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ASSESSMENT OF PRISM RESPONSES TO LOSS OF FLOW EVENTS*

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ABSTRACT

The Nuclear Regulatory Commission (NRC), with Brookhaven National Laboratory providing technical support, is continuing a preapplication review of the 471 MWt, Advanced Liquid Metal Reactor (ALMR), PRISM by General Electric. The revised design has been evaluated using the SSC code, for a series of loss of flow events (LOF) with and without Gas Expansion Modules (GEMs). These devices have a net worth of 69¢ and have reduced the seriousness of the LOF in PRISM. However, it was found that the extremely low probability case of an instantaneous loss of 4 EM pumps without scram could lead to sodium boiling even with the GEMs.

1. INTRODUCTION

The Nuclear Regulatory Commission (NRC), with technical support provided by the Brookhaven National Laboratory (BNL), is continuing a pre-application review of the 471 MWt PRISM advanced reactor design. The evaluation of the initial design has already been released,[1] with the supporting technical analyses performed by BNL.[2-3] Among the findings of the report was the determination that there were apparent vulnerabilities in the passive shutdown response to certain improbable, unscrammed, events. In particular, events which involved a rapid flow reduction without scram were of concern since the margin to sodium boiling was not large enough to compensate for all the uncertainties. Since boiling of the sodium could introduce positive reactivity and possibly trigger a power excursion, any potential for sodium boiling is a cause for concern.

In response to the initial findings by the NRC,[1] General Electric (G.E.) revised the PRISM design.[4] In several cases, the changes were made to directly address the NRC concerns. Other changes were dictated by the U.S. Department of Energy (DOE) as design improvements or revisions to enhance the economics of the plant. As the result of these changes, nearly all of the BNL analyses were repeated to factor in the new components and the revised operating conditions. Assessment of the behavior of PRISM during loss of flow events (LOF) will be summarized in this paper.

2. THE ALMR DESIGN

The ALMR plant, as presently proposed by G.E., consists of three identical power blocks of 465 MWe, for a total plant electrical rating of 1395 MWe (Table 1 and Figure 1). Each power block is comprised of three reactor modules with individual thermal ratings of 471 MWt. Each module has its own steam generator which is combined in each power block to feed a single turbine generator. The reactor module (Figure 2) is about 19 meters (62 feet) high and about 6 meters (20 feet) in diameter, and is placed in a silo (i.e., below grade).

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Under normal operating conditions, four EM pumps draw sodium from the cold pool and drive it through eight pipes to the core inlet plenum. The sodium is heated as it passes upward through the fuel assemblies (hexagonal cans containing wire wrapped pins) and into the hot pool above the core region. The heat is transferred to the intermediate loop sodium by the intermediate heat exchanger (IHX), as the primary sodium passes from the hot pool to the cold pool.

The core design is illustrated in Figure 3. A "limited free bow" restraint system is utilized to assure an outward bow in the active core region of the assemblies as long as the peak temperatures are in the core center and decrease radially. The bowing is only one of several reactivity feedbacks that are significant. The other significant feedbacks are Doppler, sodium density, fuel expansion, core radial expansion (via grid plate and above core load pads), the control rod drive line expansion, and the Gas Expansion Modules (GEMs). Most of these feedbacks are negative for off nominal conditions, since increasing the power and core average temperature causes the core criticality to decrease. This characteristic gives the core power the tendency to transition to a lower level at an elevated temperature (unless the sodium boils). Predictive calculations are performed to determine the rate, direction, and magnitude of the reactivity feedback components during postulated transients.

Table 1. ALMR Plant Design Data

Reactors Modules Per Power Block:		Three
Number of Power Blocks:		One/Two/Three
Electrical Output:		465/930/1395 MWe
Reactor Power:		471 MWt
Turbine Throttle Conditions:		7.58 MPa (Saturated)
Primary Sodium	Inlet:	610K
	Outlet:	758K
Secondary Sodium	Inlet:	555K
	Outlet:	716K
Peak Fuel Pin Linear Power:		305 W/cm
Peak Fuel Burnup:		135 MWd/kg
Refueling Interval:		18 months

3. PRISM SHUTDOWN AND HEAT REMOVAL

The PRISM reactor shutdown system includes six control rods, with each control rod capable of shutting the reactor down. Extensive diversity in the system, i.e., each rod has its own electrical system, independent drive motors, and a gravity drop alternative insertion capability, greatly reduces the likelihood of this system failing entirely.

In the center of the core is a hollow assembly beneath a container holding B_4C balls. The dropping of these balls is another independent means of terminating power. G.E. calls this system the "Ultimate Shutdown System" (USS). However, this is a comparatively slow system, requiring a minute or two to achieve a reactor shutdown.

The "passive-shutdown" response can reduce the reactor power in response to most postulated unscrammed events. While these events are quite improbable, the potential for positive reactivity insertions, through sodium boiling or fuel relocation, dictates that this class of events be evaluated. Detailed analyses of the reactor system behavior provides the timing at which events progress and gives insight into the inherent response of the system during unscrammed events.

If there is failure to remove enough heat from the primary system, the sodium will expand, and the hot pool sodium will eventually (a few hours) spill over the reactor vessel liner. This will establish an alternate flow path and will increase the heat being rejected off the reactor vessel surface to the atmosphere (Figure 2). Once this has occurred, the Reactor Vessel Auxiliary Cooling System (RVACS) becomes fully functional and removes the decay heat load efficiently enough to prevent damage to the reactor vessel and other key components. Even under normal operating conditions, there is substantial parasitic heat removal by conductance through the vessel liner, vessel, and containment vessel.

In the PRISM design, the use of sodium coolant and the relatively small reactor power facilitates this type of passive decay heat removal. Both the normal cooling and auxiliary cooling system (ACS is a natural circulation air jacket around the steam generator) can work well under natural circulation conditions. However, the RVACS is believed to have higher reliability than the normal plant cooling systems. It is very difficult to completely fail this system, even if partial blockages of atmospheric air are postulated. Even if all three heat removal systems fail completely, it would be at least twelve hours before significant core damage would result, because of the large thermal mass of the system.

4. U-10Zr-27Pu FUEL

The "ternary" metal fuel composition currently proposed for PRISM is U-10Zr-27Pu. The initial data indicates that the burnup response of the ternary metal fuel is more complex than that for the original U-Zr metal fuel, showing axial expansion early in the burnup cycle, as well as fuel component migration. In theory, the behavior of the three component ternary fuel should be more complex than binary fuel, and it should require some time to characterize fully. Argonne National Laboratory (ANL) will be obtaining more data on the ternary fuel within the next year and hopes to address some of the questions that have resulted from the data obtained from the first few pins. The radial migration of the uranium and zirconium components during burnup is very substantial and causes large changes in local fuel thermal conductivities and significant changes in the fuel solidus and liquidus temperatures. In addition, there may be some radial relocation of the plutonium component, which could change the radial power distribution within the fuel pin. Thus, some problems with the ternary metal fuel have been identified, but judgement will be reserved until more conclusive data have been compiled.

5. PRISM MODELING

The SSC[5] and MINET[6] codes were used in this analysis for complimentary purposes. SSC was developed at BNL for analyzing LMR transients. SSC can model core regions in detail, as well as the primary system, the IHX, intermediate loop, steam generator, and the major components of the ternary loop. However, alternate flow patterns that may develop during loss of heat sink events or certain loss of flow events can become very complicated, which requires the MINET flexibility for that part of the analysis.

5-1. SSC model

In Figure 4 a schematic drawing of the PRISM model is shown. The core was represented using 7 channels: fuel (or driver), internal blanket, radial blanket, control assembly, shield assemblies, hot driver, and hot internal blanket. Each channel includes 2 axial nodes in the lower shield, 6 axial nodes in the fuel region, and 4 nodes to represent the upper gas plenum.

5-2. Reactivity feedback models

Several reactivity feedbacks are important in the passive shutdown response for the metal cores. Because of the smaller Doppler feedback in the metal core, reactivity feedbacks having little importance in oxide cores are significant in the metal core. The main reactivity feedbacks are as follows:

5-2-1. Doppler feedback

As the fuel temperature increases, more neutrons are parasitically absorbed in the resonance energy range. For metal fuel, Doppler feedback is smaller than it is for oxide fuel because of the harder neutron energy spectrum, which places fewer neutrons in the resonance energy range. Also, due to high thermal conductivity, metal fuel operating temperatures are much lower than those in oxide fuel cores. This allows the power and temperature defects in a metal core to be small ($\sim$\$1.20), allowing the criticality and power level of the system to be influenced by other natural feedbacks.

5-2-2. Axial fuel expansion

Metal fuel expands axially when it heats up. Axial expansion increases the core height and decreases the effective density of the core material. This increases the probability that a neutron will escape the core, giving a negative reactivity feedback.

All analyses performed using SSC assumed that the fuel is in contact with the HT9 clad. This is the most common state for the equilibrium core since only 25% of the core will be reloaded at each refueling, and the fuel is in an unlocked state, i.e., below 2 atom per-cent burnup, only briefly. Axial expansion is dominated by the clad after lockup since metal fuel is weak (i.e., small Young's Modulus). The fuel elongations in SSC calculations were calculated by using an average strain, weighted with Young's modulus.

5-2-3. Sodium density feedback

Thermal expansion of the sodium is the only significant positive reactivity feedback, except for the long term withdrawal of the control rod drive line with vessel heatup. The thermal expansion results in fewer sodium atoms within and surrounding the core. The dominant effect is the reduction of the collisions between neutrons and sodium atoms, which hardens the neutron energy spectrum and yields a net positive reactivity feedback effect from the increased neutron importance.

The feedback formulation was set up to reference the sodium density at the refueling temperature. Each node was given equal weight within a given category (i.e., driver, internal blanket, and radial blanket).

5-2-4. Control rod drive line and vessel thermal expansion

The magnitude of this feedback is dependent upon the initial position of the control rods on the control rod worth curve. The control rod drive lines, which are in the upper internal structure, expand when they are heated, inserting the control rods further into the reactor, adding negative reactivity.

The thermal expansion of the reactor vessel ultimately limits the amount of negative reactivity inserted by the control rod drive line. The reactor vessel is cantilevered from the top, and expands down and slowly withdraws the control rods from the core up to the control rod stop positions. The time constant for the reactor vessel is about 700s, while the control rod drive line expansion time constant is around 28s. Thus, the initial response to increased sodium outlet temperatures is a negative feedback, while the long term effect could end up being positive.

Control rod and vessel expansion are calculated in SSC using single node temperatures for the vessel and control rod drive line masses. The total elongated length is calculated by subtracting the vessel expansion from the control rod drive line expansion to determine the net control rod expansion into the core.

5-2-5. Radial expansion

The radial dimension of the core is determined largely by assembly spacing. This spacing is determined by the grid plate below the core and by two sets of load pads above the core. When the structures heat up and expand it increases the core radius, which reduces the core average density in the radial direction. The effect increases neutron leakage and generates a negative feedback response.

SSC tracks the radial expansion of the core from thermal expansion only. This is accomplished by tracking the structure temperatures at the above core load pads (just above the fueled area) and at the grid plate. In the SSC calculation, no credit was given for the thermal bowing of the assemblies. It is noted that the bowing effect may reduce the risk associated with several severe accident sequences. However, the total worth of the (limited-free) bowing carries significant uncertainties. Bowing should add negative reactivity to the system when core temperatures rise. At this time, it doesn't appear that bowing can insert any positive reactivity during any significant portions of the postulated accidents reviewed to date. Hence, neglecting it is generally a conservative assumption.

5-2-6. GEM modeling

The GEM is essentially an empty assembly duct, sealed at the top, open at the bottom and connected to the high pressure in the inlet plenum of the core. The range of operation of the GEMs tested in FFTF can be seen in Figure 5. A hexagonal cross section duct, with a wall thickness slightly greater than the standard fuel and blanket duct, forms the unit. When the pumps are at full flow, the plenum pressure (minus the static heat to the GEM level) compresses the gas in the GEM cavity to the portion of the GEMs above the core. This causes more neutrons to be scattered and deflected back into the core, as compared to when the gas is adjacent to the core. When the flow decreases, the trapped helium expands and drops the sodium level into the core region. As a result, fewer neutrons are scattered back into the core region. The effect increases as the gas expands into the core, until the gas-liquid interface drops below the core. At this point the maximum negative reactivity of 69 cents (i.e., 23 cents each) is imposed. This device offers a passive negative feedback which can help maintain an acceptable power-to-flow ratio during sudden loss of flow events.

6. ANALYSIS OF EVENTS

The transient responses of the PRISM module to various loss of flow (LOF) events were evaluated using SSC, and augmented using MINET calculations when necessary. SSC has been benchmarked against both ARIES (G.E.) and SASSYS (ANL)[7] over the years, and has consistently generated similar results for similar events. The calculations contained in this section used models that were generally more conservative than those used by the vendor.

The range of events analyzed to date is shown in Table 2. The first four events in Table 2 have been published[3] and the results of those analyses will be briefly summarized here. The last two events have recently been evaluated in order to complete the analyses of the LOF sequence of events. The prediction of these events will be presented in some detail.

Table 2. Loss of Flow (LOF) Events Evaluated to Date

1. ULOF with GEMs
2. ULOF with no GEMs
3. ULOF from 2 pump seizure
4. ULOF with 1 pump failure (no coastdown)
5. ULOF with a 4 pump failure (no coastdown)
6. LOF with a 4 pump failure with SCRAM (with GEMs)
7. LOF with a 4 pump failure with SCRAM (without GEMs)

6-1. ULOF with and without GEMs

The unscrammed loss of flow (ULOF) is initiated by a trip from full power and coastdown of the EM pumps (provided by "synchronous machines" which always run in parallel with the EM pumps). The initial conditions correspond to the full power conditions shown in Table 3.

Table 3. Table of Initial and Key Operating Procedures

Description	PRISM	SSC
Power (MW_t)	471	471
Cover Gas (kPa)	99.3	99.3
Primary Flow (kg/s)	2513	2507
Primary Sodium Inlet (K)	610.9	610.9
Primary Sodium Outlet (K)	758.1	758.0
Core Height (m)	1.342	1.3462
Peak Fuel Pin/Average Fuel Pin	1.31	1.31
Fuel Pin OD (m)	.00668	.00668
Driver Fuel Pins/Assembly	331	331
Intermediate Sodium Flow (kg/s)	2293	2275
Intermediate Sodium IHX Inlet (K)	555.4	557.0
Intermediate Sodium IHX Outlet (K)	716.5	720.0

6-1-1. ULOF with GEMs

The synchronous machines provide a prescribed core flow rate vs. time after a pump trip. The loss of pressure at the core inlet causes the GEM sodium level to drop. The GEMs act like passive control rods and insert a fast negative 69¢. The reduction in flow causes an initial heat up of the core and activates the thermal expansion (negative) feedbacks and decreases the power. By 500 s, the power is reduced to the decay heat level. Thus, this is a benign event for the revised design.

6-1-2. ULOF without GEMs

The same sequence of events were prescribed as above, except the GEMs were also assumed not to function. The increasing power-to-flow ratio resulted in the core rapidly heating up. This activated the negative reactor feedbacks. The (negative) feedbacks from Doppler, axial, radial and control rod drive line expansion were larger in magnitude than the positive feedback from the decrease in sodium density. The result was a net negative reactivity insertion for about 600s. The reactor then returned to a critical condition. By this time, the power of the system had dropped to about 10% of nominal, but at higher temperatures.

6-1-3. Conclusions for the ULOF cases

The SSC predictions[3] show that PRISM would be able to withstand the nominal ULOF event (4 coastdowns), both with and without the aid of the GEMs (without GEMs some localized damage at the fuel-cladding interface could occur). The GEMs can dominate the neutronic feedbacks, and can bring the power down to the decay heat level within 500s, with a margin to sodium boiling of about 300K. The fuel temperatures decrease, and fuel damage is not a significant risk for this event. The case without the GEMs show the usual heat up of the structures, which activates the reactivity feedbacks, thus reducing the power. This causes the power to stabilize around 10% of rated, and temperatures are about 150K higher than when the GEMs are functioning (i.e., ~ 850K versus ~ 700K).

6-2. ULOF from 1 pump seizure

The EM pumps are pipe type devices with electrical coils placed outside the duct walls to supply the magnetic field required to develop the pump head. It was speculated that the coils suddenly stopped functioning during normal operation without any response from the system. The two pipes at each pump exit, which are connected to the inlet plenum of the core, would act as a short circuit in the system. The high pressure developed in the inlet plenum, from the three remaining pumps, would cause a bifrication in flow. Flow would be split between either continuing through the core or returning back to the cold pool by way of the seized pump.

The flow split was calculated using MINET. It was determined that 49% of the initial flow would continue through the core while 51% would be returned back through the seized pump. SSC was then supplied the normalized sodium flow rate to predict the system response.[3] The power dropped down to about 40% of rated within 100s and stabilized at about 60% after 400s. A review of the reactivity feedbacks showed the GEMs caused the power to drop. This forced the reactivity feedbacks to re-establish criticality at a power and flow below normal operating conditions. The fuel temperatures and core outlet sodium temperatures dropped below typical operating conditions.

6-3. ULOF/ with 1 pump failure

In this case, it is assumed that a normal ULOF event occurs with one pump coastdown missing entirely. Our calculations[3] for this event were compared against those provided by the applicant (i.e., G.E.). The sequence was as follows: One EM pump and its synchronous machine is assumed to fail. The signal to scram and trip the pumps is sent, but only the pumps responded (the pumps are not supposed to trip before the scram occurs). No control rod movement is assumed to have occurred.

Analysis of this event was complicated by the need to calculate the sodium flow rate through the reactor power using the MINET Code, and to then calculate the reactor power using SSC. Since the reactor power level and the sodium flow rate are closely coupled, a couple of iterations were needed to assure the two calculations were consistent.

In the MINET modeling, the pumps were represented individually, using the fairly detailed pump head and torque curves provided by GE. Some of the complexity is caused by the stoppage of one pump, which creates an open pipe-like pathway for the sodium to short-circuit back to the inlet of the other pumps. Normally, the flow through each pump quickly drops from 630 kg/sec. to about 300 kg/sec., and then coasts down. Instead, the flow per pump transitions to about 500 kg/sec., and the coastdown from that level is more protracted. With the line open, the circuit flow resistance is sharply reduced, leading to the surging in the pumps that are coasting down, and the reduced torque that causes the coastdown to be stretched out. As a result the coastdown of sodium flow rate through the reactor is not nearly as severe as one might otherwise anticipate.

The power dropped to decay heat levels by 500s. The GEMs supplied the largest amount of negative reactivity. After 1000s of calculations, the total reactivity was below 1 dollar negative, at which time the core exit sodium temperature reached about 950K. No fuel damage was predicted for this event.

6-4. Instantaneous stoppage of all pumps without SCRAM

This event goes well beyond bounding events in probability space and could best be described as "exceedingly unlikely". In principle, a massive earthquake might be postulated that causes the loss of all energy flow (including that from the synchronous machines) to the EM pumps and completely incapacitates the scram system. If, in addition, all this occurs simultaneously and the operator doesn't trigger the USS, then this exceedingly unlikely event could occur.

The SSC calculations were driven using the pump head and flow rate predicted by MINET. It is noted that initially the power decreases, although not nearly as fast as the flow decreases. By 20 seconds, the power is increasing, and a sodium-boiling driven power excursion develops. The total reactivity is initially dominated by the feedback from the GEMs, which quickly add 63¢ of negative reactivity, but is later dominated by the sodium density/void feedback. The sodium appears to be largely subcooled through the first 14 seconds, but large scale sodium boiling develops thereafter. Most of the other feedbacks are much smaller, although the Doppler feedback is accelerating at the end. The one crucial feedback that would have limited the severity of the event is the axial expansion of the fuel. However, our model is based on thermal expansion and does not include the rapid "prefailure extrusion" (rapid axial fuel expansion) that ANL predicts for rapid temperature increases.

This is the most serious accident analyzed for PRISM. Furthermore, this scenario is believed to be unrealistic since it requires failure of 4 EM pumps, six control rods, and four synchronous machines.

6-5. LOF with SCRAM with and without GEMs

In order to complete the LOF series of calculations, we ran several cases where scram did terminate the event. The (reference event) scenario was the same as the instantaneous loss of all EM pumps, including the coastdowns, except the control rods began to move with a 0.8s delay after the pumps were lost. The control rods (worth ~$18) required two seconds to complete their stroke. This scenario was analyzed for a core with, and without, the GEMs.

6-5-1. GEMs case

The event was initiated by an instantaneous loss of the EM pumps. At 0.8s, the control rods began to move and were completely inserted by 2.8s. (The insertion rate is likely faster, but we used the lowest credible rate.) Figure 6 shows the predicted power and flows. SSC indicates that the power does not rise during the delay to SCRAM, and that the fuel temperatures were not high enough to give cause for concern. A key parameter is the margin to sodium boiling, and Figure 7 indicates that the margin was maintained around 500K.

6-5-2. Without GEMs

The same transient was predicted as in Section 6-4-1., except the core model did include the GEMs reactivity contribution. The SSC predictions essentially duplicated the case with GEMs.

7. SUMMARY

The addition of the GEMs to PRISM has improved the predicted response to loss of flow events. The GEMs act like a passive control rod, and can substantially reduce the core power. Although the GEMs are worth only 69 cents, this is more than half of the core power and temperature defect (which is about $1.20). In the previous design configuration, the ULOF was thought to be the most serious event. The margins to sodium boiling and the predicted peak fuel temperature were greatly reduced when the GEMs were added. (G.E. also switched from a 271 fuel pin driver assembly to a 331 pin assembly, which also contributed to the changes.)

The peak temperatures predicted for the loss of flow events are presented in Figure 8. The figure indicates that some amount of cladding damage occurs, but since the graph doesn't indicate time, it must be added that the effects are minor. The exception is the instantaneous seizure of all four pumps. Although the event is far beyond realistic, the calculation does reveal that the event leads to sodium boiling and a large positive feedback.

Finally, the last series of predictions by SSC was for the loss of all four pumps with a scram. The case was analyzed with and without GEMs. The results indicated that if the scram occurs, the event will not be a major challenge for the design.

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