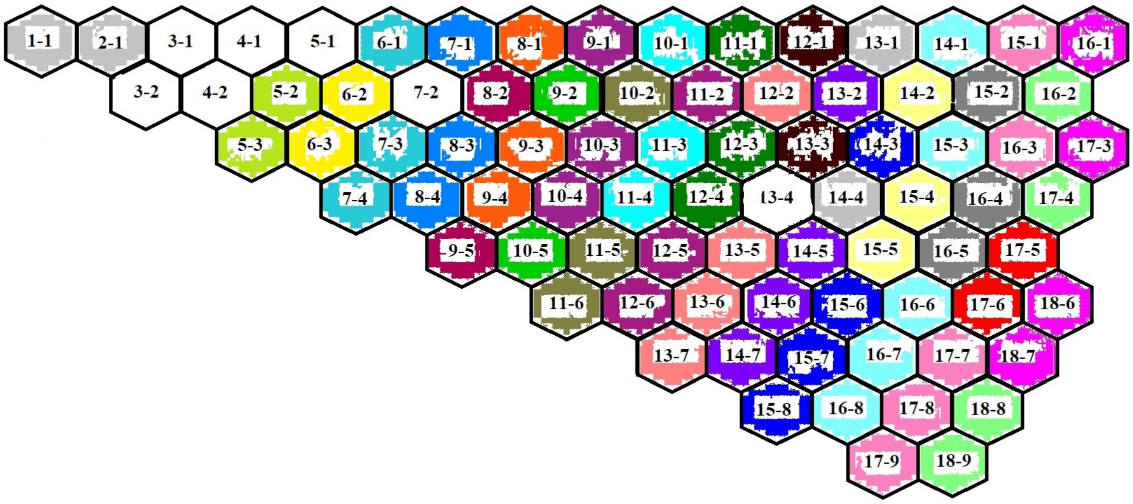


Figure 53 Refueling Port Determination

With 32 ports, all fuel, blanket, and control assemblies can be accessed.. While this may not be the most elegant approach, the alternative is 75 ports or an IVTM, an auxiliary handling machine to handle the IVTM, a reactor vessel that is longer by at least 6-8 ft., and two rotating plugs. The reactor vessel diameter would need to be increased to at least 40 ft. to rotate the UIS out of the way and allow access to the entire core., the diameter of which is about 18' 7". If the shield assemblies are to be accessed, the vessel diameter would be at least 44 ft.

This discussion has been silent on access to the shield assemblies. Three rows of shield assemblies increases the diameter of assemblies to be accessed by about 3' 8", more ports would be required (about 6-8), the rotating plug would need to be 22 ft. in diameter, and less room would be available for an optional DRACS. Reactor vessel size would not be affected. This capability would not come at excessive cost, and should be incorporated into the design.

Appendix 3 Actinide burning

Most of the long lived radioactive components of nuclear waste are the actinides, shown in the list below. With the exception of U236 and U238, all these actinides tend to build up in thermal spectrum reactors but are readily fissionable in a fast spectrum reactor. There are certain aspects of using the LMFBR concept for this purpose that require further treatment. The table below shows the reproduction (neutrons produced per neutron absorbed) for the actinides of interest:[106]

Isotope	η (LWR)	η (LMFBR)
U 235	2.04	1.98
U 236	0.065	0.071
U 238	0.25	0.66
Np 237	0.045	0.92
Pu 238	0.19	2.09
Pu 239	1.86	2.52
Pu 240	0.016	1.64
Pu 241	2.19	2.59
Pu 242	0.037	1.58
Am 241	0.035	1.09
Am 243	0.45	1.34
Cm 242	0.31	2.82
Cm 244	0.2	2.18

Table 12 Reproduction rates for actinides

Since the quantity η is a measure of the number of neutrons produced per absorption of the isotope in question, it is apparent from the table above that it might be feasible to fuel an LMFBR with minor actinides without any contribution from either uranium or plutonium, saving those fuels for the LWRs and LMFBRs that are not committed to actinide burning. However, doing so could possibly require that the actinide burner fuel form be metal.[107]

From the table above, it is reasonably clear that the minor actinides will tend to accumulate in LWRs while in LMFBRs they would be more likely to fission. It is instructive to consider how the isotopes in the table are formed in reactors. In the figure below, which is the portion of the chart of the nuclides of interest for this subject, long lived isotopes are shown in dark blue color while the shorter lived appear as successively lighter. The only significance of the isotope outlined in red (Cm^{246}) is it is in the middle of the table. For example, Am^{244} has a half life of just about 10 hours, so it will not build up in the reactor but it will decay to Cm^{244} which has a half life of about 18 years. The Np^{237} is formed from double neutron radiative capture by U^{235}, first to U^{236} then to U^{237} which decays with a 6.7 day half life to Np^{237}. Another path to Np^{237} is through an (n, 2n) reaction of U^{238}. About 60% of the minor actinides formed in LWRs are in the form of this one isotope.

106Drawn from *Characteristics of a Minor Actinide Fueled Reactor*, FFTF Internationalization Symposium, Rockwell International, May 28, 1991

107It should be pointed out that Table 11 assumed metal fuel was being used in the LMFBR, which would yield a harder spectrum than an oxide fueled system. The reproduction numbers are slightly lower for the softer spectrum associated with an oxide fueled reactor.

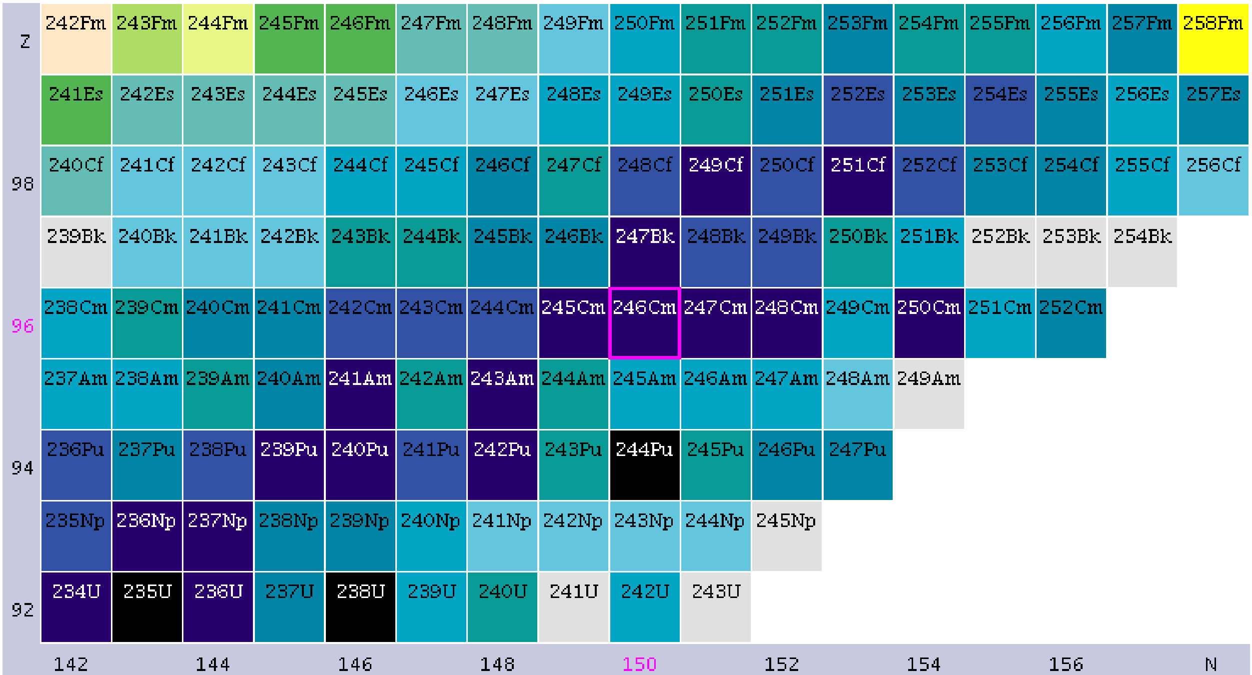


Figure 54 Chart of the nuclides – region of interest

The americium and curium isotopes are formed by successive neutron captures followed by beta decay of Pu243 and Am 244. About 35% of the minor actinides in LWRs are americium and less than 5% are curium. There will be trace amounts of berkelium and californium but not much since the curium isotopes are all long lived up to Cm^{249} so eleven neutron captures above Pu^{239} would be required for berkelium formation – an unlikely prospect.

All three of these minor actinides, neptunium, americium, and curium wind up in the waste stream following PUREX reprocessing, so means would need to be employed to remove them from the waste stream. Electrochemical methods have been developed for their separation and various chemical methods (e.g. TRUEX) have been proposed. Reprocessing spent fuel followed by removal of these three minor actinides results in waste stream radioactivity equal to the uranium from which it was originally derived after about 700 years in contrast to 250,000 years for un-reprocessed spent fuel. Thus the incentive for actinide burning becomes clear.[108]

It is well to have an idea of the magnitudes involved. Each atom percent burnup of LWR fuel contributes about 200 grams of minor actinides per metric tonne of LWR fuel. Another way of looking at this situation is at 33,000 MWD/MTU burnup of LWR fuel, the spent fuel composition is 96% uranium, 0.8% plutonium, 3.2% fission products and 0.05% minor actinides. Most LWRs operate at a burnup somewhat higher than this today. The scale up of minor actinides is somewhat less than linear due to burnup in the reactor but it is nearly linear for the curium isotopes. A fleet of 100 1000 MWe LWR reactors operating their fuel to a burnup of 50,000 MWD/MTU will generate about 1-1 ½ metric tonnes of minor actinides per year. This would be sufficient to fuel 2-3 equivalent sized LMFBRs.

108Further improvement can be obtained by separating the long lived fission products Tc^{99} and I^{129} and converting them through neutron irradiation.

An oxide fueled LMFBR running on this concoction alone would possibly not be capable of achieving criticality and would require some spiking with plutonium, probably on the order of 10-15 %. The neptunium would be, in essence, the fertile isotope, converting to Pu^{238}, which is fissile in a LMFBR. Because these isotopes are all highly radioactive and biologically hazardous, fabrication of fuel would not be a trivial matter, but probably only somewhat more difficult than fabrication of plutonium bearing fuel. Since the difference in η between the fertile and fissile isotopes is nowhere near as large as is the case between U^{238} and Pu^{239}, such reactors will be ineffective breeders, and will not be capable of operating for long periods without refueling. Because of this shortcoming, the attractiveness of committed actinide burners diminishes somewhat. Committed actinide burners would be possible, but they would likely require frequent (annual) refueling with all that implies to the plant design.

It may be preferable to mix relatively smaller amounts of the minor actinides into a uranium plutonium mix. This could also be accomplished by fabricating selective assemblies exclusively out of the minor actinides. Tailoring the reactor core with these substances could possibly offer opportunities that are not present in a reactor fueled with uranium and plutonium alone, particularly if the neptunium, americium, and curium streams are kept separate from one another. Since each of the minor actinides has different neutronics characteristics, if tailored core assemblies were fabricated it doesn't require much of a stretch of imagination to conclude that each may be particularly well suited to a particular region of the core. One could, for example, place neptunium assemblies in the lowest flux regions of the internal blanket in the interest of flux flattening or americium in high flux fuel assembly locations for the same reason. The core designer would be suddenly granted several new degrees of freedom that did not theretofore exist. This might make an interesting topic for a PhD thesis.

There are a few issues that recycling actinides raise that would need to be addressed. First, since the minor actinides have half lives that are relatively short compared with Pu^{239}, they will generate heat in unirradiated assemblies and may require cooling. Second, some of the minor actinides are alpha emitters and will release helium into the fabricated fuel pins. If the pins are vented, this would not be a problem in the reactor but would require some form of treatment during shipping. Third, it is likely that the decay heat from fuel fabricated from the minor actinides would be higher than from uranium/plutonium fuel. Although the majority of the minor actinide isotopes will fission, there will be many that will undergo radiative capture to isotopes with intermediate half lives that contribute to the decay heat of shutdown assemblies. This would be an issue when the core is being cooled by natural circulation and will pose additional constraints on refueling and spent fuel shipment.

Appendix 4 Primary steam generators

In the pool vs. loop discussion it was remarked that the primary steam generator was seen by some as the "holy grail" of LMFBR design. Conventional wisdom suggests that elimination of the intermediate loops would be desirable from at least two points of view. First, the capital cost of the plant would potentially decrease simply by elimination of all the IHTS components and real estate needed to house them – the IHXs, the IHTS pumps, and the associated piping. Second, the steam conditions would be better. If reactor outlet temperature is 1000°F and there is no IHX, steam temperature can be increased by whatever LMTD exists across the IHX – in the case of the approach proposed, 40°F. Raising the temperature of 2500 psig steam from 900°F to 950°F raises its enthalpy by about 35 BTU/lb, which, if returned to the steam plant would translate into about 3% better thermodynamic efficiency. A 3% improvement in thermodynamic efficiency would increase electric power output more than 7%, so a 1200 MWe nominal plant would become a 1300 MWe nominal plant. This is a sizable payoff. The payoff is increased further by the elimination of the power required to drive the intermediate system pumps.

However, there is a cost involved. First, with the departure of the IHXs the steam generators become a part of the primary system and would need to be enclosed within the containment boundary. Bringing the steam generators inside containment raises the specter of a steam or hot feedwater leak inside containment likely compromising the position taken earlier that the containment need not be designed for high pressures. Second, from the section 7 discussion of the heat transport system, primary steam generators may very well translate into four primary loops as opposed to the two in the design approach advocated to this point. The result is the containment boundary grows. Moreover, the isolation valves in the IHTS system now become part of the primary system as does the sodium water reaction products system (SWRPS) and its associated tanks. The SWRPS flare stack introduces another problem since it would need to be contained, which would be no trivial matter as the combustion gases following flaring would be very hot. It is probably reasonable to conclude that if a SWRPS is needed, a primary steam generator is simply not a tractable idea.

There was one sodium cooled reactor design that probably had primary steam generators and that was the Sea Wolf submarine's first reactor plant and its associated prototype. (The prototype was designated "S1G" and the plant on the Sea Wolf "S2G"). The word "probably" is used because not much is known about that plant design except to the people who were involved in its design, construction, and operation since it remains classified. But from what little that has been disclosed, it appears there was an evaporator, steam drum, and superheater all fabricated of type 347 stainless steel. The choice of stainless steel was a mistake and the superheater wound up being out of service for the life of the S2G Sea Wolf plant probably because of caustic stress corrosion, but absent the superheater, the plant was still capable of making 80% of full power. The individual units in the steam generating system probably had double walled tubes and a double tube sheet with NaK between the tubes and the tube sheets. NaK would have had to have been used as the intermediate fluid to allow the steam generators to have been placed into something approximating wet layup, assuming that the sodium side of the steam generators could have been drained somehow. It is unlikely that Sea Wolf had any system for accommodating sodium water reactions. Other than the problems with the superheater, the only significant problem experienced on S2G was a primary sodium leak.

A double walled steam generator design was developed as a part of the CRBRP project by Westinghouse and was tested to well in excess of 15,000 hours at ETEC without incident. The CRBRP project had no intention of using the Westinghouse design as anything other than a backup steam generator in case difficulties developed with the reference hockey stick design. There was no serious thought given to its use as a primary steam generator. The unit had provisions for monitoring the interspaces between the tubes but there was no double tube sheet and there was no NaK between the tubes. Although the unit did not show any evidence of leakage or degradation of any kind from either the water or the sodium sides, 15.000 hours of testing is a far cry from the 60 or more years that should represent a typical nuclear plant lifetime, so the ETEC testing is considered inconclusive, at least as far as application of the design for use as a primary steam generator is concerned. Moreover, there was no post test evaluation of the unit. A much smaller unit (about 3 MWth) furnished by the Japan Atomic Power Company (JAPC) was tested in parallel with the Westinghouse unit at ETEC. After 10,000 hours of testing, it was returned to Japan for post test evaluation. JAPC did not make the results of their post test evaluation public, but presumably they are available.

A double walled unit fabricated of 304 SS was installed in the SRE with mercury as the intermediate fluid. Hallam also had a double walled unit. Neither SRE nor Hallam was intended to be a primary steam generator. The Russian SVBR lead bismuth reactor was designed with duplex tubes. EBR-I had stainless steel double walled tubes with a copper layer between the tubes. The tubes were connected to a header rather than a tube sheet. The design led to a physically large unit in comparison to the amount of steam produced and it was not economically practical for application to large units. The Dounreay Fast Reactor in the U.K. had parallel tubes in a copper heat transfer block. Although effective in preventing sodium water reactions, the design would not scale up economically. EBR-II had 2¼ Cr 1 Mo double walled tubes in a recirculating configuration with eight evaporators, two superheaters of equal sizes and a single a steam drum. The units had double tube sheets with front faced tube to tubesheet welds[109]. Four of the evaporators used mechanically bonded tubes while the other four were metallurgically bonded. Because of the bonding, there was no realistic way to monitor the inner space between tubes for leakage. Unit performance was generally satisfactory[110] but the design features (metallurgical bonding, mechanical bonding, front faced tube to tubesheet welds, double tube sheets, lack of rapid detection of a leak to the inner-space) have generally fallen from favor. The choice of the material of fabrication, equally sized superheaters and evaporators, and the recirculating configuration influenced the CRBRP. Otherwise, there is no known double wall tube experience worldwide that is directly applicable to LMFBRs.

109To accomplish a "front faced weld" the tube is passed through a hole in the tubesheet and a fillet weld is made between the outside of the tube and the tubesheet. The main problem with this procedure is the crevice between the tube and the tubesheet where contaminants such as halides can hide out and later cause stress corrosion cracking of the tubes. "Back faced welds" are made by machining tube stubs onto the inside surface of the tubesheet then butt welding the tubes to these stubs. The resulting welds are more inspectable, have better integrity, and the crevice is eliminated.
110Buschman, H. W.; Longua, K. J.; Penney, W. H.; Operating Experience of the EBR-II Intermediate Heat Exchanger and Steam Generating System; ASME/IEEE Joint Power Generation Conference; September 25-29, 1983.

It is instructive to consider a double walled tube concept along the lines supposed to have been used on S1G/S2G with a double tube sheet and NaK as the intermediate fluid. Presumably, a modern version of such a concept would be fabricated from some kind of high chrome ferritic steel as opposed to the 347 SS used on S1G/S2G. It would probably be a straight tube unit, which would cause it to be longer than its helical coil counterpart. It would be necessary for the NaK to be a fully contained system on each steam generator probably connected to some sort of surge tank covered above by an inert gas. The pressure in the surge tank would probably be maintained somewhere near atmospheric. A system for detection of leaks between the NaK and the surrounding air would need to be provided, which might argue for maintaining the NaK pressure slightly below atmospheric, however air in-leakage into the NaK system could be difficult to detect (remember Superphénix). More likely the system would be maintained slightly above atmospheric to prevent air contamination of the NaK. The cover gas system would probably not require monitoring except for the control of its pressure.

A leak from either the water side or the sodium side would be immediately detectable. If the leak were from the sodium side, the pressure would increase to the IHTS pressure at the site of the leak which would be no less than the minimum IHTS pressure, probably at least 20 psig. If there were a leak from the water side, there would be a strong chemical reaction between the water and the NaK. The NaK surge tank would need to be provided with some sort of relief protection in order to be able to accommodate the reaction products from such an event. Because of the relatively limited quantity of NaK in each of the steam generators, flaring of the generated hydrogen would probably not be required, but the surge tank relief would probably discharge to some kind of reaction products tank. The steam generator would be immediately isolated on both the water and sodium sides once any leak had been detected, regardless whether from the sodium or the water side.

Recovery from any leak would involve a protracted outage and could require replacement of the steam generator, which would be complicated by the fact that it is located inside containment. The plant operator would be obliged to wait the ten days necessary for the Na^{24} activity to decrease low enough to permit entry into the cell. The primary sodium side would need to be drained. The failed tube(s) would need to be located somehow. Plugging tubes, particularly those connected to the inner tube-sheet would not be a simple task, particularly since the tube-sheets would have been designed to be in close proximity to one another to minimize unit length and NaK inventory. Once the damaged tubes had been repaired, it would be necessary to restore the NaK system, which would have become heavily contaminated as a result of the leak, particularly one from the water side. This would require bringing in makeup NaK, NaK purification equipment, and some sort of NaK pumping capability along with connecting piping, valves, required safety provisions, and so on. NaK has a bad reputation of being many times more difficult to handle than sodium and is not the sort of thing that would be welcomed inside containment.[111]

It is from all these considerations that the primary steam generator is not a very appealing idea, at least with the technology that exists at the current time. As has been stated earlier, one of the most important advantages of sodium is that it is benign to the materials that contain it. The

111NaK forms a superoxide, KO_2, which is potentially explosive.

same cannot be said of water. So long as sodium is the only working fluid inside the piping systems within containment, there is reasonable confidence that the interior of the containment will be a relatively uneventful place. Once water is introduced into the containment the picture changes.

If confidence can be developed in materials, welding, and fabrication techniques of the future such that designs can be developed which are certain not to fail in the hostile environment posed by the LMFBR concept, perhaps the primary steam generator idea could be revisited. However, at the current time, the idea isn't worth spending more time than has been expended writing this section, and there are much better ways to achieve economies in the LMFBR concept, many of which are explored in this paper.

Appendix 5 Pool vs. loop controversy

One of the options available in LMFBR design involves whether or not to enclose the major components of the primary system fully within a single tank. Designs that do so are referred to as “pool-type” and those that do not are “loop-type”. One of the earliest LMFBRs, EBR-2 was of the pool design. SEFOR, Fermi, and FFTF were all loop-type reactors. In the 60s and 70s in the U.S., there was the belief that while a pool might be a practical approach for a small experimental reactor, as the plant size is increased, loop-type reactors were considered more economic. However, outside the U.S., the pool-type design was preferred. While the German and Japanese demonstration plants incorporated the loop concept, the French, British, and Russians all moved quickly towards the pool concept. Superphénix, the largest LMFBR built worldwide to date, was a pool. Even the Italians had once started construction on a demonstration plant invoking the pool concept. As of the date the CRBRP Project was cancelled in 1983, the U.S., Japan, and Germany were in the loop camp while the U.K., France, and Russia were pool advocates. Since that time, what little work continued in the U.S., viz. EPRI’s Large Scale Prototype Breeder (LSPB), Rockwell’s Sodium Advanced Fast Reactor, and GE’s Power Reactor Integral Safe Module all incorporated the pool concept although a loop-type version of the LSPB was developed at the conceptual design level. When SNR-300 was cancelled, the Germans cast their lot with France joining a European consortium focused on the pool. LMFBR work undertaken in India, China, and Korea is all focused on the pool concept. Only the Japanese remain interested to the loop concept to the extent that there remains any significant LMFBR program in Japan today.[112]

Figures 49 & 50 are cutaways of the Superphénix reactor pool. Sodium from the pool enters the upper end of the IHX tube bundle and exits at the bottom. To provide a driving force for flow in the IXH, the pool is separated into two regions by a horizontal baffle also know as a “redan”. The primary pumps take suction on the cooler region below the redan. The pressure difference across the redan will be equal to the pressure drop across the IHX tube bundle. The primary pump discharges into piping that enters a plenum below the reactor core in the lower internals structure. Sodium then flows through the fuel assemblies and back to the pool.

112During the 1980s, senior representatives of the Central Research Institute for the Electric Power Industry (CRIEPI) were known to favor the pool concept. The government-sponsored entity Power Reactor & Nuclear Fuels Corporation (PNC) tended to favor the loop.

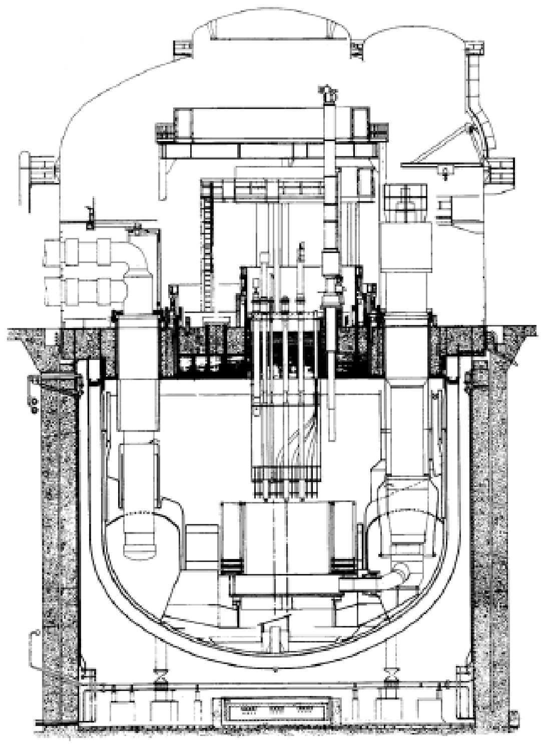


Fig. 55 Superphénix pool & dome

Although most of the primary system is contained within the pool vessel, there are of necessity some systems which must operate outside the pool. Systems involved in refueling result in transfer of primary system sodium outside the pool tank. The primary sodium must be cold trapped to remove impurities, and those associated components along with plugging temperature indicators are outside the pool. In addition, cover gas processing must be performed outside the pool. If it were desired to have decay heat removal directly off the primary system, all those associated components would be outside the pool. Of course, all these components require piping to connect the pool with the components.[113]

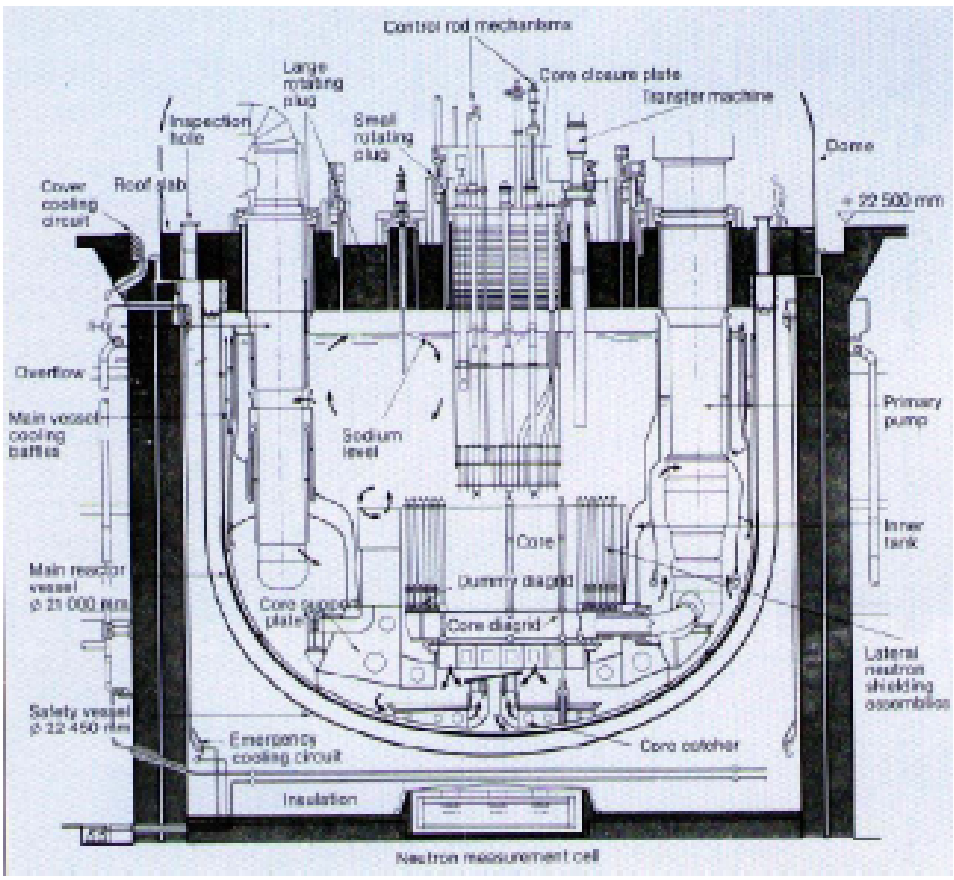
The weight of the core, the shield assemblies, and the internal structure below the core as well as the total primary sodium inventory is carried by the tank wall.[114] Because of the load carried by the tank wall, it is necessary to protect it from the hotter sodium at the core outlet. To accomplish this, an inner tank is installed above the redan. Cold leg sodium flow is directed to this region directly off the pump discharge. The upper internals structure which houses the control rod drive mechanisms and provides backup hold down for the fuel assemblies, the IHXs, and the primary system pumps are supported by the tank closure. The periphery of the tank closure also supports the tank itself as well as its guard vessel. This closure must therefore be a very massive structure.

113The Russians planned to install cold traps and a decay heat removal heat exchanger directly into the pool in their BN-1600 design.

114The Russians support their pool vessels at the bottom using a skirt welded to the vessel. This approach takes much of the load off the vessel wall, but results either in a movable deck or a requirement to accommodate motion between the pool and the deck. In the case of the Russian design, the shroud surrounding the pumps and IHXs must be provided with a bellows where it attaches to the pool structure.

There are no penetrations of the vessel wall. All penetrations are made through the closure (which is sometimes referred to as the "roof"). Eliminating vessel penetrations eliminates a source of failure and makes the vessel stress designer's task simpler.

The tank closure or deck actually consists of two offset rotating plugs, one inside the other. This feature is necessary to permit refueling. By manipulating the rotating plugs and using the transfer machine, fuel assemblies may be moved to a refueling station near the tank wall then outside the tank using, as in the case of the Superphénix plant, an A-frame device.



Fig, 56 Superphénix pool

Although it may be appear intuitive that the loop concept is simpler than the pool, the pool does have the advantage of allowing the primary system components, the intermediate heat exchangers (IHXs) and the primary pumps to be brought in much closer to the reactor than is possible in a loop since there is no need for piping expansion loops to accommodate the thermal expansion attendant with heat-up from ambient conditions to operating temperatures. This means that the containment diameter can be made considerably smaller. The Superphénix containment is about 85 ft. in diameter which contrasts starkly with the 186 ft. CRBRP containment – a plant with just slightly more than ¼ the electric output of Superphénix.[115] Since

115CRBRP containment compares poorly with Fermi-1 whose containment diameter was 72 ft. Scale-up from Fermi-1 to CRBRP size would suggest the CRBRP containment diameter should have been 135 ft.

space within the containment comes at a very high premium, the pool concept is presumed to be more economic than the loop, all other things being equal. Moreover, only a single guard vessel is required (as opposed to seven for example on the CRBRP loop plant) and the reactor vessel is essentially eliminated (or becomes the pool vessel itself depending on one's point of view) along with the boundary vessels for the IHXs and the primary pumps.

The sodium inventory of pool-type plants is considerably greater than for equivalently sized loop plants, by a factor of three or more. This larger inventory tends to reduce the effect of thermal transients on components both within and outside the pool. Since transients are generally more severe in LMFBRs than LWRs owing to the greater temperature rise across the reactor and the higher outlet temperature, features that reduce thermal transient stresses on components are potentially desirable.

The pool concept enables a very compact reactor unit, which is reliable with regard to cooling of the core and confining radioactivity. Pipelines with high temperature coolant, operating under stress are excluded, as well as the cumbersome electric trace heating cables and the sealed concrete cells for location of the primary equipment. The whole issue of protection from loss of primary coolant accidents is essentially eliminated. The hot leg vs. cold leg pump controversy is eliminated in favor of the preferable cold sodium location. Less metal is used for the components, and the total amount of construction work is greatly reduced. There are no nozzles on the tank wall with all penetrations being directly through the head. The surface area of load-bearing walls separating radioactive sodium from the external environment is significantly reduced. Absolute leak-tightness of the main primary circuit pipes is not required, as leaks would be confined within the pool vessel. The repairs on loop type plants of primary system leaks are time consuming and potentially hazardous. The loop may require draining, the cell in which it is contained must be de-inerted to permit personnel entry, care must be taken to ensure air doesn't get into the unaffected parts of the primary system and that adjacent cells remain inerted. Provisions to accommodate draining a primary loop include tanks with cover gas, connecting piping, valves, pumping equipment, trace heating and instrumentation. Each of the above is a very powerful argument favoring the pool and taken together, they make a good case that the matter is decided, there is no contest.

But hold on a minute. The pool concept does have certain important disadvantages. In the case of Superphénix, the main tank diameter is nearly 69 ft. Since Superphénix was at commercial size, one would expect it to be representative of what is to be expected for any commercially sized pool concept plant. In fact, one might expect larger pool sizes to emerge in the quest for ever larger plant sizes. This compares with a reactor vessel diameter for a commercially sized loop plant as described in this monograph of about 28-32 ft. A vessel of the pool size can only be assembled on site from pieces fabricated elsewhere. The same statement applies to much of the vessel internals including the inner vessel and lower baffle. Of course, the guard vessel would require on site assembly as well. Fabrication on site is always more difficult (and riskier) than shop fabrication where the environment is better controlled, better skilled labor and engineering support are available and appropriate machinery is close at hand. It may be possible to make the argument that the EVST leak on Superphénix was a direct consequence of on-site fabrication.

As is obvious from Figure 50, the tank, its internal structures, and the deck are massive and complicated. Complex structures which will be exposed to a challenging environment must be carefully engineered into the system with considerable attention to the plant duty cycle. There are fewer options available for instrumentation and failed instrumentation is potentially less accessible. Since it is unlikely that all the events to which the plant will be exposed during its lifetime will not have been anticipated by the designer, questions arise as to the response of all this complexity to the unknown. The complexity arises from all the tasks the pool is obliged to perform. When too much is asked of a single machine, there will eventually come the point where the machine fails all its tasks. The question is how much is too much?

Support is another issue. Seismic response becomes more challenging as the vessel size is increased. Thickening of the vessel wall to address seismic requirements complicates the structural response to thermal transients, increases costs, and makes on site fabrication more difficult. Seismic isolation is an option, but was not chosen for the Superphénix plant for reasons that are not widely known, but possibly because seismic isolation was not a developed technology at the time.

Accommodation of thermal expansion of the coolant is another issue. In a loop type design the vessel is provided with an overflow nozzle to an overflow tank. As the sodium heats up the overflow tank fills. Since there are no nozzles on a pool, all thermal expansion must be accommodated by the pool vessel itself in the form of increased vessel height.

In order to prevent activation of the intermediate sodium passing through the IHXs, it is necessary to shield the core in a pool reactor to a greater extent than is required in a loop reactor. In fact, in a loop reactor, the only equivalent assemblies are the reflectors, part of whose purpose is to conserve neutrons with its shielding function being to protect the core barrel from excessive neutron fluence, a much less demanding requirement. This requirement for shielding would increase the diameter of the pool by about ten feet in the case of Superphénix had it not already been set by the IHXs.

Although the pool may involve less total construction, there is tremendous construction activity centered on the pool area. The scheduling of activities to prevent interferences during construction will inevitably lead to a longer critical path than would be the case for a loop design. The critical path on a pool type plant is almost certainly established by the pool itself. For a capital intensive construction project such as a power plant, there is a strong incentive to reduce construction time to a minimum so as to reduce the carrying cost of the financing required to support construction. Time is money when it comes to nuclear power plant construction. The figure below shows the upper part of the Superphénix tank during a phase of the construction

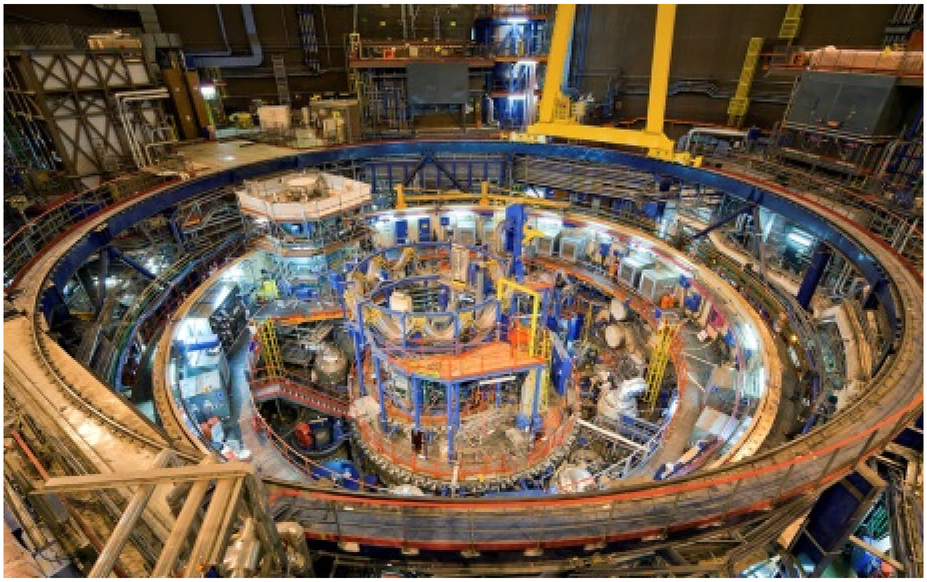


Figure 57 Superphénix during construction

Combined with the complexity of this area during construction is the complexity of the end product that results from imposing so many functions to be performed in such a small area. Figures 49 & 50 omit all the auxiliary functions than need to be crammed into this space. It would be revealing to place a photograph here to make the point, if one could be readily located. Suffice it to point out that complexity in the head area was a huge problem for CRBRP and was much greater for Superphénix.

The IHX is yet another issue for the pool. Since it is located within the pool vessel itself and directly impacts the pool diameter, there is an incentive to minimize its heat transfer area. For the case of Superphénix, the IHX log mean temperature difference (LMTD) was 59°F. Lowering the LMTD to say 40°F enables either better steam conditions, lower HTS flow rates, or a combination of the two, both of which have economic impact. An IHX with a LMTD of 40°F would have a heat transfer area 47.5% greater than an IHX with a 59°F LMTD. This greater heat transfer area is likely to pose a much smaller economic impact for a loop plant than it would for a pool.

If one were motivated to use an EM pump for primary flow, it could not be buried in the pool and would probably need to be mounted on the head requiring long discharge piping. The incentive for adopting an EM pump for the PHTS is probably non-existent in a pool reactor in contrast to the loop where the incentive is great.

Earlier in this section the greater sodium inventory of the pool was given as one of its advantages. In fact, pool advocates routinely advance this argument. Mainly, this argument is centered on the response of the plant to a reactor trip after which the hot leg experiences a significant down temperature transient owing to the long coastdown time of the primary system centrifugal pumps. If the pressure drop across the core is reduced and the primary pumps are replaced with EM pumps, this issue vanishes. Following a reactor trip, if EM pumps are used the primary system flow rate declines promptly to natural circulation flow rate and the hot leg transient is greatly ameliorated. Other transients such as loss of heat sink and transient overpower are terminated with reactor trips and tend to be benign. A large sodium inventory is a liability rather than an asset. It results in components that are larger, heavier, and more expensive. Another problem with a large sodium inventory relates to air intrusion events such as the one that resulted in a two year outage at Superphénix. The large sodium inventory on that plant made it more difficult to detect that air intrusion was occurring. Once it was detected much time was required to clean up the system because of the large quantities of sodium involved. It would be much preferred for the designer to be motivated to minimize sodium inventory rather than maximizing it. The argument about increasing sodium inventory to mitigate transients crept into the CRBRP design with unfortunate results.

Another factor weighing on the pool vs. loop controversy relates to trends in the development of the LMFBR concept. There has long been an interest to push core burnup up to ever higher levels for economic reasons. Since the core creates new fuel in the process of operation, there is no neutronics reason why peak burnup of 30% or even higher could not be achieved. Combining heterogeneous core designs with high burnup capability, it is possible to design a core that would require refueling only infrequently. If, for example, one could design a core that would require refueling only once every ten years, a totally different approach to refueling system design is created as described in section 6. There is less incentive for exploring such options in a pool since all pool reactors have two rotating plugs and can accommodate frequent refuelings.

Earlier (Section 3), an Argonne National Laboratory feasibility study of a 10,000 MWth plant was discussed. Although this study could not be identified as anything more than a concept, nonetheless, it was embodied in a loop type design. The reactor vessel was 40 ft. in diameter and 64 ft. high. The point to be made here is one of limits. Eventually, it will become impractical to increase the size of components further in the quest for ever increasing plant sizes. In this regard, there is much more maneuvering room for plant size increases if one is starting with a 28-35 ft. diameter reactor vessel as opposed to a 69 ft. diameter pool.

To achieve capital cost improvement, there is no reason why a loop-type plant can't be designed with fewer loops. For the pool, since it is desirable to minimize the pool diameter, there is an incentive to adopt small IHXs and fit them tightly inside the vessel. Superphénix had eight IHXs, which is relatively typical. Two IHXs were connected to a loop resulting in four primary loops. Loop type plants have no particular incentive to hold down the IHX size, opening the possibility of two loop plants.

The loop concept does not require that the reactor vessel be in the center of the containment. The vessel can be offset if there is a design advantage in doing so. In Japan, there has been interest in integrating the IHXs and the pumps into a single component. Doing so would eliminate the crossover pipe between the pump and the IHX. Another cost saving measure would be to eliminate the elevated loop concept and accommodate piping system expansion in a downward vertical loop contained within double-walled piping as was done on the Fermi-1 reactor. Interestingly, when Fermi-1 was being designed, consideration was given to the pool but the loop was chosen because of better access for maintenance of components, flexibility of design, and expectation that the loop would be less costly.[116] These are examples of the design flexibility afforded by loop-type plants.

In a relatively recent IAEA conference on the breeder[117], the Japanese representative present stated that Japan is continuing to develop the loop focusing on a 1500 MWe two loop concept. There is no *a priori* reason why loop type reactors must have three or four loops. The CRBRP had three loops because the original design called for the decay heat removal system to be taken off the steam generators through the Steam Generator Auxiliary Heat Removal System (SGAHRS). If one loop were inoperative, there had to be two more to provide decay heat removal redundancy using SGAHRS – thus three loops. As things turned out, the original CRBRP decay heat removal concept was not accepted by the NRC and a second system with air cooled heat exchangers off the primary system was installed (see section 8). Provided that decay heat removal is directly off the reactor or primary system, it is even possible that a single loop concept would be workable if it were to prove to be more economic. Both the SRE and SEFOR were single loop plants.

There is a design approach called the top entry concept[118] which provides for decay heat removal directly off containment vessels surrounding the reactor and loop component vessels. This concept would be workable for two loop plants and has been, in fact, adopted for the JSFR-1500. The top entry concept with separate redundant decay heat removal loops could be used on a single loop plant. Even without top entry, a single loop concept with separate reactor vessel nozzles for a redundant decay heat removal system would be workable. If the pump and IHX are integrated to fit into a single vessel, and a single loop concept is adopted, one would be left with just two vessels rather than seven as on CRBRP, each of a much more manageable size than is the case with a pool type plant. The spacing of these two vessels could be chosen in such a way so as to optimize cost and improve constructability. Since there is a single reactor and a single turbine, having single heat exchange components in between, if feasible, could be simpler than having multiple IHXs and steam generators, although for plant reliability reasons, it may be desirable to retain multiple steam generators. The economic incentive for reducing the number of steam generators is not as great as is the case for the primary loops.

Another example is the reactor vessel height. On CRBRP, the interior dimension of the reactor vessel was 59 feet – all for housing a core with an active length of just over 5 ft. There are several steps that can be taken to reduce the length of the reactor vessel, some of which were described in section 6. Does a similar opportunity apply to the pool as well? The answer is, yes but probably not to the same extent, since it would be more difficult to eliminate in-vessel transfer in a pool and the pool must accommodate the IHXs and primary pumps. All of the foregoing suggests that there is considerable unrealized potential for improvement of the loop-type design which is less obvious for pool-type plants.

The following table trades off the advantages of each of the concepts:

116*Fermi-1 – New Age for Nuclear Power*, E. P. Alexanderson, ed., American Nuclear Society, 1979
117*Liquid Metal Cooled Reactors: Experience in Design and Operation*, IAEA-TECDOC-1569, December, 2007
118*Passive cooling system for loop-type top entry liquid metal cooled reactors*, Patent Application EP 0533351 A2, C. E. Boardman et al, General Electric Co., March 24, 1993.

Pool Advantages	Loop Advantages
Eliminates separate vessels for IHXs & pumps	On-site fabrication minimized
Eliminates thermal expansion loops	Fewer critical path interferences
Eliminates overflow vessel	Better scale-up to larger sizes
Single guard vessel	Better IHTS separation
Close in containment	Better T/H optimization of IHXs
Reduced volume to be shielded	Use of PHTS EM pumps accommodated
No side penetrations	Air intrusion easier to detect
Maximizes sodium inventory	Minimizes sodium inventory

Note that the last item shows up on both sides – it is an advantage for the pool in providing greater thermal inertia and an advantage for the loop when EM pumps are deployed and a fast acting naturally circulating DHRS is available. In-vessel storage of spent fuel could be considered an advantage for the pool. CRBRP had some limited in-vessel storage, but the "design concept" does not. In-vessel storage allows recovery of the decay heat from spent fuel assemblies and makes such assemblies easier to handle once they are transferred outside the reactor vessel. The first six pool advantages could be summarized as "compact PHTS" and the seventh advantage is not quantifiable. The loop advantages are better constructability, operability, and scale-ability.

The loop appears to be generally more adaptable to evolving design approaches than is the case with the more greatly constrained pool. Moreover, the economic argument favoring the pool appears to be vulnerable and certainly didn't materialize on Superphénix. Because of the above considerations, the "design approach" is based on a loop design, albeit one very different from CRBRP.

Before leaving this subject, it needs to be acknowledged that the Russian BN-800, a pool reactor, was reported to have been completed (in 2016) for the equivalent of $2B, which would be competitive if the same could be accomplished in the U.S. Of course, a report in a technical meeting is not the same as an audit to some acceptable accounting standard. The applicable differences in labor, material, energy, and manufacturing costs would all need to be accounted for. The ability of the Russian design to pass licensing requirements in the U.S. is an uncertainty.

Appendix 6 The hot leg vs. cold leg primary pump controversy

The decision of whether to place the primary system pump in the hot leg or the cold leg is one that occupied considerable attention on CRBRP. The cold leg would generally be preferred as it operates 250°F lower in temperature than the hot leg and is exposed to a much less severe transient environment. Even for a simple reactor scram, the hot leg will experience a down temperature transient on the order of 200°F in the space of less than a minute as a result of the prolonged coastdown of the centrifugal primary system pumps. Since stainless steels, having both low thermal conductivity and high thermal expansion, are widely used in the primary circuit, thermal shock should be avoided to the extent possible. It is worthy of mention that this thermal shock problem was explicitly and successfully dealt with on the SRE, (which operated at full power with a reactor ΔT of 460°F.) and the Hallam reactors with a device that was called an eddy current brake.[119] This "eddy current brake" was essentially an EM pump working backwards, that was activated on the occasion of reactor trips and it effectively eliminated the hot leg temperature transients. The precedent for placing the pump in the hot leg actually was set by the SRE and followed by Hallam. The eddy current brake was located in the cold leg.

Lower temperature in the cold leg means higher density and lower volume rate of flow for a given weight rate of flow, simplifying the pump design. In a tightly packed reactor core such as is found on breeder reactors, the core pressure drop traditionally has been on the order of 100 psi. If the pump is in the hot leg, the primary side of the IHX will be the highest pressure point in the primary system next to the pump discharge itself. Since the secondary side of the IHX must be maintained at a higher pressure than the primary side, putting the primary pump in the hot leg raises the secondary side pressure by 100 psi above what it would have been with a cold leg primary pump. Increasing secondary side pressure complicates steam generator leak detection and imposes greater challenges on the design of the sodium water reaction products system (SWRPS). It also increases the consequences of any secondary system leak.

This controversy is closely related to the discussion in section 7 and is the result of the interplay between two perceived requirements. First is the desire for as much of the primary piping system as possible to be inspectable, which means that much of the primary piping system will be unguarded. Second is the need to accommodate the postulated double ended pipe break in the design. The reactor and pump cover gas systems are connected and both operate slightly above atmospheric pressure so as to prevent the reactor level from dropping below the outlet nozzles in the event of a double ended pipe break somewhere in the PHTS. The pump is of a sump suction design so as to eliminate the sodium seals in the pump that proved so troublesome on the SRE plant. In a sump suction pump, the shaft seal is actually sealing the cover gas. The sodium level in the pump is determined by the head losses between the reactor outlet nozzle and the pump itself. With the pump in the hot leg, the pump level drawdown is about 12 feet, which rises to over thirty feet for the cold leg pump location. While 12 feet is considered manageable, a 30 ft. drawdown would result in a much longer shaft and was generally considered on the CRBRP

119 R.E. Durand, *Sodium Reactor Operating Experience*, Chemical Engineering Progress, Vol. 57, No. 3, Mar 1961. See also R.J. Beeley , J.E. Malmeister, *Operating Experience with the SRE and its Application to the Hallam Nuclear Power Facility*, Atomics International, 1961

project to be an unrealistic option. For these reasons, on both CRBRP and FFTF, the pump was located in the hot leg.

Earlier sodium cooled reactors avoided this problem in various ways. For both SRE and Hallam, mechanical pump seals were used that were directly exposed to sodium. In the case of the SRE, the mechanical seals directly led to a flow blockage in the core when the oil (tetralin) that was used to lubricate and cool the seals found its way into the primary sodium system and blocked flow to core assemblies. The fuel failures that occurred in that reactor are celebrated by anti-nuclear activists in the San Fernando Valley to this day. Since the SRE experience, there hasn't been much interest in deploying pumps with sodium seals. This is somewhat academic anyway since the PHTS pumps were located in the hot leg on both SRE and Hallam.

The Fermi-1 designers solved the drawdown problem by installing double-walled piping throughout the primary circuit. With double walled piping, there is no incentive to maintain reactor cover gas pressure at atmospheric pressure or to equalize the pump and reactor cover gas systems. The reactor cover gas pressure was allowed to increase as the primary pump speed was increased; essentially eliminating pump drawdown to levels even less than the hot leg pump plants. The Fermi-1 approach had the additional advantage of allowing vertical expansion loops in the primary circuit resulting in a significantly more compact containment than was achieved on FFTF and CRBRP. The Fermi-1 containment was just 72 ft. in diameter. In contrast, the FFTF containment had a diameter of 135 ft. The designed thermal power of Fermi-1 was 300 MW which compares well with FFTF's 400 MW. If one assumes that containment footprint should be proportional to thermal power level, then its diameter should increase with the square root of power level and FFTF's containment diameter should have been about 83 ft. At 83 ft. diameter, FFTF's containment footprint would have been about 62% smaller that it turned out to be. While there may have been other reasons why the FFTF designers did not adopt the PHTS layout approach used at Fermi-1, the inspectability of the primary system welds played a role as was discussed earlier in section 7. In addition to the FFTF, there had been a precedent for a hot leg pump. The Karlsruhe reactor, KNK-1 which was modified to KNK-2 had a hot leg pump. The German follow-on plant, SNR-300, also had a hot leg pump as well as the then planned German commercial sized plant.

An interesting concept proposed and patented by Westinghouse[120] involves the use of a restricting barrier placed between the outlet nozzles and the sodium surface inside the reactor vessel. The sodium above the restricting barrier drained into a reactor coolant reservoir tank which had a cover gas that was equalized with the reactor and the PHTS pump. The reservoir tank was connected to the PHTS pump suction through a control valve. How the reactor was to be refueled, and where the UIS was to be placed was not described.

Another avenue that might be explored would be to eliminate the double ended pipe break from the design basis. The precedent for establishing the double ended guillotine pipe break as a design basis event was established by the licensing of Light Water Reactors (LWRs). It would be a straightforward matter to argue that a double-ended guillotine failure of low pressure sodium piping is mechanistically impossible and therefore not applicable to LMFBRs because of their much lower primary system pressure and lack of sufficient energy in the coolant for a pipe

120 U.S. Patent 3,951,738, Nuclear Reactor Coolant and Cover Gas System, George, J.R. et al, April 20, 1976

failure to propagate into the double-ended guillotine type failure. The argument would be made that if a PHTS leak were to occur, it would be quickly detected, the plant would be shutdown and cooled down, the leak would be isolated by draining the effected part of the PHTS to the overflow or sodium drain tank, and repairs would be made. Some effort was committed to this approach by at least one of the national labs during the time the CRBRP project was underway, but it was not pursued to a conclusion. This item has been included in appendix 9.

Any future loop-type plants are likely to incorporate the primary pump into the cold leg despite the precedent set by FFTF and KNK-2. The motive for retaining the elevated primary system piping concept is suspect and the advantages of the hot leg location are exceeded by the arguments favoring the cold leg location. An addition to double-walled piping would be to adopt an EM pump for the primary circuit. EM pumps have no shaft seal and therefore require no cover gas. While the use of EM pumps for heat transport system applications has been avoided in most sodium cooled reactors to date because of their poor conversion efficiency of electric power input to pumping power (typically 40% at best), they are compact, have no penetrations, require no cover gas, require no lubricants, and are likely to require little if any maintenance. Reducing PHTS head requirements makes EM pumps more attractive. Such pumps could fit nearby, directly underneath, or be incorporated into the IHXs further simplifying containment design. There is no requirement for equalizing cover gas pressure between the PHTS pump and the RV if there is no pump cover gas. A slight positive pressure in the RV may be necessary to provide adequate NPSH for the pump during plant operation. That pressure can be reduced to atmospheric when the reactor and the PHTS pumps are shut down or running at low flow.

As a final note on this subject, maintaining reactor cover gas at atmospheric pressure during operation does have a drawback. In 1992 at Superphénix there was a cover gas leak caused by failed diaphragms in a compressor used to transport cover gas to a radiometer. The systems at the plant did not detect the leak for three weeks. By the time the problem was discovered, 400 Kg. of sodium oxide had been formed in the primary coolant. Restoring oxide purity required two years and required replacement of the cartridges in the cold traps, all the time with the reactor shut down. If there were a leak in the cover gas system, it would be much better for the gas to leak out of the system than for oxygen to leak in. Out leakage of cover gas will be readily detected by radiation air monitors whereas in leakage is much more difficult to detect, particularly at low rates. Small quantities of oxygen readily combine with the hot sodium before it reaches detectable levels in the cover gas space and the nitrogen in the air can cause nitriding of structural components.

Appendix 7   LMFBR development in decline

The cancellation of the CRBRP in 1983 was the watershed event that essentially halted all development of the LMFBR in the U.S. and subsequently in every European country except Russia. Prior to that event, development activity was relatively high and technological progress was being made on a variety of fronts. There were five more or less independent forces at work that led to the CRBRP cancellation.

First, a widespread malaise had set in to the LWR industry. Throughout the latter half of the 70's aggressive regulation by the NRC led to significant increases in plant cost beyond anything that had been anticipated by the utility industry users of the technology. The early nuclear plants, although somewhat more expensive than equivalently sized coal plants, more than made up for their higher capital cost with significantly lower fuel cycle costs. As the plants became significantly more expensive, utility companies began experiencing resistance from their state public utility commissions when it came time to put the plants into the rate base. As plant construction schedules became impacted by mandated retrofits and unanticipated changes many utility companies decided they were in over their heads and cancelled plans for new nuclear capacity. At the same time, the same companies began to experience a decline in load growth. While widespread installation of central air conditioning along with other home appliances had led to load growth in the 4-6% range for the preceding 20 years, the 70s saw load growth dropping to the 1-2% range, removing the need for many of the plants utility companies had in early stages of construction.

Second, the 1970s witnessed the unbridled growth of the environmental movement in the U.S. The word "ecology" did not come into widespread use before the late 1960s. By the early 70s, the movement was at a full gallop with the creation of the EPA and a host of environmentally motivated private entities such as the National Resources Defense Council whose focus was activism. Pre-existing societies such as the Sierra Club rebranded themselves from a focus on conservation to environmental activism. By the mid 70s, there were a large number of these entities, and every one without exception became an anti-nuclear crusader. The central themes of these crusaders were (and by and large, still are) that the plants were not adequately safe and there was no way to dispose of nuclear waste. Their activism clearly had an effect on the NRC, which was under just as much attack as were the utilities building the plants. The NRC responded in the fashion to be expected from any government bureaucracy by turning the screws tighter on the utility companies.

Third was the decision in 1974 to split the Atomic Energy Commission (AEC) into separate promotional and regulatory bodies. Thus was born the Energy Research and Development Administration (ERDA), the predecessor of the Department of Energy (DOE) and the Nuclear Regulatory Commission (NRC). Prior to that split, the AEC carried out the spirit and the law of the atoms for peace initiatives launched by the Truman and Eisenhower administrations. It both promoted and regulated nuclear power and attempted to maintain a balance between the two, being careful not to over-regulate for fear of frustrating nuclear power development. Once the NRC was turned loose, there was little in the way of a moderating influence over its regulatory activism. Coincident with the split of the AEC, the Joint Committee on Atomic Energy was disbanded. The Joint Committee on Atomic Energy had previously been a powerful body in congress composed of senior senators and members of the House that promoted nuclear power development in the U.S. Its disappearance left a huge vacuum.

Fourth was the accident at Three Mile Island in 1979. All the environmental activists saw their opportunity and immediately went on the attack. Every utility company in the country that had committed to building or was operating nuclear plants became the object of a media blitz and was forced to answer why they were continuing with their construction programs or operating facilities that were "demonstrably unsafe". When the molten fuel zirconium cladding at Three Mile Island reacted with the water coolant to produce hydrogen which collected at the top of the reactor vessel, the media had a field day. Everyone knew about hydrogen bombs in those days without having a ghost of an idea how they worked, but the media made the connection anyway setting off a panic. The whole episode scared the pants off a lot of utility company executives and their board members. Following the Three Mile Island accident, there wasn't a new order for a nuclear plant for 30 years and all those that had been ordered after 1973 were cancelled.

Fifth was the non-proliferation bugaboo. In 1977, then President Carter ordered a halt to CRBRP licensing on the basis that it could contribute to nuclear proliferation.[121] The technical merit for this position was apparently based on the observation that the plutonium that was bred in the outer reached of the axial blanket was almost pure Pu^{239}, making it ideal for use in a weapon. There was no effort on the part of the administration to find a technical solution for this issue. The axial blanket thickness could have been reduced, the blanket material could have been changed to Thorium, or the blanket material could have been poisoned with small quantities of Pu^{241} if this alleged concern had been legitimate. In reality, the issue was really a stalking horse to placate Carter's environmental activist buddies and prevent progress on the plant.

By 1981, Carter was out of office and President Reagan ordered resumption of CRBRP licensing. At some point during the previous administration, some politico coined the term "technological turkey" to describe the CRBRP. What was meant by this was never made particularly clear, but the term stuck and somehow implied that the technology was no longer relevant or the plant design was behind the times or the plant cost was greater than could be justified. An unholy alliance was formed between anti-nuclear democrats and budget conscious republicans and the plant was cancelled in 1983.

Following the CRBRP cancellation, the European programs in Germany, the U.K. and France were all terminated one by one, leaving only a modest continuing effort carried on by the Russians. Most notable among these European countries is France, the host country for Superphénix. Superphénix was plagued with numerous operational problems. In addition, the responsible French design entity, notably Novatome, was not able to satisfy itself that the LMFBR concept as developed could compete with LWRs.

121Carter simultaneously ordered a halt to the licensing of the Barnwell reprocessing plant on the same basis, which was the only commercial reprocessing plant under construction at a time when no others were operating. The owners of Barnwell subsequently cancelled the project. There has been no reprocessing plant construction of any kind, commercial or government, in the U.S. since.

The most logical plant design for the follow-on to CRBRP would have been in the 1000-1500 MWe size range, oxide fueled, 1000-1050°F outlet temperature, probably of the loop design, probably with helical coil once through steam generators and possibly EM primary system pumps. The core design would probably use a somewhat larger fuel pin diameter, possibly adopting the CRBRP heterogeneous core design to take advantage of its flux flattening, better breeding, and better utilization of blanket assemblies. The primary heat transport system probably would have abandoned the elevated loop concept although the designs in existence in 1984 retained this concept. The fuel assemblies would probably have been wire wrapped hexagonal cross section with more pins per assembly. In the main however, the plant would be a scale-up from CRBRP. In fact, such a design was underway under a joint DOE/EPRI project known as the Large Scale Prototype Breeder, LSPB.

The LSPB Project began in 1982 and was managed out of an office formed by EPRI in Naperville, Illinois and staffed primarily by personnel loaned to EPRI by NSSS vendor companies and architect engineering companies associated with CRBRP. When CRBRP was cancelled late in 1983, DOE personnel decided they could no longer justify supporting a project that had been advertised as a CRBRP follow-on. Nonetheless, work continued on the project through 1986 with a focus on pool-type designs until the DOE arrived at a strategy for sustaining some activity on the LMFBR concept that would be politically acceptable. That strategy began to emerge in 1986.

The basic idea of the new strategy was to focus efforts on "small innovative LMRs". The "FB" was dropped from the LMFBR acronym because was seen as being particularly annoying to the political opponents, who were mostly arrayed around the environmental movement and non-proliferation concerns. "Innovative" appears to have been intended to convey the notion that somehow the CRBRP was an old fashioned scale-up of the FFTF and the DOE intended to break ranks with the industrial conglomerate supporting the earlier effort and plow new ground. The origins of this counterintuitive (and counter common sense) idea of "small" reactors are not known – one could speculate but to no advantage – but the DOE picked up on this theme with their revised LMFBR program and solicited proposals from the three reactor vendors. At the same time, they withdrew support for the LSPB causing EPRI to similarly withdraw support and close the office that had been set up to pursue the effort.

The DOE invited proposals from the three LMR reactor vendors that had been engaged in the design of CRBRP, Westinghouse, General Electric, and Atomics International. The DOE decided to go forward with the proposals submitted by General Electric and Atomics International. In 1988 the DOE down selected to the General Electric design for further development, which was a small modular pool design. About three years later, DOE funding dried up for the General Electric design, but the company continued to support a token activity surrounded around this concept. Although there have been some initiatives and activity at the national labs, neither the DOE nor the electric utilities have done anything of substance to advance the cause of the breeder reactor in over 30 years.

Five prevailing conditions from the 1970s affecting LMFBR design

There were five conditions that were characteristic of the 1970s that had a negative effect on breeder reactor design and development. None of the five were lasting nor, in retrospect, should any of them have had the effect they did. The importance of these prevailing conditions and their impact on decision making was not widely recognized at the time, but as always seems to be the case, hindsight is 20-20.

First, there was a cold war going on and both the U.S. and the Soviet Union were furiously expanding their weapons inventories creating a strong demand for plutonium. There was almost no commercial reprocessing, so the only significant source of plutonium was from weapons stocks, which were jealously husbanded. In fact, at the time, CRBRP couldn't even use weapons plutonium because of a statutory requirement to separate the weapons program from civilian nuclear power. The only stocks of plutonium available for CRBRP were from the DOE civilian program and they were very limited. Highly enriched uranium could have been used as the fissile material, but it is very expensive, and has a poor neutron reproduction rate resulting in a lower breeding ratio. The use of enriched uranium would also have invalidated much of what was intended to be demonstrated. This led to the need to minimize fissile inventory in the core.

A requirement for low fissile inventory inevitably leads to small diameter fuel pins. In fact, both the FFTF and the CRBRP had 0.23-in. diameter pins, which are small in comparison to other LMFBRs that were operating or under construction at the time. Cores fueled with small diameter pins will have a low internal breeding ratio with much of the breeding taking place in the blankets and they will experience fairly rapid reactivity loss with operation minimizing the burnup that is achievable and requiring frequent refueling. The initial approach on CRBRP was to refuel one third of the fuel assemblies each year and a smaller fraction of the radial blanket assemblies. In the case of the initial CRBRP homogeneous core design, one year of operation resulted in about 3½ atom percent peak fuel burnup. Peak burnup of the fuel at discharge was about 11 atom percent. With heterogeneous core designs, plutonium builds up quickly in the internal blankets causing them to generate considerable power at the end of an annual cycle, potentially making it necessary to shuffle them to peripheral regions where the flux is lower during refueling outages. There is more detail on homogeneous and heterogeneous core designs in Appendix 2C, particularly including the change from the homogeneous to the heterogeneous core design on CRBRP.

A second and related condition from the 1970s was the emergence of the uranium cartel. In 1972, representatives of the five major uranium producing nations (at that time Canada, Australia, the former Soviet Union, Niger and Namibia) formed a cartel that drove yellowcake (U_3O_8) prices sharply higher. There was an expectation that there was no other way to defeat the cartel except with a nuclear concept that was virtually independent of uranium supply – viz. the breeder reactor. This, combined with the fact that 15-20 years of operation of a LWR is necessary to produce enough plutonium for the initial fuel loading of a single equivalently sized LMFBR meant that if there were going to be a surge in orders for LMFBRs, plutonium supplies would be pinched. A further strain on plutonium resources is the fissionable material tied up in the fuel cycle – the time required to allow for spent fuel decay heat to reduce sufficiently for shipment to the reprocessing facility, and the time tied up with shipping, reprocessing, and new fuel fabrication. The expectation was that it would be at least three years between the time that LMFBR spent fuel is removed from the reactor until it is returned again in the form of new fuel.

To fit into the scheme of low fissile inventory with good doubling time, it was necessary to have a refueling system that was fast and could complete an annual refueling cycle in about two weeks. Thus, through the head refueling became the standard for LMFBRs involving a complexity of rotating heads, in-vessel transfer positions between the core barrel and the reactor vessel wall, in-vessel transfer machines (IVTM), ex-vessel transfer machines (EVTM) and an ex-vessel sodium filled spent fuel storage tank (EVST). This subject has been extensively treated in section 6.

The fundamental point to be made from the above is that fissile inventory needs to be low on the list of design considerations and future designs should be focused on operating at high burnup with long periods of time between refueling. Importantly, the plant design needs to be simpler and more economic so that it has a decent chance of being competitive with LWRs. At some point it may become necessary to refocus on fissile inventory in LMFBRs, but that point is a long way off and only after several LMFBRs have been brought on line and begin to strain the fissile supply. For now, the world is awash with plutonium drawn from weapons stockpiles.

A third condition prevailing in the 1970s was the dependence of the U.S. on foreign oil simultaneous with limited and depleted quantities of domestic natural gas – a period long before hydraulic fracturing came into the picture. It was generally believed that nuclear power was needed soon to supply the nation's energy requirements. This situation elevated the importance of design conservatism to raise confidence in technological success. This was particularly evident with the FFTF reactor. Chronologically, the FFTF followed the Enrico Fermi reactor that was built and operated by a host of utility companies led by Detroit Edison. The Fermi-1 plant experienced two serious mishaps – unreliable steam generators and a partial meltdown caused by a loose part in the reactor. The rest of the plant operated without serious incident. In particular, the design of the Fermi plant was on a track that could have led to a reasonably competitive version at commercial sizes. Nonetheless, the Fermi plant was perceived to be a failure and rather than correcting its few woes, the designers of FFTF seem to have started with a clean slate and took a different path, particularly with the heat transport system. [122] One of the key factors involved here was the desire to eliminate the double walled primary system piping system of Fermi. This move was probably motivated by the perception that the double walled system was not inspectable and never would be. The result was an increase in the cost of the plant beyond that which would have shown a promise for economic competitiveness with LWRs. Other factors weighing heavily on the FFTF design were the provisions of closed loops which turned out never to have been used and the absence of a steam generating system. The early FFTF decision makers apparently did not wish to deal with the problem of developing a satisfactory steam generating system while they had so many other irons in the fire (such as the closed loops). This turned out to have been a big mistake. The problem of resolving the steam generator design was left for the CRBRP project to solve, which contributed to excessive CRBRP project costs in funding three separate steam generator designs and a conservative design approach. Moreover, when the FFTF finally made it into operation, it had no revenues from power production to offset operating costs. The result was the operating costs became oppressive and the plant was shut down prematurely.

122 It is of some interest to note that the heat transport system of the German SNR-300 is remarkably similar to both FFTF and CRBRP. It is not clear which one influenced the other.

A fourth condition characterized by the 1970s was the development of the LMFBR more or less in parallel with LWRs. There was certainly a tendency during CRBRP development to compare its features, capabilities, and limitations with those of the LWR. Annual refueling is one example. Another example may be the presence of an operating floor within containment. For the case of the LWR, containment can be entered minutes following shutdown. Because of Na^{24}, it is not possible to quickly enter the spaces carrying primary sodium in a LMFBR. Early liquid metal plant designers all the way back to SRE compensated for this perceived shortcoming in the concept's operability by providing an operating floor inside containment, the region below housing the inaccessible primary system while the region above being accessible immediately following shutdown. The control rod drive mechanisms, the primary pump motors, and the refueling equipment are located in this space above the operating floor. As a result, the operating floor plus the reactor vessel head are heavily shielded, which adds to the cost and complexity of the plant. If the requirement for refueling in two weeks following shutdown were to be removed, there would be no logic supporting the operating floor concept and it could be eliminated, along with the reactor head shielding. Maintainability issues, although important, also need to be considered (and moderated) from this point of view, since because of the sheer nature of the beast, there will be some things that are feasible on LWRs which cannot be accomplished on LMFBRs and vice versa. The containments on EBR-2, SEFOR, Fermi-1, FFTF, and CRBRP are a carryover from LWRs. There was no reason for doing this other than the precedent set by LWRs

The fifth, and probably the most important condition that prevailed in the 1970s was the widespread notion that the LMFBR was the follow-on design concept after the LWR had matured, rather than a competitor of the LWR. That notion tied the fate of the LMFBR to the LWR and served as an apology for why the LMFBR was more expensive than the LWR – it was a concept to be realized after the LWR became obsolete as a result of uranium resource limitations. When one considers that sodium cooled thermal reactors were once considered as potential competitors with LWRs, this follow-on notion makes no sense at all. There is not that much difference between sodium thermal reactors and LMFBRs that would appreciably affect the capital cost of one over the other. The entire design approach needs to be looked at from a different angle. The basic principles of sodium coolant and breeding need to be better capitalized upon. In both design and licensing, the LMFBR must be dealt with as a very different option from LWRs. The LMFBR is an alternative to the LWR and should compete with it along with other sources of electric power generation. The features which make the LMFBR unique need to be capitalized upon in a way that makes it considerably more economic and attractive to utility company users as a near term and attractive alternative to LWRs.

Appendix 8 Uranium resource picture

In section 2 it was stated that there is sufficient uranium in the U.S. to power fleets of breeder reactors supplying basically the entire nation's energy needs for thousands of years. The purpose of this appendix is to provide the basis for that statement.

First, as was also stated in section 2, the breeder reactor makes use of uranium fuel at least twenty times more efficiently than a LWR. A corollary to this statement would be that one LWR would provide sufficient feed stock for twenty equivalently sized LMFBRs. That feed stock would be in the form of the tailings from the enrichment plants needed to enrich the fuel for the LWRs and the LWR spent fuel itself. With all of the tailings and spent fuel from the world's fleet of LWRs, it would be a long time before it would become necessary to mine additional uranium to support a growing number of LMFBRs.

Nonetheless, the question of limitations must ultimately be addressed. The world has so much oil, so much natural gas, and so much coal. All of it will be exhausted at some point – doesn't the same argument apply to nuclear power with the breeder reactor? How much uranium is there in the world and how long would it last if it were to be relied upon to supply a sizeable fraction of the world's energy needs?

The authoritative reference for uranium resource supply and demand is a document referred to as the "red book" that is published approximately biennially. It is a joint report of the OECD Nuclear Energy Agency and the IAEA.[123] On the resource side of the equation, the red book quotes resources as a function of cost of extraction. Costs of extraction are expressed in US dollars per kilogram uranium metal and categorized as shown in the table below.

Resource category	Estimated resource, 1000 MTU
$40/kgU	680.9
$80/kgU	3078.5
$130/kgU	5327.2
$260/kgU	7096.6

Table 13 World uranium resources

On the demand side, the red book reported that the 440 reactors operating worldwide in 2010 represented an installed capacity of 375.2 GWe, generated 2623 TW-hr of electricity and required 63,875 MTU. The red book also projects demand out to 2035 expecting it to increase to between 97,645 and 136,385 MTU/yr. At high resource utilization, consumption will exhaust supply in less than 70 years. Since nuclear plants now being built are expected to have a useful operating lifetime of at least 60 years, one wonders where the uranium will come from. The answer of course is that more expensive sources of uranium will be called upon.

123*Uranium 2011: Resources, Production, and Demand*, OECD 2012, NEA No. 7059

Nature's vehicle for the manufacture of uranium is the supernova. Thus, the only source of uranium on earth is space detritus originating from various galactic supernova, which only occur about once every 500 years in our milky way galaxy. Uranium is therefore a rather rare substance. Nonetheless a great deal of uranium has managed to accumulate on the planet. The oceans, for example, contain about 3 ppb of dissolved uranium in its hexavalent state. While 3 ppb doesn't sound like much, when multiplied by the mass of the planet's oceans this puny concentration adds up to a total of $5.9 \cdot 10^9$ MTU. There are various mechanisms for reducing (reducing here used in the context of decreasing the valence state of the ion) the dissolved uranium in the oceans that involve decaying organic material and hydrogen sulfide from its hexavalent state to its quadravalent state which leads to its precipitation. That is the reason why most of the so-called marine black shales which contain petrochemicals also contain fairly generous amounts of uranium.

Earlier in this monograph it was stated that the Marcellus shale contains about 25 ppm of uranium. Assuming an average thickness of 300 ft over its 90,000 sq mi extent, there are over $1.3 \cdot 10^9$ MTU in the Marcellus shale. Other shales offer more promising sources because of their higher concentrations of uranium. For example, the Chattanooga shales that run through eastern Tennessee contain 80 ppm of uranium. The Chattanooga shales are far less extensive than the Marcellus shales, covering about 1000 square miles and having an average thickness of about 20 ft. Nonetheless, these shales contain about $6.5 \cdot 10^6$ MTU, about the same as the total world resource quoted in the red book.

Yet another shale formation that has been much in the news recently is the Bakken formation, which has so enriched the state of North Dakota in the past several years. The Bakken formation is actually primarily sandstone with relatively thinner layers of shale both above and below. There is relatively little uranium in the sandstone, but the upper shale formation averages 6 ft in thickness and contains an average of 42 ppm uranium while the lower shale formation averages 13 ft in thickness and contains an average of 62 ppm uranium.[124] Given that the extent of the Bakken formation is about 200,000 square miles, the lower shale formation contains $1.6 \cdot 10^9$ MTU with another $0.5 \cdot 10^9$ MTU from the upper shale.

If one uses the current consumption and production figures in the red book, LWRs require about 25 MTU/TW-hr. Accounting for their better fuel utilization (the factor of 20 quoted above) and their higher thermodynamic efficiency, LMFBRs can reasonably be expected to require just 1 MTU/TW-hr. Actually, this number is probably way too high. The well known relationship between uranium consumption and power production of 1 gram = 0.95 MWth-days = 22.8 MWth-hr would imply that for a 40% thermodynamic efficiency and a 60% fuel utilization rate, 1 gram should yield 5.5 MWe-hr. Thus a TWe-hr of production would consume only 0.2 MTU, about a hundredth of current LWR consumption rate. At this rate of consumption, the cost of uranium extraction can be measured in the thousands of dollars per kilogram and the plants would still be economic from a fuel utilization point of view. $10,000/kg U would translate into about $20/lb of coal so even at such a high price it would still be a bargain.

A thousand 1000MWe plants would come close to supplying all the U.S. electric power requirements for many years to come. At the uranium consumption rates described above, such

124*The Uraniferrous Bakken Shales of North America*, IAEA presentation, November 12, 2009 a fleet of reactors would consume about 1750 MTU per year. At this consumption rate, the Chattanooga shale alone would be a 3700 year supply. The Marcellus shale would be good for millions of years. One could argue that over such lengthy periods other sources of energy such as fusion should certainly have been developed to the point that they can be used as reliable sources of power production. Such arguments miss the point. The point is that resource limitations cannot be used as an argument for continuing the halt on nuclear power development. Moreover, while the environmental impact of nuclear power is known and is capable of being significantly reduced, there is no knowledge – just speculation – of the environmental impact of any future technology that might be offered to replace it.

Appendix 9 Opportunities for further analyses or R&D

- There is need for better characterization of oxide fuels operating in a load following mode than currently exists (Appendix 2)
- By analysis, examine core design approaches that best reduce reactivity swing with burnup and enable very long lived core designs (Appendix 2C)
- Determine the optimum split in thickness between upper and lower axial blankets that accounts for control rod shadowing. Account for heterogeneous core in analyses (Appendix 2C)
- Development of a remote controlled variable flow control device for inner blanket assemblies that is self actuated (Appendix 2D)
- As an alternative to the above, develop a device that is self actuated that regulates the temperatures of both fuel and blanket assemblies (Appendix 2D)
- Control and operation of a naturally circulating LMFBR (Appendix 2E)
- Core restraint for a core loaded with ductless fuel (Appendix 2C)
- Sort out the details of adopting the SRE refueling approach to a large LMFBR (Section 6)
- Design of a reliable lower closure valve for a refueling shroud (Section 6)
- Detailed literature survey of the SEFOR refueling system approach (Section 6)
- Evaluate heat loads and sodium deposition rates within the refueling cell during refueling (Section 6)
- Review of the SRE and Hallam reactor bottom support to determine how obvious design questions concerning this approach were resolved on those plants (Section 6)
- Determine the scalability of the SRE/Hallam reactor vessel support design approaches to large sized plants (Section 6)
- Quantify the economic incentive for adoption of the bottom mounted reactor vessel concept (Section 6)
- Determine by analyses if it is feasible to eliminate the reactor vessel thermal liner in the bottom mounted configuration (Section 6)
- Examine the potential for combining the functions of the reactor vessel and the core barrel thus eliminating the core barrel (Section 6)
- Update earlier tradeoff studies comparing argon and helium cover gases given that the only primary system component requiring cover gas is the RV (Section 6)
- Experimental activity to explore the effects of thermal striping on prospective UIS materials (Section 6)
- Identify materials most suitable for use in a compact PHTS (Section 7)
- Devise a solution for the RV bypass flow problem when deploying DRACS for shutdown heat removal. (Section 8)
- Explore the options for core tailoring using minor actinide fuels (Appendix 3)
- Perform literature survey on self-actuated shutdown systems, perform tradeoff and evaluation of alternatives, identify weaknesses and develop indicated improvements (Section 10)
- Determine the practicality of fractional distillation as a means of removal of Cs137 from the Na coolant (Section 12)

- Evaluate the rate of migration of fission gasses in vented fuel from the location of formation to the space above the upper axial blanket (Section 12)
- Evaluate opportunities for improvements to the SWRPS that would reduce its cost (Section 12)
- Rethink the RAPS assuming helium is used as the reactor cover gas (Section 12)
- Identify a coolant, e.g. an advanced version of Dowtherm J, that would be a suitable replacement for NaK (Section 12)
- Explore the sodium void reactivity associated with the thorium cycle (Section 4)
- Develop a scholarly argument for the elimination of the double ended guillotine pipe rupture from the PHTS design basis founded on the argument that there is insufficient energy it the primary system to propagate any pipe leakage to double ended break. (Appendix 6)

Appendix 10 Cost reduction measures

1. Shorten fuel assemblies
2. Eliminate Lower Inlet Modules and associated Bypass Flow Modules
3. Reduce core pressure drop
4. Eliminate hydraulic hold-down
5. Eliminate gas tagging and associated failed fuel monitoring system
6. Eliminate head shield
7. Replace ellipsoidal lower head with flat bottom
8. Bottom mount RV
9. Eliminate SRP
10. Adopt 10-year refueling interval
11. Eliminate RV volume reserved for in-vessel transfer
12. Eliminate CCPs
13. Eliminate IVTM
14. Eliminate IRP
15. Reduce EVST volume due to CCP elimination
16. Eliminate Containment to RSB hatch
17. Reduce size of RSB to a transfer room, fuel handling cell, and shipping room
18. Eliminate LRP (OVR)
19. Eliminate UIS jacking mechanism (OVR)
20. Eliminate transfer positions in RV
21. Eliminate need for valves mounted on RV and EVST (OVR)
22. Eliminate Plug Handling Machine (OVR)
23. Eliminate need for purging space between valves (OVR)
24. Eliminate EVTM (OVR)
25. Eliminate Auxiliary Handling Machine
26. Eliminate EVTM rails through containment and RSB
27. Use Superphénix steam generating system – eliminate separate evaporators and superheater, recirculation pumps, steam drum
28. The elevated unguarded PHTS piping concept is abandoned, permitting greatly reduced containment volume. IHXs are brought in closer to reactor reducing requirements for expansion loops
29. Centrifugal pumps are replaced with EM pumps in the primary and intermediate circuits.
30. Two primary loops down from three.

31. Elimination of requirement for pony motors on PHTS and IHTS pumps
32. Check valves are eliminated from PHTS.
33. Guarded PHTS piping eliminates need for liners on PHTS vaults.
34. Adopt rectilinear vs. cylindrical containment structure and eliminate the requirement for a single elevation basemat.
35. Significantly reduce containment volume
36. Eliminate requirement for single elevation operating floor. The requirement to have a floor that is accessible during operation inside containment is unnecessary.
37. Reduce containment design pressure from 10 psi to 5 psi or any pressure that will permit reliable leak testing
38. Eliminate separate confinement building
39. Eliminate containment cooling system
40. Eliminate air filtration processes that extend beyond CAPS
41. Eliminate all cell liners not required for containment leak testing
42. Significantly reduce 1E loads
43. Eliminate requirement for electric power to maintain safe shutdown
44. Eliminate requirement for IHTS and SGS to be safety related
45. Eliminate two Primary Sodium Storage Tanks in RSB, each having 60,000 gallon capacity
46. Reduce size of Overflow Vessel to size required to accommodate heat-up from refueling conditions to full power allowing margin for TOP transients at power.
47. Use HVAC cooling (vs. NaK) for the primary cold traps.
48. Combine primary system makeup pump drain vessel with primary system drain tank.
49. Elimination of tritium removal unit from CAPS
50. Eliminate unnecessary features from the IGRP system
51. Replace NaK with an updated version of Dowtherm J everywhere it exists in the plant, where possible.
52. Defer or eliminate radwaste system for treatment of effluent from the spent fuel cleaning facility
53. Eliminate control rod ejection accidents from the design basis.

Appendix 11 List of acronyms used

AEC	Atomic Energy Commission
AFMS	Alternative fuel management scheme
AHM	Auxiliary handling machine
ALM	Auxiliary Liquid Metal
ANL	Argonne National Laboratory
APDA	Atomic Power Development Associates
ASLB	Atomic Safety and Licensing Board
ATWS	Anticipated transients without scram
BRC	Breeder Reactor Corporation
CAPS	Cell Area Processing System
CCP	Core component pot
CRBRP	Clinch River Breeder Reactor Plant
CRDM	Control rod drive mechanism
CRIEPI	Central Research Institute for the Electric Power Industry (Japan)
CRM	Cost Reduction Measure
CRRNM	Collapsible rotor-roller nut mechanism
DBA	Design Basis Accident
DF	Decontamination Factor
DFBR	Demonstration Fast Breeder Reactor (Japan)
DHRS	Decay Heat Removal System
	Direct heat removal service
DNB	Departure from nucleate boiling
DRACS	Direct reactor auxiliary cooling system
DOE	Department of Energy
EBR	Experimental Breeder Reactor
EM	Electromagnetic
EPRI	Electric Power Research Institute
ERDA	Energy Research & Development Administration
ESF	Engineered safety feature
EVST	Ex-vessel storage tank
EVTM	Ex-vessel transfer machine
ETEC	Energy Technology Engineering Center
FFTF	Fast Flux Test Facility
FHC	Fuel handling cell
FTM	Fuel Transfer Machine
GV	Guard Vessel
GWe	Giga watt electric
HCDA	Hypothetical core disruptive accident
HTS	Heat transport system
IAEA	International Atomic Energy Authority
IGRP	Inert Gas Receiving & Processing system
IHTS	Intermediate heat transport system
IHX	Intermediate heat exchanger

IRACS	Intermediate reactor auxiliary cooling system
IRP	Intermediate rotating plug
IVTM	In-vessel transfer machine
LMFBR	Liquid metal fast breeder reactor
LMTD	Log mean temperature difference
LRP	Large rotating plug
LRW	Liquid Radioactive Waste
LOF	Loss of flow
LSPB	Large Scale Prototype Breeder
LWR	Light water reactor
MTHM	Metric tonnes heavy metal
NRC	Nuclear Regulatory Commission
JAPC	Japan Atomic Power Company
JSFR	Japanese sodium fast reactor
MWD	Megawatt days
MWe	Megawatts electric
MWth	Megawatts thermal
MTU	Metric tonnes of uranium
NPSH	Net positive suction head
NSSS	Nuclear steam supply system
OHRS	Overflow heat removal system
OV	Overflow Vessel
OVR	Open vessel refueling
PACC	Protected air cooled condenser
PFR	Prototype fast reactor (UK)
PHTS	Primary heat transport system
P&ID	Process and Instrumentation Diagram
PMC	Project Management Corporation
PNC	Power Reactor and Nuclear Fuel Development Corporation (Japan)
PRA	Probabilistic risk assessment
PRACS	Primary reactor auxiliary cooling system
PRDC	Power Reactor Development Company
PSAR	Preliminary safety analysis report
PSER	Preliminary safety evaluation report
PUREX	Plutonium uranium extraction (reprocessing plant)
PWR	Pressurized water reactor
RAPS	Radioactive Argon Processing System
RCB	Reactor Containment Building
RFP	Request for proposals
RFTP	Reactor fuel transfer port
RGT	Rotating guide tube
RDT	Reactor Development & Technology (Division of the AEC)
RSB	Reactor Services Building
RV	Reactor Vessel
SCC	Standard Cubic Centimeters
SEFOR	Southwest Experimental Fast Oxide Reactor

SGAHRS	Steam generator auxiliary heat removal system
SGB	Steam Generator Building
SGS	Steam Generating System
SRE	Sodium Reactor Experiment
SRP	Small Rotating Plug
SS	Stainless steel
STP	Standard temperature and pressure
SWRPS	Sodium water reaction products system
THDV	Thermal hydraulic design value
TOP	Transient overpower
TRUEX	Transuranic extraction (reprocessing plant)
TWe	Terra watt electric
UIS	Upper internals structure
WVN	Wet Vapor Nitrogen