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Small Liquid Metal Cooled Reactor Safety Study

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1. Summary

The Small Liquid Metal Cooled Reactor Safety Study documents results from activities conducted under Small Liquid Metal Fast Reactor Coordination Program (SLMFR-CP) Agreement, January 2004, between the Central Research Institute of the Electric Power Industry (CRIEPI) of Japan and the Lawrence Livermore National Laboratory (LLNL)[1]. Evaluations were completed on topics that are important to the safety of small sodium cooled and lead alloy cooled reactors. CRIEPI investigated approaches for evaluating postulated severe accidents using the CANIS computer code. The methods being developed are improvements on codes such as SAS 4A used in the US to analyze sodium cooled reactors and they depend on calibration using safety testing of metal fuel that has been completed in the TREAT facility. The 4S and the small lead cooled reactors in the US are being designed to preclude core disruption from all mechanistic scenarios, including selected unprotected transients. However, postulated core disruption is being evaluated to support the risk analysis. Argonne National Laboratory and the University of California Berkeley also supported LLNL with evaluation of cores with small positive void worth and core designs that would limit void worth. Assessments were also completed for lead cooled reactors in the following areas: (1) continuing operations with cladding failure, (2) large bubbles passing through the core and (3) recommendations concerning reflector control. The design approach used in the US emphasizes reducing the reactivity in the control mechanisms with core designs that have essentially no, or a very small, reactivity change over the core life. This leads to some positive void worth in the core that is not considered to be safety problem because of the inability to identify scenarios that would lead to voiding of lead. It is also believed that the void worth will not dominate the severe accident analysis. The approach used by 4S requires negative void worth throughout the core life, which leads to large reactivity worth in the control systems. The conclusions from the evaluations support the high level of safety that can be achieved with small liquid metal cooled reactors using either approach.

2. Introduction

This report documents results from activities conducted under a formal agreement between the Central Research Institute of the Electric Power Industry (CRIEPI) of Japan and the Lawrence Livermore National Laboratory (LLNL), the Small Liquid Metal Fast Reactor Coordination Program (SLMFR-CP) [1]. The agreement was established under Memorandum of Understanding between LLNL and CRIEPI to cooperate on research programs with a primary focus on advanced technologies on the Nuclear Fuel Cycle technologies, including small fast reactors. [2].

The report focuses on the safety topics identified in Table 2-1. The reported research on these topics was support by ongoing activities at Argonne National Laboratory (ANL) and the University of California, Berkeley (UCB) through informal agreements with LLNL.

Table 2-1 Areas of Liquid Metal Fast Reactor Safety

Research items	CRIEPI	LLNL/ANL
Sodium coolant reactor		
(a) Reactivity insertion events (unprotected transient overpower, UTOP) including an Hypothetical Core Disruptive Accidents (HCDA) analysis	x	
(b) Preliminary investigation of core design with a positive coolant void reactivity	x	x
Lead or lead-bismuth coolant reactor		
(a) Feasibility of continuous operation in case of cladding breach		x
(b) Safety feature at large bubble passing through a core		x
(c) Recommended reactor reflector control and shutdown systems for heavy metal cooled reactor.		x

CRIEPI addressed the first two issues identified in Table 2-1 based on the sodium cooled reactor designs including the Super Safe, Small and Simple (4S) design. The LLNL team focused on small heavy metal cooled reactors with some attention also given to sodium cooled reactors. Each of the topics in Table 2-1 is addressed in the following sections. In addition, a summary and discussion related to the options for U.S. NRC review and licensing of first-of a-kind reactors, such as the 4S is provided in Section 5.

3. Sodium Cooled Fast Reactor

3.1 Reactivity Insertion Events

There was a concern identified in [3] related to the amount of excess reactivity that was included in the 4S design and the potential to postulate reactivity accidents that could lead to core disruption. CRIEPI completed research and development on analysis methods that address this issue and reported on these during two meetings at LLNL. [4, 5]. Appendix A provides the status report presented in the February 2, 2005 meeting.

The methods and analyses are similar to methods developed during the NRC licensing of the Clinch River Breeder Reactor modified to account for the fact that 4S uses metal fuel. The failure of this fuel during the postulated overpower transients is based on research performed by ANL and used in the NRC Preliminary Application Evaluation of the ALMR and SAFR reactors [6, 7]. The preliminary analysis in Appendix A demonstrates good progress on the development of the methods for analyzing the HCDA but is incomplete in addressing all the concerns related to the 4S. The fact that 4S has a negative void worth coupled with the results of the preliminary analysis concerning fuel failure and motion supports that no energetic HCDA is likely to occur. The scope of the analysis does not include control failure that may cause a transient overpower accident. This initiator, although very improbable, also needs to be addressed. The high reactivity worth of the controls and the operational characteristics of the slowly moving reflectors may introduce fuel failure sequences different from those already studied. Also the need to reposition the central absorber rod at selected points in the core life presents another opportunity for a control failure initiated transient involving a high worth control assembly. This operation should be conducted with the reactor shut down and far sub critical, similar to the original startup operation.

3.2 Positive Coolant Void reactivity

Typically, light water reactor designs operate with negative coolant void reactivity worth, so that if coolant is removed from a region of the core the reactor power is reduced. This characteristic is important to the control of Boiling Water Reactors (BWRs) and has received increased attention in all reactor designs since the Chernoybl reactor accident. The 4S design efforts have been conducted using a criterion that the core shall retain a negative void worth throughout the core life. The void patterns used to evaluate the design options against this criterion have been selected to maximize the void worth without consideration of the physical processes that might produce the voiding. This is a very conservative requirement for several reasons. First liquid metal cooled reactors operate at temperatures far below their boiling points at essentially atmospheric pressure. Thus, voiding from coolant vapor formation is not likely to occur. Gas formation and entrainment in the coolant, an alternate postulated path to voiding, is also very unlikely since the sources of such gas entrainment are unlikely to be sufficiently large and coherent to generate positive reactivity effects. Voiding as a consequence of fuel failure is a third mechanism of voiding. This can be the consequence of release of retained fission gas from the fuel pins or from local over heating from overheated fuel during postulated accidents. These sources are also very unlikely to produce voiding in such away as to insert large amounts of positive reactivity. In spite of these facts, CRIEPI and Toshiba have continued to use this conservative criterion because of the extensive experience Japanese utilities have with BWRs and objective that 4S have clearly demonstrable improved safety characteristics over alternative designs.

To achieve a negative void worth in a liquid metal reactor requires the core design to either be very flat (core diameter much larger than the fuel height) or very tall (fuel height much larger than core diameter). Either configuration results in a high level of neutron leakage and a negative reactivity effect with the assumption of voiding. These approaches move the core design away from the more optimum economical configuration of a core height more nearly equal to the core diameter. MONJU has a core height that is slightly more than half the effective core diameter, more typical of liquid metal cooled reactor designs.

The current U.S. GEN IV lead cooled fast reactors, similar to earlier (S-PRISM, SAFER) U.S. sodium cooled fast reactor designs seek to maintain a low (in some cases less than $1 of positive void worth in the most geometrically positive void configuration) void worth and have not applied the constraint used in 4S. In the case of the GEN IV small lead reactor designs this has lead to the ability to design reactors with conversion ratios very near 1.0 and by doing so requiring a very small margin (less than $1) of excess reactivity when at full power. This is also a very significant safety characteristic and has been the preference in U.S. designs from both a safety and performance considerations.

Table 3-1 provides a comparison of small low power density sodium and lead cooled reactor core designs that were developed by UCB [8]. From this comparison it is seen that core designs with very similar reactivity (except for coolant expansion coefficient and P/D) and performance characteristics can be obtained with both coolants.

Table 3-2 provides a comparison of the safety implications of voiding in both cores. The volume of gas from a single failed fuel rod is about the same but the much higher pressure of the hydrostatic head of lead keeps the gas bubble from expanding. The volumes shown in Table 3-2 are for the volumes of the displaced coolant, which is the volume of importance from a reactivity standpoint. Thus one of the more probable sources of voiding will have much less of a reactivity effect in the lead cooled reactor. The power increases are estimated on a conservative basis and are readily protected without damage to the core. However, if a larger number of pins were to fail simultaneously then more serious consequences would occur in the sodium cooled core. This provides some motivation for maintaining smaller void worth in the sodium-cooled core than in the lead-cooled core.

Table 3-1
Small Low Power Density LMR Core Design Parameters

Performance Parameters	Na Coolant	Pb Coolant
P/D ratio	1.16	1.36
Average volumetric power density (W/cc)	35.8	26.0
Pu wt%	11.87	12.20
Burnup reactivity swing (%dk)	0.195	0.221
Peak-to-average power density	2.017	1.829
Average burnup after 20EFPY (GWd/tHM)	51.0(51.0[a])	50.80(53.1[a])
Peak fast neutron fluence(n/cm^2) at 20EFPY	4.012E+23	3.829E+23
Initial conversion ratio	1.0380	1.0446
Doppler effect (dk/dT)	-6.5842E-06	-5.2442E-06
Axial fuel expansion (dk/dT)	-4.4541E-06	-4.6379E-06
Coolant expansion (dk/dT)	+5.8844E-06	+1.6747E-06
Grid plate radial expansion (dk/dT)	-8.2501E-06	-8.0679E-06
Void reactivity (%dk)		
Voiding inner core/+gas plenum	+1.907/+1.685	+2.718/+1.516
Voiding middle core/+gas plenum	+0.654/+0.521	+0.689/-0.068
Voiding outer core/+gas plenum	-0.133/-0.202	-0.694/-1.045
Voiding total core/+gas plenum	+2.469/+2.048	+2.555/+0.424
Peripheral absorber reactivity worth (%dk)	1.099	1.990
Central absorber reactivity worth (%dk)	4.174	4.138
Peripheral + central absorber worth (%dk)	5.545	6.811
Total heavy metal inventory (kg)	17507	17507

a Corresponds to peak fast neutron fluence of 4.0×10^{23} n/cm^2

Table 3-2 Safety Implications of Voiding

Estimated Parameters	Lead cooled core	Sodium cooled core
Volume of gas released from one fuel rod (hot pin) at EOL (cc)	314	3810
Moving velocity of coolant (m/sec)	0.447	0.651
Passing time through active core (sec)	2.796	1.920
One unit channel coolant volume (cc)	248.3	115.6
Number of channels corresponding to the amount of the gas	1.3	33
Void fraction (the void is assumed to be only in inner core)	0.0384%	1.0028%
Reactivity inserted by the above void (pcm)	1.022	17.88
Power increase (%) due to prompt jump	0.3	5.2

4. Lead/Lead-Bismuth Cooled Fast Reactor

4.1 Continuous Operation with Clad Breach

In most reactor systems, especially LWRs, it is not desirable to run with cladding breaches in which fission products are leaking and the coolant has access to the fuel. Typically when this condition is detected the fuel assembly with the damaged cladding is located and replaced. This operation helps to control the level of radioactive contamination in the primary system and also simplifies core safety analysis by removing the need to establish the safety adequacy of the core to operate with cladding breaches. It also simplifies the refueling operations by reducing the transfer of radioactivity that would build up in the coolant throughout the fuel handling and storage systems. Fortunately, it has been demonstrated that LWR fuel can be manufactured with a very high reliability that reduces the need for removal of leaking fuel to an acceptably low frequency. The LWR fuel will typically resides in the reactor for 3 to 4 years without failure. Also, because the reactor is shutdown for refueling about every 18 months there are many opportunities to remove damage fuel without unscheduled shutdowns.

In the case of small reactors with core lifetime objectives of 30 years there is a greater incentive to continue to operate with a limited amount of breached fuel. It is desirable for both economic and proliferation reduction reasons to eliminate the usual onsite fuel handling and storage systems in these long life designs. This means that when cladding breaches are identified that are unacceptable, it will require a lengthy unplanned shutdown period to replace the core or in some designs the reactor assembly.

Although it will still be desirable to limit the extent of failed fuel in the small long life reactors, there are significant differences between LWRs and sealed long life liquid metal cooled systems that support a realistic expectation to safely operate with some amount of breached fuel.

1. The primary system is low pressure and the primary coolant boundary is required to be absolutely leak tight, especially if the coolant is sodium. Penetrations of the primary system boundary are usually restricted to the reactor cover which is not in contact with the coolant. Thus, fission products leaked from the fuel are confined to the primary system and its connected cover gas and coolant service systems that are also low pressure and leak tight.
2. There are differences between lead cooled and sodium cooled systems that may impact considerations about operating with breached fuel. The lead alloy cooled systems intend to eliminate the intermediate coolant loop required in sodium systems. The high pressure boundary containing the working fluid (water, helium or S-CO_2) of the power conversion system penetrates the primary system that then requires an overpressure relief system on the primary system. The implications this may have on operating with breached fuel may not be substantial but will need to be considered.
3. The sodium coolant has been demonstrated to be very benign, easily maintained and producing no damage to the coolant boundary material over 30 years of operation in the case of EBR-II. Thus the integrity of the coolant boundary containment is assured. This same performance has not been demonstrated with lead alloys but may be possible if oxygen control in the coolant can be perfected or if economical materials resistant to heavy metal corrosion can be developed.
4. Compatibility of the coolants with the exposed fuel is clearly assured in the case of sodium and metal fuel and appears to be similar for lead and UN. These fuels will be bonded with the coolant material and therefore are exposed to it statically throughout life. The ability of the fuel to resist potential deterioration from many years of exposure to flowing coolant will need to be confirmed.
5. The fact that fuel will be replaced at intervals of 10-30 years and will not be stored on site means that there should be greater tolerance to handling damaged fuel. There are issues with this depending on the design approach. If the whole reactor is replaced then the presence of failed fuel may not be as much of a challenge as when the whole core is replaced. If individual fuel assemblies are replaced, but not stored on site, similar to the current LWR operations, then the assemblies with failed fuel may be packaged for shipment in containers different than used for non-failed fuel. The same can not be said for a whole core replacement. If the core contains leaking fuel it may require a double containment for licensed shipment.

The actual power operation of the fuel may be the least challenging issue. The key will be the resistance of the exposed fuel to flowing coolant over the long exposure. The fact that the long life fuel is operating at low linear power levels and will achieve relatively low burnup during its long life should reduce the potential for fuel deterioration. This is an area that will require demonstration with bounding or prototypical conditions.

4.2 Large Bubble Passing Through Core

Qualitatively considerations of a large bubble passing through the core are considered in this section. There are no identified realistic mechanisms for a large bubble to pass through the core. Speculative mechanisms can be postulated such as a large release of fission gas from the fuel. This case requires highly speculative fuel failures to occur to produce gas at the core inlet in a coherent manner. This is inconsistent with the physical reality. Entrainment of cover gas and accumulation of a bubble at some location in the coolant path until its release can also be postulated but this is also not realistic. There would have to have been an incredibly gross error in the reactor design for this situation to occur. Both the entrainment of gas and avoidance of flow passages that could accumulate gas are eliminated by design. Vapor formation in the core during a severe mismatch in power and flow is also a postulated mechanism for forming a bubble in the core but this scenario has many complexities and is related more to scenarios for severe accidents than a bubble passing through the core. The temperature margin to vapor formation in the core is many hundreds of degrees in both sodium and lead alloys. A power to flow mismatch that produces vapor requires unrealistic assumptions such as instantaneous stoppage of flow at full power. This is not possible in the current lead alloy designs that employ natural circulation. In the case of 4S it requires postulated failure in both primary coolant pumps and failure to scram. This extremely unlikely failure sequence is predicted to lead to severe fuel disruption and reactor shutdown prior to formation of significant voiding and therefore is not considered further here.

Answering the “what if” question of what happens if a large bubble passes through the core provides an assessment of safety margin the core has to a combined thermal hydraulic and the reactivity disturbance. The reactivity effects of the bubble will depend on the void worth throughout the core. If the void worth is everywhere negative, as in 4S, then the effects of the bubble are more easily predicted because the power will be known to decrease until the bubble begins to exit the core and the coolant reenters. In this case, which applies to 4S, the power will decrease some depending on how negative the void worth is and will return to full power when the coolant reenters. The impact of the event on the fuel will depend largely on the time required for the bubble to transit the core. A large bubble will not pass through the core as easily as the liquid coolant because of the larger resistance of gas flow through the fuel assemblies. Assuming that the inlet pressure remains constant, the time for the bubble to transit the core may be several seconds. This time may be of the same order of magnitude as the time required for the fuel decay power to reduce and thus the reduced cooling from the bubble gas may result in a fuel temperature increase even though the power is decreasing. At the time the coolant starts to reenter there will be positive reactivity additions and the power will begin to increase while a portion of the bubble is still in the core. The level of the fuel temperature increase will depend on the bubble core transit time and power level. The power to heat removal ratio may still be increasing because the heat removal by the gas is so poor. If the transit time is in the order of several seconds the event may result in cladding damage, even though the power is decreasing at the start. However, more realistically the bubble is likely to contain some small amount of entrained liquid and therefore will be a more effective coolant.

The problem becomes more complex in a sodium cooled core with the same power level and temperature, but with regions of positive void worth. To have regions of positive void worth, the core will need to be shorter than is the case of 4S and therefore the bubble transient time will be less. However, the fact that there will be positive and reactivity effects as the bubble reaches the mid-plane of the core means that the average power through the transient can likely to be higher. There will also be a combination of positive and negative reactivity effects as the bubble exits and the coolant reenters the core. Given the shorter bubble transient time, it is difficult to predict just how much more severe the event will be without specific analysis. Certainly the amount of positive void worth will be important and prompt criticality must be avoided or the situation is clearly more severe. If the maximum positive void worth is less than $1 the prompt criticality will be avoided and even if it is somewhat greater than $1 prompt criticality will be avoided because the bubble passage will introduce negative reactivity before the positive effects are realized and therefore prompt criticality is avoided even with a somewhat higher void worth. Beyond being more difficult to analyze, it is not obvious that the consequences of a bubble passing through the core with a small (approximately $1) positive void worth will be significantly different than is the case for a core with negative void worth throughout the core. The higher average power level may be compensated by the shorter transient time.

In the case of lead alloy cooled core with the same power and temperature as the sodium cooled core the situation is not much different. However, on the basis of an equal gas mass the bubble is going to be volumetrically smaller by as much as a factor of ten than in the case of sodium and therefore its reactivity effects, both positive and negative, are much smaller. If one arbitrarily specifies the same volumetric bubble size the consequences may be somewhat less because of the larger flow area used in naturally circulating lead and therefore less resistant to the bubble passage and a shorter transit time.

Detailed analysis of this postulated event may be necessary to support licensing reviews or support the risk assessment but it should not be considered within the design basis because of the improbability of its occurrence. It is not clear from these qualitative considerations that the postulated passage of a bubble through the core should form a basis for requiring negative void worth throughout the core. However, from a defense-in-depth standpoint it appears desirable to avoid designs that would achieve prompt criticality from a postulated large bubble passing through the core.

4.3 Recommended Reflector Control and Shutdown

4.3.1 Comparison of the LFR and 4S control Requirements

The questions concerning control and shutdown systems for liquid metal cooled reactors are based primarily on experience with sodium cooled reactor control mechanisms and the design objectives for the core. The choice is also influenced by the fact that within GEN IV only small LFRs are being considered that permit using control elements on the perimeter of the core. If large cores were being considered then the option of using control mechanisms on the perimeter of the core would not be available. Based on the Clinch River Breeder Reactor (CRBRP) experience and the desire to assure a very high reliability in the shutdown function, two separate and diverse mechanisms are provided in the LFRs considered to date and it is very unlikely that this requirement will be relaxed in any future designs. The reliability and therefore redundancy that may be required in the two separate and diverse mechanisms will be determined on the basis of the detailed design characteristics of the mechanisms, and the control and shutdown response requirements.

Systems have been proposed that are essentially the same as the designs being evaluated for 4S, i.e. radial reflectors and a single central shutdown mechanism. Other systems being considered would use neutron poison materials on the perimeter as well as in the central location. In either approach the actuator mechanisms for the central control and the peripheral control would be diverse from one another. In the approach being used in the LFR, core designs with high conversion ratios, the reactivity compensation for burnup is very small compared to 4S. There is less than \$1 of reactivity required for burnup compensation in the current designs. This design approach is possible because there has not been a constraint for a negative void worth. In fact the void worth in the designs being considered may be more than several dollars. This difference between the 4S approach and the LFR approach has a greater impact on the peripheral control mechanisms than on the central control, although both systems have a reduced total worth requirement and there is no need to move the central control more than once at the start of operations. Following the removal of the central absorber rod the peripheral control is positioned to make the reactor critical at the operating temperature. Over the 20-30 year life of the core and once critical at the operating temperature the peripheral control is required to move only a small distance to account for the uncertainty in the burnup calculation. Thus, in the LFR designs the control worth and the control drive requirements are simplified from those required in 4S where the central absorber remains in the core early in life and must be repositioned during the core lifetime.

The other advantage of the LFR approach is that postulated reactivity accidents are limited to the amount of reactivity associated with the uncertainty in the burnup reactivity over the life of the core. This means that even at the beginning of life the reactivity accidents from full power are limited to much less than \$1 and even with a postulated insertion of all available reactivity it may be possible to have a safe inherent shutdown of the reactor. This leads to a reduced safety importance of the active control systems and adequacy of the non-redundant central absorber rod.

4.3.2 Recommendations Concerning Drive Mechanisms

The choice of the drive mechanisms for the LFR must await greater design detail than has been developed to date, however the constraints on the selection of the mechanisms are similar to those for 4S. This is assuming that the outlet operating temperatures are the same for both reactors. Currently the GEN IV LFR has an operating outlet temperature of 560C. The following discussion assumes this temperature will be reduced 50C or more. Most importantly, the designs must meet very demanding reliability requirements. The second most demanding requirement is associated with integrating the drive mechanisms with the small reactor closure assembly inherent in a small reactor. Other constraints associated with maintaining the temperature and achieving the necessary reactivity worth appear to be a less demanding. Maintaining position and alignment throughout the long life may also be a challenge but the design efforts on the LFR have not reached this level of detail.

The demanding reliability requirements cause one to select drive mechanisms that have a proven operational history such as those used in FFTF or EBR-II or those developed for CRBRP. There will likely be a need to modify these designs to fit in the constrained space on the small reactor closure. There is also the complication of the close proximity of the working fluid penetrations in the closure. In the current LFR GEN IV design the working fluid is supercritical CO_2 but water/steam has not been ruled out. In either case the piping is high pressure, versus the low pressure nozzles of the secondary sodium system in the 4S. Because of this complexity there is incentive to consider innovative approaches that would simplify the reactor closure assembly design and manufacturing. The electromagnetic mechanisms being developed by Toshiba is one such system. However, there are concerns about the reliability and development cost associated with these devices. The development to date has been limited and would require considerable more effort that could ultimately prove unsuccessful. Thus, the deployment schedule for the reactor prototype will effect the decision. The deployment schedule for a sodium cooled reactor can be much closer in time than for the LFR which needs to give more attention to the selection of coolant, fuel and structural materials and coolant chemical control than is the case for sodium. Selection of the control mechanisms is secondary at this time. Assuming the closure assembly geometry can be resolved with proven mechanical control drives, these would likely be selected. Even, these would require some level of development testing because of the likely design changes required to integrate them into the closure assembly. Innovative drives would need to be considered if proven mechanisms could not be accommodated.

The proven designs may not be acceptable because of the difficult closure interface in an LFR, or even for the 4S to a lesser extent. In addition, the alignment over the long life will be a concern. An innovative design like the electromagnetic system could help if these issues were to be a problem, since the length of mechanical components requiring alignment will be shorter. There may be other innovative designs that can address the closure penetration and alignment challenges, such as systems that use hydraulic or pneumatic pressure to operate the peripheral assemblies. In the case of the LFR the lead is a very good reflector and simply displacing it with void can be used as the control mechanism. This may be difficult against the very high hydraulic head present at the core level with a lead system. All innovative systems will have developmental issues that it would be good to avoid if possible.

In the case of the GEN IV the control decisions are in the future and are likely to be influenced by the decision on 4S assuming it is built and operated before an LFR prototype. If a 4S prototype were to be built in the very near future, we would recommend that proven mechanisms be adapted unless it has been determined that they are unacceptable. If there is both time and money to develop and demonstrate the reliability of an innovative alternative that better optimizes the design then this should continue to be developed and demonstrated. In the case of 4S, because of the requirement for the planetary gear mechanism and the very slow drive rate for the reflector there is really no proven design. This is more an issue for the operational reliability than the safety but never the less makes the recommendation for a mechanical design for 4S tentative on reliability demonstration of the prototype control.

The reflector control is fundamental to the 4S design and a similar design can be used in the LFR. Designs considered to date would not require the extremely slow micro drive used in 4S and could possibly use simpler systems applied in to more conventional designs of sodium cooled systems. It is also possible to use neutron absorber assemblies at the core perimeter. Selection of the type of assembly for the LFR is still open and again could be influenced by the 4S experience if this were to occur prior to building an LFR prototype. Alignment and the guiding the motion of the smaller absorber assemblies is expected to be easier than the challenge of maintaining the true motion of the larger close fitting reflector assembles. Here again the importance of experience will have influence the decision on the LFR.

5. Nuclear Regulatory Review Options

The US Nuclear licensing requirements are specified in Title 10 of the US Code of Federal Regulations (CFRs). Part 50 (10CFR50) and Part 52 (10CFR52) address the licensing and safety documentation requirements for nuclear reactors. These were discussed by the NRC at a meeting concerning the Galena, AK interest application of a 10MWe power plant [8]. Part 50 along with 10CFR20, 71 and 100 contain most of the technical requirements applicable to reactors. The operating nuclear power reactors licensed to date in the US have all been licensed under 10CFR50.

Part 52 specifies procedural requirements to obtain Early Site Permits (SubpartA), Standard Design Certification (Subpart B) and Combined Licenses (COL) (Subpart C) for nuclear power plants. Part 52 also contains Appendices M, N, and O that provide procedures applicable to several approaches to standardization, such as: manufacturing multiple reactors for installation at different sites (Appendix M), application for approval of a standard design for use at multiple sites, standard design review and approval (Appendix O). The difference between Appendix O and Subpart B is that the design approved under an Appendix O review is not certified with a code of federal regulation rule concerning future use of the design on a suitable site. Three LWRs have been designed certified under Subpart B; ABWR by GE, 10CFR52 Appendix A, System 80+

by ABB-CE, 10CFR52 Appendix B and AP600 by Westinghouse Electric, 10CFR52 Appendix C.

Application for a construction permit for an advanced reactor or a request for a pre-application review could be made with the intention of using either Part 50 or Part 52 in future applications. If the ultimate application for construction and operation were made under 10CFR50, one would have the confidence that the regulatory procedures have been used for licensing many nuclear facilities including the 103 currently operating power plants. Application for a construction permit under 10CFR52 is done under Subpart C through which one would obtain a combined license to construct and operate the plant. The Subpart C procedure has not been applied to any power plants and contains uncertainties concerning its use. Currently, there are initiatives under the DOE 2010 program that may lead to the regulations being applied to one or more advanced LWR designs. Application of Subparts B or C to reactor types other than LWRs is permitted but no such application has been completed. The Exelon Generation Company proposed a plan in its pre-application review to apply for a COL for a ten module plant under Subpart C. Following construction and operation it planned to apply for a standard design certification under Subpart B. This was a very aggressive plan and would have required a number of revisions to various parts of 10CFR50 in addition to being the initial applicant to test this part of 10CFR52. Many of the issues with 10CFR50 had to do with the fact that they were planning to request a single license for a plant that consisted of multiple reactors. This possibility has not been addressed in the current regulations. Each reactor on a site has its own license. The pre-application review of the PBMR was terminated without resolving many of the issues. Their approach to building and operating the plants prior to seeking design certification clearly has some merit since operating experience is an implied prerequisite to design certification of advanced designs that are not LWRs.

The ACR700 pre-application review and plan for design certification may be more representative of an approach that could be considered for 4S. Since there is considerable world experience and some US NRC experience with licensing sodium cooled reactors it is similar in experience status to the ACR700. The ACR700 plan relies on considerable worldwide operating experience with CANDU reactors and to some extent on PWR experience for the SGs and the balance of plant. This example is useful only if design certification is the intended objective. It provides little help for seeking a construction and operating license. The ACR700 example provides little encouragement for design certification of an LFR because of the fact that the only operating experience with this type of reactor is with a Russian design. Availability and adequacy of the documentation concerning this experience is highly questionable.

The Safety Evaluation Report (SER) resulting from a pre-application review is not likely to depend on which approach to the operating license one ultimately intends to use. As stated in 52.83 with a few specific exceptions all parts of 10CFR50 apply to COLs issued under 10CFR52. Similarly, the SER from a pre-application review is not likely to differ much if the intention is to ultimately obtain a standard design certification. 10CFR52 contains specific requirements for issuance of a design certification to reactors that depart significantly from LWR designs. These can be expected to apply in a pre-application review of this type of reactor, independent of the type of ultimate licensing application. Because of the NRC desire for improved safety in future plants the following excerpt can be expected to be applicable to any new design that departs significantly from LWR experience.

> "52.47 b(2)(i) Certification of a standard design that differs significantly from light water reactor designs described in (b)(1) of this section, or utilizes inherent, simplified, passive, or other innovative means to accomplish safety functions will be granted only if
>
> (A)(1) The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience or a combination thereof;
> (2) Interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience or a combination thereof;
> (3) Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions; and
> (4) The scope of the design is complete except for site-specific elements such as service water intake structures and the ultimate heat sink; or
> (B) There has been acceptable testing of an appropriately sited, full-size, prototype of the design over a sufficient range of operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. If the criterion in paragraph (b)(2)(i)(A)(4) of this section is not met the testing of the prototype must demonstrate that the non-certified portion of the plant cannot significantly affect the safe operation of the plant."

Licensing the prototype identified in the paragraph (B) option could presumably be completed under 10CFR50 or with a COL application under 10CFR52. The previous qualifications concerning the uncertainties with the currently untried COL would favor 10CFR50. The disadvantage of 10CFR50 and what precipitated the development of 10CFR52 Subpart C is the requirement for two steps of licensing public hearings, one at the point of the construction permit and one and the point of issuing and operating license.

The 4S and a future LFR qualify to be considered under the identified sections of 10CFR52. Therefore the analysis, appropriate test program and experience should be identified in the pre-application review submittals to assure the scope of any additional test program, or full-size prototype test, necessary to complete a license application review is identified. To license a prototype test it will be more straight-forward to use 10CFR50 than to apply for a COL because of the of the innovative design features that may be untested until the prototype operations are completed.

6. References

1. Small Liquid Metal Fast Reactor Coordination Program (SLMFR-CP) Agreement, January 2004

2. Memorandum of Understanding between CENTRAL RESEARCH INSTITUTE OF ELECTRIC POWER INDUSTRY and The Regents of the University of California, Managers and Operators of the U.S. Department of Energy's LAWRENCE LIVERMORE NATIONAL LABORATORY

3. N. Brown, J. Choi, J. Stewart, C. Smith, D. Wade, A. Minato, Joint Preliminary Feasibility Study Final Report, UCRL-TR-201726, December 2003

4. Small Liquid Metal Cooled Reactor Safety Meeting at LLNL, September 23, 2004

5. Small Liquid Metal Cooled Reactor Safety Meeting at LLNL, February 2, 2005

6. General Electric, PRISM Preliminary Safety Information Document, GEFR-00793 UC-87Ta, November 1986

7. U.S. Nuclear Regulatory Commission, "Summary of Advanced Reactor LMR Evaluations –PRISM and SAFR", NUREG/CR -5364 (BNL-NUREG-52197), October 1989

8. NRC Meeting, Discussion of Galena, AK, Interest in 4S Application, February 2, 2005

Appendix A Methods and Analysis of 4S HCDA

1

4S Safety Evaluation and Related Issues, Present Status of CDA and PSA Evaluation

February 2, 2005
N. Ueda
CRIEPI

2

Contents

- 4S safety evaluation procedure
  - BDBE analysis (ATWS, PLOHS, etc)
  - PSA (level 2,3)
- CDA analysis code CANIS
  - Sample output
  - Parametric analyses
- Status of 4S CDA analysis
  - Kinetic parameter

2

Background

- 4S near-term deployment
  - Licensing procedure
  - SSTAR, Alaska

- Safety R&D scoping
  - R&D priority
    - FR generic, 4S unique
  - Required budget

3

4S Safety Strategy

➢ General issues

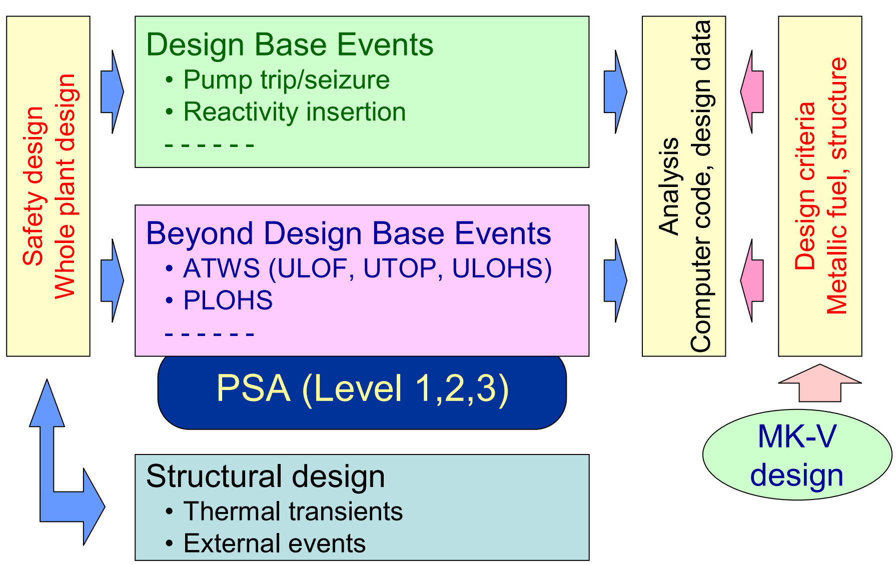
- Follow the previous reactor procedures

➢ 4S Unique issues

- Required to develop new or advanced procedures
  - Metallic fuel
  - Passive shutdown
  - Passive decay heat removal
  - Structure design standard (mod.9Cr-1Mo), ISI

3

Safety evaluation of 4S



Safety evaluation sharing

3

CRIEPI	Toshiba
BDBE ATWS PLOHS Local fault	DBE Design modification Thermal transients
Level 2 Energy generation Source term Level 3 Siting evaluation	Level 1 PSA Identification sequences FT analyses ET analyses

BDBE Evaluation

3

- ATWS
  - CERES: without fuel failure
  - CANIS: fuel failure (CDA)
- PLOHS
  - CERES: passive DHRS (RVACS)
    - Fuel failure with RVACS?
- Local fault
  - Sub-channel blockage
  - TIB ?
    - Analytical model and procedure

CDA Initiator of 4S

3

- ULOF
  - Coolant boiling -> no positive reactivity
  - Reactor power gradually decrease
  - Loss of coolability -> fuel melting
  - Cladding failure -> ex-pin motion (dispersion)
- UTOP
  - Coolant boiling -> no positive reactivity
  - Fuel melting -> in-pin motion (extrusion)
  - Cladding failure (after low power shift)
- PLOHS (loss of RVACS)
  - Loss of RVACS?
  - Similar consequence as ULOF?

CANIS code Development

3

- Previous FBR licensing procedure
  - CDA evaluation was required to assess the risk profile (no cliff edge).
- SAS code
  - Developed by ANL (U.S.)
  - Used in CRBL and MONJU (SAS3D)
- Code for metallic fuel core
  - SAS4A metal version was developed, but it is not available in Japan.
- SAS like code: CANIS by CRIEPI

3

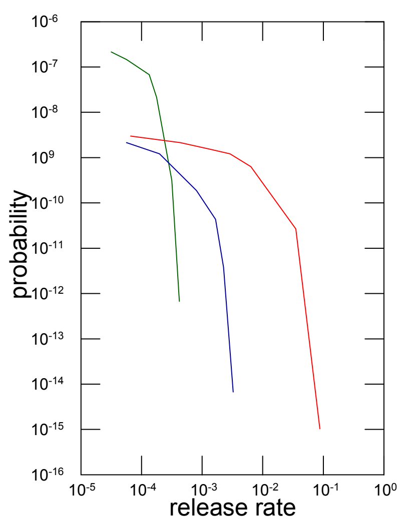
CDA Energetics

- Initiating phase
  - Prompt criticality
- Post initiating phase (transient phase)
  - recriticality
    - Molten fuel pool in-core region
    - Molten fuel pool ex-core region
- Mechanistic energy

Evaluation of molten fuel relocation and heat removal

6

Level 2 PSA



CCDF

- CCDF
  - Event sequence
  - FP category
- FP performance
- Structure response
  - RV/GV
    - GV: containment boundary
  - Roof slab
  - Top dome
- ET analysis
  - Branch identification

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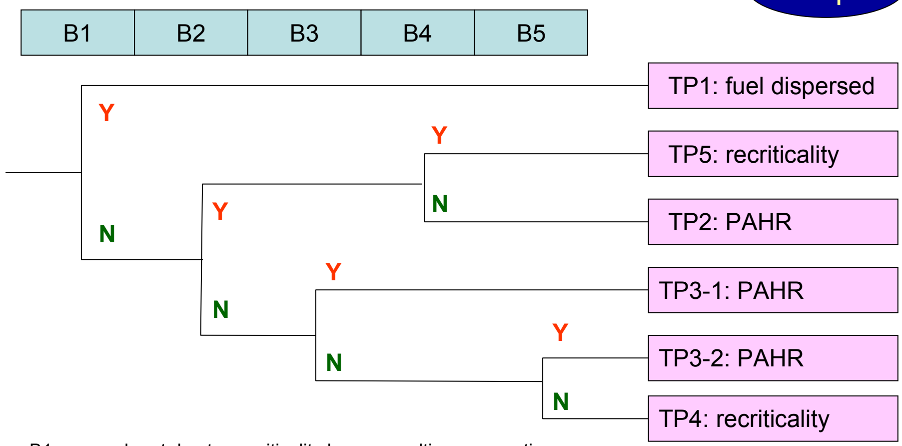
4

Event Tree Analysis

sample



B1: energy burst due to re-criticality by core melting compaction
B2: molten fuel discharge to core inlet plenum in early stage
B3: molten fuel discharge to core inlet plenum in latter stage
B4: molten fuel (fuel particle) cohering by fluid motion
B5: sloshing

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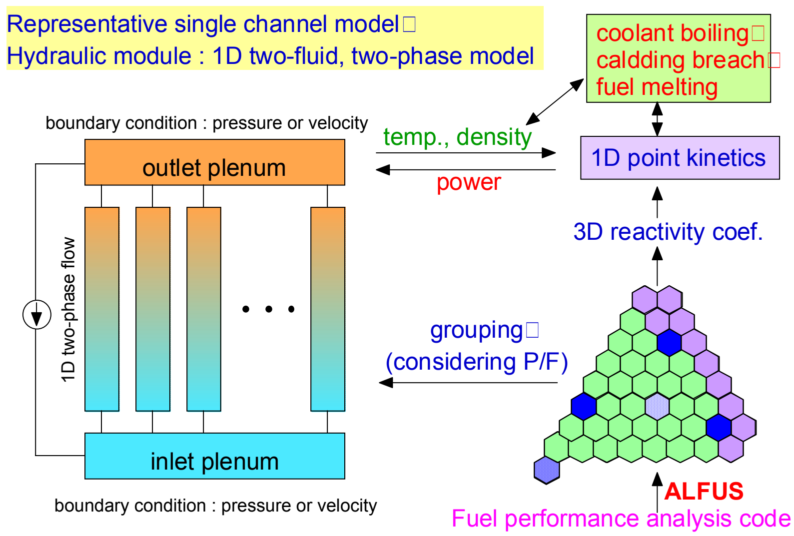
4

Example of CANIS Output

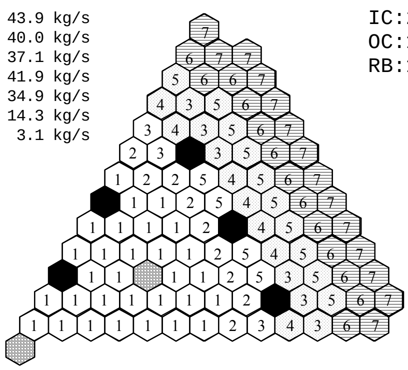
- CANIS code has been adopted to evaluate the upper limitation of the whole core coolant void reactivity.
- This reactivity is the important index of the performance and the safety.
- A large size and a middle size cores were analyzed in ULOF transient.

CANIS Configuration

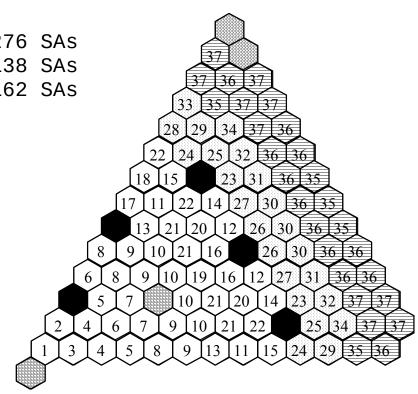


1500MWe Core Configuration

1 : 43.9 kg/s
2 : 40.0 kg/s
3 : 37.1 kg/s
4 : 41.9 kg/s
5 : 34.9 kg/s
6 : 14.3 kg/s
7 : 3.1 kg/s



IC:276 SAs
OC:138 SAs
RB:162 SAs



Flow distribution grouping

Lumping for CANIS

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4

Large Core Specifications

Power	3900 MWth/1500 MWe
Inlet/Outlet tem.	355/510 Deg-C
Core height	100 cm
Configuration	Homogeneous (2 regions)
Peak LHR	478/433 W/cm (*)
Pu enrichment	14.7/20.3 wt%(HM) (*)
Ave. Burn-up	~150 MWd/t
Fuel dia. (inner/outer)	7.1/6.1 mm
No. of fuel elements	271 pins/SA
Duct size	14.77 mm

(*) inner SA/outer SA

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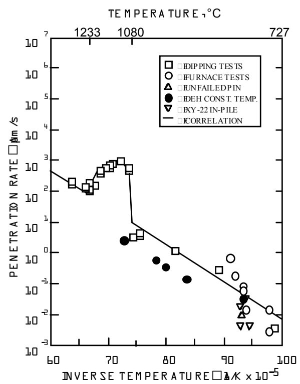
4

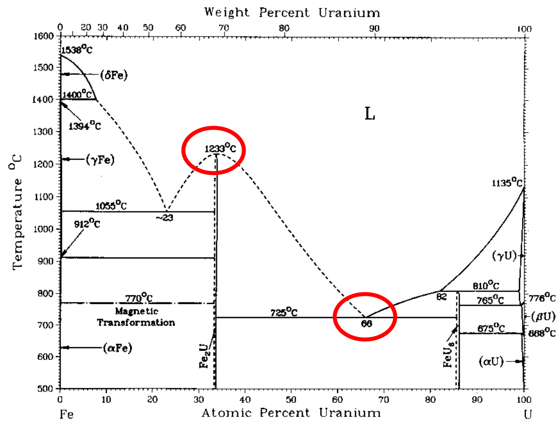
Calculation Conditions in ULOF

- Flow halving time: 5.5 sec.
- Passive reactivity effects:
  - With fuel axial expansion
  - Without core radial expansion
- Coolant void worth: 8~12 $ (original: 7.7 $)
- Coolant boiling temperature: ~950 Deg.-C
- Cladding failure criteria: 1000~1250 Deg.-C
- Fuel melting point: 1180 Deg.-C
- Fuel dispersion model: mechanistic flow model

4

Eutectic and Penetration Rate





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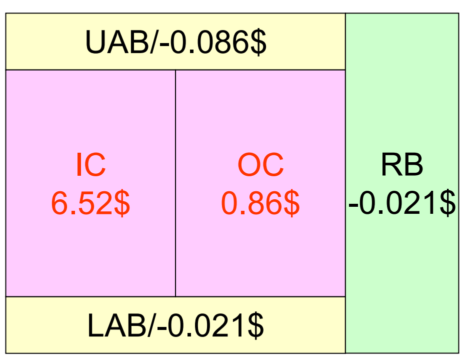
Void Reactivity



Voiding area:
Coolant flow channel
except bonding Na and
gap Na (inter wrapper Na)

In calculation, sodium density coefficients are linear-extrapolated with the factor of 7.7 \$ to 8 ~ 12 \$.

Ref) Voiding area:
Coolant channel and bonding Na
8.3$: diffusion theory
7.7$: transport correction

4

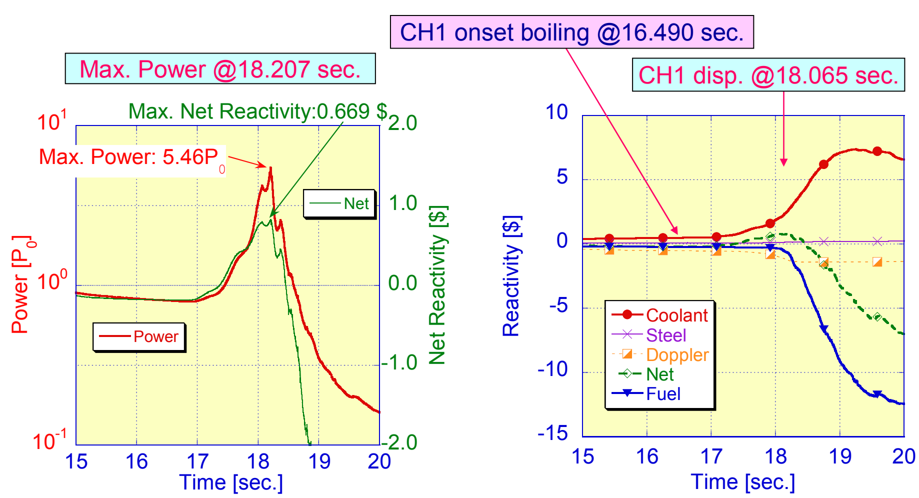
Typical Sequence

- Fuel element failure timing at the highest P/F channel
  - Boiling: 16.490
  - Dryout: 17.402
  - Melting: 17.554
  - Dispersion: 18.065 (cladding failure)

1500 MWe large metallic fuel core
Reference case: 8$ void reactivity, 1200 Deg-C cladding failure

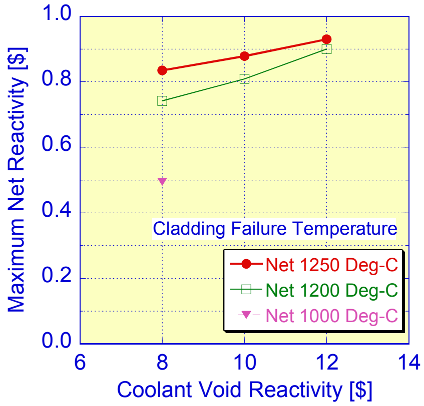
4

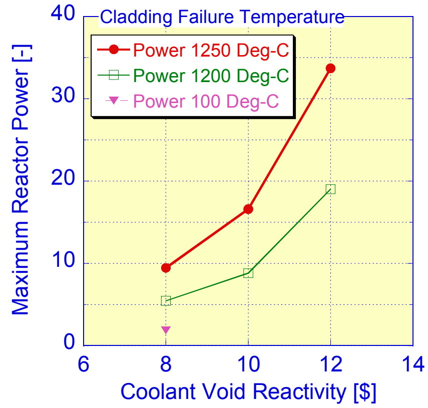
Power and Net Reactivity



Reference case: 8$ void reactivity, 1200 Deg-C cladding failure

Initiating Phase Summary

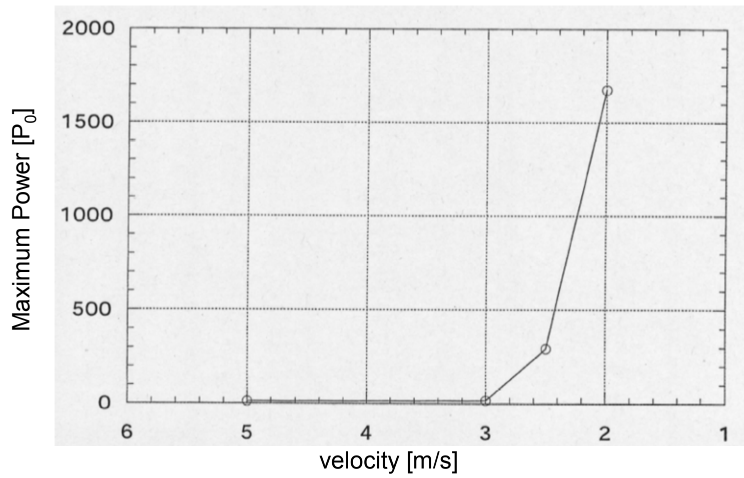




Reference case: 8$ void reactivity, 1200 Deg-C cladding failure

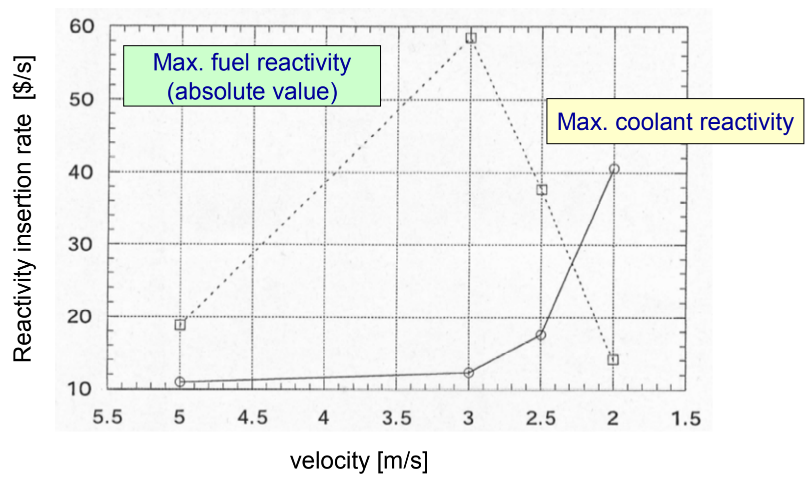
Parametric Survey (1)

- dispersion velocity
  - non-mechanistic model (constant velocity)



Parametric Survey (1)

➢dispersion velocity



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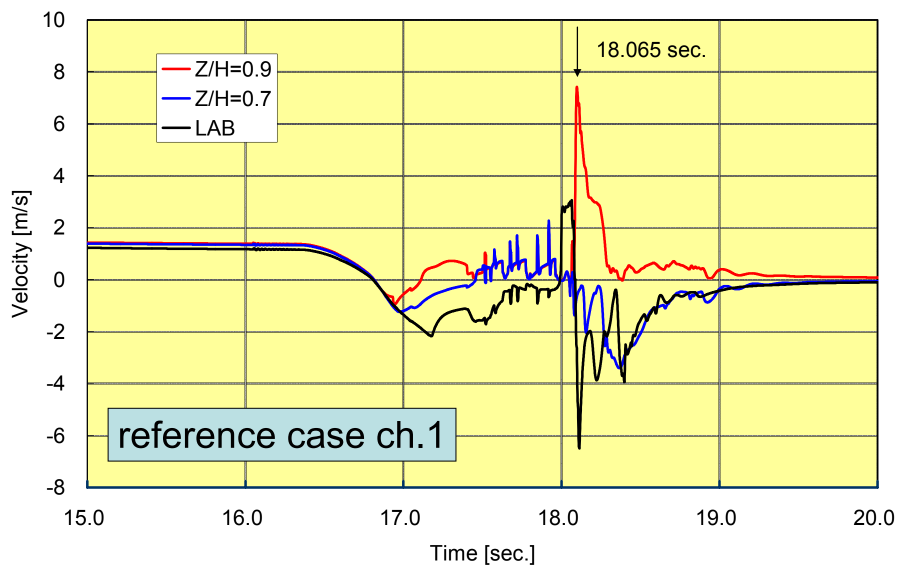
Fuel dispersion velocity



Parametric Survey (2)

➢Open area of cladding breach

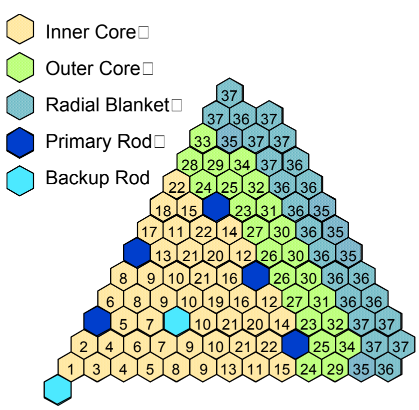
- Reference area: ¼ of coolant flow area

	area X 10^{-3} m^2	P_{max} (P_0)	$\rho_{net,max}$ ($)	Dispersed fuel(*1)
reference	2.485	3.458	0.6684	~100 kg
50%	1.2425	3.485	0.6896	~100 kg
10%	0.2485	8.200	0.8192	~60 kg

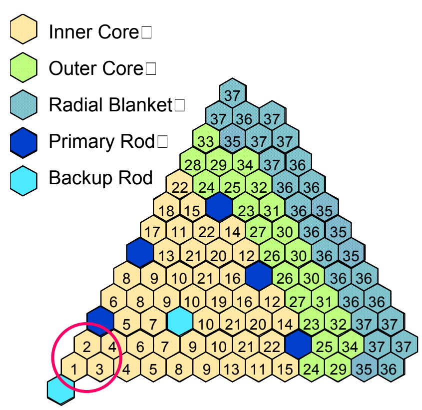
(*1) amount of dispersed fuel in 0.5 second after cladding breach

4

Lumping condition (1/2)



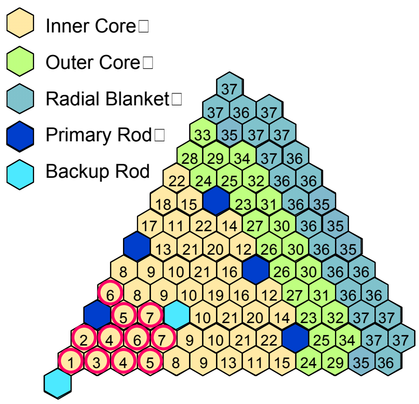
reference



Case 1

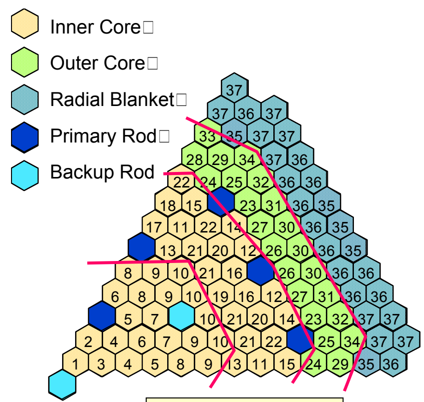
chs 1~3 lumped

Lumping condition (2/2)



Case 2

chs 1~7 lumped



Case 3

chs 1~10 lumped
chs 11~22 lumped
chs 23~34 lumped

Parametric Survey (3)

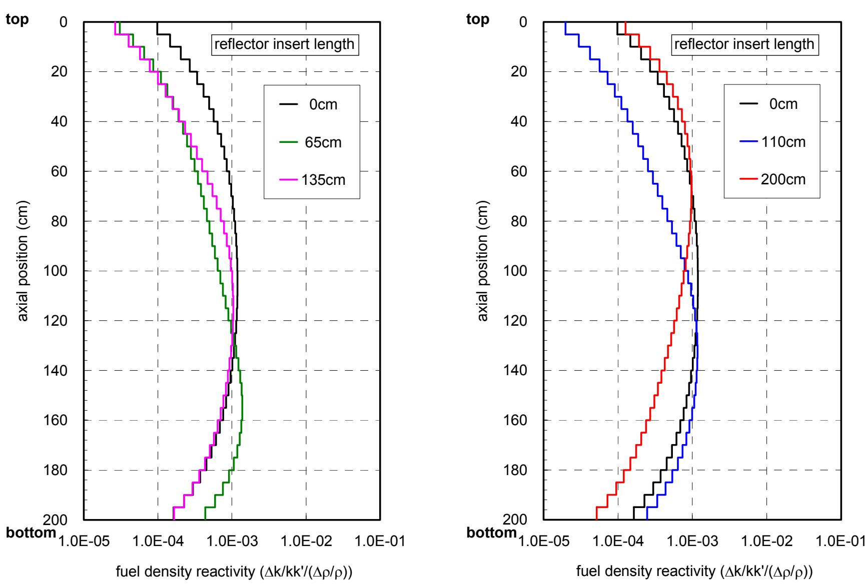
- Coherency
  - Increasing lumping assemblies

	coherency	P_{max} (P_0)	$\rho_{net,max}$ (\$)	$\rho_{cl,max}$ (\$)
reference	-	3.458	0.6684	5.124
Case 1	3/52 of IC	4.9144	0.7487	5.157
Case 2	11/52 of IC	10.052	0.8247	6.285
Case 3	4 zone	38.983	0.9327	6.706

Target Analytical Condition for CDA

- 4S core is controlled by reflector
  - Axial fuel worth distribution is strongly affected by the reflector position.
    - Fuel worth of standard core is not so changed during core life.
  - There may be severe reflector position in ex-pin fuel dispersion.
  - In-fuel motion reactivity is not affected.
- Coolant reactivity feedback
  - Large negative in BOC and nearly zero in EOC

Fuel Relocation Reactivity



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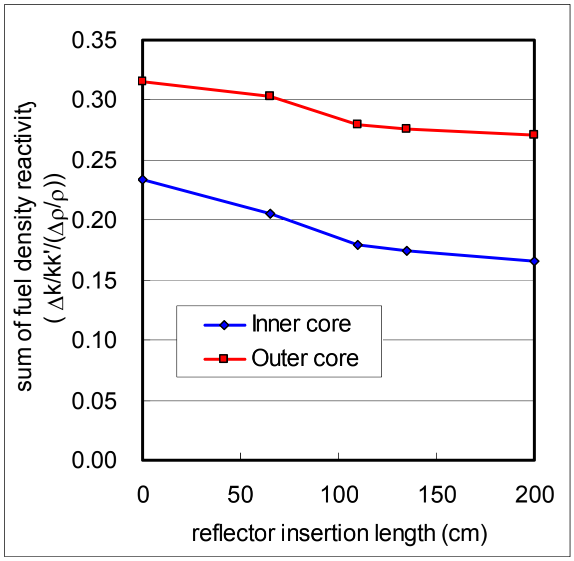
Effect of Reflector Position

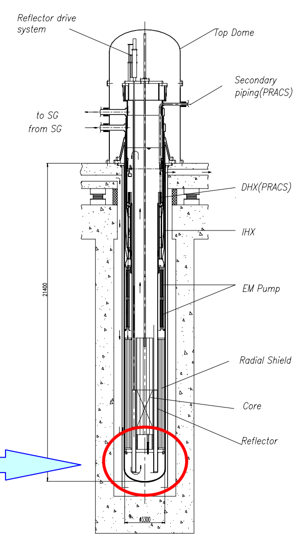


4

Molten Fuel Relocation

Recriticality in Post IP

- 4S core features
  - High Pu enrichment
    - easier to critical
  - Tall configuration
    - handicap in axial compaction
  - Lower configuration
    - cramped space
    - easy to crowd



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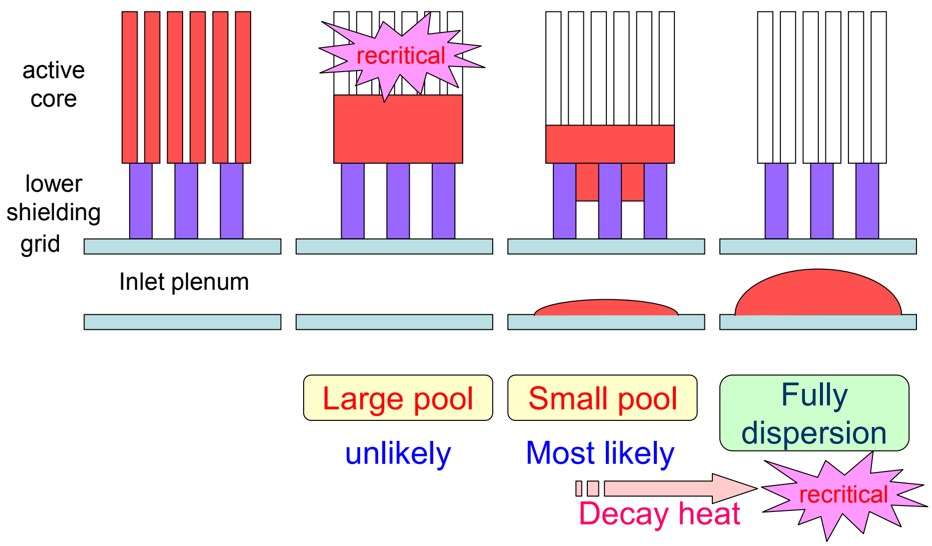
Post Initiating Phase



4

Future work (1/2)

- 4S initiating phase analysis by CANIS
  - ULOF
    - Non-energetic procedure
    - Fuel melting fraction may depend on RVACS
  - UTOP
    - Power transient / quick fuel dispersion
    - Extrusion? Dispersion?
  - PLOHS
    - Loss of RVACS?
  - LORL
    - RV/GV double penetration

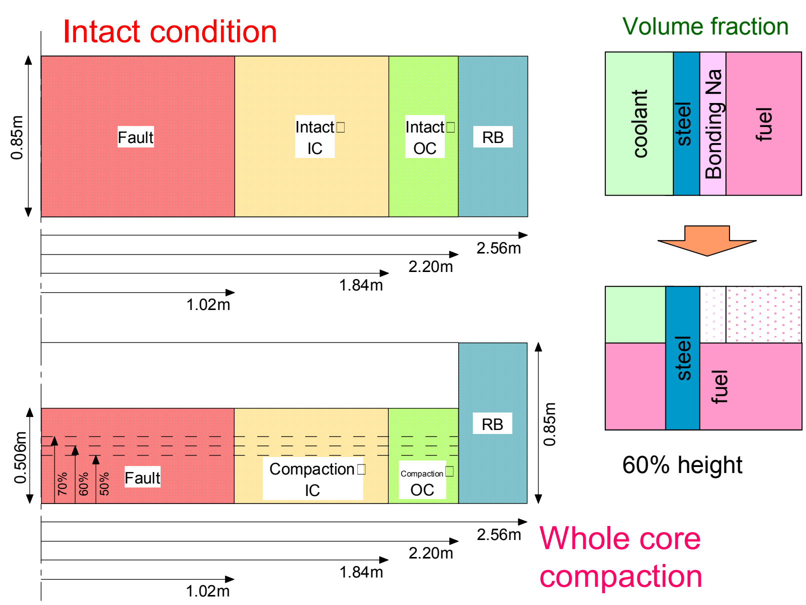
4

Future work (2/2)

- Level 2 PSA
  - Branch identification in each initiating event
  - Estimation of branch occurrence
    - Energy estimation
    - Structure response
- Level 3 PSA
  - Research generic analysis procedure
  - Site condition

4

Axial Core Compaction (1500MWe)



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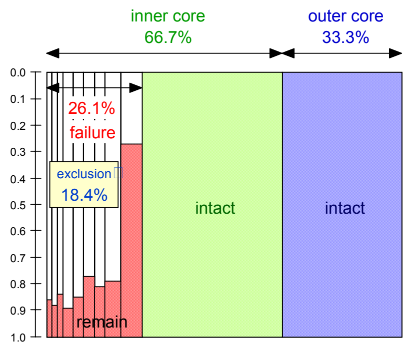
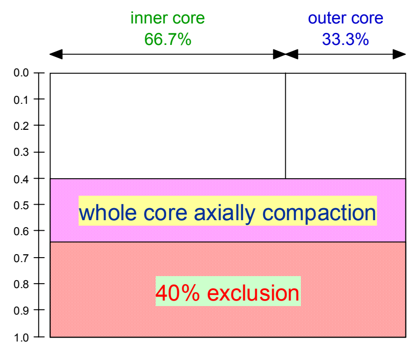
Fuel Dispersed Condition



CANIS result



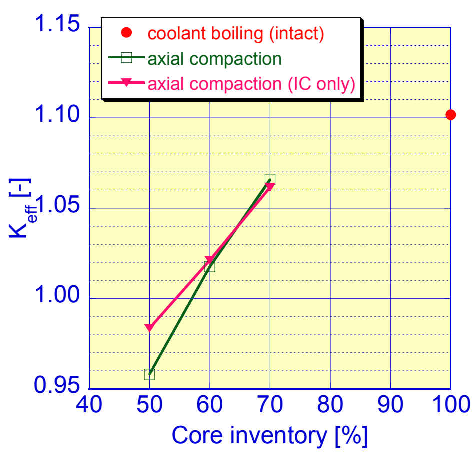
Deterministic compaction

- CH1-9 faults from 1 to 9
- CH9 is fault relatively latter stage

- Core height becomes 60% after compaction.
- 40% fuel inventory is required to maintain sub-criticality.

4

K_{eff} Analytical Results



Analytical condition

- Code: MCNP 4C
- Nuclear data: JENDL-3.2 (Am & Cm are from 3.1) 300K
- Histories: 2,500,000
- Fuel density: swelling from 75% to 100%
- Fuel composition: EOEC without FP
- Structure: no mixture with fuel
- Fuel dispersion area: coolant flow path
- Doppler effect: correction to steady state temperature

About 40% fuel inventory is required to be excluded to maintain the sub-criticality after core axial compaction.
Larger inventory than above is required in only inner core compaction condition.

4

Summary(1/3)

- 4S safety features
  - Well designed
    - Active reactor shutdown and protection system
    - Passive decay heat system
  - Reliability development
    - Passive shutdown mechanism
    - Passive decay heat capability
- Analytical codes
  - Well developed by real test
    - Passive system modeling

4

Summary (2/3)

- CDA consideration
  - Decision of design policy against CDA
    - Why is CDA evaluation required?
    - Is there any inconsistency with 4S safety design policy?
  - CDA criteria
    - Do we step into recriticality problem in post IP?
  - CDA analytical tools
    - Do we have sufficient tools verified by real data?
  - Design modification
    - Is it a tug of war between licensing body and us?

4

Summary (3/3)

- Strategy for early deployment
  - To find the issues to clarify in each stage
    - Demonstration reactor (to establish new technologies)
    - Commercial reactor
  - Manpower, Budget and Facilities

We will try to gain the maximum results as far as we can!

5

Summary

- Peak power and the maximum net reactivity are strongly affected by the cladding failure condition.
  - Time duration between
    - the boiling onset
      - positive reactivity insertion and
    - the first cladding failure (fuel dispersion)
      - Negative reactivity insertion.

5

Summary (contd.)

- Neither prompt criticality nor energetics is predicted.
- The upper limitation of the whole core void reactivity is evaluated to be 8$ with conservativeness.
- This value has been used in “Feasibility Study on Commercialized Fast Reactor Cycle Systems” carried by JNC and Japanese utilities.
- The 6$ limitations is applied to MOX fuel cores.

5

Safety Analysis Code

- Plant dynamic analysis code
  - SASSYS (ANL) flow-network
  - SSC (ORNL) flow-network
  - CERES (CRIEPI) 3D plenum
- CDA analysis code
  - SAS4A (ANL) initiating phase
  - CANIS (CRIEPI) initiating phase
  - SIMMER-III/IV (JNC) transition phase

5

Plant Dynamics Code "CERES"

CERES is a multi-dimensional plant transient simulation code for LMRs developed by CRIEPI.

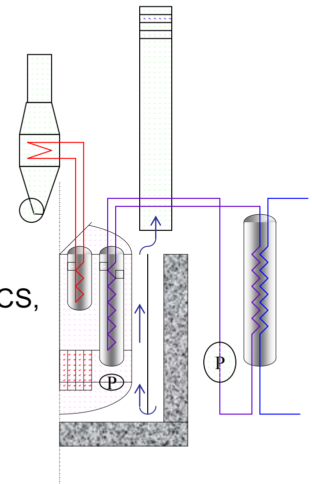
Core Formed with plural average (nominal) channels. One-point kinetic model
2D reactivity feedbacks
Hottest pin model (for safety evaluation)

Hot /Cold plena Multi-dimension

Components (one-dimension)
Pipes, Pumps, SGs, IHXs, DHRSs (DRACS, PRACS, RVACS, IRACS), ACSs

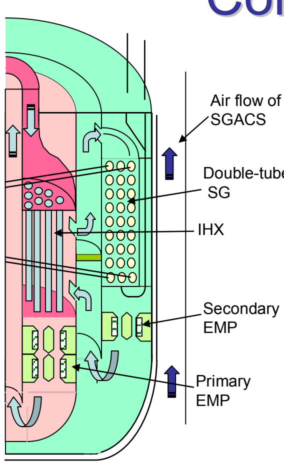
Operation model: Interlock system,
Plant control system

Fluid : Sodium, Water, Lead, Lead-Bismuth Eutectic



5

Reactor System Concept

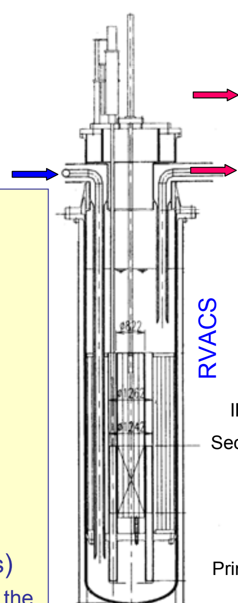


Concept of the Integrated equipment

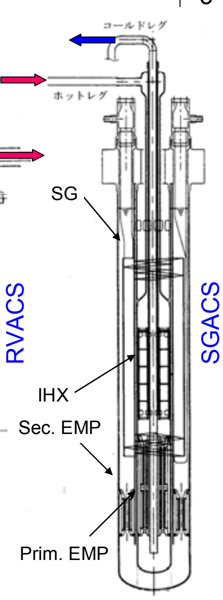
Compact core
⇨ Small R/V
⇨ Loop type

One loop

DHRS
small power
⇨ RVACS
⇨ SGACS
(Natural air cooling outside the vessels)
⇨ Remove the A/C of the PRACS



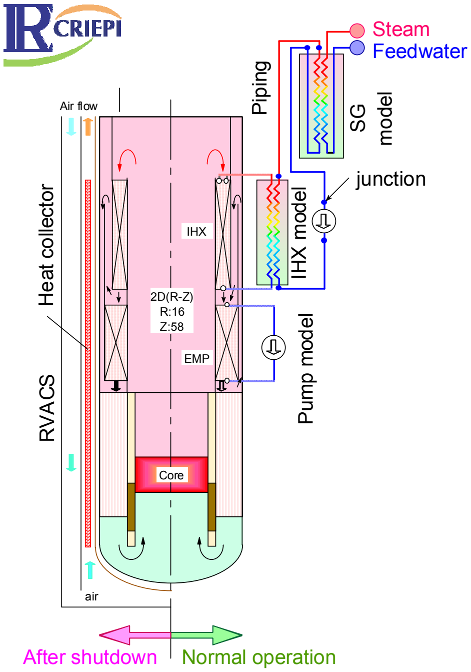
Reactor vessel



Integrated component

5

CERES Modeling



Core
Reactor plenum (2D or 3D)
Primary components
Secondary components
Feed water line/steam
DHRSs

Point kinetics
R-Z reactivity coefficients
Passive reactivity models

Hot channel model

1.5m x 17.86m (16 x 58 nodes)

5

Active Reactor Protection System

- Shutdown system
  - Lowering reflectors (quick and automatic)
  - Absorber rod insertion (automatic? manual?)
- Reactor trip signals
  - Simplified design
    - Reactor power (NIS), EM pump electricity
    - Independency in two shutdown systems
- Cooling system for RPS
  - Passive system

5

Passive Shutdown

- Core support grid expansion
- Core radical expansion (bowing)
  - complex mechanical phenomena
  - Radiation effects
  - Plastic thermal deformation

4S has only 18 sub-assemblies!
Mechanical behavior can be confirmed by real test.

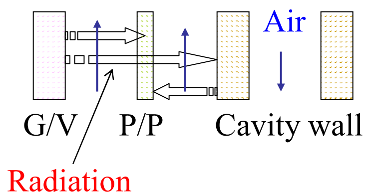
5

Passive Decay Heat Removal System

- RVACS
  - Fully passive (natural convection: Na and air)
  - How to demonstrate the capability through 30 years operation
  - Guard vessel is the containment boundary
    - How to fight against an assumption of both RV and GV failures
- SGACS (SG Air Cooling System)
  - Same concept as RVACS but in IHTS
  - Large surface required

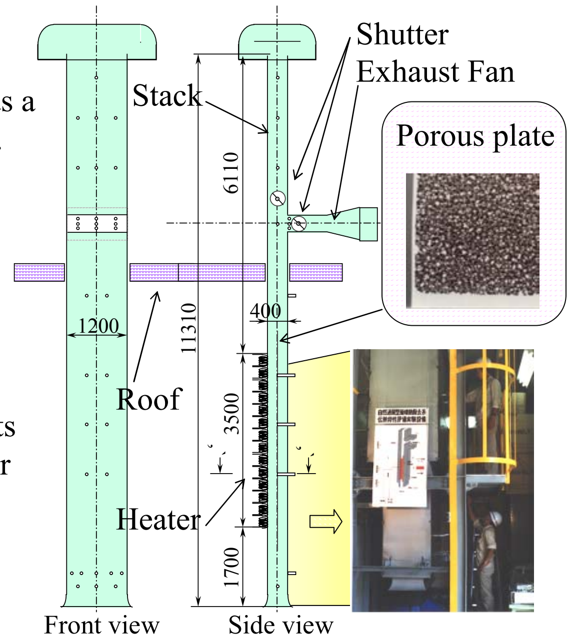
RVACS

A porous plate is proposed as a heat collector in the air riser.



Based on these experiments correlation for heat transfer have been proposed.

Experimental apparatus in CRIEPI



CDA Evaluation

- Objectives

  To evaluate the potential to eliminate the re-criticality of the metallic fuel core in hypothetical CDA sequence

  - Prevention of the prompt criticality in initiating phase
  - Elimination of the re-criticality in post initiating phase

CRIEPI

5

CDA Problem in 4S

- 4S reactor is designed to aim to exclude the CDA discussion.
- Licensing body may require a CDA evaluation for 4S reactor as a quasi-deterministic assumption (a priori assumption).
- In presence, no CDA evaluation has been done.
- Assumptions and analytical conditions are to be discussed form the point of view of the safety policy of 4S.

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Fuel Performance Data

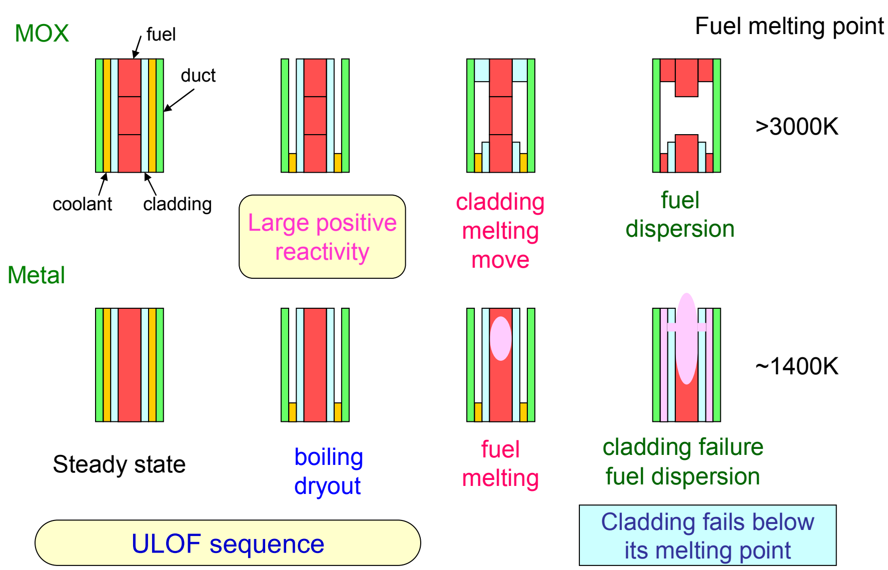
- Irradiation data
  - EBR-II
  - FFTF (U-xPu-10Zr/D9, U-10Zr/HT9)
- Transient fuel motion data
  - TREAT M series
    - M2-4: U-5Fs/316SS
    - M5-7: U-19Pu-10Zr/D9, U-10Zr/HT9
  - Whole-Pin Furnace
    - FM-1: U-10Zr/HT9
    - FM2-6: U-19Pu-10Zr/HT9
    - FM7: U-19Pu-10Zr/316SS

Fuel Element Degradation

- Cladding failure
  - Creep failure with thinning by eutectic
- In-pin fuel motion
  - Molten fuel cavity formation and extrusion
- Ex-pin fuel motion
  - Molten fuel dispersion
    - Penetration length
    - Interaction with coolant
    - FP gas effect

Analytical code must be verified by safety tests up to failure with irradiated fuels.

CDA Progression Differences



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Comparison of CDA Feature

- MOX fuel
  - Molten cladding relocation -> positive reactivity
  - Fuel starts to move after melting or breaking
- Metallic fuel
  - Failure timings are very close.
  - T_{bp} (1250K) ~ T_{mp} / cladding failure (~1450K)
- 4S reactor core
  - 18 sub-assemblies -> strong coherency
  - Large positive reactivity insertion rate

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Code Verification for 4S

- Key failure behaviors
  - In-pin fuel motion (extrusion)
  - Cladding failure mechanism including eutectic
  - Ex-pin fuel motion (fuel dispersion)
    - LOF type experiment (furnace heating tests with sodium?)
- Is irradiated long fuel pin required?
  - Using EBR-II irradiated fuel pins
    - Extrapolation to 4S fuel pin (2.0~2.5m)
  - Experimental accuracy
  - Sufficient number of fuel pins? (U-Pu-Zr/HT9)

Appendix B Comparison of the Lead and Sodium Cooled Long-Life Low-Power Density Cores

A CONSISTENT NEUTRONIC COMPARISON OF The LEAD and SODIUM COOLED LONG-LIFE LOW-POWER-DENSITY CORES

In Support of the CRIEPI Small Liquid Metal Fast Reactor Coordination Program

Ser Gi Hong, Ehud Greenspan
Department of Nuclear Engineering
University of California, Berkeley

LLNL/ANL/UCB-CRIEPI Meeting
Livermore, CA 2/2/05

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1

Introduction

- Study done for the Encapsulated Nuclear Heat Source (ENHS). Reference design is for 125MWt
  - Low-power-density blanket-free core
  - Natural circulation
- A consistent core neutronic comparison of Lead and Sodium cooled cores
  - Same values of power, fuel rod dimensions and linear heat generation rate
  - The core life is constrained by the clad radiation limit (i.e., peak fast neutron fluence of 4.0x10E+23 n/cm^2)
  - The lattice P/D ratio is determined so as to have nearly zero (~1$) burnup reactivity swing over core life.

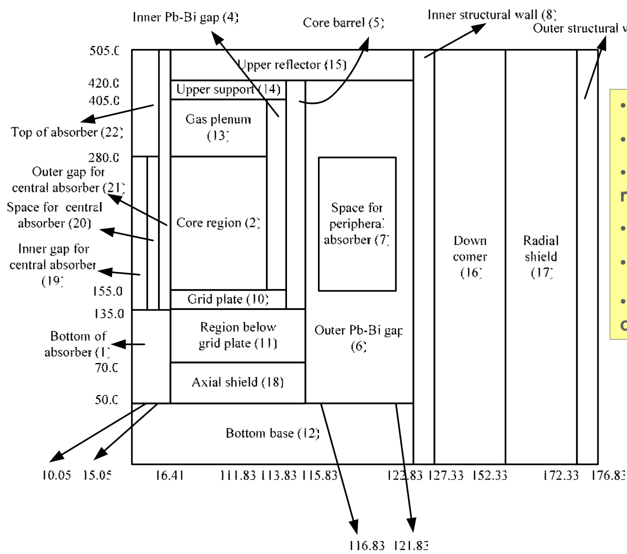
3

Computational Methods

- The ENDF/V-B based multi-group cross sections are prepared with NJOY and TRANSX Codes.
- The core depletion analysis is done with REBUS-3/DIF3D using 80group DIF3D calculations in RZ geometry.
- The reactivity coefficients and reactivity worth are analyzed using 80 group DIF3D calculations.
- The fission products are represented by one lumped fission product for each fissionable heavy metal nuclide.
- The decay chain modeled spans the range from ^{234}U to ^{246}Cm.

3

Reference Pb/NA Cooled Cores



- 125MWt Power
- 125cm active core height
- 101.4W/cm linear heat generation rate
- 0.65cm clad (HT-9) inner radius
- 0.13cm clad thickness
- 5cm central and peripheral absorbers of 40%B_4C+40%W+20%HT-9

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Main Core Performances of Reference ENHS Cores (BOL)

Performance Parameters	Na Coolant	Pb Coolant
P/D ratio	1.16	1.36
Average volumetric power density (W/cc)	35.8	26.0
Pu wt%	11.87	12.20
Burnup reactivity swing (%dk)	0.195	0.221
Peak-to-average power density	2.017	1.829
Average burnup after 20EFPY (GWd/tHM)	51.0(51.0[a])	50.80(53.1[a])
Peak fast neutron fluence(n/cm^2) at 20EFPY	4.012E+23	3.829E+23
Initial conversion ratio	1.0380	1.0446
Doppler effect (dk/kk'C)	-6.5842E-06	-5.2442E-06
Axial fuel expansion (dk/kk'C)	-4.4541E-06	-4.6379E-06
Coolant expansion (dk/kk'C)	+5.8844E-06	+1.6747E-06
Grid plate radial expansion (dk/kk'C)	-8.2501E-06	-8.0679E-06
Void reactivity (%dk)		
Voiding inner core/+gas plenum	+1.907/+1.685	+2.718/+1.516
Voiding middle core/+gas plenum	+0.654/+0.521	+0.689/-0.068
Voiding outer core/+gas plenum	-0.133/-0.202	-0.694/-1.045
Voiding total core/+gas plenum	+2.469/+2.048	+2.555/+0.424
Peripheral absorber reactivity worth (%dk)	1.099	1.990
Central absorber reactivity worth (%dk)	4.174	4.138
Peripheral + central absorber worth (%dk)	5.545	6.811
Total heavy metal inventory (kg)	17507	17507

[A] Corresponds to peak fast neutron fluence of 4.0x1023 n/cm^2

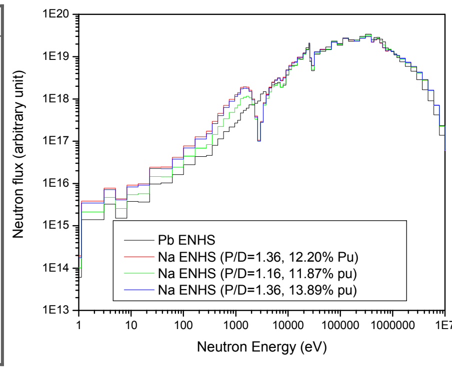
The lead cooled core has

- Much larger P/D ratio (better natural circulation capability)
- Flatter power distribution (larger fuel burnup and longer core life)
- Much less positive reactivity coefficient of coolant expansion
- Slightly less negative Doppler coefficient
- Slightly less negative coolant void reactivity only for active core but much less positive one for active core plus gas plenum regions
- Larger reactivity worth of the peripheral absorber and combined insertion of the peripheral and central absorbers

5

Comparison of Neutron Balance and Core Spectra

Characteristic	Lead-R	Sodium-I	Sodium-II	Sodium-R
P/D ratio	1.36	1.36	1.36	1.16
Pu wt%	12.20	12.20	13.854	11.87
k_{eff}	1.00475	0.934583	1.00571	1.00562
Leakage	0.178	0.239	0.233	0.210
Absorption				
In HM	0.778/0.434	0.729/0.409	0.736/0.392	0.758/0.414
In coolant	0.010	0.002	0.002	0.001
In Zr	0.013	0.009	0.009	0.010
In HT-9	0.021	0.021	0.020	0.021
Total	0.822	0.761	0.767	0.790
Fission	0.344	0.320	0.344	0.344
ν	2.921	2.915	2.921	2.916
η	1.292	1.270	1.365	1.323
ICR	1.0445	1.0449	0.8939	1.0382



- The lead cooled core has hardest neutron spectrum, smallest leakage probability, largest absorption by coolant (its value is small), and largest absorption by HM.
- The sodium cooled core (Sodium-I) having the same P/D ratio and Pu wt% as the lead cooled core has much smaller k_{eff} because of its larger leakage probability.
- Sodium-II having the same P/D ratio (1.36) but Pu wt% needed for criticality at BOL has smallest initial conversion ratio (ICR) because of its larger Pu-239/U-238 ratio.

4

Sensitivity of Control Elements Reactivity Worth

Central absorber; r=10.5-15.5 cm

Parameters	Pb cooled core	Na cooled core
	40%B_4C+40%W+20%HT-9	
5cm thick	4.143%dk	4.174%dk
7cm thick	4.310%dk	4.354%dk
10cm thick	4.385%dk	4.448%dk
15cm thick	4.408%dk	4.460%dk
	5cm thick	
40%B_4C+40%W+20%HT-9	4.143%dk	4.174%dk
50%B_4C+30%W+20%HT-9	4.299%dk	4.378%dk
60%B_4C+20%W+20%HT-9	4.418%dk	4.535%dk
70%B_4C+10%W+20%HT-9	4.511%dk	4.664%dk
	10cm thick	
40%B_4C+40%W+20%HT-9	4.385%dk	4.448%dk
50%B_4C+30%W+20%HT-9	4.430%dk	4.615%dk
60%B_4C+20%W+20%HT-9	4.511%dk	4.742%dk
70%B_4C+10%W+20%HT-9	4.573%dk	4.841%dk

Peripheral absorber

Parameters	Pb cooled core	Na cooled core
	Cavity (20%HT-9)	
5cm thick	0.371%dk	0.115%dk
7cm thick	0.413%dk	0.120%dk
10cm thick	0.469%dk	0.126%dk
15cm thick	0.547%dk	0.132%dk
	40%B_4C+40%W+20%HT-9	
5cm thick	1.998%dk	1.099%dk
7cm thick	2.071%dk	1.154%dk
10cm thick	2.100%dk	1.173%dk
15cm thick	2.110%dk	1.187%dk
	5cm thick	
40%B_4C+40%W+20%HT-9	1.998%dk	1.099%dk
50%B_4C+30%W+20%HT-9	2.082%dk	1.197%dk
60%B_4C+20%W+20%HT-9	2.146%dk	1.273%dk
70%B_4C+10%W+20%HT-9	2.196%dk	1.333%dk

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Sensitivity of Control Elements Reactivity Worth

- For both cores, the increase of the central absorber thickness has a small effect on its reactivity worth.
- For example, an increase of the central absorber thickness from 5cm to 15cm gives a reactivity worth increase by 6.4% and 6.9% for, respectively, the lead and sodium cooled cores.
- A composition change of the 5cm thick central absorber from 40%B_4C+40%W+20%HT-9 to 70%B_4C+10%W+20%HT-9 increases the reactivity worth by 8.9% and 11.7% for, respectively, the lead and sodium cooled cores.
- The peripheral absorber reactivity worth is much larger (~2 times) in the lead cooled core than in the sodium cooled core.
- A cavity type (20%HT-9) peripheral absorber has sufficient reactivity worth to control the burnup reactivity swing in the lead cooled core, but not in the sodium cooled core.
- An increase of the peripheral absorber thickness from 5cm to 15cm increases the reactivity worth by 5.6% and 8.0% for the lead and sodium cooled cores, respectively.
- A composition change of the 5cm thick central absorber from 40%B_4C+40%W+20%HT-9 to 70%B_4C+10%W+20%HT-9 gives a reactivity worth increase by 9.91% and 21.3% for the lead and sodium cooled cores, respectively.

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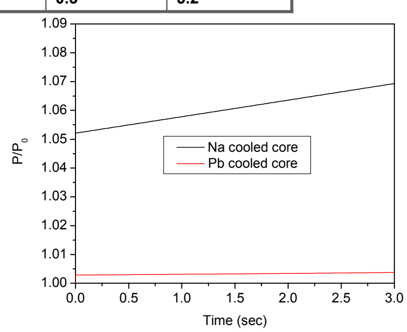
Safety Implications of Positive Void Coefficient -- Response to Fission Gas Release

Estimated Parameters	Lead cooled core	Sodium cooled core
Volume of gas released from one fuel rod (hot pin) at EOL (cc)	314	3810
Moving velocity of coolant (m/sec)	0.447	0.651
Passing time through active core (sec)	2.796	1.920
One unit channel coolant volume (cc)	248.3	115.6
Number of channels corresponding to the amount of the gas	1.3	33
Void fraction (the void is assumed to be only in inner core)	0.0384%	1.0028%
Reactivity inserted by the above void (pcm)	1.022	17.88
Power increase (%) due to prompt jump	0.3	5.2

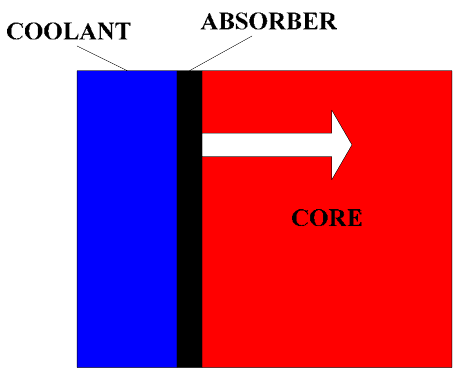
- **The volume of gas released from one hot fuel rod is much smaller for lead-cooled core than for sodium-cooled core.**
- **The resulting positive reactivity insertion in the sodium-cooled core is ~17 times larger than in the lead-cooled core.**
- **The power increase resulting from the above reactivity insertion is much larger in the sodium-cooled core than in the lead-cooled core.**



Power increase estimation using a point kinetic model without temperature feedback

5

Extension of the Central Absorber Region - Rationale



- Increase of fraction of neutrons leaking through inner core surface
- Increase of fraction of neutrons absorbed by central absorber
- Reference location: 10.5 to 15.5 cm

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Extension of the Central Absorber Region - Results

Outer radius of central absorber	20cm		25cm		30cm	
Parameters	Pb cooled	Na cooled	Pb cooled	Na cooled	Pb cooled	Na cooled
P/D ratio	1.34	1.12	1.32	1.09	1.30	1.05
Pu wt%	12.13	11.75	12.13	11.71	12.15	11.62
Burnup reactivity swing (%dk)	0.251	0.182	0.260	0.257	0.343	0.336
3D peaking factor (BOL/EOL)	1.775/1.810	1.937/1.961	1.711/1.744	1.903/1.919	1.655/1.687	1.890/1.900
Conversion ratio (BOL)	1.0459	1.0408	1.0428	1.0362	1.0373	1.0334
Average discharge burnup (GWd/tHM)	50.7 (54.4)	51.0 (53.1)	50.7 (56.5)	51.0 (54.8)	50.8 (58.5)	51.0 (55.8)
Peak discharge burnup (GWd/tHM)	96.7 (103.7)	105.5 (109.8)	93.1 (103.7)	103.3 (111.0)	90.0 (103.7)	102.3 (111.9)
Peak fast neutron fluence (n/cm^2)	3.731E+23	3.843E+23	3.592E+23	3.723E+23	3.473E+23	3.657E+23
Central absorber worth (%dk)	4.681	5.134	5.349	5.801	5.977	6.402
Peripheral absorber worth (%dk)	2.113	1.171	2.299	1.289	2.516	1.367
Combined absorbers worth (%dk)	7.569	6.642	8.519	7.445	9.439	8.130

- ❑ The reactivity worth of the control elements significantly increases for both coolants.

  Increase of the central absorber region radius from 20cm to 30cm leads to increase of the worth of the central absorber, peripheral absorber and their combined effect, respectively

  - Pb: 26.7%, 19.1%, and 24.7%
  - Na; 24.7%, 16.7%, and 22.4%

- ❑ For all cases, the sodium cooled cores have larger central absorber reactivity worth but smaller reactivity worth of the peripheral and combined absorbers
- ❑ The increase of the central absorber region leads to a reduction of the hydraulic diameter. This reduction is small in the lead-cooled core but very significant in the sodium-cooled core.
- ❑ These cores have flatter power distribution, longer core life and larger discharge burnup.

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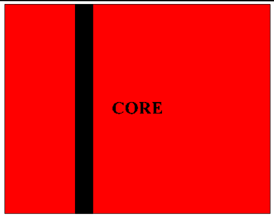
In-Core Control Rods -- An Annular Absorber



- Higher importance location
- No central cavity of coolant
- Annular – for calculation simplicity

Outer radius of annular absorber	20cm		30cm		40cm	
Parameters	Pb cooled	Na cooled	Pb cooled	Na cooled	Pb cooled	Na cooled
P/D ratio	1.34	1.12	1.32	1.09	1.30	1.05
Pu wt%	12.13	11.75	12.13	11.71	12.15	11.62
Burnup reactivity swing (%dk)	0.251	0.182	0.260	0.257	0.343	0.336
3D peaking factor (BOL/EOL)	1.775/1.810	1.937/1.961	1.711/1.744	1.903/1.919	1.655/1.687	1.890/1.900
Conversion ratio (BOL)	1.0459	1.0408	1.0428	1.0362	1.0373	1.0334
Average discharge burnup (GWd/tHM)	50.7 (54.4)	51.0 (53.1)	50.7 (56.5)	51.0 (54.8)	50.8 (58.5)	51.0 (55.8)
Peak discharge burnup (GWd/tHM)	96.7 (103.7)	105.5 (109.8)	93.1 (103.7)	103.3 (111.0)	90.0 (103.7)	102.3 (111.9)
Peak fast neutron fluence (n/cm^2)	3.731E+23	3.843E+23	3.592E+23	3.723E+23	3.473E+23	3.657E+23
Annular absorber worth (%dk)	4.681	5.134	5.349	5.801	5.977	6.402
Peripheral absorber worth (%dk)	2.113	1.171	2.299	1.289	2.516	1.367
Combined absorbers worth (%dk)	7.569	6.642	8.519	7.445	9.439	8.130

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In-Core Control Rods -- An Annular Absorber

- As the radius of the 5cm thick annular absorber increases
  - The optimal P/D ratio decreases. This change is smaller for lead than for sodium coolant
  - The power peaking increases → shorter core life and smaller discharge burnup.
  - For same outer radius, the power peaking of cores with a central cavity (absorber) is smaller than of cores with an annular absorber (in-core control elements)
  - For same outer radius, the reactivity worth of the in-core absorber is somewhat lower than of the central absorber. The same trend is observed for the peripheral absorber and for the combined worth.
  - For the range of the annular type absorber position considered here, the peripheral absorber worth doesn't change monotonically but the reactivity worth of the combined insertion of the annular type and peripheral absorbers increases monotonically.
- For the extreme cases considered here, the combined insertions of in-core and peripheral absorbers give 16.557%dk (i.e., k_{eff} <0.84) and 17.383%dk (i.e., k_{eff}<0.831) for lead and sodium coolants, respectively – adequate for safe shutdown.

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Comparative Analysis of New Lead Cooled Core Options

- The detailed core performances of the following four lead cooled ENHS cores are analyzed and inter-compared :
  - DESIGN-I : Extension of the central cavity region
    - Absorber inner and outer radii: 24cm and 29cm
    - Absorber composition: 1%HT-9 + 99%Pb
    - P/D ratio: 1.30
  - DESIGN-II : Like DESIGN-I but
    - Absorber composition: 5%HT-9 + 95%void
    - P/D ratio: 1.28
  - DESIGN-III : Use of an in core annular absorber
    - Absorber inner and outer radii: 40cm and 45cm
    - P/D ratio: 1.34
  - REFERENCE : the reference lead-cooled ENHS core (P/D=1.36)

Comparative Analysis of New Lead Cooled Core Options

Performance Parameters	DESIGN-I	DESIGN-II	DESIGN-III	Reference
Core inner/outer radius (cm)	30.0/109.91	30.0/108.35	0/40/45/110.93	16.41/111.83
P/D ratio	1.30	1.28	1.34	1.36
Average volumetric power density (W/cc)	28.47	29.37	26.79	26.01
Pu wt%	12.15	12.08	12.19	12.20
Burnup reactivity swing (%dk)	0.343	0.435	0.203	0.221
Peak-to-average power density	1.655	1.530	2.002	1.829
Peak burnup after 20EFPY (GWd/tHM)	90.0(103.7[a])	82.6(100.9[a])	112.7(102.9[a])	99.89(104.4[a])
Average burnup after 20EFPY (GWd/tHM)	50.8(58.5[a])	50.8(62.0[a])	50.9(46.5[a])	50.80(53.1[a])
Peak fast neutron fluence(n/cm^2) at 20EFPY	3.473E+23	3.275E+23	4.383E+23	3.829E+23
Initial conversion ratio	1.0373	1.0345	1.0432	1.0446
Doppler effect (dk/kk'C)	-5.7176E-06	-5.5320E-06	-5.4502E-06	-5.2442E-06
Axial fuel expansion (dk/kk'C)	-4.1232E-06	-4.1612E-06	-4.0428E-06	-4.6379E-06
Coolant expansion (dk/kk'C)	+1.7895E+06	+1.6855E-06	+2.1435E-06	+1.6747E-06
Grid plate radial expansion (dk/kk'C)	-7.2605E-06	-7.0031E-06	-7.0238E-06	-8.0679E-06
Void reactivity (%dk)				
Voiding inner core/+gas plenum	+2.146/+1.258	+2.269/+1.029	+1.238/+0.65[b](+1.528/+0.696)	+2.718/+1.516
Voiding middle core/+gas plenum	+0.750/+0.123	+0.839/+0.197	+0.302/-0.230	+0.689/-0.068
Voiding outer core/+gas plenum	-0.663/-0.991	-0.615/-0.952	-0.611/-0.878	-0.694/-1.045
Voiding total core/+gas plenum	+2.175/+0.489	+2.414/+0.351	+2.362/+0.375	+2.555/+0.424
Peripheral absorber reactivity worth (%dk)	2.506	2.671	2.017	1.990
Central absorber reactivity worth (%dk)	5.949	5.704	12.787	4.138
Peripheral + central absorber worth (%dk)	9.419	9.234	16.57	6.811
Total heavy metal inventory (kg)	17505	17506	17507	17505

[A] Corresponding to peak fast neutron fluence of 4.0x10^{23} n/cm^2, [b] inner most core region

Comparative Analysis of New Lead Cooled Core Options

- DESIGN-III core (annular in-core absorber) has
  - largest reactivity worth of in-core absorbers and of all absorbers
  - smallest reduction of P/D ratio
  - largest power peaking → shortest core life and smallest discharge burnup
- All new core designs have somewhat smaller core outer radius and slightly larger burnup reactivity swing than the reference ENHS core
- DESIGN-I and II cores have
  - lower power peaking → longer core life, larger discharge burnup
  - larger reactivity worth of control elements than the reference ENHS core
  - negative reactivity coefficient of coolant expansion when the coolant temperature varies over all reactor regions (not given in Table)???
- In particular, DESIGN-II has least peaked power distribution →16.8% increased discharge burnup relative to the reference ENHS core, and least positive coolant void reactivity

Comparative Analysis of New Sodium Cooled Core Options

Performance Parameters	DESIGN-I (Extension of central absorber region)	DESIGN-II (Use of annular in core absorber)	Reference
Core inner/outer radius (cm)	30.0/90.52	0/40/45/94.2	14.0/95.39
P/D ratio	1.05	1.13	1.16
Average volumetric power density (W/cc)	43.6	37.7	35.8
Pu wt%	11.62	11.81	11.87
Burnup reactivity swing (%dk)	0.336	0.156	0.195
Peak-to-average power density	1.890	2.286	2.017
Peak burnup after 20EFPY (GWd/tHM)	102.3(111.9)	129.2(104.7)	110.3(110.3)
Average burnup after 20EFPY (GWd/tHM)	51.0(55.8)	51.2(41.5)	51.0(51.0)
Peak fast neutron fluence(n/cm^2) at 20EFPY	3.657E+23	4.935E+23	4.012E+23
Initial conversion ratio	1.0334	1.0442	1.0382
Doppler effect (dk/kk'C)	-7.1915E-06	-6.9289E-06	-6.5842E-06
Axial fuel expansion (dk/kk'C)	-4.5699E-06	-4.4191E-06	-4.4541E-06
Coolant expansion (dk/kk'C)	+2.6957E+06	+5.1214E-06	+5.8844E-06
Grid plate radial expansion (dk/kk'C)	-7.4381E-06	-8.0907E-06	-8.2501E-06
Void reactivity (%dk)			
Voiding inner core/+gas plenum	+0.770/+0.683	+1.124/+1.005(+0.832/+0.714)	+1.907/+1.685
Voiding middle core/+gas plenum	+0.418/+0.362	+0.263/+0.190	+0.654/+0.521
Voiding outer core/+gas plenum	-0.0532/-0.0868	-0.112/-0.154	-0.133/-0.202
Voiding total core/+gas plenum	+1.141/+0.965	+2.150/+1.804	+2.469/+2.048
Peripheral absorber reactivity worth (%dk)	1.367	1.059	1.099
Central absorber reactivity worth (%dk)	6.402	15.674	4.174
Peripheral + central absorber worth (%dk)	8.130	17.383	5.545
Total heavy metal inventory (kg)	17505	17507	17505

Comparative Analysis of New Sodium Cooled Core Options

- DESIGN-I (having extended central cavity) has:
  - tightest lattice; P/D ratio of 1.05
  - Smallest power peaking
  - less positive reactivity coefficient of coolant expansion and coolant voiding
  - significantly increased reactivity worth of control elements -- by 24.4%, 53.4% and 46.6% for the peripheral, central and their combined effect
- DESIGN-II (having an in core annular absorber) has:
  - slightly reduced P/D ratio of 1.13

  in comparison with the sodium cooled reference ENHS core:
  - more peaked power distribution
  - slightly reduced reactivity worth of the peripheral absorber but
  - drastically (213.5%) increased reactivity worth of combined insertion of all absorbers
  - less positive coolant void reactivity

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Lead versus Sodium Cooled New Option Cores

- In comparison with the sodium cooled cores, **the lead cooled cores** have **much larger reactivity worth of the peripheral absorber**, similar reactivity worth of combined insertion of all absorbers, **less peaked power distribution (longer core life and larger discharge burnup)**, **smaller change of optimal P/D ratios** from reference design, and **less positive coolant void reactivity and reactivity coefficient of coolant expansion**.

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62.5MWt Pb/Na Cooled ENHS Core Designs

- It is shown that the same conditions (core height, fuel dimensions, average linear heat generation rate) did not give the feasible long life cores both for lead and sodium coolants because of significant increase of neutron leakage.
- The fuel volume fraction is increased by changing the fuel dimensions (clad inner radius; 0.65cm→0.83cm, clad thickness;0.13cm→0.21cm).
- For lead coolant, following two long-life cores (125cm high) are designed;
  - Lead-I :
    - Determined so as to have the same specific power as the reference ENHS core
    - High linear heat generation rate of 259.6W/cm, tight lattice of P/D=1.07, 20EFPY core life
  - Lead-II :
    - Determined so as to have the same volumetric power density as the reference ENHS core
    - Relatively low linear heat generation rate of 196.6W/cm, loose lattice of P/D=1.16, 28EFPY core life
- For sodium coolant, there were no feasible designs of long-life cores with keeping the specific power.
  - Sodium-I :
    - Determined so as to have the same volumetric power as the reference ENHS core
    - High linear heat generation rate of 208.4W/cm, tight lattice of P/D=1.04, 22EFPY core life
    - Central absorber region composition of 10%HT-9 (90%void) for mitigation of the central power peaking

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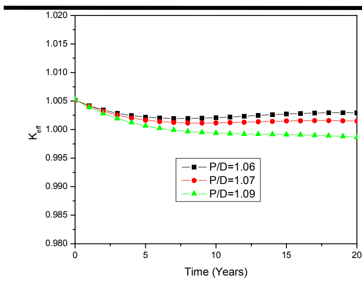
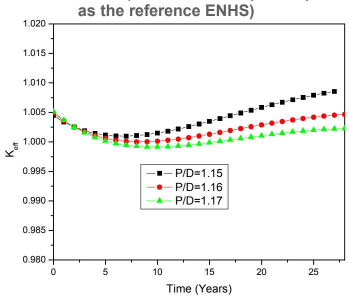
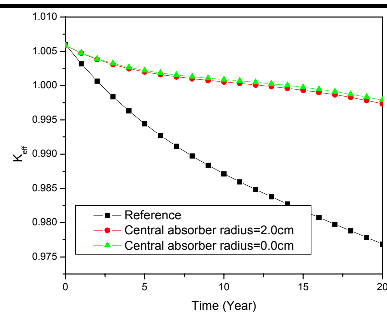
62.5MWt Pb/Na Cooled ENHS Core Designs



Lead cooled cores (with the same specific power as the reference ENHS)



Lead cooled cores (with the same volumetric power density as the reference ENHS)



Sodium cooled cores (With the same specific power as the reference ENHS, P/D=1.01)

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62.5MWt Pb/Na Cooled ENHS Core Designs

Performance Parameters	LEAD-R	SODIUM-R	LEAD-I	LEAD-II	SODIUM-I
Central absorber region radius (cm)	16.41	14.0	16.41	16.41	6.0
P/D ratio	1.36	1.16	1.07	1.16	1.04
Linear heat generation rate (W/cm)	101.4	101.4	259.6	196.6	208.4
Average volumetric power density (W/cc)	26.0	35.8	41.9	27.0	35.6
Specific power (kW/kg)	7.140	7.140	7.141	5.409	5.733
Pu wt%	12.20	11.87	11.68	11.63	11.49
Core life (EFPY)	20(20.9)	20(19.4)	20(21.6)	28(27.5)	22(22.2)
Burnup reactivity swing (%dk)	0.221	0.195	0.404	0.489	0.558
Peak-to-average power density	1.829	2.017	1.680	1.738	2.016
Peak burnup after 20EFPY (GWd/tHM)	99.89(104.4)	110.3(109.9)	91.2(98.4)	99.3(97.6)	97.1(98.0)
Average burnup after 20EFPY (GWd/tHM)	50.80(53.1)	51.0(50.8)	51.0(55.0)	54.0(53.1)	45.1(45.5)
Peak fast neutron fluence(n/cm^2) at 20EFPY	3.829E+23	4.012E+23	3.709E+23	4.070E+23	3.965E+23
Initial conversion ratio	1.0446	1.0382	1.0320	1.00878	0.99493
Doppler effect (dk/kk'C)	-5.2442E-06	-6.5842E-06	-5.2942E-06	-5.2736E-06	-5.2805E-06
Axial fuel expansion (dk/kk'C)	-4.6379E-06	-4.4541E-06	-4.8163E-06	-4.4632E-06	-5.0550E-06
Coolant expansion (dk/kk'C)	+1.6747E-06	+5.8844E-06	+3.6422E-07	+1.0292E-06	+2.5416E-06
Grid plate radial expansion (dk/kk'C)	-8.0679E-06	-8.2501E-06	-6.6883E-06	-7.0443E-06	-7.9059E-06
Void reactivity (%dk)					
Voiding inner core/+gas plenum	+2.718/+1.516	+1.907/+1.685	+0.696/+0.498	+1.314/+0.888	+0.847/+0.768
Voiding middle core/+gas plenum	+0.689/-0.068	+0.654/+0.521	+0.144/-0.002	+0.305/+0.010	+0.261/+0.223
Voiding outer core/+gas plenum	-0.694/-1.045	-0.133/-0.202	-0.375/-0.473	-0.525/-0.700	-0.0696/-0.473
Voiding total core/+gas plenum	+2.555/+0.424	+2.469/+2.048	+0.451/+0.0359	+1.053/+0.216	+1.040/+0.987
Peripheral absorber reactivity worth (%dk)	1.990	1.099	4.519	3.353	2.268
Central absorber reactivity worth (%dk)	4.138	4.174	7.879	6.141	3.093
Peripheral + central absorber worth (%dk)	6.811	5.545	13.267	10.380	5.609
Total heavy metal inventory (kg)	17507	17507	8751	11554	10901

62.5MWt Pb/Na Cooled ENHS Core Designs

- The new small power rating cores have higher core average linear heat generation rate, tight lattice and more or less increased burnup reactivity swing.
- For lead coolant, LEAD-I using the same specific power has much less peaked power distribution (i.e., longer core life and larger discharge burnup), nearly zero coolant void reactivity, much less positive (or negative for whole regions coolant temperature changes) reactivity coefficient of coolant expansion, drastically enhanced reactivity worth of control elements, but tight lattice of P/D=1.07 (giving smaller amount of heavy metal loading).
- LEAD-II using the same volumetric power has less peaked power distribution, less positive coolant void reactivity, less positive (or negative for whole regions coolant temperature changes) reactivity coefficient of coolant expansion, significantly enhanced reactivity worth of control elements, and relatively loose lattice of P/D=1.16.
- For sodium coolant, it is much difficult to design a small long life core due to its inferior neutron balance relative to lead coolant.
- There were no feasible long life cores with keeping the specific power.
- SODIUM-I using the same volumetric power density has very tight lattice of P/D=1.04, hard neutron spectrum resulting in much less positive coolant void reactivity worth and coolant expansion reactivity coefficient (negative for whole regions coolant temperature changes), enhanced reactivity worth of the peripheral absorber but smaller reactivity worth of the central absorber, and larger reactivity worth of the combined insertion of all absorbers than original sodium cooled ENHS core.

Summary and Conclusions

- The core performances including the safety related reactivity coefficients for the lead and sodium cooled ENHS cores are analyzed and inter-compared in a consistent manner.
- Relatively to the sodium cooled reference ENHS core, the lead cooled reference ENHS core has (1) much larger P/D ratio but much harder neutron spectrum, (2) less peaked power distribution, (3) more positive coolant void reactivity for active core but much less positive one for active core plus gas plenum, (4) much less positive reactivity coefficient of coolant expansion, (5) larger reactivity worth of the peripheral absorber and total absorbers, but smaller one of the central absorber
- The complete release of all gases from one hot fuel rod at EOL leads to much more significant situations in sodium cooled core than in lead cooled core by the following aspects : (1) much larger volume of the leased gas, (2) much larger positive reactivity resulting from the leased gas, (3) much larger power increase resulting from the leased gas
- Two new design options for enhancing the reactivity worth of the control elements are very effective both for lead and sodium coolants.
  - For the options using the extension of the central absorber region, the sodium cooled cores have significantly reduced P/D ratios but the lead cooled cores have slightly changed P/D ratio, much less peaked power distribution, much less positive coolant void reactivity and reactivity coefficient of coolant expansion.
  - For the options using an annular type absorber in core internal, the new cores have slightly changed P/D ratios, more peaked power distributions but drastically increased reactivity worth of total absorbers both for sodium and lead coolants.

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Summary and Conclusions

- With changes of fuel rod dimensions, it was possible to neutronically the 62.5MWt ENSH cores cooled either lead or sodium coolant to have long life over 20EFPY.
- These new cores have tighter lattice and reduced core diameter leading to much less positive (or negative when the coolant temperature changes are applied to all regions) reactivity coefficient of coolant expansion and coolant void reactivity.
- In particular, with lead coolant, it was possible to design the long life cores to have a nearly zero coolant void reactivity, significantly enhanced reactivity worth of control elements, and much less peaked power distribution (longer core life and larger discharge burnup).
- It was much more difficult to design a small long life core using sodium coolant due to its inferior neutron balance relative to lead coolant.

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