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SAFETY ASPECTS OF THE U.S. ADVANCED LMR DESIGN

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ABSTRACT

The cornerstones of the United States Advanced Liquid Metal Cooled Reactor (ALMR) program sponsored by the Department of Energy are: the plant design program at General Electric based on the PRISM (Power Reactor Innovative Small Module) concept, and the Integral Fast Reactor program (IFR) at Argonne National Laboratory (ANL). The goal of the U.S. program is to produce a standard, commercial ALMR, including the associated fuel cycle. This paper discusses the (1) U. S. regulatory framework for design of an ALMR, (2) safety aspects of the IFR program at ANL, (3) the IFR fuel cycle and actinide recycle, and (4) the ALMR plant design program at GE.

## 1. INTRODUCTION

The Integral Fast Reactor (IFR) program at ANL is responsible for the irradiation performance, advanced core design methodology, safety, and the fuel cycle (including the fuel cycle facility) for metal fuel for the ALMR. An engineering-scale fuel cycle facility is being designed by ANL for installation in the Hot Fuel Examination Facility-South (HFEF/S) adjacent to EBR-II scheduled for hot operation in fall 1990. General Electric is responsible for managing the reactor plant design involving wide participation by U. S. industry, National Laboratories, and international participation. The work is in the advanced conceptual design phase, and key development and testing tasks are proceeding in parallel.

The design for the ALMR is based on the PRISM (Power Reactor Innovative Small Module) concept originated by General Electric (Ref. 1) and on the IFR (Integral Fast Reactor) metal fuel concept developed by ANL (Ref. 2). The basic elements of the concept are: (1) metallic fuel, (2) liquid sodium cooling, (3) modular, pool-type reactor configuration, (4) an integral fuel cycle, based upon pyro-metallurgical processing and injection-cast fuel fabrication, with the fuel cycle facility collocated if so desired. The ALMR concept is particularly responsive to long-term energy supply needs by nature of metal fuel's high breeding capability and to the need for a secure fuel cycle with actinide recycling to ease the long-term waste disposal task.

In the ALMR concept, the liquid sodium coolant operates at atmospheric pressure, and maintains a design point margin to boiling greater than 400K (700°F). This eliminates the need for a pressurized primary system and thick-walled pressure vessels. With its high thermal conductivity and specific heat capacity, liquid metal cooling enables the ALMR to operate at decay heat levels in natural circulation, without the need for forced flow. Liquid metal cooling permits a compact core configuration that complements the neutronic advantages of metal fuel and an enhanced fast neutron energy spectrum.

## 2. REGULATORY FRAMEWORK

The ALMR Program is carried out taking into account continuing changes in the regulatory environment; changes which aim at achieving both improved safety and simpler, more predictable licensing of nuclear power plants. Key aspects are discussed below.

## Advanced Nuclear Power Plant Policy

The U.S. Nuclear Regulatory Commission (NRC) established a policy on Advanced Nuclear Power Plants. This policy encourages interaction between the designers of advanced concepts and the NRC at early stages of the design process. The policy states that advanced reactors should provide at least the same degree of protection of the public and the environment as is required for current reactors, and that advanced reactors are expected to provide enhanced margins of safety. The following desirable characteristics are suggested in the policy for advanced reactors.

- Highly reliable, less complex shutdown and decay heat removal systems; use of inherent or passive means are encouraged.
- Longer time constants to allow more time before reaching safety system activation.
- Simplified safety systems, reduced requirements for operator actions.
- Reduced potential for severe accidents and consequences by inherent safety, reliability, redundancy, diversity, and independence in safety systems.
- Reliable equipment in the balance of plant, or safety system independence from the balance of plant, to reduce challenges to the safety system.
- Easily maintainable equipment and components.
- Reduced radiation exposure to plant personnel.
- Defense in depth by multiple barriers to radiation release and by reducing the potential for and consequences of severe accidents.
- Features that are based on existing technology or which can be established by development programs.

## Safety Goal Policy

A policy on Safety Goals has also been established by the NRC. The central principle of this policy is that the risk posed by nuclear power plants to the neighboring population should not exceed one-tenth of one percent of the accidental fatality and cancer risk resulting from all other causes, and thus represent not a significant additional risk. The specific implementation of this policy has not yet been established by the NRC; however, the Advisory Committee for Reactor Safeguards (ACRS) has made its recommendations on this subject. The ACRS recommends as a general guideline that the likelihood of a large radiation release be less than $10^{-6}$ per reactor year. The ACRS also recommends separate guidelines for prevention of severe core damage and for mitigation, implying that some mitigative capability be required even if the safety goal is shown to be met by preventive means alone. The recommended guideline for mitigation is a minimum of less than one chance in ten for a large radiation release for the entire family of core melt scenarios.

## Standardization

In view of the serious difficulties and delays experienced in the nuclear power plant licensing process, there has been a general agreement that the U.S. licensing process requires reform and that a key ingredient is the certification of standard plant designs. The NRC has recently completed a new regulation titled Early Site Permits; Standard Design

Certifications; and Combined Licenses for Nuclear Power Plants (Code of Federal Regulations Title 10, Part 52). This regulation establishes the process for standard plant design certification. For new designs, which differ significantly from the established light water cooled reactor technologies, it calls for operation of a full-size prototype. The first commercial plant, licensed through the conventional licensing process, could serve the role of the prototype.

Evacuation Planning

A major contributor to regulatory delays in the U.S. has been emergency planning. Particularly troublesome aspects have been the detailed off-site evacuation plans and exercises, involving numerous local agencies, and the provisions for rapidly alerting the neighboring population to prepare for evacuation. It is generally agreed that on-site emergency planning is prudent, and also that provisions for off-site actions, such as communication links with certain local agencies are reasonable. However, the situation could be much simplified if requirements for detailed off-site evacuation plans and exercises and provisions for early warnings, such as sirens, could be eliminated. While the NRC has not reached a formal position on this subject, the NRC staff has proposed to consider eliminating the troublesome aspects mentioned above if the plant meets certain criteria. The NRC Staff proposes that these criteria be that the probability be less than $10^{-6}$ per year that the lower level Protective Action Guidelines (1 REM whole body,5 REM thyroid) are exceeded at the site boundary for 36 hours after an accident, considering all accident events.

## 3. INTEGRAL FAST REACTOR PROGRAM AT ANL

The two major goals of the IFR development effort are improved economics and enhanced safety. The enhanced safety goal has focused on designing for reliance on inherent processes to provide neutronic shutdown and reactor cooling in response to accident initiators. While they are not considered to be part of the reactor design basis, the consequences of unprotected (i.e., without scram) accidents have traditionally played a significant role in the evaluation of safety performance and the determination of containment requirements for licensability of liquid metal-cooled reactors.

The essence of the passive safety is to provide for intrinsic LMR performance characteristics that maintain the balance between reactor cooling capability and power production and prevent core disruption in instances when engineered safety systems have failed. These response characteristics are achieved by use of inherent mechanisms, hydraulic, and neutronic reactor system properties, which are determined by the choice and arrangement of reactor materials.

However, the most significant safety aspects of the IFR program result from its unique fuel design. A ternary alloy of uranium, plutonium, and zirconium, developed at Argonne, based on experience gained through more than 20 years operation of the EBR-II reactor with a uranium alloy metallic fuel. In the IFR concept, the ternary fuel is injection cast as cylindrical slugs and placed inside the cladding. Liquid sodium bond in the fuel-cladding gap provides an efficient heat transfer medium that, along with the high fuel thermal conductivity, maintains low fuel operating temperatures. The fuel-cladding gap is sized for a low smear density (typically 75%) to accommodate irradiation-induced fuel swelling to permit high burnup. Fuel elements of U-19Pu-10Zr in HT9 (Martensitic) and D9 clad have been irradiated in EBR-II to 14.4 at.% and 18.4 at.% burnup, respectively as of June 1989.

IFR Fuel Thermal Performance

Many of the superior safety performance characteristics of the IFR ternary alloy fuel design can be traced to its thermal and mechanical properties. The low temperature gradient across the fuel gives a correspondingly small zero power-to-full power Doppler reactivity swing, resulting in reduced control reactivity requirements and less external reactivity available for accidental insertion. The low operating temperature also yields a smaller positive Doppler reactivity input in unprotected transients on power reduction. This permits other reactivity feedbacks such as axial and radial core thermal expansion, to overcome the small positive Doppler input associated with power reduction, resulting in self-adjustment of the reactor core power to equal available decay heat removal capacity in loss-of-heat-sink (LOHS) and loss-of-flow transients (LOF).

Under accident conditions, transient heating of metallic fuel produces cladding loading dominated by the plenum pressure. The similarity of the fuel and the cladding thermal expansion and the compliance of the porous fuel lead to negligible Fuel-Cladding Mechanical Interaction (FCMI) cladding damage. Fuel melting in a metallic fuel element does not result in a significant clad loading because of the available porosity and small fuel density decrease on melting. The high thermal conductivity of metal fuel results in the hottest fuel being located near the core exit. Six experiments M2 to M7 have been performed in the TREAT transient reactor to determine margins to fuel pin failure, failure location, associated mechanisms and consequences and to characterize pre- and post-failure fuel relocation. A full range of fuel burnup and fuel and clad compositions are to be investigated. Tests M2, M3, and M4 were carried out using EBR-II driver fuel pins with U-5 Fissium fuel. Nine such pins were tested under slow overpower transients, with burnup and peak heating conditions being the key test parameters. Three of the pins were tested to cladding breach.

Three similar transient overpower tests (M5, M6, and M7) have been performed, using five D9-clad U-19 Pu-10Zr fuel pins with burnups up to 10 at.% and one low-burnup HT9-clad U-10Zr fuel pin. Two of the ternary fuel pins were tested to failure. Posttest analyses and examinations of the test pins from those tests have been completed. Additional TREAT tests will be performed to expand the database for IFR reference fuels to higher burnups, HT9-cladding, and to evaluate the impact of high Pu fuel.

The general results of the tests are that metal fuel has a large margin to pin failure (about 4 times nominal power in an 8 second period overpower transient), and significant molten fuel extrusion into the plenum region. In the experiments where pin failure occurred, considerable sweepout and dispersal fuel was observed without blockage formation. Fuel extrusion can provide a significant source of negative reactivity feedback in preventing severe core melt accidents (Ref. 3).

Metallic fuels interact metallurgically with iron-based cladding materials. During normal operation, the rate of solid-state inter-diffusion is no greater than the wastage in ceramic pins due to fission product attack of the inner cladding wall. During transient heating, cladding penetration by liquid fuel-cladding eutectic can contribute to cladding failure; however, the effect appears to weaken the cladding only by thinning the wall. Two major out-of-pile test programs on irradiated fuel are underway at ANL to investigate the impact of fuel-clad eutectics. They are the Fuel Behavior Test Apparatus (FBTA) and the Whole Pin Furnace (WPF) Tests. The FBTA apparatus tests a short segment (~ 1 cm) of an irradiated element to determine the cladding penetration by the fuel clad eutectic. The WPF program can test the combined effects of clad thinning, eutectic penetration, and fission gas loading upon clad integrity.

IFR Fuel Neutronic Performance

The metallic fuel form also offers favorable neutronics properties. Specifically, the absence of low mass moderating atoms in the fuel leads to a hard neutron spectrum, increasing the neutron production per neutron absorbed in the pin. This occurs both because of the higher $\eta$ value for $Pu^{239}$ with the harder neutron spectrum and because of the enhanced fast fission in $U^{238}$. The combined effect increases the number of neutrons available for breeding and parasitic losses ($\eta_{eff}$-1) from about 1.65 for oxide systems to about 1.95. Moreover, the effective heavy metal density is increased by use of the metallic fuel relative to the traditional oxide fuel form. Both of these characteristics can be used to increase core internal conversion ratio to a point where zero burnup reactivity swing is achievable in a three or four batch core with 12 to 20 month refueling interval.

The harder neutron spectrum attendant the metallic fuel form has two important effects on reactivity feedback coefficients. The negative Doppler reactivity coefficient, T dk/dT, is reduced by about a third relative to oxide systems. The positive sodium density coefficient becomes more positive by about 1/3. The net effect of the lower temperature rise across the pin radius and the shifts in reactivity coefficients is to make the coolant temperature rise component of the power coefficient larger than that which is vested in the fuel temperature rise. This partitioning of the power coefficient components (which is opposite to that of oxide fuel) is the key to favorable passive reactivity shutdown response attainable in the IFR.

IFR Fuel Local Faults Tolerance

Loss of cladding integrity of a fuel element during normal steady-state full power operation should not occur during the design lifetime of the fuel because of the margins included in the design of the fuel and cladding. However, stochastic fuel element failure must be anticipated, due to a random cladding defect which goes undetected during manufacture and inspection or due to random localized thermal, hydraulic or mechanical conditions within the fuel assembly.

Metallic fuel elements have a range of features that enhance their tolerance to local fuel failure events. These features include:

a. Fuel compatibility with sodium - no chemical reaction products.
b. High thermal conductivity of metal fuel - This results in very low fuel centerline temperatures, and reduced hot-spot temperatures for distorted geometries.
c. Low fuel clad mechanical interactions - reduction in clad loading.
d. Easy to fabricate fuel-allows easy attainment of high quality reprocessed fuel.

The major fraction of the original EBR-II experience was with uranium-fissium alloy. Recent experiments with ternary alloy fuel are confirming the anticipated excellent performance. Six Run Beyond Clad Breach (RBCB) experiments with predefected metal fuel have been completed with breach time of up to 223 days without observable fuel loss or opening of the breach site.

Anticipated Transients Without Scram

In the full spectrum of unprotected accidents, three specific initiators have emerged to serve as quantifiers of safety margins. They are: (1) the loss-of-flow (LOF) accident, in which power to the coolant pumps is lost, (2) the transient overpower (TOP) accident, in which one or more inserted control rods are withdrawn, and (3) the loss-of-heat-sink (LOHS) accident, in which feedwater supply to the steam generators is lost. For all three initiators, it is also assumed that the plant protection system fails to insert the shutdown control rods. These events are generally classed as anticipated transients without scram (ATWS). The key to successful prevention of core disruption under these conditions is the provision in the design for reactor performance characteristics that: (1) limit mechanisms leading to reactor damage, and (2) promote mechanisms responding to the upset condition and acting to restore the balance between reactor power production and cooling. An example of the first is the minimumization of the control rod TOP accident. This is achieved by maximizing the breeding potential and conversion of fertile uranium into fissile plutonium. This reduces the total burnup reactivity swing, the control reactivity requirement, and thus the available insertion reactivity.

Avoidance of both short- and long-term core disruption in ATWS events depends on (1) providing sufficient negative reactivity feedback to overcome the power-to-cooling mismatch and return the system to equilibrium at slightly elevated system temperatures, or alternately, (2) reducing the positive reactivity feedback components acting to resist the transition to system equilibrium. In this second respect, metallic fuel provides superior inherent safety performance in ATWS events, due to the reduced positive Doppler reactivity feedback associated with the small radial temperature gradient in the fuel (high thermal conductivity).

Full scale unprotected LOF and LOHS transients have been carried out in EBR-II (Ref. 4). These tests have confirmed the capability of the metal fueled IFR concept to respond to unscrammed accidents without core (coolant boiling or fuel failures) or system damage.

Severe Accidents

The probability of core meltdown is exceedingly remote; however, despite all possible design measures taken, a theoretical possibility of core meltdown (e.g., from complete and sudden loss of flow without scram or from complete, long-term loss of all decay heat removal systems) remains. Work to date has revealed three characteristics of particular importance to reduction of risk for these extreme scenarios: (1) the adiabatic Doppler feedback rate for metal fuel is equal or greater (more negative) than for oxide fuel, and (2) metallic fuel disperses upon melting giving rise to a powerful reactivity shutdown mechanism, and (3) resolidified molten metal fuel debris beds are highly porous and are coolable.

## 4. IFR FUEL CYCLE - ACTINIDE RECYCLE

The waste management potential of the IFR concept is promising but has yet to be demonstrated. The key technical elements of the IFR fuel cycle technology are based on metallic fuel and pyroprocessing. Pyroprocessing is radically different from the conventional PUREX reprocessing developed for the LWR oxide fuel. Chemical feasibility of pyroprocessing has been demonstrated. The next major step in the IFR development program will be the full-scale pyroprocessing demonstration to be carried out in conjunction with EBR-II. IFR fuel cycle closure based on pyroprocessing can also have a dramatic impact on the waste management options, and, in particular, on the actinide recycling.

For discussion of high-level waste management, it is convenient to categorize the nuclear waste constituents into two parts: fission products comprised of hundreds of various isotopes, and actinides comprised of uranium and the transuranic elements--neptunium, plutonium, americium, curium, etc.

The relative radiological risk factors for the fission products and actinides contained in the LWR once-through spent fuel waste are plotted in Fig. 1 as a function of time after discharge from the reactor. The radiological risk factor of the spent fuel is normalized to the cancer risk associated with the original natural uranium ore. Figure 1 illustrates the dominance of the long-term radiological risk of actinides over all other fission products. In a time span of the order of 200 years, the fission products decay to a sufficiently low level that their radiological risk factor drops below the cancer risk level of their original uranium ore. Actinides, on the other hand, have long half-lives and their radiological risk factor remains orders of magnitude higher than that due to fission products for tens or hundreds of thousands of years. From this point of view, therefore, there is a strong incentive to separate actinides and recycle them back into the reactor for in-situ burning.

The benefit is in the fact that the effective lifetime of the nuclear waste is reduced from millions of years to about 200 years. This would have an enormous impact on assuring the integrity of high-level waste for its lifetime and ultimately on the public acceptance of the nuclear power. But even if the actinides are removed and the lifetime of the high-level waste is reduced to hundreds of years, the need will remain for a geological repository.

IFR pyroprocessing makes actinide separation practical. In the IFR process, most of the actinide elements accompany the plutonium product stream. Those that do not, initially stay with the rare earths. But pyrochemical processes can separate rare earths from actinides remarkably well, in contrast to the difficulty of these separations in the PUREX process. The hardened IFR neutron spectrum is better for the actinide burning than that of any other reactor type. Thus the potential of the IFR concept to achieve the actinide recycling is very promising, although further research and development work is needed to fully establish its feasibility.

Material Control

The ALMR concept with the Integral Metal Fuel cycle is also responsive to concerns over the control of plutonium, the so-called "nonproliferation issue." In this fuel cycle, the plutonium is never completely separated from uranium, nor from much of the fission products, so that it remains in a diluted and highly radioactive form, making diversion extremely difficult, and requiring extensive processing to separate the plutonium. The plant layout also incorporates an optional on-site fuel cycle facility. With this option, fuel transport and the attendant diversion concerns can be even further reduced.

## 5. ALMR PLANT DESIGN AT GE

Safety Approach

The overall objective of the ALMR program is to develop a system with improved safety and competitive economics for the long term. The design is to

- be responsive to long term resource, waste, and security requirements,
- be responsive to the new directions in regulatory requirements, and
- incorporate lessons learned from past practices where certain design characteristics caused problems in licensing, public acceptance, and economic viability.

Examples of design characteristics found in the past to be troublesome in the licensing area are as follows.

- Reliance on multiple active systems to maintain a safe state.
- Resultant vulnerability to loss of electrical power and questions of reliability, especially that of decay heat removal.
- Reliance on operator actions to perform safety functions.
- Vulnerability to operator errors, especially in interfering with automatic safety actions.
- Difficulty in meeting anticipated transient without scram (ATWS) requirements.
- Vulnerability to large seismic events causing multiple failures.

- Vulnerability of the reactor system to faults in the balance of plant.

The safety goals and requirements established for the ALMR program include the conventional ones established in the past for nuclear power plants, and for sodium cooled power plants in particular, such as leak protection, fire mitigation, protection from natural phenomena, etc. However, in response to the recently evolving regulatory framework in the U.S. and the lessons learned from past experience, a number of additional safety goals were established for the ALMR program which may not have been used, or at least not emphasized, in the past. The most important of these are listed below.

- Passive decay heat removal, not vulnerable to operator errors.
- Strong inherent negative reactivity feedback for core reactivity control to maintain a safe state in the event of anticipated transients without scram (ATWS).
- Reactor protection (scram) system, well separated from the plant control system, with failure to shutdown to be less than $10^{-6}$ per demand.
- No operator action required to reach and remain in a safe state.
- Operator action cannot inhibit or override safety actions.
- High margins in ultimate seismic capability.
- Very low core damage probability, below $10^{-6}$ per year.
- Accidental radiation release probabilities and characteristics such that detailed off-site evacuation planning, exercises, and early warning will not be required.
- Passive and other innovative safety characteristics to be demonstrable in a prototype without damaging the plant.

## Innovative Plant and Safety Features

### Modular Reactor in Underground Silo

The ALMR plant layout is shown in Fig. 2. It consists of three identical power blocks of 465 MWe, for a total plant net electrical rating of 1395 MWe. Each power block consists of three reactor modules with individual thermal ratings of 471 MWt, each reactor module has its own steam generator which jointly supply steam to a single turbine generator. The reactor modules and the intermediate heat transport systems are underground, providing improved protection from tornadoes, missiles, and sabotage. The small thermal ratings of the individual reactor modules ease the task of decay heat removal, and reduce the potential consequences of a core damaging accident.

### Seismic Isolation

The reactor module with all the safety related systems rests on 20 seismic isolators (Fig. 3). The isolators essentially decouple the system from horizontal accelerations in the high frequency range which are of greatest importance in establishing design margins. Work is in progress to determine the cost of raising the safe shutdown earthquake capability from the original 0.3 g requirement to the range 0.5 to 0.75 g. The primary sodium circulation pumps are electromagnetic, with synchronous machines providing backup power for flow coastdown in the event that normal power to the pumps is lost. The synchronous machines are on the seismically isolated platform, and thus the potential for a seismic event degrading the flow coastdown capability is minimized.

Passive Decay Heat Removal

Normal decay heat removal is done by passing steam from the steam generator to the condenser. Two passive backups are provided. The first of these is natural circulation of the intermediate heat transport system and natural circulation atmospheric air flow through a shroud surrounding the steam generator. The ultimate decay heat removal is by the reactor vessel auxiliary cooling system: atmospheric air, naturally circulating around the containment vessel in the underground reactor silo, as indicated in Fig. 3. This system is always in operation, and is highly immune to human interference and to structural failure. Because of the large and multiple air passages, a 90% blockage can be tolerated with temperatures remaining below ASME level D limits (700C, 1300F).

Limited 1E Power Requirements

Because of the passive characteristics of the design, the requirement for Class 1E power are low, approximately 50 kilowatts for a nine module plant, and can be supplied entirely from batteries. Continuous power is required only for the reactor protection system sensors, electronics, and monitoring displays, together with basic lighting and ventilation for the operators.

Reactivity Shutdown and Control

The reactivity shutdown system consists of six control rods, associated drives and electronics. The requirement established for the system is that the probability of failure to shut down be less than $10^{-6}$ per demand. The insertion of any one of the six rods will bring the core to cold shutdown conditions. Each rod can be inserted into the core three different ways: rod run-in by the plant control system, fast run-in initiated by the reactor protection system, and gravity drop also initiated by the protection system. The reactor protection system is safety grade, automatic, well separated from the nonsafety grade plant control system, and located entirely in the reactor module vaults, away from the control room.

A key requirement placed on the ALMR design is that it maintain a safe state, through passive means, for anticipated transient without scram (ATWS) events involving loss of primary flow, loss of heat sink, and control rod runout. This is achieved through strong negative temperature coefficients, conservative power rating, and a core design with low excess reactivity. A key contributor to these characteristics is the metal fuel. The metal fuel has excellent negative feedback characteristics and furthermore, it provides superior neutron economy, so that the core can be designed for very low reactivity swing during the refueling cycle, and thus, with very low excess reactivity held down by control rods.

Figure 4 shows the reactor behavior under a combined condition of loss of forced primary flow and loss of heat removal by the intermediate heat transport system at full power and without scram. Through the inherent reactivity feedbacks and the passive air flow heat removal, the core outlet sodium temperature settles below the established temperature limit of 700C (1300F) and high margin to sodium boiling is maintained. While the fuel-clad interface temperature exceeds the limit for several minutes, no fuel failures are expected.

Figure 5 shows the reactor behavior under an accidental rod withdrawal at full power without scram, adding 40 cents reactivity at the maximum capability of the control system, 2 cents per second. The power peaks at about 170% and settles at about 135% of rated. The sodium and fuel temperatures again settle out below the 700C (1300F) limit, and no fuel failures are expected. Various options are under considerations for limiting the anticipated inadvertent rod withdrawal to about 40 cents. The ideal situation would be if the total core excess reactivity at full power and with all rods withdrawn to their mechanical limits, could be limited to this figure. In the event that this proves to not be possible, because of core reactivity changes during the refueling cycle and because of uncertainties, other options are under consideration to limit reactivity addition by inadvertent rod withdrawal.

Prototype Test

The ALMR effort includes the planning and design of a prototype plant test program to demonstrate the passive reactivity control and decay heat removal features, even under failure to scram conditions, and to provide a basis for standard design certification. While the exact route to the prototype test can remain flexible in response to possible changes in the need for the ALMR, the current judgement is that the lowest cost and risk approach is the building of a single reactor module, preferably on a U.S. Government site, for the purposes of the prototype test. This could be done in two phases: building only the reactor module and intermediate heat transport system for a safety test phase, and then adding the turbine-generator for a power operation phase. Alternately, the prototype testing could be performed on the first commercial power producing plant consisting of a single power block.

NRC Regulatory Review of the ALMR

Both the ALMR design team and the NRC recognize the desirability of interaction with each other during the design process to assure regulatory approval of the final product. Such interaction has been an integral part of the ALMR program plan in the form of regulatory review cycles, starting at the conceptual design stage. The first review cycle was accomplished during 1987 and 1988. A second review cycle is planned for the advanced conceptual design phase during the next three years. More formal regulatory review will begin with the preliminary design phase leading to preliminary design approval, and will continue during the final design phase, leading to final design approval and the licensing of the prototype plant. Finally, for the completion of the safety testing in the prototype, standard design certification will be obtained, opening the door to commercialization. The current reference schedule reaches certification about the year 2003; however, the schedule is flexible and can be adjusted to meet changing requirements.

A Preliminary Safety Information Document (PSID) was submitted to the NRC for review in November 1986. This document is similar to a Preliminary Safety Analysis Report (PSAR), but with less detail because of the conceptual nature of the design. During 1987 and 1988, numerous meetings and discussions were held among the design team, the NRC Staff, and the Advisory Committee for Reactor Safeguards (ACRS) in the course of the review. The results of the review are the draft Safety Evaluation Report (SER) prepared by the NRC Staff (Ref. 5) and the review letter by the ACRS (Ref. 6) reporting the findings.

The safety evaluations by the NRC Staff and by the ACRS are generally favorable. Overall, they find that the design is responsive to the NRC's Advanced Reactor Policy, namely, that the design has the potential for a level of safety at least equivalent to current plants, and that the design provides several passive and other desirable features enhancing the safety of the power plant. The passive reactivity feedback and decay heat removal features are recognized and credited by the reviewers, as are the long response time and low risk of core damage under many severe challenges to the plant, and the reduced dependence on and vulnerability to human actions and errors. The NRC recommendation was that the design and development continue.

As expected from any safety evaluation, and particularly from a preliminary, first-round evaluation, a number of issues and concerns were identified by the reviewers. Most of these are of a nature such that they can be addressed and satisfied as the design work progresses and more information and detail become available. Examples are the role and protection of the operators, sabotage resistance, sodium fire protection, and shutdown system diversity.

The issues raised in the review which have the potential for significant impact on the design and the overall approach are severe accidents and containment. That these major issues would be raised was not completely unexpected. The submittal for the first round of NRC review emphasized core damage prevention. The analyses show that the inherent reactivity effects and the passive decay heat removal reduce the probability of sodium boiling and fuel melting to a level sufficiently low to meet the NRC safety goal by means of core damage prevention alone, namely, that the probability of a significant radiological release is less than $10^{-6}$ per year. Nevertheless, analyses of radiological release were made to show compliance with regulatory site limits. The analyses conservatively assume that the containment boundary, which is completely sealed during operation, leaks directly to the atmosphere at a rate of 0.1% per day, and that all the fuel cladding has failed. The results show that with 100% of the noble fission gases, 0.1% of the other fission products, and 0.01% of the actinides released at the rate of 0.1% per day, the regulatory limits of 25 REM whole body and 300 REM thyroid dose are met for 2 hours at 0.5 mile and for 30 days at 2 miles. The results also show that with a still conservative but reduced release fraction of solid fission products and fuel materials, the 1 REM whole body and 5 REM thyroid Protective Action Guideline doses are met for 36 hours at the site boundary.

The submittal for the first round of NRC review was made before the ACRS made the recommendation that in meeting the safety goal, a mitigation capability of at least a factor of 10 be required for the entire family of core melt scenarios. In the first submittal, specific analyses were not made of potential impact on the containment boundaries by core energetic events or molten fuel movement. Such severe events were considered only in the probabilistic risk assessment and only in a simplified manner.

Both the ACRS and the NRC Staff have expressed concern about the unconventional containment concept used in the design, that is, the absence of a separate strong, pressure and temperature resistant containment structure completely surrounding the primary system, as additional protection for very low probability and unforeseen accidents. In the current design, the containment boundary is the reactor vessel head and the containment vessel (often called guard vessel) which completely surrounds the primary reactor vessel. The vessel head boundary is backed up by the head access area at a secondary confinement boundary. Concerns were also expressed about the positive sodium void effect and the possibility of fuel failure propagation in the event of fuel assembly blockage, however low the probability of such events may be.

Future Regulatory Review

The current phase of the ALMR effort, the advanced conceptual design, will lead to the next round of regulatory review by the NRC and the ACRS. For this phase of the work, the safety approach and goals of the ALMR program described earlier are retained, but are augmented in response to the evaluation received in the first round of regulatory review.

The goal of very low core damage probability through passive means is retained. In addition, very low probability events which could lead to severe core damage will be explicitly considered and their potential impact on the system boundaries will be analyzed to investigate the capabilities of the design to limit radiation releases. Various alternative containment concepts with additional or improved barriers to radiation release will be considered and evaluated, while retaining the highly reliable and passive decay heat removal system. The advanced conceptual design phase also includes numerous tasks which are responsive to other issues raised in the regulatory review and which aim to reduce the cost and improve the commercial aspects of the design.

## 6. ACKNOWLEDGEMENTS

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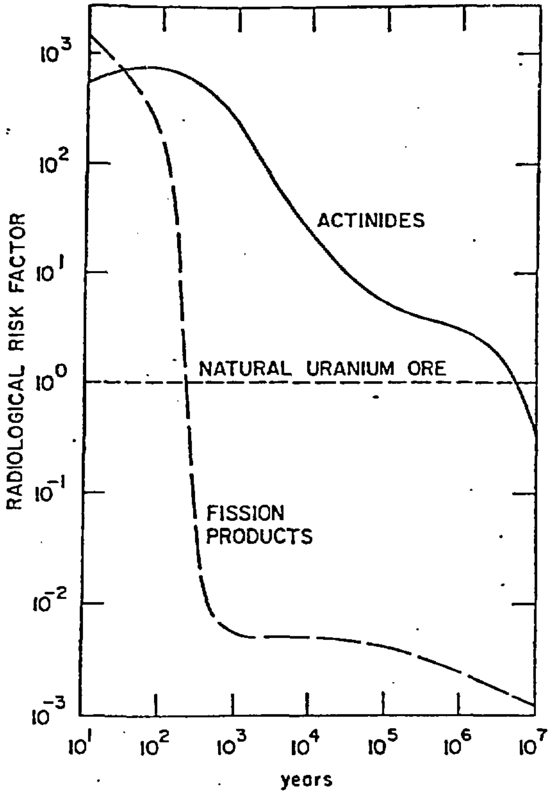


Figure 1. Radioactive Decay of High Level Reactor Waste

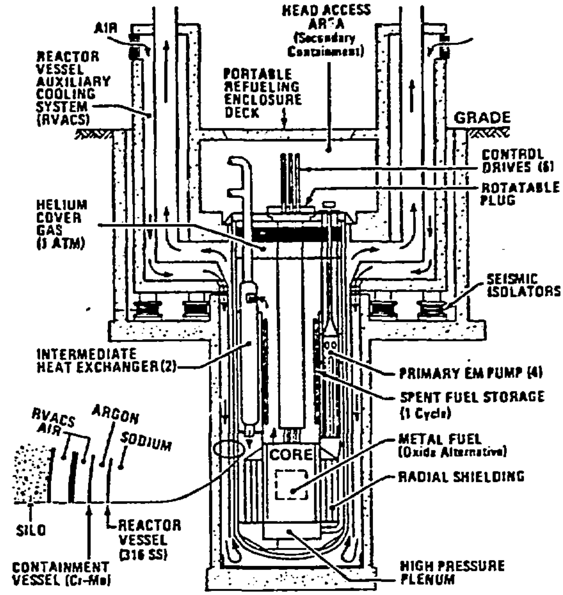


Figure 3. Reactor Module in Below-Grade Silo

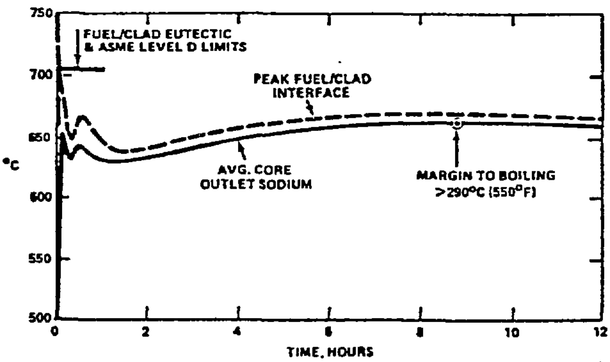


Figure 4. Sodium and Fuel Temperatures After Loss of Primary Forced Flow and Loss of Cooling by the Intermediate Heat Transport System at Full Power and With Failure to Scram

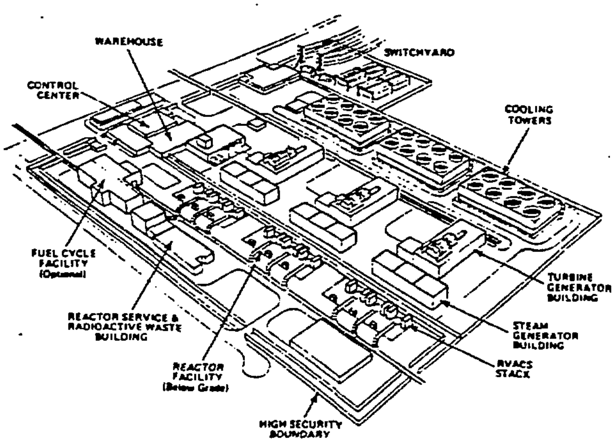


Figure 2. PRISM ALMR Power Plant-1395 MWe

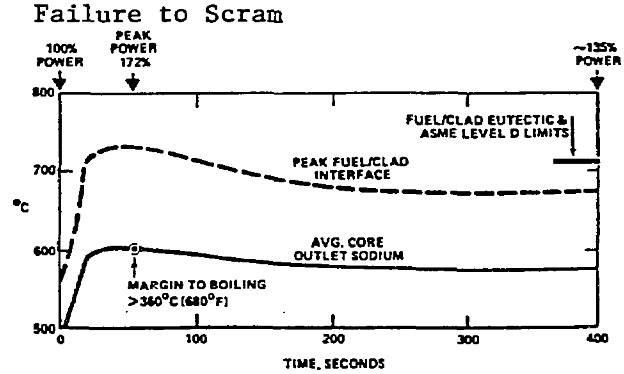


Figure 5. Sodium and Fuel Temperatures Resulting from Withdrawal of Control Rods at Full Power and With Failure to Scram (40¢ Reactivity Insertion at 2¢/Second)