
Preapplication Safety Evaluation Report for the Sodium Advanced Fast Reactor (SAFR) Liquid-Metal Reactor

U.S. Nuclear Regulatory Commission

**Office of Nuclear Reactor Regulation
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T. L. King, R. R. Landry, E. D. Throm, J. N. Wilson



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ABSTRACT

This safety evaluation report (SER) presents the results of a preapplication design review for the standard modular Sodium Advanced Fast Reactor (SAFR) liquid metal reactor (LMR), NRC Project No. 673. The SAFR conceptual design was submitted by the U.S. Department of Energy (DOE) in accordance with the U.S. Nuclear Regulatory Commission's (NRC) "Statement of Policy for the Regulation of Advanced Nuclear Power Plants" (51 Federal Register 24643). This policy provides for early Commission review and interaction with designers and licensees. The standard SAFR design consists of four identical reactor modules, referred to as "power paks," each with a thermal output rating of 900 MWt, coupled with four steam turbine-generator sets. The total electric output is 1400 MWe. The reactors are liquid sodium cooled and use metallic-type fuel, a ternary alloy of U-Pu-Zr. The design includes passive reactor shutdown and decay heat removal features.

The staff and its contractor, the Brookhaven National Laboratory, have reviewed this design placing emphasis on those unique features in the design that accomplish the key safety functions for reactor shutdown, decay heat removal, and the containment of radioactive materials.

This SER presents the NRC staff's preliminary technical evaluation of the safety features in the SAFR design, including the projected research and development programs required to support the design, and the proposed testing needs. It must be recognized that final conclusions in all matters discussed in this SER require approval by the Commission.

During the NRC staff review of the SAFR conceptual design, DOE terminated work on this design in September 1988. This SER documents the work done to that date and no additional work is planned on the SAFR.

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PREFACE

This safety evaluation report (SER) for the Sodium Advanced Fast Reactor (SAFR) liquid metal reactor (LMR) is being issued to document the work done to date on the review performed by the U.S. Nuclear Regulatory Commission (NRC). This review was performed at the request of the U.S. Department of Energy (DOE) consistent with the NRC's advanced reactor policy statement (51 Federal Register 24643). DOE terminated work on the SAFR design and the NRC plans no further work on the SAFR design. This SER must be used with caution and no conclusions on the overall acceptability of the SAFR concept should be drawn from this SER.

The staff positions and preliminary conclusions on all matters discussed in this SER are not final. In particular, it should be noted that the Commission has not endorsed SER sections pertaining to the key policy issues discussed in SECY 88-203, July 15, 1988, of (1) the selection of the accidents to be analyzed, (2) the use of a mechanistically derived radionuclide source term for plant siting evaluations, (3) the acceptability of a design without a conventional containment structure, and (4) the acceptability of the proposed offsite emergency planning. This review was limited to the SAFR concept only, and potential onsite fuel cycle facilities for an LMR are not addressed.

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ABBREVIATIONS

A	
ACI	American Concrete Institute
ACRS	Advisory Committee for Reactor Safeguards
AGB	auxiliary generator building
ALARA	as low as is reasonably achievable
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	anticipated operational occurrences
APTS	automated plant trip system
Ar	argon
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
B	
BDBA	beyond design basis accident(s)
BDBE	beyond design basis event(s)
BE	bounding event(s)
BNL	Brookhaven National Laboratory
BOP	balance of plant
BTP	branch technical position
B ₄ C	boron carbide
C	
CAM	containment atmosphere monitor(s)
CAS	compressed air system
CB	control building
CCP	core component pot(s)
CCTV	closed-circuit television
CE	Combustion Engineering
cfm	cubic feet per minute
CFR	Code of Federal Regulations
CIS	containment isolation system
CRBR	Clinch River Breeder Reactor
CRBRP	Clinch River Breeder Reactor Project
CVCS	Chemical and Volume Control System
D	
DBA	design-basis accident(s)
DBE	design-basis event(s)
DBEWS	design-basis event without scram
DBL	design-basis leak
DEG	double-ended guillotine
DOE	Department of Energy
DOT	Department of Transportation
DRACS	direct reactor auxiliary cooling system
DRHX	direct reactor heat exchanger

Abbreviations

EBR	experimental breeder reactor
EC	event category(ies)
ECCS	emergency core cooling system
EER	electrical equipment rooms
EFDS	equipment and floor drainage system
EOP	emergency operating procedures
EPA	Environmental Protection Agency
EPZ	emergency planning zone
ERDA	Energy Research and Development Administration
ETEC	Energy Technology Engineering Center
EVTM	ex-vessel transfer machine

FCF	fuel cycle facility
FEMA	Federal Emergency Management Agency
FFTF	Fast Flux Test Facility
FHC	fuel handling cell
FHM	fuel handling machine
FMEA	failure modes and effects analyses
FRS	floor response spectrum
FSAR	Final Safety Evaluation Report
FTC	fuel-transfer cell

GDC	general design criterion (criteria)
gpm	gallons per minute

HEDL	Hanford Energy Development Laboratory
HEPA	high-efficiency particulate air
HP	health physics
H&V	heating and ventilation
HVAC	heating, ventilating, and air conditioning

IEEE	Institute for Electrical and Electronics Engineers
I&C	instrumentation and controls
IFR	integral fast reactor
IGRPS	inert gas receiving and monitoring system
IHTS	intermediate heat transfer system
IHX	intermediate heat exchanger
IS	interim storage
ISI	in-service inspection
IVHM	in-vessel handling machine

LMFBR	liquid-metal fast breeder reactor
LMR	liquid-metal reactor(s)
LOCA	loss-of-coolant accident
LOF	loss of flow
LOHS	loss of heat sink
LOOP	loss of offsite power
LPZ	low population zone
LWMS	liquid waste management system
LWR	light-water reactor(s)

M	
McR	main control room
MEFS	moderate energy fluid system
MGSS	miscellaneous gas supply system
MSSS	main steam supply system
Mwd	mega-watt days
N	
Ncws	normal chilled-water system
NDHX	natural-draft heat exchanger
NEMA	National Electrical Manufacturers Association
NFPA	National Fire Protection Association
NIMB	nuclear island maintenance building
NNS	non-nuclear safety
NRC	Nuclear Regulatory Commission
NSFPS	non-sodium fire protection system
NSSS	nuclear steam supply system
O	
OB	operating basis earthquake
OSHA	Occupational Safety and Health Agency
P	
PAG	protective action guideline
PAM	post-accident monitoring
PASS	post-accident sampling system
PCS	plant control system
PCSS	primary control shutdown system
PDC	principal design criterion (criteria)
PFR	prototype fast reactor
PHTS	primary heat transport system
PLOHS	protected loss of heat sink
PPS	plant protection system
PRA	probabilistic risk assessment
PRISM	power reactor innovative small module
PSAR	Preliminary Safety Analysis Report
PSID	Preliminary Safety Information Document
PSWS	plant service water system
PWR	pressurized water reactor
Q	
QA	quality assurance
R	
RACS	reactor air cooling system
R&D	research and development
RCB	reactor containment building
RG	regulatory guide(s)
RI	Rockwell International (Rocketdyne Division)
RSB	reactor service building
RV	reactor vessel
RWS	radioactive waste system

Abbreviations

S AFR	Sodium Advanced Fast Reactor
SAR	safety analysis report
SAS	secondary alarm station
SASS	self-actuating shutdown system
SB	site boundary
SC	safety class
SER	safety evaluation report
SFPS	sodium fire protection system
SG	steam generator
SGB	steam generator building
SHRS	shutdown heat removal system
SRP	standard review plan
SRSS	square root of the sum of the squares
SS	stainless steel
SSCs	structures, systems, and components
SSE	safe shutdown earthquake
SSST	site suitability source term
STP	safety test plan
SWMS	solid waste management system
SWS	sanitary water system
SWRPRS	sodium-water reaction pressure relief system

T BD	to be determined
TGB	turbine generator building
TID	technical information document
TMI-2	Three Mile Island Unit 2
TOP	transient overpower
TREAT	transient reactor test facility
TSC	technical support center

U IS	upper internal structure
ULOF	unprotected loss of flow
ULOHS	unprotected loss of heat sink
UPS	uninterruptable power supply
UTOP	unprotected transient overpower

Z PGA	zero peak ground acceleration
ZPPR	Zero Power Plutonium Reactor

1 INTRODUCTION AND SUMMARY

1.1 General

The staff of the U.S. Nuclear Regulatory Commission (Commission or NRC) prepared this safety evaluation report (SER) to document its review of a conceptual standardized design for a modular sodium advanced fast reactor (SAFR). The Preliminary Safety Information Document (PSID) for the SAFR design was submitted for early review and Commission interaction by the U.S. Department of Energy (DOE) in accordance with the NRC's "Statement of Policy for the Regulation of Advanced Nuclear Power Plants" published in the Federal Register (Ref 1.1). DOE provided the design documentation (Ref. 1.2) and later supplemented and amended it as the review progressed. The staff's review is considered a preapplication review for the purpose of providing guidance early in the design process on the acceptability of the SAFR design. This review follows the process and guidelines given in NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants" (Ref 1.3).

This SER does not constitute an approval of the SAFR design but rather documents the Commission's preliminary guidance regarding licensing requirements, including the acceptability of the DOE-proposed supporting research and development programs. This SER is intended to guide the development of further documentation to support licensing of the SAFR concept. The Commission can only make a final decision regarding licensing after the staff has found the SAFR design acceptable and an applicant has complied fully with the administrative processes of nuclear reactor licensing, including public notification and participation, as required in Title 10 of the Code of Federal Regulations (CFR), entitled "Energy."

A plant utilizing the SAFR design concept will consist of one or more independent, power-generating units called "power paks." A typical site layout for a 4-module unit is shown in Figure 1.1. Each power pak will consist of a sodium-cooled reactor system that will provide heat from the primary coolant through two intermediate coolant loops to two steam generators that will send steam to drive the single turbine generator. A standard SAFR module will consist of one power pak which will produce 900 MWt (350 MWe). The power pak reactor system will employ a compact pool-type design fueled by a metallic alloy of uranium, plutonium, and zirconium (U-Pu-Zr) contained within a sodium bond in a special stainless steel (HT-9) cladding. The SAFR safety design is based on passive reactor shutdown and decay heat removal systems. The major operational specifications for a single module of the SAFR design are given in Table 1.1. An isometric view of the important features of a single module in the SAFR design is shown in Figure 1.2. A simplified schematic plan of the overall plant flow diagram is provided in Figure 1.3. Figures 1.4 and 1.5 show the elevation and plan views, respectively, of the nuclear steam supply system (NSSS). Elevation and plan views of the reactor assembly are shown in Figures 1.6 and 1.7, respectively.

SAFR was designed by the Rocketdyne Division of Rockwell International (RI) in cooperation with Bechtel National, Inc., and Combustion Engineering, Inc. (CE). The Argonne National Laboratory (ANL), an associate of the SAFR team, has provided major analytical and test support. RI is responsible for system engineering and integration, the reactor system, the heat transport systems, the refueling systems, safety and licensing, and the marketing and commercialization assessment. CE provides support for licensing and the marketing and commercialization and is also responsible for the reactor vessel and internals, heat exchangers, and control room design. Bechtel also provides licensing and marketing and commercialization support and is responsible for building and structures, power conversion systems, and auxiliary systems.

1.2 SAFR Approach and Objectives

The stated goals of the SAFR program are to develop a conceptual design that minimizes plant cost and maximizes passive (inherent) safety. These goals are

- Develop a widely usable standard plant having limited financial risk, a short onsite construction schedule, and competitive energy costs.
- Minimize the complexity of managing the project by using a simplified design and minimizing on-site, license-related work.
- Ensure public and investment protection by avoiding the need for fast operator action and by obviating the need for evacuation because of passive (inherent) safety features.
- Ensure minimum development risk by using existing technology.
- Provide for a high-capacity factor, long plant life, low personnel radiation exposure, and effective security and safeguard measures.

The third goal defines the general safety objectives for SAFR. The overall specific safety objective was stated as: to provide, at the lowest practical cost, a safe design that (1) responds passively to all credible events, (2) minimizes the potential for severe accidents, and (3) eliminates the need for extensive offsite evacuation planning by demonstrating its low risk. The four guiding principles used by the designer to translate the overall objective into specific safety criteria are

- Capitalize on passive safety attributes of the pool concept and metal fuel.
- Design safety systems to be independent of power-generation systems.
- Emphasize accident prevention.
- Keep the design simple.

In response to these safety criteria, the SAFR designers have reduced the number of systems, components, and structures classified as safety related in comparison to both light-water reactors (LWRs) and other liquid-metal reactor (LMR) designs. The main control room and balance-of-plant (BOP) items are proposed to meet commercial, industrial-grade standards, as well as are many items associated with the nuclear island, such as diesel generators and cooling water systems. DOE contends that because of these characteristics, certain key changes in traditional approaches to safety are justified. Specifically, for SAFR, DOE has proposed

- a siting source term based on radioactive materials released from the melting of a single fuel subassembly (rather than the traditional TID-14844 releases)
- no conventional containment building
- no requirements for preplanned offsite emergency evacuation or drills

1.3 Comparison With Other LMRs

The designers of SAFR considered worldwide LMR experience to date. This experience consists of operation of eight facilities. The major facilities are listed in Table 1.2. Each of these facilities uses a unique combination of shutdown systems, shutdown heat removal, and containment or confinement. Operating experience with the newer smaller facilities, such as EBR-II, FFTF, and Phenix has been very good. The largest facility, Super Phenix, has had operational problems in the two years since full-power operation began. In general, the SAFR designers have tried to incorporate into the design what the nuclear industry has learned from world-wide experience.

1.4 Scope of Review

The DOE supplied the following documents on the SAFR conceptual design and the staff reviewed them:

- Preliminary Safety Information Document (PSID), Volumes 1-4 (Ref. 1.4).
- Safety Test Plan (STP) (Ref. 1.5).

These and other DOE documents and information supplied by DOE contractors are formally identified (as are all references) at the end of each chapter in which they are used. It should be noted, however, that DOE classifies many of the documents as "Applied Technology." Documents so classified are not currently in the public domain and any such documents should be requested from DOE.

Because of the conceptual nature of the SAFR design, this review concentrates on those features, issues, and research and development activities considered key to its safety and viability. Issues of a policy nature have been identified according to the way in which the SAFR designer proposes to meet NRC safety requirements that will require Commission review and guidance. These are

- the selection of events that must be considered in the design
- siting source-term calculation and use
- adequacy of containment
- adequacy of emergency planning

These issues are discussed in SECY 88-203 (Ref. 1.6), which the staff sent to the Commission on July 15, 1988. At the present time, the Commission has not taken a position on these issues. The approach and conclusions described in this SER are consistent with the criteria presented in SECY 88-203. The Advisory Committee for Reactor Safeguards (ACRS) also reviewed the key issues (Ref. 1.7). Many other technical issues were identified during the review. These are addressed in the appropriate sections of this SER, as detailed in Section 1.6.

Each chapter or major section within each chapter of this SER identifies the scope of the staff review and points out the review limitations and the items deferred. The staff directed its review principally in the areas of fuel design, reactor physics, reactor vessel integrity, the passive heat removal system, and the safety analysis. Although less effort was expended in areas of instrumentation, control, electrical systems, auxiliary systems, occupational exposures, human factors, safeguards and security, and balance-of-plant items, the limited review in some of these areas led to the identification of important issues. The staff did not review the areas of mechanical equipment design, the modeling of fission-product transport, and other phenomena involving chemical processes for which experimental data are key to the staff acceptance of any models proposed.

1.5 Review Approach and Criteria

In reviewing the SAFR design, the staff was guided by the recent Commission policies on advanced reactors (Ref. 1.1), severe accidents (Ref. 1.8), safety goals (Ref. 1.9), and standardization (Ref. 1.10). Further guidance on the use of these policies and on the review process is provided in NUREG-1226 (Ref. 1.3). In general, the staff used an approach that parallels the review approach used on LWRs. The staff evaluated the many factors that contribute to LWR safety (such as conservative design practices directed toward accident prevention and the use of redundancy and diversity in accomplishing key safety functions) to ensure that similar factors or adequate substitutes were provided for the SAFR design. The staff did not determine acceptability of the design by measurement against a single factor (such as the safety goals) or by comparing PRA results with LWR results. Although probabilistic risk assessment (PRA) analysis is a useful tool in evaluating a design, the staff does not consider it to be developed to the point where it can be used as the primary measure of reactor safety or acceptability. The staff relied primarily on a deterministic review to

Introduction and Summary

ensure that adequate conservatism and defense in depth are maintained in the design. This review also serves as the basis for making a judgment on the potential of the SAFR design to provide protection to the public and the environment at least the equivalent of that offered by the current generation of LWRs.

Central to the staff's evaluation was the treatment of the policy issues discussed in Sections 1.4 and 3.2. These policy issues arose from the different approach used in SAFR to accomplishing key safety functions. The staff's approach in these areas can be summarized as follows. Because of the high potential to prevent core damage, a mechanistic analysis of radionuclide releases for a range of low-probability events (equivalent to severe accidents in LWRs) was substituted for the traditional nonmechanistic large source term (which is representative of a source term from a core-melt accident) utilized in LWR siting. The safety goal policy helped the staff define the range of low-probability events which need to be considered. However, provisions were maintained for engineering judgment to bound uncertainties in the selection of these events. Similarly, the review of a design without a conventional containment building was based on a mechanistic analysis of a range of low-probability events and on the potential for demonstrated capability of the design (via prototype testing) to perform as predicted. Inherent in this approach is a shift in emphasis in defense in depth from accident mitigation to accident prevention and plant protection. With respect to emergency planning, the potential for a long response time before core damage and the release of radiation was considered. The long response time may compensate for certain emergency planning requirements.

Consistent with the above, the review followed the general approach of a construction permit review as described in the Standard Review Plan (SRP) (Ref. 1.11) but was far less comprehensive, emphasizing only items believed to be important to feasibility and safety and deferring to a later stage of review those items judged less significant. The chapter-by-chapter organization of this document, as well as the PSID submitted by DOE, follows generally the organization of the SRP.

The staff's review was aided by independent analysis at Brookhaven National Laboratory (BNL). This independent analysis was directed toward confirming the potential of the key safety features of the design to perform their functions and to look for vulnerabilities in the design through sensitivity studies. A summary of this independent analysis is included in Chapter 15 and in Appendix A. BNL also reviewed selected topics in fuel design, thermal-hydraulics, reactor physics, and safety which have contributed to this SER. The Advisory Committee on Reactor Safeguards reviewed the conceptual design and issued a letter with its views on January 19, 1989. The ACRS letter is included in Appendix D.

In reviewing the SAFR design, the staff developed definitions for various event categories to be used in the evaluation. Specifically, the staff defined four event categories (ECs) which, in general, correspond to traditional LWR event categories as follows:

- EC-I, Anticipated Operational Occurrences
- EC-II, Design-Basis Accidents
- EC-III, Severe Accidents
- EC-IV, Emergency Planning-Basis Events

These categories were developed to avoid confusion over which events need to be considered in the design and how they are to be selected. The consideration of Event Category III in the design is intended to ensure that low-probability events beyond the traditional design-basis envelope are considered in the design so as to provide a sufficient challenge to the plant to allow the use of a mechanistic siting source term. The consideration of such events in the design also meets the intent of the Commission's severe accident policy statement. The staff described these event categories and their use in SECY 88-203 (Ref. 1.6).

1.6 Identification of Safety Issues

During its review of SAFR, the staff identified a number of design and safety issues. The major issues are listed in Table 1.3. They are also identified in the fifth subsection (i.e., X.X.5) of each major section of this report along with other issues the staff noted. The staff also identified significant research and development items necessary to support licensing the SAFR design and reached agreement on them with DOE. These are identified in Table 1.4.

The Phenix Gas Bubble Problem

In the summer of 1989, three emergency shutdowns were activated at Phenix (a pool-type sodium reactor operating in France since 1974) by negative reactivity signals (Ref. 1.12). An analysis of these events leads to the following scenario:

- During a 4-month inspection of the plant, especially during the inspection of the safety vessel at reduced temperatures, sodium oxide impurities accumulated at the sodium pool surface.
- These impurities accumulated at small gas-purging "weep holes" located below the reactor core in the diagrid, holes that normally allow entrained gas in the coolant to leak past the grid and through the core (while still of small volume and importance).
- The resulting blockages caused gas to accumulate in pockets below the reactor core.
- Once the gas bubbles became large enough, they were discharged, mainly through the peripheral zone of the core, thereby producing a negative reactivity effect.

Key factors in the event were (1) the accumulation of sodium oxide impurities at the pool surface and the failure to filter them before restarting the pumps, (2) the importance of the gas-purging weep holes in preventing gas bubbles from building up below the core, and (3) the available pathway for the gas bubbles to escape, which fortunately passed along the outside of the core and not through the center (where this reactivity feedback would have been positive). The French solution to this problem entailed preventing system startup with significant sodium oxide impurities in the system and periodically testing that the gas-purging mechanism is functioning properly.

Importance to LMR Designs in the United States

This is the type of problem that, once recognized, can be largely avoided by design. In particular, the design of the Phenix core inlet plenum creates such pockets, as the inlet nozzle through which the sodium passage is significantly lower than the point at which the assembly and the diagrid intersect. Because of the likelihood of entrained gas accumulating in these pockets, the French drilled the weep holes to allow the gas to bypass constantly. Additionally, if it is anticipated that such gas pockets will accumulate, then the design should ensure that any gas bubbles would exit around the core and not through its center.

To prevent such problems, the designer should (1) prevent the buildup of such contaminants by periodically cleaning up and filtering the coolant, (2) avoid creating such pockets below the core, (3) design the weep holes with greater care, perhaps with some sort of monitoring system, and (4) ensure that any accumulated gas bubbles would be purged along the outside of the reactor and not through the center of the core.

Because this problem has only recently been recognized, it has not yet been explicitly factored into the design of LMRs in the United States. However, it should not be difficult to modify these designs, if necessary, to eliminate this as a safety concern. This issue would be addressed for the SAFR design in any future design review.

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1.7 Conclusions

On the basis of the results of this review, the staff concludes that the SAFR design, as generally conceived, has built upon and utilized experience and technology developed in previous sodium reactor programs and is responsive to the Commission's advanced reactor policy statement. However, the following issues are of fundamental concern to the SAFR design:

- the potential for certain accident scenarios to lead to significant fuel damage, sodium boiling, and subsequent positive reactivity feedback transients with a potential large early release of radioactive material
- the potential for an in-core flow blockage (of as little as one fuel assembly) to escape detection and lead to fuel melting and an energetic reactivity accident
- the potential for sodium leaks in the steam generator building to impinge directly on the secondary containment wall
- the lack of an active safety-grade reactor shutdown system

These concerns interfere with the ability of the proposed SAFR containment concept and offsite emergency planning provisions to meet the criteria in Chapter 3 and are discussed further in Chapters 6, 13, 14, and 15. With satisfactory resolution of these fundamental concerns and the other issues identified in this SER, the staff considers the SAFR design to have many good points and to have the potential to achieve a level of safety at least equivalent to that of current-generation LWRs, with greater safety in certain areas. This conclusion is based on

- the potential for only minor core damage and fission-product release for many severe challenges to the plant
- the reduced dependence on human actions and the reduced vulnerability to human error
- the long response time of the reactor under many accident conditions, providing time for evaluation and corrective action
- the capability to demonstrate by test the significant safety features and performance of the plant over a wide range of events
- the results of independent analysis by BNL, which indicated good agreement with the designers' predicted performance

Of course, final determination on the acceptability of the SAFR design is contingent on

- satisfactory resolution of all of the issues identified in this SER
- completion of a final design and a licensing review by NRC
- successful design, construction, testing, and operation of a prototype reactor as described in the Safety Test Plan (Ref. 1.5).

1.8 References

- 1.1 NRC "Regulation of Advanced Nuclear Power Plants: Statement of Policy," 51 Federal Register 24643, July 8, 1986.
- 1.2 Letter from F. X. Gavigan, DOE, to T. P. Speis, NRC, transmitting Preliminary Safety Information Document for SAFR and PRISM, November 17, 1986.

- 1.3 NRC, P.M. Williams and T.L. King, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," NUREG-1226, June 1988.
- 1.4 DOE "SAFR - Sodium Advanced Fast Reactor," Preliminary Safety Information Document, AI-DOE-13527, Initial Issue - October 1985, with amendments. (DOE Applied Technology.)
- 1.5 DOE, "SAFR - Sodium Advanced Fast Reactor," Safety Test Plan, 149TP000001, October 21, 1987. (DOE Applied Technology.)
- 1.6 NRC, "Key Licensing Issues Associated With DOE Sponsored Advanced Reactor Designs," SECY 88-203, July 15, 1988.
- 1.7 Letter from W. Kerr, Chairman, ACRS, to L. W. Zech, Jr., Chairman, NRC, "Report on Key Licensing Issues Associated With DOE Sponsored Reactor Designs," July 20, 1988.
- 1.8 NRC, "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," 51 Federal Register 32138, August 8, 1985.
- 1.9 NRC, "Safety Goals for the Operation of Nuclear Power Plants," 51 Federal Register 28044, August 4, 1986.
- 1.10 NRC, "Nuclear Power Plant Standardization, Policy Statement," 52 Federal Register 34884, September 15, 1987.
- 1.11 NRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," NUREG-0800, July 1982.
- 1.12 J. Bouchard and C. Le Rigoleur, "The History of Fast Reactor Safety in France," 1990 International Fast Reactor Safety Meeting, Snowbird, Utah, American Nuclear Society Meeting, Volume I, pgs. 85-95.

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Table 1.1 Major plant operational parameters

Item	Parameter
Thermal output (rated)	900 MW
Electrical output (rated)	377 MW gross 350 MW net
Total primary coolant flow rate	36.7×10^6 lb/hr
Total secondary coolant flow rate	36.4×10^6 lb/hr
Total steam production	3.6×10^6 lb/hr
Steam temperature at turbine	850 °F
Steam pressure at turbine	2700 psia
Average fuel burnup	102,000 MWd/T
Reactor core cycle	365 days
Minimum overall breeding ratio	> 1
Plant design life	60 years
Plant availability	80 %

Table 1.2 Major world LMRs (operating or under construction)

Reactor	Thermal power	Pool/Loop	Date critical	Country
EBR-II	62.5 MW	Pool	1963	U.S.
Phenix	563 MW	Pool	1973	France
PFR	559 MW	Pool	1974	U.K.
FFTF	400 MW	Loop	1980	U.S.
SNR 300	736 MW	Loop	-	FRG
Monju	714 MW	Loop	-	Japan
Super Phenix	3000 MW	Pool	1983	France
Joyo	100 MW	Loop	1977	Japan
BN-350	1000 MW	Loop	1973	USSR
BN-600	1470 MW	Pool	1980	USSR

Table 1.3 Major SAFR design and safety issues

Section	Topic	Description
6.1.4 15.3.5	Containment	Acceptability of proposed containment design remains an open issue. Certain bounding events can potentially lead to core-melt, energetic events, and reactor/containment vessel penetration.
4.6.5 7.2.4 7.7.5 15.2.2	Shutdown Systems	APTS should be considered to be safety grade. Testability and safety classification of SASS need to be developed.
	60-Year Plant Life	NRC is currently limited to licensing plants for lifetimes of 40 years. Aging and degradation must be properly understood for 60-year life; for example, irradiation damage and long-term exposure to high temperatures and sodium.
4.3.5 6.1.4	Sodium Void	Positive sodium void coefficient results in the potential for positive reactivity insertion events under ATWS conditions. Design alternatives to reduce the magnitude should be explored in conjunction with eliminating events that could lead to sodium boiling.
4.4.5 6.1.4 9.1.5	Flow Blockage and Control Rod Flotation	Lack of in-core instrumentation to detect assembly flow blockages (such as due to manufacturing errors) results in a potential fuel melting/reactivity insertion initiator. Control rods should be designed so as not to be able to be lifted by inadvertent operation of the primary coolant pumps during refueling.
5.2.5 5.4.5 5.5.5 15.2.3	IHX Integrity	The IHX has intermediate sodium on the shell side and the steam generation system uses single-wall tubes. With a very high steam pressure (2800 psi), there is a potential for SG tube leaks to results in IHX failure. The integrity of the IHX needs further assessment.
5.3.5	Reactor Coolant Pressure Boundary	The reliability of the hard-faced primary pump rotating seals in a sodium environment must be demonstrated.
5.4.5 8.3.2	Pump Coastdown	Steps must be taken to ensure the primary pump coastdown exceeds one minute.

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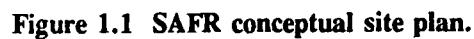
Table 1.3 Major SAFR design and safety issues (con't.)

Section	Topic	Description
5.5.5	Sodium Water Reaction Pressure Relief System (SWRPRS)	Should be safety grade.
7.3.1	Containment Isolation	Non-Class 1E power can compromise the isolation function due to lower reliability.
3.3 7.5.1 9.4.3	Control Room	Control room and remote shutdown area (including their habitability systems) should be safety grade and the post-accident monitoring displays and manual scram function in each area should also be safety grade.
7.7.4	Safety Classification of Sodium Leak Detection System, Failed Fuel Detection System, and Sodium-Water Reaction Detection System	These systems should be considered safety grade.
13.1.5 15.3.5	Emergency Preparedness	Acceptability of proposed ad hoc offsite emergency planning remains an open item.
13.3.5	Security	The SWRPRS and control room must be appropriately located for safety-grade equipment.
14.2.1	Safety Tests	Commitment is necessary regarding the scope of the prototype facility for testing. Any reductions in full-scope plant may require acceptability of certain features to be demonstrated using separate effects facilities.
9.3.5 15.1.1	Common Wall Between Containment at the Steam Generator Building	Sodium leakage from the IHTS piping in the steam generator compartment and ensuing fire may challenge the structural integrity of the secondary reactor containment building.
15.3.5 15.10.5	Safety Analyses	Use of PRA for accident selection is not acceptable. Potential for plant response to certain events to lead to sodium boiling and positive reactivity feedback accidents.
Appendix C	Risk Assessment	The use of the PRA is limited due to uncertainty in system reliability, phenomenology, and unmodeled contributors to risk.

Table 1.4 Major research and development required for the SAFR

Area	Required Development	National Laboratory Support	Status
Metal Fuel	- Fuel performance, HT-9 cladding tests - Reprocessing (ternary fuel)	- EBR-II^a, TREAT^b, FFTF^c - ANL^d-reprocessing	- Proceeding - Proceeding
Passive Safety	Confirm passive response and safety margins	- EBR-II tests (ATWS) - ANL-reactor flow tests - ANL-RACS heat-removal test - Prototype reactor test	1986 results - Proceeding - Proceeding - Proceeding
In-Vessel Refueling Machine	- Design definition (EBR-II, FFTF, PEC^e, and PFR^f technology) - Under-sodium performance	- ANL-design support - Prototype test-TBD^g	- Proceeding - Planned
Control Drive	Scram performance and reliability	Prototype test-TBD	Planned
Failed Fuel Detection	Sodium and cover gas monitoring with hermetically sealed system	- ANL-under-head delayed neutron monitor - HEDL^h-passive fission gas monitor	- Proceeding - Pending
Reactor Physics	- Validate analytical tools - Confirm rod worths, power distribution, etc.	ANL-ZPPRⁱ	Pending

^aEBR-II - Experimental Breeder Reactor-II^bTREAT- Transient Reactor Test Facility (Idaho National Engineering Laboratory)^cFFTF - Fast Flux Test Facility^dANL - Argonne National Laboratory^ePEC - Prova Elementi Combustibili^fPFR - Prototype Fast Reactor^gTBD - To be determined^hHEDL - Hanford Engineering Development LaboratoryⁱZPPR - Zero Power Plutonium Reactor



Important NSSS Design Features

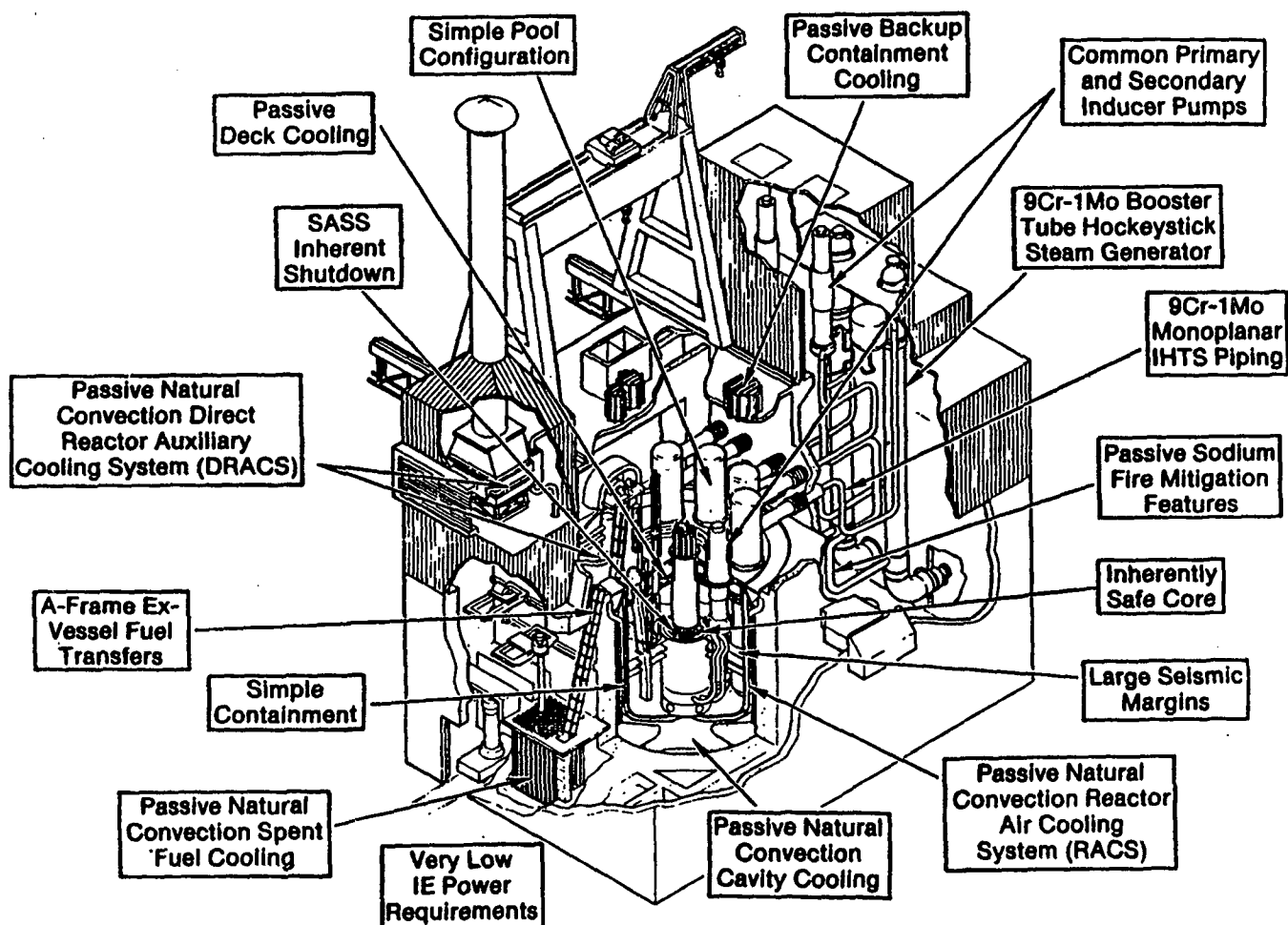


Figure 1.2 Important NSSS design features.

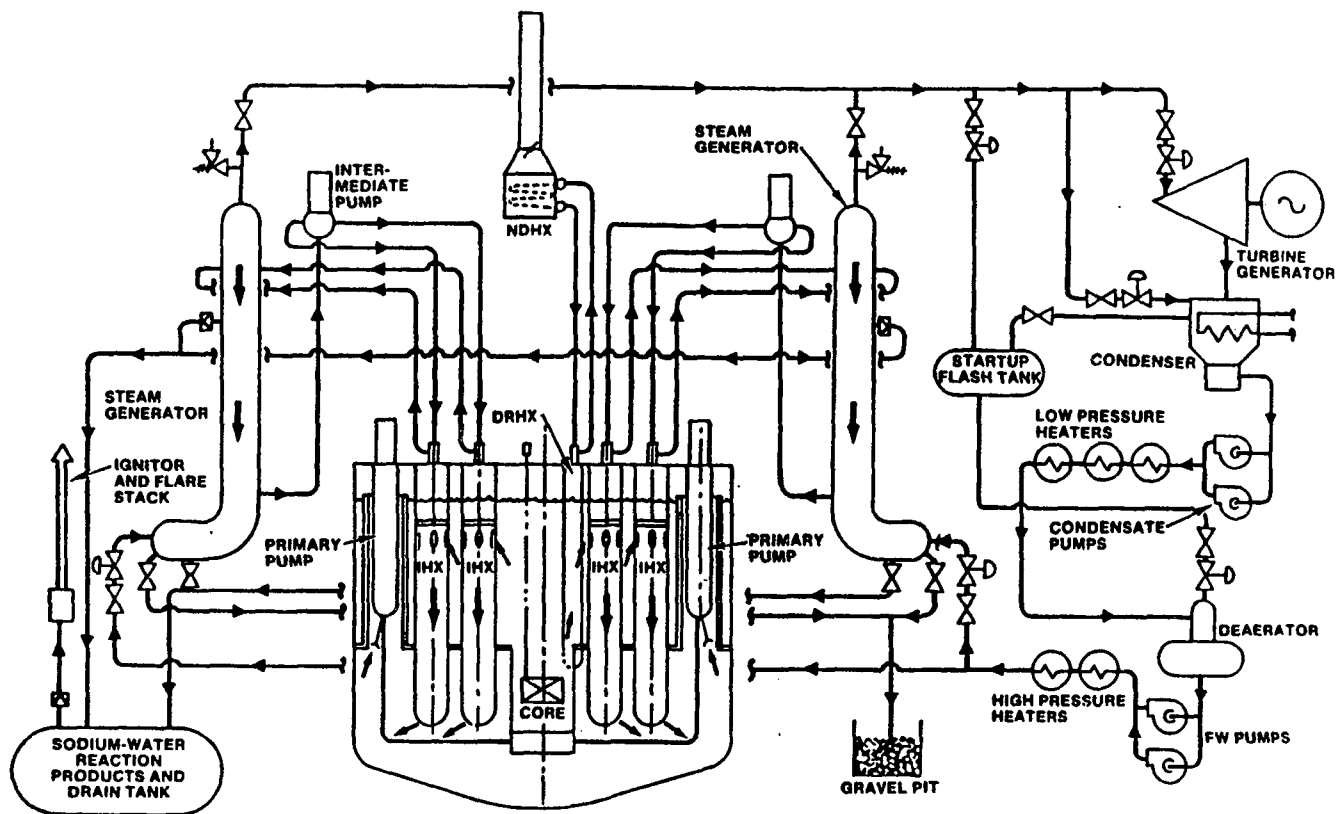


Figure 1.3 Simplified power pak flow diagram.

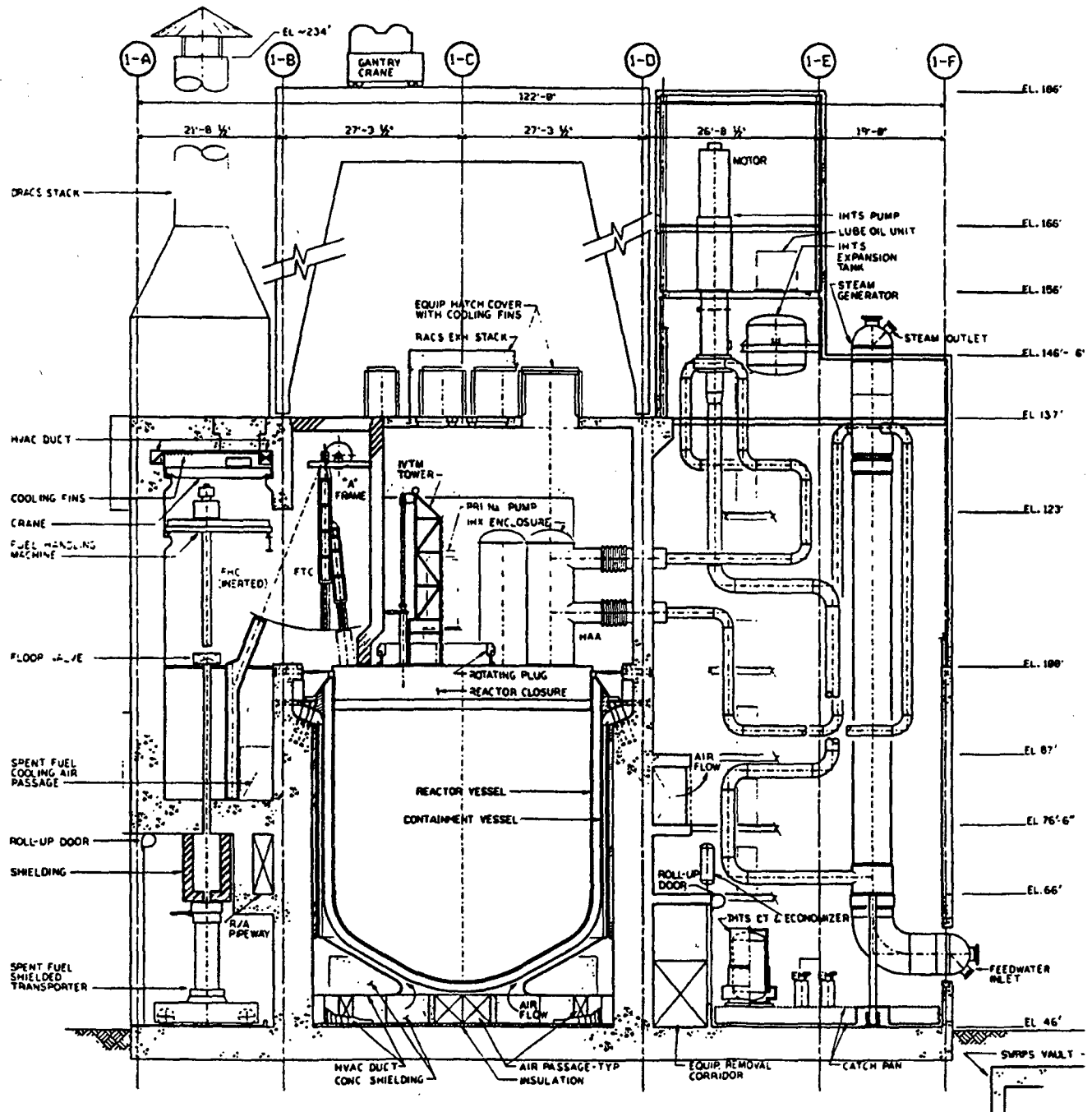


Figure 1.4 Nuclear island elevation view.

Introduction and Summary

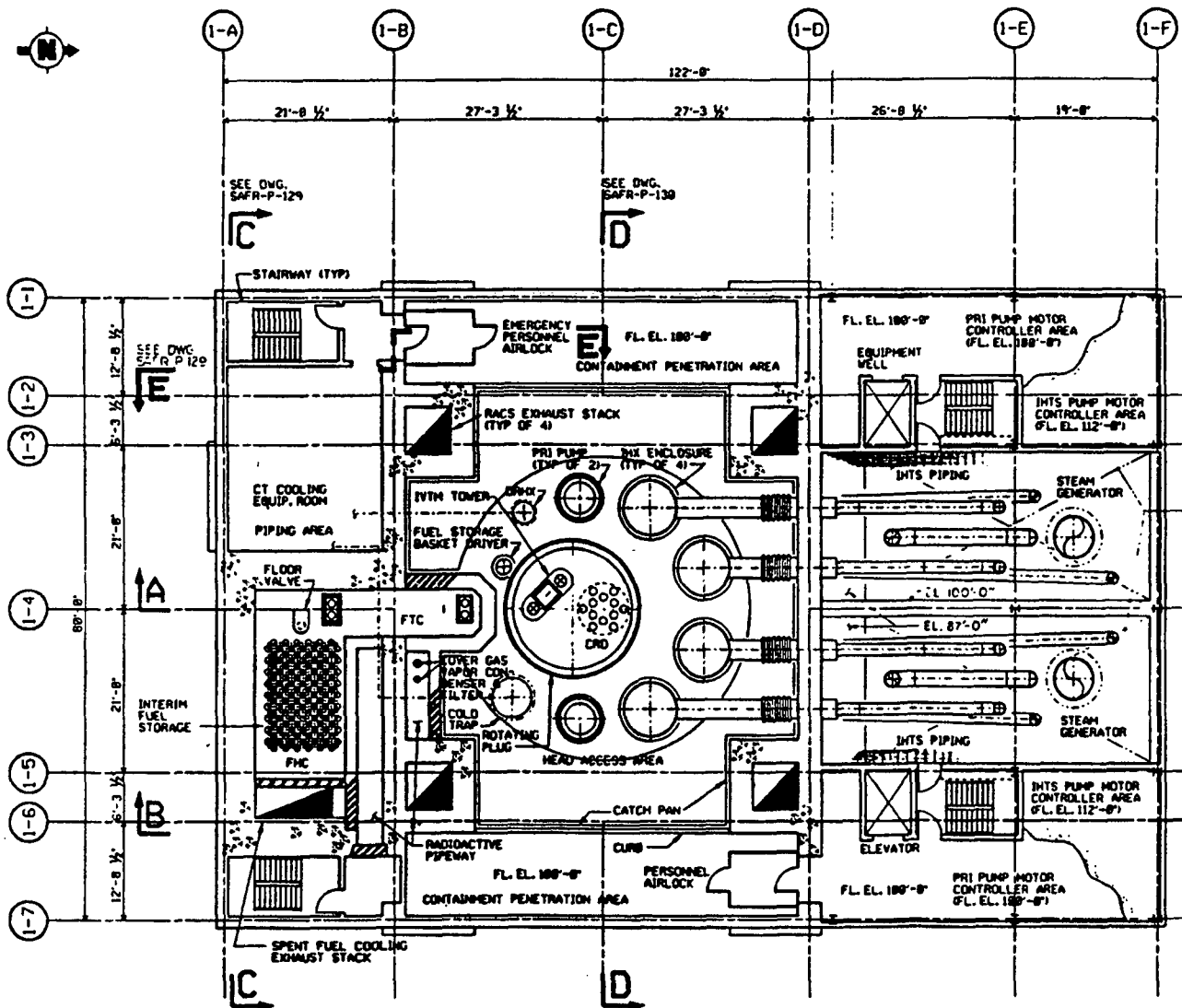


Figure 1.5 Nuclear island plan view (120 feet below grade).

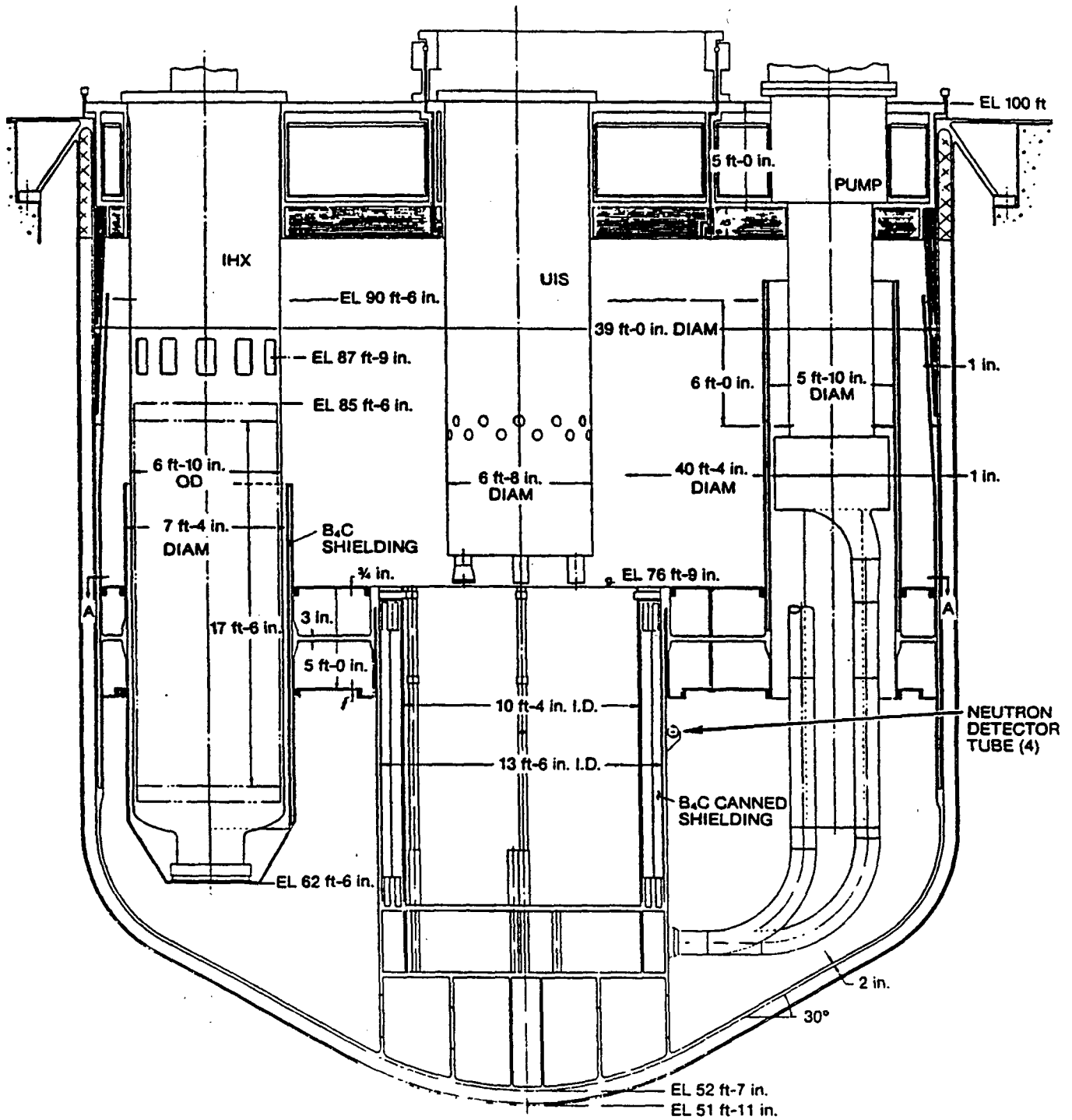


Figure 1.6 Reactor assembly elevation view.

Introduction and Summary

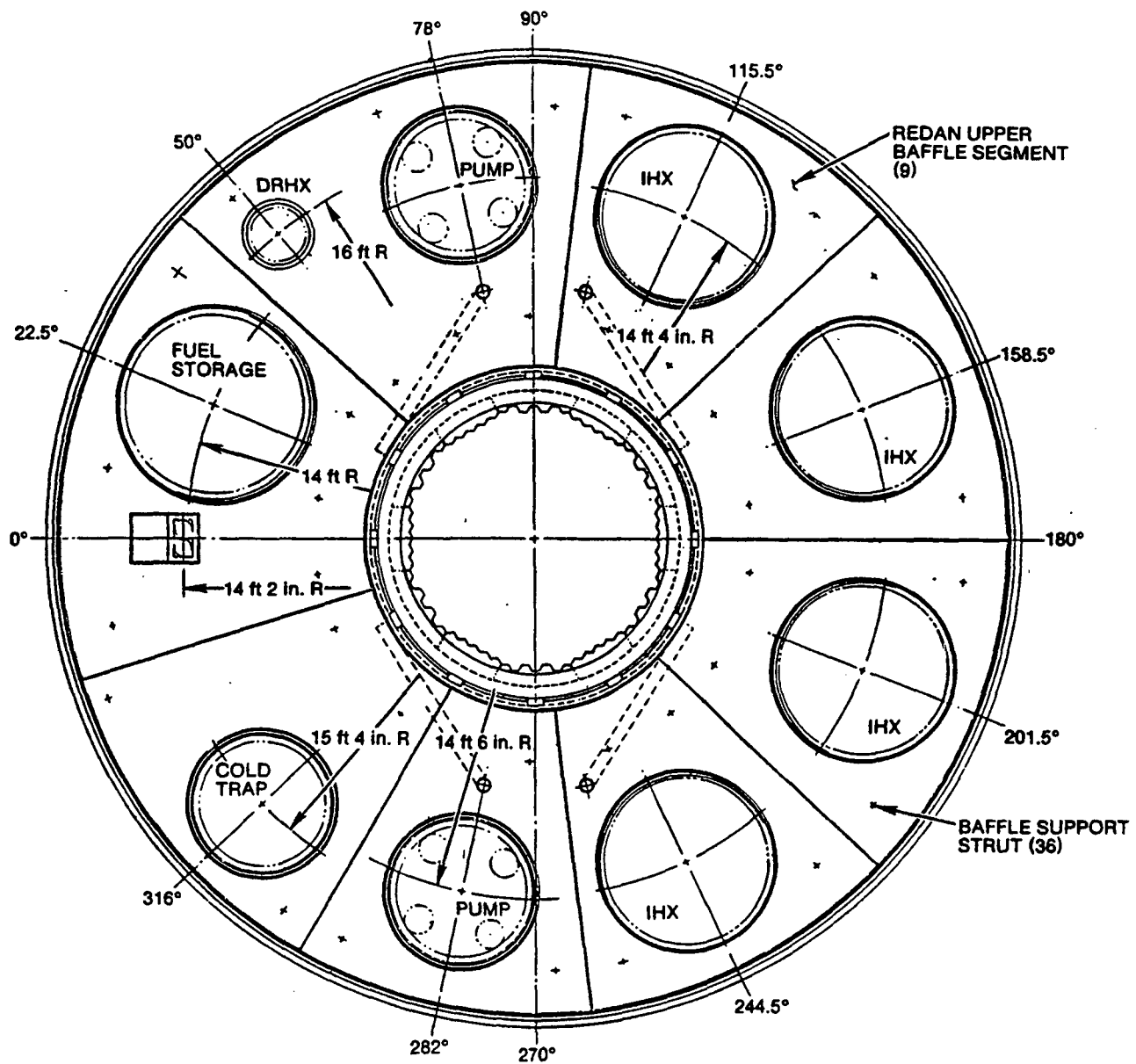


Figure 1.7 Reactor assembly plan view at elevation 78 feet.

2 SITE LOCATION AND DESCRIPTION

No specific site has been selected for the SAFR plant. Chapter 2 of the PSID (Ref. 2.1) describes the standard site characteristics assumed by DOE for evaluating site-suitability factors for the SAFR design concept. According to the PSID, the SAFR plant is directly applicable to more than 75 percent of the sites in the United States believed capable of receiving such large shipments as a shop-fabricated reactor vessel. The site would not be located near commercial or military airports or near an aircraft corridor. Consideration will be given to Regulatory Guide 1.91 (Ref. 2.2) with regard to the proximity of routes for explosives carriers.

The plant has been designed for a safe-shutdown earthquake (SSE) of 0.3 g and an operating-basis earthquake (OBE) of 0.1 g. An OBE less than half the SSE deviates from the usual interpretation of the requirements for the OBE. The staff comments on this subject in Section 3.4 and its seismic design evaluation appears there too.

Meteorology diffusion limits were developed consistent with Regulatory Guide 4.7 (Ref. 2.3). Characteristics for precipitation, snow and ice loads, temperature and humidity, cooling tower design, safety-related ultimate heat sink, and the windload and height for safety-related structures were included for site evaluation purposes. Consideration of the design-basis tornado is addressed in Section 3.5. The maximum groundwater level is assumed to be 10 feet below plant grade. DOE has committed to provide flood protection to plant grade level. Makeup water was assumed to range in seasonal temperature from 33°F to 100°F.

Integrated population exposures resulting from normal operation and accident conditions must reflect suitable population distributions around the plant. The SAFR design was proposed as being licensable on at least 75 percent of existing LWR sites in the United States that could be reached by barges. The maximum enveloping population distributions were developed by DOE and RI from data in NUREG-0348 (Ref. 2.4). That report gives population distributions around all 111 LWR sites projected to the year 2000.

Table 2.1 shows the total population and population density surrounding the 75th-percentile LWR site in the year 2000. These data were used for dose calculations.

The site boundary radius is 0.4 mile and represents an exclusion radius; the low-population zone radius is 3 miles.

2.1 Conclusion

The proposed site envelope appears to be acceptable pending resolution of the questions regarding the selection of the OBE and tornados. These items can be resolved at a later design stage.

2.2 References

- 2.1 DOE, "SAFR - Sodium Advanced Fast Reactor," Preliminary Safety Information Document, AI-DOE-13527, Initial Issue - October 1985, with amendments. (DOE Applied Technology.)
- 2.2 NRC, "Evaluation of Explosives Postulated to Occur on Transportation Routes Near Nuclear Power Plants," Regulatory Guide 1.91, Rev. 1, February 1978.
- 2.3 NRC, "General Site Suitability Criteria for Nuclear Power Stations," Regulatory Guide 4.7, Rev. 1, November 1975.
- 2.4 NRC, "Demographic Statistics Pertaining to Nuclear Power Reactor Sites," NUREG-0348, November 1979.

Table 2.1 Maximum enveloping population densities surrounding the SAFR plant

Radius from plant (mi)	Total population ^a (persons)	Population density (persons/mi ²)
1	330	105
10	73,000	232
50	3,300,000	420

^a Projected population surrounding the 75th-percentile LWR site in the year 2000 (NUREG-0348, Ref 2.1).

3 REVIEW APPROACH AND CRITERIA

3.1 Introduction

The criteria applied by DOE and RI to the SAFR design concept include, with modifications, the full range of criteria and guidance provided by the NRC for current light-water-reactor (LWR) applications for commercial power production. These include the NRC's Standard Review Plan (SRP, Ref. 3.1), regulatory guides (RGs), and the general design criteria (GDC) given in Appendix A to 10 CFR Part 50, as modified for consistency with American National Standards Institute/American Nuclear Society criteria (ANSI/ANS-54.1-1989), "General Safety Design Criteria for a Liquid Metal Reactor Nuclear Power Plant."

The SAFR PSID (Ref. 3.2) was submitted for early use before a formal preliminary safety analysis report (PSAR) for a construction permit was prepared. The level of detail of the information in the PSID is much less than that required for a PSAR. As a result, the evaluation of the SAFR design against the SRP was presented chapter by chapter, and only significant deviations from requirements in the SRP were identified. DOE has committed to produce a more detailed assessment of how SAFR meets the SRP requirements as part of a formal application (Ref. 3.2).

DOE and RI reviewed all of the regulatory guides for their applicability to the SAFR design concept. A list of those directly applicable for the SAFR conceptual design and those not directly applicable but that do apply in principle and intent is given in PSID Table 1.7-1. Those regulatory guides which DOE and RI do not believe to be applicable are omitted from that list. Those believed to be applicable were applied to the SAFR design.

For use in the SAFR design, DOE and RI have adopted the latest general design criteria for liquid-metal reactors (LMRs) as given in ANSI/ANS-54.1-1989. They have designated these the principal design criteria (PDC) for the SAFR plant to avoid possible confusion about whether they should be considered applicable, in general, to LMRs. The SAFR PDC are identical to the ANS-54.1 GDC except if the SAFR design has no system or function for which the ANS-54.1 GDC have a requirement. In that case, the requirement was deleted and an explanation justified the deletion. Table 3.1 lists the SAFR PDC and the corresponding NRC GDC. DOE and RI find the SAFR PDC supply a level of safety at least equivalent to that provided through application of the NRC GDC. The staff's evaluation of the criteria used by DOE and RI for the SAFR design is given in Section 3.3. The staff's evaluation of the approach to safety classification used by DOE and RI is given in Section 3.4. The SAFR design uses a combination of safety-related and non-safety-related systems and equipment to satisfy the PDC. Because of the unique nature of the design with regard to safety functions, the classification of these features requires considerable attention.

The staff has also reviewed the seismic evaluation methodology used for the SAFR design. Staff comments appear in Section 3.5.

The remainder of the areas normally covered in Chapter 3 of an application are discussed in Section 3.6. These include tornado, flood, missile design bases, and the dynamic effects of pipe rupture. The staff has also commented on the design of mechanical systems.

3.2 NRC Advanced Reactor Review Criteria

The design characteristics of SAFR are unique; thus, the proposed approach and the criteria to be applied in the review differ from those for conventional light-water reactors (LWRs) in several key areas. The major differences in RI's approach can be summarized as (1) desiring to use a more mechanistic approach in the selection of accidents to be considered in the design and in the calculation of source terms, (2) not including a conventional containment building in the design, (3) elimination of the need for planned offsite emergency evacuation and drills, and (4) achieving more flexibility in the approach to standardization. Each of these major differences results from the characteristics

Review Approach and Criteria

of the design that, because of the use of passive reactor shutdown and decay heat removal systems, are claimed by DOE and RI to prevent fuel damage for a wide range of accident conditions, including such very unlikely events as anticipated transients without scram, station blackout, and multiple operator errors. The staff has looked at the fundamental technical issues associated with each of these areas and has developed an approach and criteria to address each. The approach utilizes the guidance in the Commission's advanced reactor (Ref. 3.3), safety goal (Ref. 3.4), severe accident (Ref. 3.5), and standardization (Ref. 3.6) policy statements as the basis for deriving a set of decision criteria against which to evaluate the SAFR design in the four areas.

The proposed criteria are grouped into three areas. The "general criteria" that describe the overall approach and principles used by the staff in conducting the review of the SAFR concept (these general criteria could also be applied in the review of any advanced reactor design significantly different from current-generation LWRs). The "specific licensing criteria" that implement the general criteria in the following key areas:

- accident selection
- siting source term
- containment
- offsite emergency planning

Finally, the "standardization criteria" that are described in SECY 88-202 (Ref. 3.7). SECY 88-202 presents a set of criteria that the staff developed for use in the review of the DOE's plans for standardization of the sodium advanced fast reactor (SAFR), the power reactor innovative small module (PRISM) liquid-metal reactor, and the modular high-temperature gas-cooled reactor (MHTGR). These criteria address the following standardization issues:

- scope and level of detail of design to be standardized
- plant options (number of reactor modules) to be standardized
- prototype testing

The criteria proposed in SECY 88-202 were subsequently incorporated into 10 CFR Part 52. A safety test plan (Ref. 3.8) has been proposed for the SAFR design concept that includes both out-of-pile and in-pile tests at existing facilities to establish the licensing basis for the initial SAFR module. The plan also includes special tests on the prototype SAFR module to provide validation of analytical predictions for a wide spectrum of transients. The primary purpose of the tests is to provide the basis for obtaining a certification of the standard SAFR plant design. The staff reviews the safety test plan in Section 14.2 of this report.

3.3 Evaluation of the Principal Design Criteria for the SAFR

The staff has reviewed the criteria proposed by DOE and RI to be used for the design of the structures, systems, and components for the SAFR concept. The general approach is consistent with that currently used for commercial LWRs, as described below. It is clear, however, that a much more detailed treatment of the requirements specified in the staff's SRP will be required in a formal submittal. In addition, the next review stage will require an in-depth review of how the SAFR design conforms with the guidelines provided in the regulatory guides.

The approach taken by DOE and RI on the general design criteria (GDC) deviates from that currently used for commercial LWRs. Rather than use the NRC's LWR GDC defined in Appendix A to 10 CFR Part 50, DOE and RI have chosen to base their principal design criteria (PDC) on the set of GDC for LMRs developed by the ANS-54.1 working committee. The ANS-54.1 working committee used the NRC's GDC from Appendix A as a starting point, but modified those criteria for application to LMRs. DOE and RI have provided these criteria in the PSID, together with a discussion of how each one differs from, or is equivalent to, the corresponding NRC GDC and how the SAFR design complies with that PDC. A list correlating the SAFR PDC with the NRC's LWR GDC from

Appendix A (10 CFR Part 50) is given in Table 3.1 at the end of this chapter. DOE and RI have added six criteria that they find unique to LMRs and necessary for proper design management. These are:

- PDC 1.4, "Protection Against Sodium Reactions"
- PDC 1.7, "Sodium Heating Systems"
- PDC 4.1, "Assurance of Adequate Reactor Coolant Inventory"
- PDC 4.4, "Reactor and Intermediate Coolant and Cover Gas Purity Control"
- PDC 4.5, "Intermediate Coolant System"
- PDC 4.6, "Inspection and Surveillance of Intermediate Coolant Boundary"

DOE and RI have also deleted five criteria from the NRC's list of GDC on the basis that they are unique to LWRs. Those GDC that were deleted are:

- GDC 33, "Reactor Coolant Makeup"
- GDC 35, "Emergency Core Cooling"
- GDC 38, "Containment Heat Removal"
- GDC 39, "Inspection of Containment Heat Removal"
- GDC 40, "Testing of Containment Heat Removal System"

Briefly, the bases for omitting these criteria are summarized below. GDC 33 requires a reactor coolant makeup system to protect against small breaks in LWR coolant systems. SAFR utilizes a different approach (covered by PDC 4.1, "Assurance of Adequate Reactor Coolant Inventory") based on features to protect against both large and small breaks in the primary coolant system. GDC 35 requires an emergency core cooling system (ECCS) to protect against large breaks in the coolant system for LWRs. DOE and RI exclude this criterion on the same basis used for excluding GDC 33, together with the argument that PDC 4.7, 4.8, and 4.9 provide assurance of appropriate reactor residual heat removal system design, inspection and surveillance, and testing required to satisfy the intent of GDC 35 on ECCS.

The last three GDC deleted (GDC 38, 39, and 40) are related to containment heat removal system design, inspection, and testing. These criteria require a system to rapidly reduce temperature and pressure in the primary containment following a loss-of-coolant accident (LOCA) in an LWR. These criteria were deleted on the basis that there is no comparable functional requirement for the SAFR design. DOE and RI stated that the passive heat transfer systems used on SAFR for long-term cooling of the reactor closure (part of the primary containment) and the reactor building (secondary containment) are adequately covered by PDC 4.10, which addresses structural cooling systems. The inspection and testing requirements of GDC 39 and GDC 40 are said to be met by PDC 4.11 and 4.12 for structural cooling systems. The staff is currently considering the need for a containment system for the SAFR concept. Such a system could potentially require a cooling system as is required by GDC 38, 39, and 40. Therefore, the staff will defer its decision on the applicability of these GDC until that issue has been resolved.

The ANS-54.1 GDC consistently replace the term "important to safety" (where it appears in the NRC's GDC) with the phrase "safety related." The NRC's position on the use of the two terms is given in Generic Letter 84-01 (Ref. 3.9). The staff stated in that letter that the two terms are not equivalent and the NRC's regulatory jurisdiction involving a safety matter is not controlled by the use of those terms. This is an open item.

The staff believes that the approach taken by DOE and RI on the principal design criteria (PDC) for SAFR is workable. This approach is similar to that used in the Clinch River breeder reactor and is acceptable for this stage of review.

The staff has reviewed the PDC for the SAFR design against the NRC GDC given in Appendix A to 10 CFR Part 50. The results follow, keyed to the NRC GDC; the corresponding SAFR PDC (or

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position) is noted in parenthesis. How the SAFR design will meet (or need not meet) the GDC requirement is summarized briefly. In some cases there are substantial changes from the GDC. Although the staff's review should not be construed as an NRC approval of ANS-54.1 (a much broader NRC review would be required for that purpose), the staff does consider the use of these industry-developed standards for such innovative designs as the SAFR design a good starting point to guide the design and facilitate the staff's review.

GDC 1: Quality Standards and Records (SAFR PDC 1.1)

The structures, systems, and components of the SAFR design are classified according to their location and service. Recognized industry codes and standards are applied according to the safety functions required. A quality assurance program has been developed and referenced. Appropriate documentation and records are expected to be maintained throughout the operating license period.

The SAFR design appears to meet the intent of GDC 1 with minor word changes.

GDC 2: Protection Against Natural Phenomena (SAFR PDC 1.2)

A standard site has been assumed in defining the natural phenomena for structures, systems, and components important to safety. Design-basis events (DBEs) will be selected that envelope the worst of the natural phenomena consistent with plant risk goals.

The SAFR design appears to meet the intent of GDC 2 with minor word changes.

GDC 3: Fire Protection (SAFR PDC 1.3)

The design of the fire protection systems will comply with

- Energy Research and Development Administration (ERDA) Industrial Fire Protection Manual, Chapter 0552
- National Fire Protection Association (NFPA) codes
- 10 CFR Part 50, Appendix R
- NUREG-0800, SRP Section 9.5.1 (Ref. 3.1)

Special consideration will be given to fires resulting from sodium spills, sodium-water reactions, and limitation and isolation of their effects as described in SAFR PDC 1.4.

The SAFR design appears to meet the intent of GDC 3 with minor word changes.

GDC 4: Environmental and Missile Design Bases (SAFR PDC 1.5)

Structures, systems, and components, whether in a controlled or uncontrolled environment are designed to perform their intended functions for the environmental conditions possible and the spectrum of credible missiles. Spontaneous rupture of low-pressure sodium piping is not considered credible and has been excluded from consideration.

The SAFR design appears to meet the intent of GDC 4, as modified for the SAFR.

GDC 5: Sharing of Structures, Systems, and Components (SAFR PDC 1.6)

The SAFR design does not share any nuclear safety-related structures, systems, or components among the reactor modules. A single non-safety-grade control room is proposed for all reactor modules with redundant non-safety-grade shutdown capabilities at each individual module.

The concept of a non-safety-grade control room (discussed in Chapter 13) will need further definition by the applicant and review by the staff prior to staff acceptance.

GDC 10: Reactor Design (SAFR PDC 2.1)

The design, maintenance, and operation of the reactor will ensure that the core design limitations will not be violated during normal operation or anticipated operational occurrences. The fuel research and development program is designed to provide the fuel operating and limiting conditions necessary to define the normal and off-normal operating limits.

The SAFR PDC appear to meet the intent of GDC 10. The SAFR PDC include criteria to cover fuel management, flow blockage, and fuel failure propagation that are critical in an LMR to the prevention of severe core-damage accidents.

GDC 11: Reactor Inherent Protection (SAFR PDC 2.2)

The composition and design of the fuel shall be such that the overall reactivity coefficient is always negative during power operation and for off-normal transients. The overall reactivity coefficient will be dominated by a strong Doppler coefficient and various thermal expansion coefficients. The reactivity effects will be verified during the prototype facility safety test program.

The SAFR design appears to meet the intent of GDC 11. RI made additional proposals regarding feedback requirements to accommodate ATWS. The acceptability of this feature would need to be addressed at a later date.

GDC 12: Suppression of Reactor Power Oscillations (SAFR PDC 2.3)

Tight neutronic coupling of the SAFR core design will prevent spatial instability. The Doppler and other feedback coefficients should ensure a stable response to reactivity perturbations at full power, provided sodium does not boil.

The SAFR design appears to meet the intent of GDC 12.

GDC 13: Instrumentation and Control (SAFR PDC 2.4)

Instrumentation is planned to monitor and control such parameters as neutron flux, control rod position, chemical composition, pressure and temperature, flow rates, levels, and leakage, as necessary, to ensure plant safety. Instrumentation will be provided in the reactor system, intermediate heat transport system, steam generation system, the power conversion system, shutdown heat removal system, and radwaste and auxiliary systems. Instrumentation readouts are planned to give the operator information necessary to support operation of the plant, respond to upset conditions affecting operation, prevent radiological release, detect fire hazards, and prevent potential damage to the plant.

The SAFR design appears to meet the intent of GDC 13 with minor word changes.

GDC 14: Reactor Coolant Pressure Boundary (SAFR PDC 2.5)

The reactor coolant pressure boundary is designed and will be manufactured to recognized industry codes and standards for pressure vessels. Appropriate codes and standards will be applied in accordance with the limiting conditions for the plant.

The SAFR designs appears to meet the intent of GDC 14 with minor word changes.

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GDC 15: Reactor Coolant System Design (SAFR PDC 2.6)

All components and systems associated with fluid transport and containment are designed according to their requirements for safe operation and maintenance of safe shutdown conditions. These systems are designed according to the appropriate ASME codes and standards.

The SAFR design appears to meet the intent of GDC 15 as modified for LMRs.

GDC 16: Containment Design (SAFR PDC 2.7)

The SAFR primary containment system will consist of the containment vessel surrounding the reactor vessel, the closure head, and certain elements that the intermediate heat exchange systems share with primary pumps. The containment system is designed to limit leakage and release of radioactive material during any postulated accident within the guidelines of 10 CFR Part 100. The containment vessel and associated components are designed according to the appropriate codes and standards to meet the anticipated function and limiting conditions. The secondary containment/confinement will be provided by the reactor containment building (RCB). The RCB will be hardened against external events and will have a restricted leak rate under nominal conditions.

This PDC represents a significant modification to the intent of GDC 16. This criterion permits the combination of a low-volume, low-pressure, primary containment and low-pressure, secondary confinement structure to meet the requirements of GDC 16. The functional requirements for containment systems in advanced reactors, including the acceptability of the non-conventional SAFR containment designs, are currently under consideration. Therefore, the acceptability of this PDC remains an open item.

GDC 17: Electric Power Systems (SAFR PDC 2.8)

The onsite power systems will consist of both ac power subsystems and dc power subsystems. Each main turbine generator will supply power to the respective reactor modules. Each reactor module will be supplied with high-voltage ac power from the plant normal or secondary offsite sources. Non-safety-grade gas turbines will provide backup power to protect plant investment. The reactor modules will also be equipped with battery dc power sources. The Class 1E systems, equipment, cable design, conduits, raceways, and so forth, are designed according to the criteria defined in the appropriate IEEE codes and standards. The appropriate quality assurance program will be applied for Class 1E equipment.

The SAFR design appears to meet the intent of GDC 17. As proposed, the SAFR PDC modifies GDC 17 to account for the long-response time of the plant and its effect on safety-related power demands and to address the use of the passive shutdown and decay heat removal capabilities of this plant in responding to station blackout.

GDC 18: Inspection and Testing of Electric Power Systems (SAFR PDC 2.9)

The ac and dc power systems are designed and will be tested during operation in accordance with the appropriate Institute of Electrical and Electronics Engineers (IEEE) standard and Regulatory Guide 1.118 (Ref. 3.10). Tests will include pre-operational tests, initial system tests, and periodic equipment and systems tests. Tests will be performed to verify performance and to determine degradation of systems and equipment.

The SAFR design appears to meet the intent of GDC 18 with minor word changes.

GDC 19: Control Room (SAFR PDC 2.10)

The control room and the central control center are not designed as safety-grade structures or equipment. The passive safety characteristics will be relied upon to ensure that the reactor facility will shut down independent of operator action, should an abnormal event occur. Remote independent shutdown capabilities at the site will be available for manual nuclear shutdown monitoring and communication tasks that may follow an abnormal event. These functions will be capable of being performed from two separate locations for each module.

The staff will require further evaluation to reach a decision on this criterion. The staff's reservations are based on the importance of the operator to safety (as discussed in Chapter 13) and the staff's historical position that the control room and the remote shutdown areas should be designated safety grade and should provide protection to the operator.

GDC 20: Protection System Functions (SAFR PDC 3.1)

The SAFR plant will use three nuclear shutdown systems. Only one, the passive reactivity feedback effects, is classified as a safety-grade system. None of these systems is required to initiate the passive, post-shutdown heat removal. The plant protection system (PPS) will initiate closure of the safety-related containment isolation valves. Plant investment will be protected by the non-safety-grade automatic plant trip system (APTS). The PPS and APTS will combine to prevent exceeding fuel design limits. The self-actuated-shutdown-system (SASS) and the passive feedback effects will combine to prevent exceeding fuel damage limits for ATWS events. The APTS will send a signal to the PCS to initiate primary and intermediate pump coastdown and shutdown of the steam generator, feedwater and turbine systems. It also will actuate dampers and valves required for operation of the non-safety-grade DRACS. The safety-grade RACS will always be in operation.

The language of PDC 3.1 appears to be the same as GDC 20, with minor word changes. As DOE and RI note, SAFR PDC 3.1 will meet the requirements and intent of GDC 20 by combining nuclear-safety-related and non-nuclear-safety-related systems. However, the interpretation is not straightforward. Although it may be possible to establish the adequacy of the application of multiple, mixed safety class systems to meet the requirements of GDC 20, the staff is not convinced that this is the proper approach to take. In addition, at least one safety-grade active shutdown system should be required for SAFR, as discussed in Section 4.6.5.

GDC 21: Protection System Reliability and Testability (SAFR PDC 3.2)

Both the safety-related and non-safety-related portions of the protection system are designed to permit on-line testing and calibration without degradation of the shutdown function or violation of the single-failure criterion. The system will initiate shutdown should the system failure state not satisfy minimum safety requirements.

PDC 3.2 appears to meet the intent of GDC 21 with minor word changes. The application of this criterion is subject to the same concerns expressed for GDC 20 (PDC 3.1).

GDC 22: Protection System Independence (SAFR PDC 3.3)

The protection system is designed with redundant logic trains, reactor trip devices, and engineered safety feature actuation devices that are physically separated and electrically isolated. Administrative controls, redundancy, automatic self-testing and calibration, and status indication will minimize the possibility of undetected loss of the protection function.

The SAFR design appears to meet the intent of GDC 22 with minor word changes.

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GDC 23: Protection System Failure Modes (SAFR PDC 3.4)

The protection system is designed so that channel and logic circuit failures are in a direction to result in reactor trip. There will be sufficient redundancy to prevent a single failure from blocking a reactor trip.

The SAFR design appears to meet the intent of GDC 23 with minor word changes.

GDC 24: Separation of Protection and Control Systems (SAFR PDC 3.5)

RI stated that the functions performed by the protection and control systems (e.g., containment isolation) will be completely separated from the passive shutdown function. Since RI proposed APTS and SASS as non-safety-grade systems, these systems are interpreted as not having to meet GDC 24.

The staff would require additional information and further review to determine whether the APTS and SASS have to meet GDC 24. Therefore, applicability of GDC 24 remains an open item.

GDC 25: Protection System Requirements for Reactivity Control Malfunctions (SAFR 3.6)

The protection system will prevent the reactor fuel and coolant boundary conditions from reaching design limits for any single malfunction of the reactivity control systems.

The SAFR design appears to meet the intent of GDC 25. RI added requirements for limiting coolant boundary conditions due to the potential for thermal shock in LMRs because of the large difference between hot-leg and cold-leg temperatures in LMRs.

GDC 26: Reactivity Control System Redundancy and Capability (SAFR PDC 3.7)

GDC 26 requires two independent reactivity control systems of different design, one of which is suggested to be the control rod. The systems must perform their function with a single failure. One shall limit the size of reactivity increase and either it or the other shall limit the rate of increase to prevent exceeding fuel design limits. The systems shall be able to hold the reactor subcritical at the lowest core operating temperature.

DOE and RI note the SAFR design exceeds the requirements of this criterion. The design uses three means of reactivity control (primary control rods tripped by the APTS or manually, secondary SASS shutdown rods with APTS, manual or Curie point magnet trip and the passive reactivity feedback effects). Only the passive reactivity feedbacks are classified as safety grade.

The acceptability of this approach is discussed under GDC 20 (PDC 3.1).

GDC 27: Combined Reactivity Control Systems Capability (SAFR PDC 3.8)

PDC 3.8 requires each of the reactivity control systems to be designed to independently prevent the reactor from exceeding fuel damage limits from postulated accidents assuming a single, active component failure. For anticipated transient without scram (ATWS) events, the criterion requires prevention of loss of coolability by such highly reliable features as passive actuation (SASS) or passive feedback, or design features to mitigate or accommodate such loss.

PDC 3.8 represents a substantial revision of GDC 27. The SAFR design approach to meeting the requirement of GDC 27 is quite different. The SAFR design relies on the passive feedback as the only safety-grade shutdown system. The primary control rods can be tripped by the APTS signals or manually. The secondary shutdown rods can be tripped by the APTS signals or manually and the same secondary rods can also be tripped by the Curie point magnetic latch system.

The design features provide substantial confidence that the SAFR plant will meet the intent of GDC 27, but the staff has the same concern here as expressed for GDC 20 regarding safety classifications of active shutdown systems.

GDC 28: Reactivity Limits (SAFR PDC 3.9)

The reactivity control system is designed to prevent exceeding acceptable fuel and coolant design limits and to ensure core coolability in the event of the withdrawal of a single control rod at its maximum rate of travel. The reactor vessel internal structures and core restraint system are designed to accommodate the reactivity transient from a seismically induced event. The reactivity available in the control rods will be limited by reducing the net burnup (hence control requirement) to near zero.

The SAFR design appears to meet the intent of GDC 28 with minor word changes to accommodate differences between LWR and LMR designs.

GDC 29: Protection Against Anticipated Operational Occurrences (SAFR PDC 3.10)

The protection and control systems are designed for a high probability of performing their nuclear safety functions in the event of anticipated operational occurrences. Sufficient redundancy diversity and testability should provide an acceptable level of assurance that the systems will meet their intended functions.

The SAFR design appears to meet the intent of GDC 29 with minor word changes.

GDC 30: Quality of Reactor Coolant Pressure Boundary (SAFR PDC 4.2)

PDC 4.2 implements the requirements of PDC 4.1, which requires assurance of adequate reactor coolant inventory. That requirement will be met by the reactor vessel and containment vessel design and configuration. Since the reactor vessel is the primary component needed to satisfy PDC 4.1, it is the only component addressed in PDC 4.2 (the SAFR equivalent of GDC 30). The reactor vessel will be shop fabricated in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, Class 1, augmented by applicable high-temperature code cases and additional requirements based on experience. The vessel will be fabricated and tested to acceptable standards. The containment vessel is classified as ASME Safety Class 2. However, the containment vessel will be used to satisfy part of the requirements of PDC 4.1 to ensure sodium inventory in the event of a leak.

The design includes the means of coolant leakage detection and appears to meet the intent of GDC 30.

GDC 31: Fracture Prevention of Reactor Coolant Pressure Boundary (SAFR PDC 4.13)

The reactor vessel, closure head, intermediate heat-transfer system (IHTS), and direct reactor auxiliary cooling system (DRACS) coolant boundaries will be fabricated of materials capable of meeting the deformation and fatigue failure modes in accordance with the specifications of ASME B&PV Code service levels as defined in Appendix T to the ASME Code Case N-47, Safety Class 1. The intent is to require any coolant boundary using the leak-before-break principle to avoid brittle and rapidly propagating fracture modes.

The SAFR design appears to meet the intent of GDC 31 with word changes to reflect unique LMR concerns regarding high temperature and a sodium environment.

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GDC 32: Inspection of Reactor Coolant Pressure Boundary (SAFR PDC 4.3)

External reactor vessel walls will be inspected periodically in accordance with the ASME B&PV Code, Section XI, requirements. A material surveillance program will be initiated. The annulus between the reactor vessel and containment vessel will be continuously monitored for sodium leakage by using sodium aerosol detectors. The reactor closure area will be monitored by radiation detectors.

The SAFR design appears to meet the intent of GDC 32 with minor word changes.

GDC 33: Reactor Coolant Makeup

The low-pressure system for the SAFR LMR and the nature of the pool configuration will preclude the necessity for providing primary coolant makeup water. The reactor vessel/containment vessel annulus will be sized to limit the possible sodium leakage to prevent core uncovering should the annulus fill completely.

The staff tentatively accepts this position and believes that GDC 33 is not applicable to the LMR design.

GDC 34: Residual Heat Removal (SAFR PDC 4.7)

The SAFR reactor module will be equipped with a passive, natural-draft, air-cooling system for the reactor containment vessel. Operator action will not be required to actuate the system and operator action can not deactivate the system. The system will utilize multiple intake and exhaust structures. The natural circulation flow paths and flow rates will be sufficient to remove the decay heat following ATWS, station blackout, and loss-of-heat-sink events without loss of coolable core geometry.

The SAFR design appears to meet the intent of GDC 34 which was modified considerably in PDC 4.7 to reflect differences between LWR and LMR technologies.

GDC 35: Emergency Core Cooling

The proximity of the reactor vessel and the containment vessel will preclude a large loss-of-coolant accident from occurring. Therefore, there is no need for an emergency core cooling system, as is the case for the light-water reactor.

The staff tentatively agrees and believes that GDC 35 is not applicable to this LMR design.

GDC 36: Inspection of Emergency Core Cooling System (SAFR PDC 4.8)

Although DOE and RI note an ECCS is not required for LMRs (i.e., GDC 35 is not applicable), they believe that GDC 36 (which pertains to inspection of the ECCS) is relevant since the intent is to provide continuous surveillance and periodic inspection of the plant's shutdown heat removal system (SHRS). For the SAFR plant, these shutdown heat removal systems will be the reactor air cooling system (RACS) and DRACS. All DRACS piping and critical components outside the reactor containment building (RCB) will be periodically inspected. The in-vessel heat exchanger and piping inside guard pipes will be inspected through ports. RACS inlets and outlets will be visually inspected and the air passages can be inspected by remote visual means.

The staff believes this criterion appears to meet the intent of GDC 36.

GDC 37: Testing of Emergency Core Cooling System (SAFR PDC 4.9)

Consistent with the requirements described in PDC 4.8 for GDC 36 (above), the SHRS for the SAFR plant is designed to permit appropriate periodic testing to ensure integrity and operability of the

systems. RACS will operate continuously and will be monitored for airflow and temperature. DRACS will be tested at each planned shutdown.

The staff believes this criterion appears to meet the intent of GDC 37.

GDC 38: Containment Heat Removal

This criterion requires that a system be provided to rapidly reduce primary containment temperature and pressure following a loss-of-coolant accident (LOCA) and imposes certain requirements on that system. There is no comparable function or system on the SAFR design. Hence, this criterion was not applied to SAFR. The passive heat transfer systems used on SAFR for long-term cooling of the reactor closure (primary containment) and reactor building (secondary containment) are adequately covered by PDC 4.10, which applies to structural cooling systems and is equivalent to GDC 44.

The need for a containment system for the SAFR concept remains an open item. Therefore, any final decision on the applicability of GDC 38 remains an open item.

GDC 39: Inspection of Containment Heat Removal System

This criterion imposes inspection requirements on the containment heat removal system required by GDC 38.

Since the need for a containment system remains an open item, the applicability of GDC 39 is also an open item.

GDC 40: Testing of Containment Heat Removal System

This criterion imposes testing requirements on the containment heat removal system required by GDC 38.

Since the need for a containment system remains an open item, the applicability of GDC 40 remains an open item.

GDC 41: Containment Atmosphere Cleanup (SAFR PDC 5.11)

The intent of GDC 41 is to require systems to control the composition of the containment atmosphere. The primary containment atmosphere, including the reactor cover gas, will be monitored and isolated from the environment. Any release of radioactivity to the cover gas will be processed by the cover gas purification system. For releases from the cover gas, the secondary containment/confinement building will be isolated to restrict leakage to the atmosphere.

The SAFR design appears to meet the intent of GDC 42 with word changes to reflect differences in design, including effects of sodium aerosols.

GDC 42: Inspection of Containment Atmosphere Cleanup Systems (SAFR PDC 5.12)

The radioactive gas processing systems are designed to permit periodic inspection in accordance with Regulatory Guide 1.140 (Ref. 3.11), which references ANSI N510.

The SAFR design appears to meet the intent of GDC 42.

GDC 43: Testing of Containment Atmosphere Cleanup Systems (SAFR PDC 5.13)

The radioactive gas process will undergo periodic testing and maintenance in accordance with Regulatory Guide 1.140 (Ref. 3.11), which references ANSI N510.

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The SAFR design appears to meet the intent of GDC 43.

GDC 44: Cooling Water (SAFR PDC 4.10)

The SAFR design utilizes natural-draft air and the atmosphere as the ultimate heat sink. This criterion has been expanded to require structural heat removal systems as needed under normal operating or accident conditions by any coolant, not just water. This is in addition to the SHRS requirements.

The SAFR design appears to meet the intent of GDC 44.

GDC 45: Inspection of Cooling Water System (SAFR PDC 4.11)

The heat exchange surfaces will be inspected periodically.

The staff believes this criterion meets the intent of GDC 45.

GDC 46: Testing of Cooling Water System (SAFR PDC 4.12)

The passive systems will be tested periodically to ensure they are adequate.

The staff believes this criterion appears to meet the intent of GDC 46.

GDC 50: Containment Design Basis (SAFR PDCs 5.1 and 5.2)

The containment vessel will be designed to withstand, with sufficient margin, the temperature and pressure conditions resulting from normal operation, including anticipated operational occurrences and postulated accidents. Margins will be included to account for uncertainties in the accident phenomena and calculations.

Although these PDC appear to meet the intent of GDC 50, the fact that the containment issue remains an open item leaves the acceptance of these PDC an open item.

GDC 51: Fracture Prevention of Containment Pressure Boundary (SAFR PDC 5.4)

The containment pressure boundary will be designed so that under normal operating conditions and design-basis-accident conditions the materials will behave in a nonbrittle manner and the probability of rapidly propagating fractures is minimized. The containment vessel and closure head will be designed in accordance with ASME Code Section III, Division 1, Subsection NE. The secondary concrete containment/confinement building will be designed to ACI-349.

This PDC appears to meet the intent GDC 51, although final determination would be dependent upon the decision on containment.

GDC 52: Capability for Containment Leakage Rate Testing (SAFR PDC 5.5)

Leakage rate will be tested periodically for the containment vessel and the containment/confinement buildings and their material equipment and structures. The structural tests will be done in accordance with Subsection NE-6300, Division 1, Section III of the ASME Code, and leak rate tests will be similar to those specified in 10 CFR Part 50, Appendix J.

This PDC appears to meet the intent of GDC 52, although final determination is dependent upon the decision on containment.

GDC 53: Provisions for Containment Testing and Inspection (SAFR PDC 5.6)

The reactor containment is designed to permit inspection and testing of penetrations that have resilient seals or bellows. The lower portion of the containment vessel will have no penetrations and will be visually inspected. All closure head penetrations are designed to permit testing comparable to that specified in Appendix J to 10 CFR Part 50.

The SAFR design appears to meet the intent of GDC 53.

GDC 54: Piping Systems Penetrating Containment (SAFR PDC 5.7)

Piping systems penetrating the primary containment will be equipped with leakage detection, isolation, and containment capabilities having redundancy, reliability, and testability to ensure leakage is within acceptable limits as required to meet the containment safety function.

The SAFR design appears to meet the intent of GDC 54 with modifications to account for penetrations not requiring isolation, such as the non-radioactive IHTS sodium system.

GDC 55: Reactor Coolant Pressure Boundary Penetrating Containment (SAFR PDC 5.8)

In the SAFR design, the primary coolant sodium is contained within the reactor vessel in its entirety. No piping or systems penetrate the primary containment with primary sodium coolant during normal operation. Lines connected to the reactor cover gas that penetrate primary containment are equipped with double isolation valves. The fuel transfer port interfaces with the reactor cover gas and is closed during operation with two normally closed valves.

The SAFR design appears to meet the intent of GDC 55.

GDC 56: Primary Containment Isolation (SAFR PDC 5.9)

All lines penetrating the primary containment are equipped with redundant isolation valves placed as close to the primary containment as practical. During refueling and reactor component replacement, the fuel transfer cell is inerted and isolated by two normally closed valves. The reactor will be shut down during such transfers.

The SAFR design appears to meet the intent of GDC 56 with minor word changes.

GDC 57: Closed System Isolation Valves (SAFR PDC 5.10)

Any line penetrating the primary containment that is a part of or directly connected to the reactor coolant or cover gas boundary or is connected directly to the containment atmosphere has suitable isolation capability.

The SAFR design appears to meet the intent of GDC 57 with minor word changes.

GDC 60: Control of Releases of Radioactive Materials to the Environment (SAFR PDC 6.1)

The liquid and gaseous radioactive waste processing system provides for retention of waste products until the appropriate guidelines for discharge and dilution are met. Solid radioactive waste products are retained and packaged in approved transportation casks before being transported to long-term, high-level, disposal locations.

The SAFR design appears to meet the requirements of GDC 60 with minor word changes.

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GDC 61: Fuel Storage and Handling and Radioactivity Control (SAFR PDC 6.2)

The fuel storage and handling, radioactive waste, and other systems that may contain radioactive material are to be designed to ensure adequate nuclear safety under normal operation (including anticipated operational occurrences) and postulated accident conditions.

The SAFR design appears to meet the requirements of GDC 61 with minor word changes.

GDC 62: Prevention of Criticality in Fuel Storage and Handling (SAFR PDC 6.3)

Criticality in the fuel storage and handling systems is prevented by physical systems or processes and by use of geometrically safe configurations.

The SAFR design appears to meet the requirements of GDC 62.

GDC 63: Monitoring Fuel and Waste Storage (SAFR PDC 6.4)

The fuel storage areas and radioactive waste systems are provided with monitoring systems capable of detecting loss of residual heat removal capability and excessive radiation levels and initiating appropriate actions, should those conditions occur.

The SAFR design appears to meet the requirements of GDC 63.

GDC 64: Monitoring Radioactivity Releases (SAFR PDC 6.5)

Means are provided for monitoring reactor containment and confinement atmospheres, effluent discharge paths, and plant environs for radioactivity released during normal operation, including anticipated operational occurrences and postulated accidents.

The SAFR design appears to meet the intent of GDC 64.

3.4 Classification of Structures, Systems, and Components

3.4.1 Design Description and Safety Objectives

The intention of the staff review is to follow the current light-water reactor (LWR) practice in the classification of structures, systems, and components (SSCs) for the SAFR advanced LMR concept wherever applicable. For certain SAFR systems and components, interpretation of LWR guidelines is necessary recognizing those differences in plant design and technology between LMRs and LWRs. Use of the LWR guidelines should ensure that classification of the SSCs for LMRs compares favorably with those for LWRs. The seismic classification of safety-related structures, systems, and components ensures that these plant features are designed to withstand the effects of earthquakes without loss of capability to perform their safety functions.

3.4.2 Scope of Staff Review

The staff reviewed the application of a classification system to the safety-related structures, systems, and components of the SAFR design. This review focused on those safety-related plant features identified in PSID Tables 3.2-2, 3.2-3, 3.2-5, and 3.2-6 and the identification of the appropriate construction codes for these major safety-related structures, systems, and components.

The seismic classification of safety-related structures, systems, and components was reviewed to ensure that these plant features are designed to Regulatory Guide (RG) 1.26 (Ref. 3.12) as Quality Group A. 10 CFR 50.55a also states that components classified as Quality Groups B and C must

meet the requirements for Class 2 and Class 3 components, respectively, in Section III of the ASME Boiler and Pressure Vessel Code.

3.4.3 Conformance With NRC General Design Criteria Relevant to System

Quality Group and Seismic Classifications

The principal design criteria considered by the staff in its review of the system quality group classification are GDC 1, "Quality Standards and Records" of Appendix A to 10 CFR Part 50 and 10 CFR 50.55a, "Codes and Standards." GDC 1 requires that nuclear power plant systems and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. RG 1.26 (Ref. 3.12) is the principal document used in the staff review for identifying on a functional basis the components of those systems important to safety as NRC Quality Groups A, B, C, or D. 10 CFR 50.55a identifies those American Society of Mechanical Engineers (ASME) B&PV Code, Section III, Class 1 components that are part of the reactor coolant boundary. These reactor coolant boundary components are designated in RG 1.26 as Quality Group A. 10 CFR 50.55a also states that components classified as Quality Groups B and C must meet the requirements for Class 2 and Class 3 components, respectively, in Section III of the ASME Code. DOE does not completely adhere to the guidance of the above design criteria for the SAFR plant in the safety class (quality group) classification of safety-related structures, systems, and components in PSID Tables 3.2-2, 3.2-3, and 3.2-6. The relationship between the NRC quality groups and the SAFR safety classes is summarized in PSID Table 3.2-1. PSID Table 3.2-1 also identifies the quality assurance, seismic, ASME Code, electrical, and structural requirements for each SAFR safety class. In PSID Table 3.2-2, the reactor containment building (RCB) is classified as Safety Class 3 (SC-3); however, for LWRs, the secondary containment is classified as Safety Class 2. The staff would require further review to accept the position that the RCB be classified SC-3. It has also been the staff's position that all major components of the reactor coolant boundary should be classified Safety Class 1. The staff therefore does not agree with the following SAFR statement in Section 3.2.2.1 of the PSID:

Where more than one system is capable of accomplishing a nuclear-safety function and one system, on its own, satisfies all nuclear safety-related system requirements (e.g., redundancy, diversity, capacity), the designer will classify the latter to the corresponding SC [safety class] and the other(s) as NNS [non-nuclear safety].

Implementation of this statement has resulted in the downgrading of certain systems and components in PSID Table 3.2-6. The staff considers that the safety class and the seismic class of these systems and components may not be acceptable because the SAFR statement is not consistent with the NRC "defense in depth" approach to safety for LWRs. The safety class and seismic class of these systems and components in PSID Table 3.2-6 will require further review and may need to be revised in accordance with RGs 1.26 (Ref. 3.12) and 1.29 (Ref. 3.13). Consistency with these guidelines would require changing either the primary or the SASS control rods to Safety Class 1 systems. The steam-water reaction, pressure-relief system would also be classified as Safety Class 1. The radioactive waste system would be classified as Safety Class 3 to be consistent with RG 1.26, and all systems containing radioactive materials in the inert gas system would also be classified as Safety Class 3.

The principal design criterion considered by the staff in its review of the seismic classification is GDC 2, "Design Base for Protection Against Natural Phenomena." GDC 2, in part, requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions.

The earthquake that these safety-related plant features are designed for is the safe-shutdown earthquake (SSE) as defined in Appendix A to 10 CFR Part 100. The SSE is based upon an evaluation of the maximum earthquake potential and is that earthquake which produces the maximum vibratory ground motion for which the safety-related structures, systems, and components are to

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remain functional. The plant structures, systems and components that should be designed to remain functional during an SSE are designated as seismic Category I in Regulatory Guide 1.29 (Ref. 3.13). RG 1.29 is the principal document used by the staff for identifying those plant features that, as a minimum, should be designed to seismic category I requirements. As a minimum, the following components from PSID Tables 3.2-5 and 3.2-6 should be seismic Category I:

- control building
- steam generator relief and dump system

3.4.4 Conclusions

The staff has reviewed the use of the DOE classification system in SAFR PSID Table 3.2-2, 3.2-3, 3.2-5, and 3.2-6 and, in general, finds the safety class classification of safety-related structures, systems, and components to be acceptable except for certain systems and components in PSID Tables 3.2-2, 3.2-3, and 3.2-6 that are discussed above. These PSID tables, in part, identify major components in fluid systems (such as pressure vessels, heat exchangers, pumps, piping, and valves), and mechanical systems (such as cranes, fuel-handling equipment, and other miscellaneous equipment). The SAFR PSID does not provide piping and instrumentation diagrams that identify the classification boundaries of interconnecting piping and valves. The staff agrees that this level of detail is unnecessary for this stage of review. The staff concludes that except for those certain items in PSID Tables 3.2-2, 3.2-3, and 3.2-6 that are inconsistent with RG 1.26, the safety class classification and the construction requirements for other safety-related structures, systems, and components identified in these tables are in conformance with the ASME Code, the Commission's regulations, and the guidance provided in RG 1.26 and provide assurance that component quality is commensurate with the importance of the safety functions of these plant features. This is acceptable, as it constitutes an acceptable basis for satisfying the requirements of GDC 1.

The staff also concludes that except for those certain items in PSID Table 3.2-6 that are inconsistent with RG 1.29, the seismic classification of structures, systems, and components identified in PSID Tables 3.2-2, 3.2-3, 3.2-5, and 3.2-6 are in conformance with the Commission's regulations and are properly classified as seismic Category I items in accordance with RG 1.29. This constitutes an acceptable basis for satisfying, in part, the requirements of GDC 2.

The classification system and rules used by DOE and RI for the SAFR design will require a much more broadly based NRC review at a later design stage before the design can be accepted by the NRC. There are many questions related to NRC's traditional defense-in-depth philosophy imbedded in this approach. It has the potential to affect traditional approaches to single-failure, diversity, and redundancy, and will play an important role in considering the separation of the balance of plant (BOP) from safety functions. The staff would require that these areas be addressed specifically in a PSAR that employed this approach. The staff would also initiate a broadly based NRC review of these aspects at that time.

3.5 Seismic Design

3.5.1 Design Description and Safety Objectives

The procedures and parameters used to develop seismic loadings and criteria for seismic Category I structures, systems, and components are consistent with current light-water-reactor (LWR) seismic analysis and design practice where applicable. The objective is to ensure, with a high level of confidence, that the systems or components housed in these structures will maintain their 10 CFR Part 100-related radionuclide control function under design-basis conditions.

3.5.2 Scope of Review

The staff focused its review on the identification and acceptability of key design criteria, key safety issues, and the potential of the design to meet these criteria, either through safety analyses or model tests. Neither the staff nor its contractors performed independent calculations on seismic analysis and design, design of Category I structures, or foundation design.

3.5.3 Conformance With Seismic Design Criteria

The principal design criteria considered by the staff in its review of (1) seismic design, (2) design of Category I structures, systems, and components, and (3) foundations are (1) Appendix A to 10 CFR Part 100 and (2) 10 CFR Part 50 (GDC 2, 4, 16, and 50). RI has also committed to meet the intent of the following regulatory guides for seismic design, design of Category I structures, systems, and components, and foundations:

- RG 1.12, "Instrumentation for Earthquakes"
- RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants"
- RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants"
- RG 1.69, "Concrete Radiation Shields for Nuclear Power Plants"
- RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis"
- RG 1.94, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants"
- RG 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components"
- RG 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)"

RI has also provided a SAFR-specific response to the generic safety issues described in NUREG-0933 (Ref. 3.14) in Appendix B to the PSID.

The use of these criteria as defined by applicable codes, specifications, standards, and regulatory guides provides reasonable assurance that, in the event of earthquakes, the structures, systems, and components will withstand the specified design conditions without impairment of structural integrity or safety function. Conformance with these criteria constitutes an acceptable basis for satisfying, in part, the requirements of General Design Criteria 2, 4, 16, and 50.

Seismic Input

In order to develop a standard plant design that could be sited on 75 percent of the prospective U.S. sites, data were collected from 106 existing nuclear power plant sites and appropriate seismic boundaries were developed. The standard SAFR plant will be designed for a range of site conditions and seismic input levels. Six analysis cases were developed as follows:

- soft soil
- relatively soft soil which produces high floor response spectra (FRS) peak accelerations in the low-frequency range
- intermediate stiff soil for structures and components in the intermediate-frequency range
- rigid rock site which produces high FRS peak accelerations in the high-frequency range
- rock site with very large shear wave velocity (8000 ft/sec)
- layered soft soil underlaid by a very hard rock

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Various seismic input levels for different shear wave velocities are presented in PSID Table 3.7-1 and PSID Figure 3.7-1. The range of soil limits covered in six cases mentioned above are considered to satisfy local design requirements at a majority of potential U.S. sites. DOE has committed to perform special studies to cover particular soil conditions and their impact on seismic Category I structures. When a site is chosen, the soil properties will be compared with the design envelope and the subject will be addressed at that time.

DOE also stated that an OBE equal to one-third the zero period ground acceleration (ZPGA) of the SSE will be used. The staff questioned this deviation from the requirements of 10 CFR Part 100, Section V(a)(2). See Question 2.1 of Appendix G to the PSID, "NRC Question Responses." The applicant's response is as follows:

When a site is selected for a SAFR plant, local geology and seismology will be evaluated and the appropriate values of SSE and OBE established for that site. The OBE will be chosen as one which "could reasonably be expected to affect the plant site during the operating life of the plant" in accordance with the 10 CFR 100, Appendix A requirement. If the OBE or SSE are larger than those used for design of the standard plant, then either seismic reanalysis and redesign, if required, will be done to accommodate the higher seismic input values or the plant will be moved to a different site for which the standard design is acceptable. When the OBE that is selected on the basis of return frequency for a particular SAFR site is less than 1/2 SSE for that site, the case for departure from that part of 10 CFR 100 will be made and justification presented as allowed by 10 CFR 100.

This commitment provides reasonable assurance that the seismic input will be adequately defined so as to form a conservative basis for the design of structures to withstand seismic loadings.

Damping Values

Damping values will be generally in accordance with Regulatory Guide (RG) 1.61 (Ref. 3.15), with the exception of cable raceway systems for which the damping value is 15 percent of the critical value for the SSE. The staff has accepted this high damping for cable tray systems for other LWR plants, when justified by test results.

Another exception is for piping damping values. Recent experimental data have demonstrated that the damping values of the regulatory guide are overly conservative for piping and ASME Code Case N-411 allows higher values, which are primarily based on data obtained from uninsulated or lightly insulated pipe. For SAFR piping, DOE will use damping values even larger than Code Case N-411. A justification for this will be provided on the basis of tests performed at the Hanford Engineering Development Laboratory (HEDL) and under the LMR Base Technology Program. This remains an open item.

Design Time-History

A synthetic time-history was generated by the method described in Section 2.5.1(c) of Bechtel BC-TOP-4-A Rev. 3 (November 1974) for earthquake records of the duration of strong motion. The response spectra generated from a synthetic time-history meet the requirements of Regulatory Guide 1.60 (Ref. 3.16) for input response spectra.

Design Response Spectra

Horizontal design response spectra will be selected in accordance with RG 1.60 (Ref. 3.16).

Vertical ground acceleration will be in accordance with NUREG-0800, SRP Section 3.7.1.II.1a (Ref. 3.1).

Seismic Analysis Method

A lumped-parameter model approach will be used in accordance with BC-TOP-4-A. DOE has stated that a summary of natural frequencies, mode shapes, modal responses, and response loads determined by seismic analyses will be given in the final safety analysis report (FSAR).

Three-dimensional earthquake effects will be combined using the square-root of the sum-of-squares (SRSS) method, where practical. Where the SRSS method has inherent difficulty, combination would be done according to "the component factor method" as follows:

$$R_{\text{total}} = R_i + 0.4R_j + 0.4R_k; R_i < R_j < R_k$$

where R_i , R_j and R_k are the set of three co-directional response maxima caused by individual excitation in three directions. Modal responses are combined in accordance with the requirements of Regulatory Guide 1.92 (Ref. 3.17).

Soil-Structure Interaction

The lumped-parameter approach will be employed with the visco-elastic properties of foundation media represented by equivalent springs and dampers. PSID Figure 3.7-4 shows a schematic lumped-parameter model of the soil-structure system with equivalent foundation springs and the dampers representing the foundation impedances for horizontal seismic excitation. The equations of motion will be solved in the frequency or time domain, by a method described in Appendix D of BC-TOP-4-A.

The staff raised the question of the current SRP requirement of enveloping the results of two methods, namely lumped parameter and finite boundaries. (See Question 3.23 of Appendix G to PSID, "NRC Question Response.") The response is a commitment that when a site is chosen for a SAFR plant, site-specific analyses using both methods, including soil-structure interaction effects, will be made to show that the design of a standard SAFR plant envelopes those resulting from site-specific analyses. This commitment forms an acceptable basis for the adequacy of the soil-structure interaction analyses.

Development of Floor Response Spectra

The time-history analysis method will be used to develop floor response spectra (FRS) is described in BC-TOP-4-A. The FRS will be converted into design response spectra by a smoothing and peak broadening process consistent with Regulatory Guide 1.122 (Ref. 3.18).

Torsional Effects

Torsional effects will be considered in a lumped-mass model which will include all stiffness and inertial characteristics, torsional moment of inertia, eccentricity, mass moment of inertia, center of rotation, and center of mass.

To account for accidental torsion, an additional eccentricity of 15 percent of the maximum building dimension will be included. This is in accordance with the requirement of SRP Section 3.7.2.II.11 (Ref. 3.1).

Seismic Analysis for Equipment and Other Subsystems

Equipment will be qualified seismically by using either analysis or dynamic testing or a combination of both. Analysis will be performed as a lumped-mass and stiffness model similar to those described under the seismic analysis methods, including

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- response spectrum analysis
- time-history analysis
- equivalent static load method

This meets the intent of the requirements of SRP Section 3.7.3 (Ref. 3.1). Electrical raceway supports will be analyzed by response spectrum methods or experimental data acquired from actual seismic testing performed on support systems. This is consistent with LWR practice and provides the basis for acceptance.

Design criteria and analytical techniques that will be used for piping systems are described in BP-TOP-1, which also addresses torsional effects of eccentric masses.

The PSID is not clear on how electrical equipment will be seismically qualified. Until this item is clarified to the satisfaction of staff, it is considered an open item.

Seismic Instrumentation

The seismic instrumentation program will comply with RG 1.12 (Ref. 3.19) with certain exceptions noted in the PSID. The exceptions listed appear to be justifiable although further justification may be necessary. For example, peak strain gauges will be used in lieu of peak recording accelerographs inside the secondary containment where the temperature and the radiation limitations render accelerographs unacceptable for use.

Data obtained from such installed seismic instrumentation will be sufficient to determine the acceptability of continued operation, in the event of an earthquake.

The control room for the SAFR design is not proposed as a seismic Category I structure. Activation of seismic triggers will cause an audible and visual annunciation in the control room to alert operators that an OBE or SSE has occurred. When OBE levels have been exceeded, the reactor will be manually shut down and maintained in a safe-shutdown condition by passive systems located on the nuclear island which is a seismic Category I structure block. The acceptability of the control room not being a safety-grade (Category I) structure is addressed in Sections 3.2 (GDC 19) and 3.3.3 of this SER.

3.5.4 Design of Seismic Category I Structures

The seismic Category I structures for the SAFR design are

- reactor secondary containment building
- reactor service building
- foundation and lower walls of the steam generator building

The nuclear island consists of these three structures integrated as one, on a common basemat.

Loads and Loading Combinations

The loads and loading combinations for all seismic Category I concrete structures will be in accordance with American Concrete Institute Code ACI-349, RG 1.142 (Ref. 3.20), and SRP Sections 3.8.1 and 3.8.4 (Ref. 3.1).

The loads and loading combinations for steel structures will be determined in accordance with American Institute of Steel Construction (AISC) specifications and SRP Sections 3.8.2 and 3.8.4 (Ref. 3.1). DOE has pointed out that the NRC and the industry are attempting to significantly alter these guidelines (Ref. 3.21) and that the SAFR design will use these new criteria as they become better

defined. The staff will have to review the application of any new methods, when available. This is an open item.

Design and Analysis Procedures

The general concept for the design and analysis of structures is consistent with LWR practice. A three-dimensional, finite-element model of the nuclear island is used in the analysis.

For soil-structure interaction, the boundary conditions of the three-dimensional model are represented by linear spring elements at each node in three orthogonal directions. The spring constants are obtained by using half-space theory.

The design of all Category I structures is based on applicable sections of the following codes and standards:

- ACI-349 American Concrete Institute, "Code Requirements for Nuclear Safety-Related Concrete Structures" (augmented by Regulatory Guide 1.142)
- AISC-S32f American Institute of Steel Construction, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings"
- ASME-III American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components"

DOE stated that the applicable SAFR principal design criteria (PDC) are Criterion 1.1 through Criterion 1.6 which are described in Section 3.1.1 of the PSID.

Foundation Design

The criteria used in the design of the foundations and the stability investigation of the nuclear island are consistent with the requirements of SRP Section 3.8.5 (Ref. 3.1). Stability factors of safety given in PSID Section 3.8.5.3 are acceptable as they are consistent with LWR practice and SRP requirements. Conformance with these criteria constitutes an acceptable basis for satisfying, in part, the requirements of GDC 2 and 4.

3.5.5 Conclusions

The staff concludes that the procedures to be utilized for (1) seismic design, (2) design of Category I structures, systems, and components, and (3) foundation design are generally acceptable, subject to satisfactory resolution of the open items identified above. The use of these procedures will provide reasonable assurance that in the event of a design-basis earthquake, the structural integrity of structures, systems, and components will not be impaired.

3.6 Tornado, Flood, Missile, Dynamic Effects of Pipe Rupture and Mechanical Systems

3.6.1 Tornado Design Basis

RI has proposed that, instead of using Regulatory Guide 1.76 (Ref. 3.22) as a basis for tornado design of the SAFR plant, the ANSI-ANS 2.3 - 1983 Region I data base be used. The staff has not accepted the ANSI document as a replacement for Regulatory Guide 1.76 because the ANSI design decisions are based largely on subjective judgment. However, Regulatory Guide 1.76 is being reevaluated using the considerable quantity of tornado data now available that was not available when Regulatory 1.76 was developed. DOE and RI have discussed extensively their rationale for the tornado design basis they propose to use for the SAFR plant (Appendix D to the PSID). The staff has reviewed the proposed approach and cannot accept it at this time for the reason noted above and

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because of the interpretation of the tornado frequency (Appendix D of the PSID interprets the frequency limits as the probability of exceeding 10 CFR Part 100 dose limits as a result of tornados rather than applying a frequency limit to define wind speeds associated with tornados of that frequency). Because of the lack of supporting data and the heavy reliance on engineering judgment, the staff cannot accept the analytical methods used in the PRA evaluation in PSID Appendix C without additional information and review.

3.6.2 Flood

The SAFR design is based on a normal groundwater level of 10 feet below grade and a maximum groundwater level at grade.

When a site is chosen for a SAFR plant, Regulatory Guide 1.59 (Ref. 3.23) will be used to establish the maximum probable flood and other hydrological considerations. Flooding from internal leaks will be controlled by catch pans and drains to collection tanks, if necessary.

3.6.3 Missiles

The potential for a turbine to generate a missile (e.g., a broken blade) is reduced by use of solid-core rotors. However, the staff will require the turbines to be oriented (or barriers provided) so that such failures will not damage or affect structures, systems, and components (SSCs) or functions important to safety.

Aircraft hazards have not yet been evaluated. They will be addressed for a specific site when the site is selected. Potential sources of missiles (e.g., from high-pressure systems and rotating machinery) will be reviewed at a later design stage.

3.6.4 Dynamic Effects of Pipe Rupture

SAFR PDC 1.5 requires that nuclear-safety-related SSCs be designed for the dynamic effects from pipe breaks. SRP Sections 3.6.1 and 3.6.2 (Ref. 3.1) are used for guidance, along with several variations that are identified in Section 3.6.1.1 of the PSID. DOE and RI have committed to incorporate changes in the SAFR design to reflect recent revisions of GDC 4 regarding the dynamic effects of pipe breaks.

The SAFR design does not have high-energy systems within the nuclear island building or within the steam generator building (SGB), since the high-pressure steam generator (SG) nozzle is outside the SGB wall. RI intends to attempt to eliminate the need for large restraints on the high-pressure steam lines. If needed, pipe restraints can be incorporated.

A major or double-ended guillotine (DEG) rupture is not considered credible for sodium piping outside the reactor vessel in the SAFR design and is not used as a design basis. The leak-before-break argument is the justification for this position. Since the primary piping is within the reactor vessel and submerged in sodium, a small leak cannot be detected and a DEG rupture is analyzed for the primary sodium piping. The leak-before-break concept for sodium piping is based on

- use of leak detection and in-service inspection
- previous liquid-metal operating experience with noncatastrophic failures
- use of ductile piping materials
- low operating pressures:
 - sodium operating temperatures well below atmospheric boiling
 - continuous monitoring for detection of small leaks and subsequent corrective action before gross failure

A fracture mechanics development program is planned to validate the use of leak-before-break for the chromium-molybdenum (9 Cr-1Mo) piping used in the SAFR intermediate heat transport system (IHTS).

The design-basis leak for the IHTS is taken as a moderate energy fluid system (MEFS) leak (as defined in NRC Branch Technical Position MEB 3-1 (Ref. 3.1)) even though DOE and RI find that the leak-before-break concept is applicable. This is consistent with that used on the Clinch River breeder reactor and is acceptable. The dynamic effects of such a leak at the low operating pressure of the IHTS is limited to jet impingement forces on the piping insulation. The pipe insulation will be designed to withstand such forces.

3.6.5 Mechanical Systems

The structural integrity of Safety Class 1, 2, and 3 components, component supports, and core supports will be ensured by designing them in accordance with ASME Code, Section II, Division 1, and the appropriate high-temperature code cases. The PSID addresses the loading combinations and stress limits for the appropriate design transients and plant duty cycles. The operating conditions and transients for individual components have not been completely defined. The PSID discusses the mechanical design criteria for the major components.

As described in PSID Subsection 3.9.1, the reactor vessel and the reactor internal components (e.g., core support) will be built to ASME Code, Section III, Subsection NB and the appropriate high-temperature code cases. PSID Subsection 3.9.2 covers ASME III Code Class 2 and 3 components which includes the containment vessel and the reactor closure head.

PSID Subsection 3.9.3 addresses those non-safety-related items that are given attention because they do provide added protection for both plant investment and public safety. These are identified as the control rod drive system, the intermediate heat transport system (IHTS), and the direct reactor auxiliary cooling system (DRACS). The control rod drive housings are part of the containment vessel and are classified Safety Class 2. The self-actuated shutdown system (SASS) housing is also categorized as a Safety Class 2 system, but the application of the appropriate safety classification to core structures and components (e.g., fuel, core restraints) which are involved in the passive shutdown function have not yet been defined.

3.6.6 Conclusions

The staff concludes that the design criteria described in Section 3.5 of this SER are generally acceptable, with the exception of the design-basis requirements for tornados. However, the design requirements for many important components remain to be defined.

3.7 References

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- 3.2 DOE, "SAFR -- Sodium Advanced Fast Reactor," Preliminary Safety Information Document, AI-DOE-13527, Initial Issue - October 1985, with amendments. (DOE Applied Technology.)
- 3.3 NRC, "Regulation of Advanced Nuclear Power Plants; Statement of Policy, 51 Federal Register 24643, July 8, 1986.
- 3.4 NRC, "Safety Goals for the Operation of Nuclear Power Plants," 51 Federal Register 28044, August 1, 1986.

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- 3.5 NRC, "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," 51 Federal Register 32138, August 8, 1985.
- 3.6 NRC, "Nuclear Power Plant Standardization, Policy Statement," 52 Federal Register 34884, September 15, 1987.
- 3.7 NRC, "Standardization of Advanced Reactor Designs," SECY 88-202, July 15, 1988.
- 3.8 DOE, "SAFR -- Sodium Advanced Fast Reactor," Safety Test Plan, 149TP000001, October 21, 1987. (DOE Applied Technology.)
- 3.9 NRC, Generic Letter 84-01, "Use of Terms, Important to Safety, and, Safety Related," January 5, 1984.
- 3.10 NRC, "Periodic Testing of Electric Power and Protection Systems," Regulatory Guide 1.118, Rev. 2, June 1978.
- 3.11 NRC, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Regulatory Guide 1.140, Rev. 1, October 1979.
- 3.12 NRC, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Regulatory Guide 1.26, Rev. 3, February 1976.
- 3.13 NRC, "Seismic Design Classification," Regulatory Guide 1.29, Rev. 3, September 1978.
- 3.14 NRC, "A Prioritization of Generic Safety Issues," NUREG-0933, with supplements.
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- 3.16 NRC, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Regulatory Guide 1.60, Rev. 1, December 1973.
- 3.17 NRC, "Combined Modal Responses and Spatial Components in Seismic Response Analysis," Regulatory Guide 1.92, Rev. 1, February 1976.
- 3.18 NRC, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components," Regulatory Guide 1.122, Rev. 1, February 1978.
- 3.19 NRC, "Instrumentation for Earthquakes," Regulatory Guide 1.12, Rev. 1, April 1974.
- 3.20 NRC, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)," Regulatory Guide 1.142, Rev. 1, October 1981.
- 3.21 NRC, H. Hwang, B. Ellingwood, and S. Kagami, "Probability Based Load Combination Criteria for Design of Concrete Containment Structures, NUREG/CR-3876, Brookhaven National Laboratory, BNL-NUREG-51795, August 1985.
- 3.22 NRC, "Design Basis Tornado for Nuclear Power Plants," Regulatory Guide 1.76, April 1974.
- 3.23 NRC, "Design Basis Floods for Nuclear Power Plants," Regulatory Guide 1.59, Rev. 2, August 1977.

Table 3.1 Correlation of 10 CFR Part 50, Appendix A, general design criteria to SAFR principal design criteria

GDC	PDC	GDC	PDC
1	1.1	35	b
2	1.2	36	4.8
3	1.3	37	4.9
4	1.5	38	b
5	1.6	39	b
10 ^a	2.1	40	b
11	2.2	41	5.11
12	2.3	42	5.12
13	2.4	43	5.13
14	2.5	44	4.10
15	2.6	45	4.11
16	2.7	46	4.12
17	2.8	50 ^a	5.1, 5.2
18	2.9	51	5.4
19	2.10	52	5.5
20	3.1	53	5.6
21	3.2	54	5.7
22	3.3	55	5.8
23	3.4	56	5.9
24	3.5	57	5.10
25	3.6	60 ^a	6.1
26	3.7	61	6.2
27	3.8	62	6.3
28	3.9	63	6.4
29	3.10	64	6.5
30	4.2	c	1.4
31	4.13	c	1.7
32	4.3	c	4.1
33	b	c	4.4
34	4.7	c	4.5
		c	4.6

^a There are no 10 CFR Part 50 criteria numbered 6 through 9, 47 through 49, or 58 and 59.

^b These criteria are unique to LWRs.

^c These criteria are unique to LMRs.

4 REACTOR

4.1 Summary Description

The reactor system, which will consist of the core assemblies and the control and safety rod drive mechanisms, will be entirely contained within the reactor vessel. The reactor will produce 900 MW of thermal energy. The reactor design specifications are given in Table 4.1.

The core will include 96 fuel assemblies, 46 internal blanket assemblies, 48 radial blanket assemblies, 9 control and shutdown rod assemblies, 54 steel shield assemblies, and 126 B₄C shield assemblies. The fuel and blanket assemblies are designed for a residence time of 4 years. Uranium-plutonium-zirconium (U-Pu-Zr) metal is the reference fuel and blanket material. A core map is shown in Figure 4.1

The fuel assembly will include an array of 271 fuel pins with upper and lower shield assemblies in a hexagonal-shaped duct, an inlet nozzle assembly at the bottom of the duct, and an outlet nozzle assembly at the top of the duct. The design provides a means for handling the fuel and also provides a means for controlling the coolant flow along the fuel pins. The metal pins each contain a 38-inch column of U-Pu-Zr fuel with space above the fuel to accommodate fission gases.

The blanket assembly and pins are similar to the fuel assembly fuel and pins. The hexagonal duct of the assembly is the same as that of the fuel assembly. There are 169 pins in a blanket assembly, and each pin is larger in diameter than a fuel pin. The metal blanket pins each contain a 46-inch column of U-Zr material. The internal and radial blanket assembly designs are identical.

The radial shield assembly designs are similar in configuration to those of the fuel assembly. The steel shield assemblies contain 37 closely packed solid stainless steel rods, and the B₄C assemblies contain B₄C-filled rods. The three rows of radial shield assemblies will be designed to be sufficient to protect the adjacent permanent structures from excessive neutron damage during the 60-year plant life. Furthermore, all in-vessel shielding will contribute to the overall effectiveness of the biological shielding located external to the reactor vessel (RV).

Reactor operation and shutdown will be controlled by six primary control assemblies and three secondary shutdown assemblies. The primary control rod system will consist of the primary control rod drive mechanisms, the drivelines, and the primary control assemblies. The drive mechanism will be mounted on top of the RV closure and will provide mechanical actuation for the insertion, withdrawal, and scram functions of the control rod absorber. The secondary shutdown rod system will consist of the secondary shutdown rod drive mechanism, the driveline, and the secondary shutdown assembly. It will provide a backup shutdown for off-normal conditions and will be independent of, and diverse from, the primary system. Automatic reactor shutdown scram will be accomplished by release of the absorber bundle from an electromagnet. This magnetic release will be actuated either by the interruption of power to the magnet or the loss of magnetic flux permeability of a Curie point material caused by a core outlet overtemperature. (See Section 4.6, below, for further information on the reactivity control systems.)

4.2 Fuel System

4.2.1 Design Description

The fuel and blanket subassembly design and operational specifications are presented in Chapter 4 of the PSID (Ref. 4.1) and are not reproduced here. The fuel design specifies the ternary fuel, 71% U-19% Pu-10% Zr, and the blanket design specifies binary fuel, 90% U-10% Zr. The plutonium source assumed for the SAFR design is LWR recycle Pu. Both fuel and blanket will be clad with low-swelling HT-9 steel, and the subassembly ducts will also be fabricated of HT-9. The fuel and blanket assemblies are designed for a 48-month and 100-month lifetime, respectively. The fuel

discharged will have a maximum peak burnup of 148 MWd per kilogram (~ 15 atom percent) and the blanket will have a peak burnup of 49 MWd per kilogram. During this period of operation, considering normal and anticipated duty cycle events which include load following and run beyond cladding breach operation, no more than 0.1 percent of the pins in the (equilibrium) core are predicted to fail. The system will be designed to tolerate a set of design-basis accidents with allowable consequences ranging from no significant degradation of expected fuel lifetime to maintenance of coolable geometry (see Chapter 15 of PSID, Ref. 4.1).

4.2.2 Scope of Review

The fuel system proposed by RI, U-Pu-Zr fuel with HT-9 cladding, is a new concept with little operational experience. Despite many years of successful operation with metal fuel in EBR-II, the differences in material, geometry, and operating conditions are such that direct application of that experience is difficult without additional fuel and material testing, safety tests, and analytical model development. A research and development (R&D) program sponsored by DOE and being implemented by Argonne National Laboratory, as part of the Integral Fast Reactor (IFR) Fuel Performance Program, is currently under way to provide support for the SAFR fuel and core design. The review presented below was carried out with the recognition that a new technology is being developed and, although much research has already been done in support of the program, much remains to be done.

The staff review consisted of an assessment of the current state of knowledge with respect to the SAFR fuel system concept and a review of the R&D effort planned within the IFR program. Final judgment regarding the ability of the SAFR design to meet the design criteria of 10 CFR Part 50 and the above objectives must await a detailed review of the R&D program results. The more limited objective of the review was to identify potential problems in the design that could be ascertained at this early date and that could lead to major safety-related problems. A second objective was to determine whether the R&D program would lead to development of the experimental data base and analytical tools that will eventually be required to support licensing of the SAFR design.

The review was carried out using published literature as a basic resource, in conjunction with the ANL IFR collection of reports, some of which is DOE Applied Technology information. This literature was supplemented by several information exchange meetings presented by ANL staff to BNL and NRC personnel. In addition, a series of written exchanges in the form of questions and replies provided useful clarifications.

4.2.3 Design Criteria

The criteria to be used in fuel system safety reviews are discussed in the Standard Review Plan (SRP) (Ref. 4.2). The objectives of the review derive from 10 CFR Part 50 (GDC 10). According to the SRP, the objectives of this criterion are to provide assurance that

- The fuel system is not damaged as a result of normal operation and anticipated operational occurrences.
- Fuel system damage is never so severe as to prevent control rod insertion when it is required.
- The number of fuel rod failures is not underestimated for postulated accidents.
- Coolability is always maintained.

With minor changes in wording, the SAFR PSID (Ref. 4.1) uses these four objectives for the reactor and fuel system design criteria. These are presented in Section 3.1 of the SAFR PSID.

4.2.4 Research and Development

Analytical tools, together with a supporting experimental data base, are being developed for analysis of the fuel system response to the range of design and exposure conditions anticipated. EBR-II provides an extensive irradiation experience data base for the metal-fuel concept. The Mark II uranium-fissium driver fuel clad with Type 316 stainless steel, which is based on many years of experimental and analytical development, has been successfully irradiated to burnups near the SAFR design limit. This experience lends support to the metal-fuel concept proposed by SAFR designers. It is noted, however, that the basic materials, geometry, and exposure conditions are different in the EBR-II and SAFR fuel designs. As a result of these differences, analytical models are used to extrapolate the EBR-II data to the SAFR design concept. Much prototypic experimental data remains to be developed in order to verify the models and to establish the basic material compatibility relationships between fuel, cladding, and sodium.

In order to meet the NRC safety criteria for the proposed design, analytical tools and supporting experimental data are required in two broad areas: (1) fuel design and performance based on the specified duty cycle of the system and (2) response of the fuel to transients. The development efforts under way are discussed briefly below.

4.2.4.1 Fuel and Cladding Fabrication Technology

The SAFR liquid-metal-reactor (LMR) plant is designed to contain integral fuel reprocessing facilities in the form of a commercial-scale fuel cycle facility that will include fabrication, reprocessing, and waste treatment. At the end of its lifetime, fuel will be reprocessed using a pyrometallurgical technique to separate out fission products. This processing technique has been demonstrated on a laboratory scale. The development work leading to engineering scale demonstration of the technique is being done at the present time. Thus the ability of the reprocessing technique to produce U-Pu-Zr fuel with the requisite quality assurance standards of uniform composition from batch to batch has yet to be demonstrated on a commercial scale. There will no doubt be fissium products remaining in the fuel. The final composition of the reprocessed fuel has yet to be determined.

HT-9 steel has been chosen as the reference cladding material as a result of its demonstrated low-swelling characteristics at neutron fluences of interest to the LMR program. The staff notes that, although this material may have potentially favorable properties, little is known about its structural response with extended irradiation as planned in SAFR. Further data and response characteristics data are expected to be provided as experience with this material is obtained through the EBR-II and Fast Flux Test Facility (FFTF) irradiation programs. The planned irradiation programs appear to be adequate to provide the needed information in a timely manner consistent with the schedule proposed for the SAFR program. This steel alloy is a relatively new one and its uniformity of composition and thermomechanical properties from batch to batch has not yet been adequately demonstrated. In addition, welding and fabrication techniques have only recently been developed to suit this alloy.

Since the technology is new, particular attention needs to be paid to quality assurance of material supply, fabrication, and welding techniques.

4.2.4.2 Fuel Design and Performance Methodology

A program of analytical model development, irradiation performance testing, out-of-pile materials testing, and experimental verification is being performed as part of the ANL IFR program. This program contains the essential research efforts required to develop the technology to support the SAFR design. The brief review of the major elements of the program points out several phenomenological issues that are important to the success of the metal-fuel program. Future research needs to be followed relatively closely for new developments.

The LIFE-METAL computer code is the analytical tool being developed at ANL to model the response of the metal fuel and blanket elements to steady-state and operational transient conditions. The code, which is used as the fuel design tool, has been adapted from earlier versions that were used to analyze oxide, carbide, and nitride fuel systems. It is fully operational in its application to metal fuel at this time. It contains preliminary models or correlations of the relevant physical phenomena and incorporates the latest available thermophysical property data on metal fuels. The staff has not reviewed the LIFE code modeling; however, analysis done by ANL using this code indicates good agreement with experimental data. The staff should review LIFE-METAL as the metal-fuel LMR matures.

At the present time, ANL believes that the fuel lifetime is limited by the mechanism of creep rupture of the HT-9 cladding under internal loading caused by fission gas plenum pressurization. Cladding wastage resulting from chemical exchange processes between the fuel and the cladding is also considered. It is believed that, because of the properties of the highly porous fuel and the high strength of the cladding, the fuel-cladding mechanical interaction component of cladding loading is a physical mechanism of the second order. All of the relevant mechanisms involved in predicting fuel failure within the bounds of the above scenario are being investigated.

The global aspects of fission gas release from metal fuels have been studied and characterized to the extent that models are available to predict fission gas release and plenum pressurization. Local effects along the fuel-pin axis are currently under investigation, as are local fission gas retention and swelling. Models for fuel swelling and radial versus axial fuel strain are being developed. The LIFE code modeling of these phenomena is supplemented by more detailed modeling in the STARS code. Although early indications were that axial fuel strain terminated after approximately 5-percent burnup, more recent evidence suggests that axial strain may continue to increase up to 10-percent burnup. This evidence has a strong influence on the control characteristics of the reactor system. Close attention should be paid to research in this area.

Significant fuel restructuring has been observed to occur as a result of irradiation. Zirconium has been observed to migrate in a manner that leaves a zirconium-depleted annular zone within the center of the fuel column. Since the fuel's melting characteristics are dependent upon alloy composition, this effect could strongly influence the fuel element design characteristics, both in terms of geometry and in terms of maximum operating power density. The mechanisms of migration within fuel elements are not understood at present. The experimental observations of migration are used as empirical input to the LIFE code at the present time in order to estimate the effects on the fuel element thermal and mechanical response. DOE plans additional research to investigate this phenomenon.

In addition to fuel restructuring, irradiation and fission gas release are observed to lead to the development of a nonuniform porous fuel structure with porosities observed in excess of 60 percent on a local basis. Although the mechanisms of fission gas release are understood and preliminary models are available, prediction of porosity distributions is beyond current technology. Models for porosity distribution are being developed and evaluated.

The effects of fuel restructuring and porosity have a strong influence on the thermal-mechanical behavior of the LMR fuel element. The local thermal conductivity of the fuel depends strongly on both the local elemental concentration and the local porosity. The behavior of bond sodium in possibly filling some of the porosity is an additional unknown and is currently under investigation. The creep properties of the fuel depend on the porosity as well, although local effects may not be important. Since, at present, neither phenomenon is well understood, the experimental observations are being used to specify the material and porosity distributions in calculations of the thermal response of the fuel element. The computed temperature distribution within the fuel element must be compared with the local alloy solidus temperature in order to make judgments about reasonable design limits for power density and possible modifications of fuel design. These areas are crucial for

future decisions regarding maximum operating power.

The current hypothesis is that the porous and spongy nature of the fuel leads to negligible fuel-cladding or mechanical interaction loading of the cladding; this area, however, requires additional research. Work at ANL will address the creep behavior of the porous fuel structure so that models may be verified for the LIFE code. Additional questions relate to possible fuel growth due to the presence of solid fission products and to the possibility that near the end of life, the porosity will be reduced by the solid fission products.

Cladding breach criteria are being developed for implementation into the LIFE code. Preliminary models have been incorporated in LIFE. The models are based on out-of-pile HT-9 tube burst data developed at HEDL. Little in-pile data relevant to the failure of HT-9 is currently available. Models developed will be verified against in-pile fuel-cladding performance data.

An extensive program of fuel-cladding compatibility experiments is being carried out to characterize the chemical exchange processes at the fuel-cladding interface. An out-of-pile facility is used to test fresh and irradiated fuel samples over a range of prototypic temperatures. The current data base for HT-9 is not extensive and additional data are anticipated. A correlation developed for the cladding penetration rate has been implemented in the LIFE code. At present, it is believed that the penetration rates are small enough to have no influence on fuel lifetime. Since this phenomenon has a potentially important impact on fuel lifetime, it is important to keep track of developments.

The IFR Fuel Performance Program of fuel assembly irradiations and posttest examinations supports the analytical model development program described above. A very strong program has been carried out and is planned for the future in the EBR-II, which will be converted to the ternary fuel. Additional experiments are currently being carried out in the FFTF using experimental metal-fuel assemblies. Consideration is also being given to eventually utilizing metallic fuel elements in FFTF, although this appears to be the binary U-Zr system for the present. Ternary fuels will be tested in the FFTF as experimental assemblies. A substantial data base will be developed over the next 5 years relevant to the behavior of metal-fuel systems. At present, most of the available in-pile ternary fuel irradiation data have been obtained using D-9 cladding. Data from HT-9 fuel-cladding systems are being obtained at the present time. One fuel assembly in EBR-II has successfully achieved 9-percent burnup in ternary and binary fuel clad with HT-9.

In general, the staff considers the planned IFR program of fuel performance irradiation to be an adequate means of compiling information on the proposed SAFR fuel. The list of planned experiments is extensive and involves irradiation in EBR-II, FFTF, and TREAT. It is believed that the program will provide a very substantial data base for the modeling efforts described above. One possible weakness in the program involves experimental verification of the SAFR designers' claim that no more than 0.1 percent of the pins in the (equilibrium) core will fail during normal and transient conditions. The FFTF irradiations will come the closest to prototypic SAFR conditions. The number of planned ternary fuel elements to be irradiated to provide an adequate data base for a statistical analysis remains an open item, along with how the differences in geometry and other relevant parameters will be accounted for in such a statistical analysis.

On the basis of this discussion, the staff believes that the research and development program currently in place by DOE is capable of providing the relevant analytical tools and supporting data base to support the SAFR design. Although uncertainties exist in many of the areas described above, there appear, at present, to be no major problem areas in the area of fuel performance that are likely to prevent ultimate acceptance of the basic ternary fuels concept.

Transient Fuel Response

Research is taking place to develop a set of computer codes to predict the behavior of LMR fuel subject to transient overpower and other transient events. The FPIN2 code is a detailed thermal-mechanical model of an individual fuel element used for analyses of fuel performance under transient

conditions. The code has been modified from earlier versions used to model oxide fuel. It is currently operational using the latest thermal-mechanical properties of the metal fuel-cladding system. It has undergone some verification through comparisons with the transient overpower M-series of TREAT experimental data. Supporting this computer code are the more detailed STARS code for steady-state initialization and FRAS3 for transient fission gas behavior. A less-detailed model of fuel element thermal-mechanical response is being developed in the DEFORM-5 fuel behavior model of the SASSYS whole-core response computer code. Both of these codes are designed to predict transient events to the time of cladding rupture. The codes will predict location and timing of fuel failure. However, neither contain the capability for ex-pin fuel motion modeling. A peer review of these codes may be necessary in future reviews.

Models are being developed, implemented, and tested in the transient response codes for cladding rupture based on the HEDL test data for HT-9. Fission gas retention and distributions are obtained from more-detailed STARS calculations and from experimental data. Fuel swelling rate is calculated using a model of gas-bubble growth. The creep behavior of the porous fuel is modeled using an empirically based correlation.

The fuel-cladding interaction rate at elevated temperatures is treated empirically using data obtained from both in-pile and out-of-pile experiments. Data from experiments using HT-9 samples are now being obtained. These data appear to be consistent with previous information.

The results from the FPIN2 and DEFORM-5 codes are being compared with the results of the M-series of TREAT experiments (M2-M7) performed using both fresh and irradiated fuel elements. Comparison of code calculations with the data is providing an understanding of the behavior of the EBR-II fuel elements using U-fissium fuel and, more recently, with the ternary fuel clad with both D9 and HT-9 (test M7) cladding. The comparisons of code results with the data have provided an understanding of the mechanisms of failure of these fuel elements during "slow" overpower transients. Experiments and analyses indicate that, under the "slow" overpower conditions of the experiments, the fuel pins fail near the top of the fuel column, at which point the molten fuel is released into the coolant channel. Experimental observation indicates that the molten fuel is swept downstream from the failure location, presumably under the action of the flowing sodium. Although the arguments appear plausible, they should be verified by experiments using fuel elements more prototypic of the SAFR fuel. Additional experiments should continue to be performed with higher burnup fuel.

The modeling and fuel failure arguments that have been applied to the M-series of overpower experiments should be tested with experiments using fuel elements that are closer to the SAFR design than are the EBR-II elements. This means that the fuel elements irradiated in the FFTF should eventually be tested to provide confidence in the models and in the interpretation of the results. Consideration is currently being given to such testing as part of the IFR program. Further experiments that simulate more rapid transients are not planned at the present time. This decision appears reasonable for rod-withdrawal accident simulation. Faster transients may, however, be necessary for simulation of loss-of-flow (LOF) accidents under conditions of large reactivity insertion due to sodium boiling.

The use of TREAT tests to determine the transient response appears to be acceptable. Future testing plans in TREAT are currently being developed. The need to investigate such factors as rapid eutectic formation and cladding penetration, fuel melting and motion characteristics, and more prototypic pin and transient tests is being considered. Modeling uncertainties remain; many have already been discussed here. Research is being performed to reduce the modeling uncertainties. Transient overpower data will be available to verify the modeling. However, the staff believes that additional experiments using fuel elements more prototypic of the SAFR design are appropriate. It should be noted that current analyses of transient events rely largely on system response simulations and eutectic formation versus time and temperature correlations. Detailed analyses of fuel-pin dynamics are not generally performed.

Additional Remarks

Experiments performed thus far using breached metal-fuel elements have indicated good compatibility of the sodium with the metal system. Little erosion of metal has been observed. Additional run-beyond-cladding-breach experiments are planned in EBR-II with the ternary fuel system. The data generated should allow assessment of the potential for problems related to erosion- or corrosion-product formation.

Future analyses to confirm that fuel coolability will be maintained under operational and anticipated transients will be developed from the research programs described above. Although under extended irradiation conditions, HT-9 is not expected to swell, the fuel-irradiation performance program should provide the experimental data necessary to establish whether coolant channel closure due to ballooning will be a problem. This will include issues related to ensuring control rod insertion.

4.2.5 Safety Issues

Although no major problem (not currently being addressed) has been found, much research is needed in order to establish the safety and reliability of the specific fuel concept to the burnups planned.

4.2.6 Conclusions

The SAFR fuel system, proposed by RI, U-Pu-Zr fuel clad with HT-9, is a new concept. Many of the basic design principles have been developed from EBR-II metal fuel experience. However, because of differences in material, geometry, and exposure conditions, this experience must be extrapolated to the SAFR design through the use of analytical tools that characterize the operational history and transient responses of the fuel system. Experimental data must be obtained, both to support the model development efforts and to verify the integrated computer codes.

At this stage of the design, the staff review was carried out with the limited objective of identifying potential problems in the fuel system design that could be ascertained at this early date and that had the potential to lead to major safety-related problems. A second objective was to reach a conclusion about whether the R&D program currently in place would lead to development of the experimental data base and analytical tools that will eventually be required to support licensing of the SAFR design.

Although no new major safety-related problems in the proposed SAFR fuel system design were identified, many phenomenological uncertainties must be resolved in order to provide a set of analytical tools and the supporting experimental data base necessary for licensing. These include

- The uniformity of quality; that is, composition, thermophysical properties, and strength characteristics, resulting from production and fabrication technologies for the fuel and cladding, need to be established.
- Behavior of and the extent of fuel restructuring and porosity characteristics as a function of burnup need to be confirmed. Although first-order estimates of their effects based on experimental evidence have been made, data are needed for burnups that exceed 10 percent.
- The behavior of prototypic fuel at high burnup ($\geq 10\%$) in prototypic geometry needs to be established. Concerns are the closure of porosity due to solid fuel swelling and, perhaps, the effect of the weight of the fuel column.
- The statistical data base to support the claim of no more than 0.1 percent fuel failures needs to be developed using ternary fuel of prototypic geometry.

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- The conclusions drawn with respect to the behavior of the SAFR fuel system under "slow" overpower transients must yet be verified in experiments with fuel elements of prototypic geometry.
- The run beyond cladding breach and the potential for fuel failure propagation need to be experimentally established.

In general, the staff finds that DOE has in place the programs related to fuel-system characterization, both operational and transient, that should lead toward resolution of the technological uncertainties and development of the appropriate analytical tools.

4.3 Nuclear Design

4.3.1 Design Description and Safety Objective

The SAFR reactor will utilize metallic fuel, and will generate 900 MWt. The heterogeneous core will have the common LMR configuration, that is, wire wraps on fuel rods, which are grouped in hexagonal assembly cans. The driver fuel assemblies will have a 48-month life and an average burnup of 102 MWd per kilogram (maximum of 148 MWd/kg). About one-fourth of the core will be replaced during the annual refueling. The heterogeneous core layout will have a breeding ratio of about 1.1 and will use the one ring of radial blankets to produce enough plutonium to refuel the core in the equilibrium cycle. Only one fuel enrichment zone will be used. The burnup reactivity swing over the fuel cycle is predicted to be about \$0.37. The power fraction produced in each region is predicted to go from 72 percent to 65 percent from the beginning-of-equilibrium cycle to the end-of-equilibrium cycle in the driver fuel, while the inner blanket will shift from 10 percent to 16 percent. This can be compared to the radial blankets which shift power fractions from 17 percent to 18 percent during the same period. The power generation will be shifted to the blankets during a fuel cycle, since fuel is being bred as fissile plutonium from the depleted uranium in the blankets. This phenomenon will allow the design to have a minimal reactivity swing during a fuel cycle, which will permit operation with almost fully withdrawn control rods. Such operation will limit the amount of positive reactivity available for insertion in a rod-withdrawal accident and will contribute to the passive safety characteristics of SAFR.

RI and ANL evaluated the nuclear design of the core using codes that include the three-dimensional diffusion theory code DIF-3D and its associated burnup code REBUS-3. The nuclear coefficients were generated with a perturbation theory code, VARI3D.

The design objectives are primarily related to achieving passive safety through the "inherent shutdown" and minimal burnup reactivity swings (to minimize possible transient overpower (TOP) initiators). The breeding of plutonium is not an explicit objective of the DOE LMR program.

4.3.2 Scope of Review

The review in Section 4.3 focused on the reactor design calculations, reactivity feedback estimates, and the uncertainties associated with each. The extensive package of computer codes used by ANL is fairly standard in scope and methods. A complete, independent calculation of reactor characteristics was impractical during the current review, but such an effort must be undertaken in future evaluations of the SAFR design. The ANL calculational process was discussed with the staff. The conclusion was reached that the ANL calculations are credible, but the ANL estimates of the uncertainties may be low. The uncertainties in calculations for a metal-fuel core should be relatively low, as the very hard neutronic spectrum reduces problems often caused by the resonances in the cross-sections.

Since the "inherent shutdown" is based on reactivity feedbacks, it is important to properly estimate the values of key feedbacks, such as radial expansion, axial expansion, Doppler, sodium density, and

control rod drive-line expansion. Simplified calculations were used to verify the radial expansion and control rod drive-line expansion, axial expansion, and Doppler feedbacks as being reasonable. Sodium density was the most difficult feedback to estimate. However, judging from other LMR designs, the overall sodium void reactivity worth appears to be in the right range.

4.3.3 Design Criteria -- GDC 11, 12, 25, 26, 27, 28, and 29

10 CFR Part 50, Appendix A (GDC 11 and 12), requires that the reactor core be designed so that, in the power operating range, the net reactivity nuclear feedback characteristics will compensate in the event of a reactivity insertion and that there will be no power oscillations that can result in exceeding fuel design limits.

The requirements of 10 CFR Part 50, Appendix A (GDC 25-29), were used to assess the diverse control rod insertion methods, control rod withdrawal limitations, and use of passive feedback effects to provide a diverse means of shutdown.

4.3.4 Research and Development Programs

The major research effort for this reactor design is the metal-fuel IFR program, which is discussed in Section 4.2 of both the PSID (Ref. 4.1) and this SER. Part of this effort includes the HT-9 cladding material, which has performed very well in the FFTF (with oxide fuel), but is only now being tested in combination with the metal fuel. Additional R&D is required, using a critical facility to confirm such nuclear characteristics as power distribution, control rod worth, and reactivity feedback.

4.3.5 Safety Issue

The positive sodium void coefficient results in certain EC-III events having the potential to lead to positive reactivity insertion events (see Chapter 15). The staff is concerned about the positive sodium void reactivity coefficient, and efforts should be made to reduce its magnitude, as much as practical, even if the likelihood of sodium boiling is reduced to the point that no events which could lead to sodium boiling are in EC I-III. Efforts in this regard should be reviewed at a later design stage.

4.3.6 Conclusions

The SAFR reactor design appears to be at an appropriate level of maturity, given the status of the metal-fuel program and other R&D. Although there are uncertainties associated with the "reactivity feedbacks," it appears that the RI and ANL projections are plausible, although perhaps slightly optimistic. The approach to the core design and its supporting R&D is generally acceptable. The results of further in-reactor and criticality experiments should be followed closely and will be required to support final acceptance.

4.4 Thermal and Hydraulic Design

4.4.1 Design Description and Safety Description

The SAFR reactor is orificed so as to provide extra cooling to the hotter channels; this results in relatively flat core outlet temperatures, particularly at the end of reactor life. The 11 orifice zones include 5 drivers, 3 internal blankets, and 3 radial blankets.

Core inlet nozzles (at the bottom of the assemblies) are designed so that it is nearly impossible to block all flow to any given channel. Entrance holes all around the nozzle accomplish this objective. However, no in-core instrumentation is provided in the design to measure fuel or blanket assembly temperature for flow and thus confirm no blockage (as from a fabrication error).

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The primary system is designed to promote the natural circulation of sodium (in both the primary and intermediate systems). This gives the system the ability to remove decay heat in the absence of forced flow.

4.4.2 Scope of Review

The staff evaluated drawings of the core design, the inlet region and orificing, and the assembly nozzles and discussed the evaluation with RI and ANL. The staff performed independent calculations to assess the ability of SAFR to remove decay heat under natural circulation conditions.

4.4.3 Design Criteria

GDC 10 states that fuel design limits must not be exceeded. The equivalent SAFR criterion is PDC 2.1. In addition, it is important that a SAFR plant maintain fuel geometry for a range of transients (EC-II and III) to support the proposed containment design. This includes preventing flow blockages and sodium boiling. Also, the ability to remove decay heat via natural circulation is key to the safety of the SAFR design.

4.4.4 Research and Development Program

The ultimate testing of the orificing and the thermal-hydraulic design of the core will take place during the safety tests performed on the first module. Additional data regarding fuel response and burnup will come from the IFR program.

4.4.5 Safety Issues

Although natural circulation cooling is generally available in LMR systems, the adequacy must be confirmed for the SAFR design. This is an important issue, but it appears that natural circulation cooling will work well in the SAFR design and can be tested during prototype testing (Chapter 14). It must also be shown that the combination of inlet orificing and absorber bundle weight are such that startup of the primary system pumps during refueling will not result in absorber bundle ejection or flotation. Finally, since there is no in-core fuel or blanket assembly temperature or flow instrumentation, it is possible that an in-core flow blockage (as from a fabrication error) could go undetected until fuel damage, failure propagation, and/or a reactivity insertion has taken place.

4.4.6 Conclusions

The analyses done by the staff and the designer indicate the potential to choose thermal-hydraulic characteristics for the SAFR to achieve satisfactory cooling of the core, including conditions of natural circulation. Features have been added to prevent in-core flow blockage consistent with those used on the FFTF and the Clinch River breeder reactor (CRBR); however, the prevention and detection of assembly flow blockages due to fabrication errors remains an open item. Under most transient conditions, sufficient margin to sodium boiling is provided; however, some SAFR bounding events (see Chapter 15 and Appendix A) could lead to sodium boiling and these issues need to be resolved.

4.5 Reactor Materials

4.5.1 Structural Materials of Control Rod Drive System

Type 316 stainless steel and Inconel 718 are specified. Chromium-molybdenum (Cr-Mo) and nickel-iron (Ni-Fe) alloys are used for the magnetic and temperature-sensitive materials for the self-actuating shutdown system (SASS) electromagnet. The pressure-housing material, which forms the reactor containment boundary, uses Type 304 stainless steel.

4.5.2 Components of Reactor Core

The fuel and blanket assembly structural material is HT-9, a ferritic alloy. Control and shutdown assemblies use D-9, a stainless steel material. Removable shield assemblies will be constructed of Type 316 stainless steel, or possibly D-9.

4.5.3 Design Criteria

The materials of construction for the primary reactivity control and shutdown system and the secondary SASS are selected to satisfy the performance requirements for the duty cycle service and for the expected operating environmental conditions, such as temperature, neutron fluence, and the liquid-sodium medium.

The materials to be used to construct the core components will meet the performance requirements under the environmental effects of neutron fluence and the liquid-sodium environment.

4.5.4 Research and Development Program

A preliminary irradiation test has been performed to determine the irradiation effects on the temperature-sensitive magnetic material. Further testing is planned to demonstrate the integral performance of a SASS magnetic latch in an in-pile sodium environment. The magnetic coil resistance, resistance to ground, and holding force as a function of temperature will be measured. A full-scale water test of a typical SASS assembly with a jet pump is also planned.

The testing priorities for safety and licensing needs for all materials include the separate and combined effects of neutron and thermal aging on stainless steel welds, the use of Inconel 718 and hardfacing for sliding seals, and methods for inspecting 100 percent of the welds of all components critical to safety during and, for closure welds, after assembly of the reactor vessel, containment vessel, and internal components.

4.5.5 Safety Issues

The R&D program will be followed by out-of-pile, functional, and reliability-oriented margin tests in sodium to establish that SASS can satisfy the stringent reliability requirements of the secondary shutdown system.

The design, including complex penetrations with seals that are expected to operate within a sodium environment, contains a number of features, the operability of which has not yet been adequately demonstrated. Development studies and detailed engineering evaluations of many of these features need to be performed to ensure that the reactor vessel, containment vessel, internal structures, and redan assembly (see Section 5.3.1) maintain this integrity under the thermal and radiation environments to which they will be exposed.

4.5.6 Conclusions

The SAFR reactor materials and reactor core components are unique to this design and require demonstration that the specific components will withstand the operating environment of an LMR.

4.6 Reactivity Control and Shutdown Systems

4.6.1 Design Description and Safety Objectives

Two automatic shutdown (scram) systems are utilized in the SAFR design. Neither was proposed to be safety grade since RI considers the passive reactivity feedback characteristics as sufficient to perform the safety function of reactor shutdown. The automatic plant trip system (APTS) can drive

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in all six of the primary control rods, which have a net worth about 3.6 percent Δk (i.e., \$10.2), and can interrupt power to the electromagnetic latch and drop the six primary rods plus the three secondary control rods into the core, with an additional net worth about 2.4 percent Δk (i.e. \$6.7). The three secondary rods can be dropped in by the self-actuated shutdown system (SASS). SASS is based on a temperature-sensitive Curie point magnetic material that loses its magnetic properties at a higher-than-normal operating temperature (i.e., 1050°F) and thereby releases the secondary safety rods whenever the core outlet temperature exceeds that temperature. Jet pumps are utilized to divert hot sodium from adjacent fuel channels to the magnetic material in order to ensure a timely response from the SASS. Thus, the three secondary control rods are dropped in whenever either the APTS interrupts the power to the latch or the SASS Curie point magnet overheats significantly.

The objective of these two shutdown systems is to provide a diverse means of reactor shutdown whenever a signal is generated by the APTS or whenever sodium outlet temperature increased to the Curie point. The minimum number of primary control rods needed for a successful reactor shutdown is two out of the six. This will insert $2 \times 0.53\% = 1.06$ -percent Δk (about \$3.00), which will overcome the temperature defect (i.e., \$1.7) plus a \$1.00 shutdown margin. The secondary rods need only one rod to enter the core in order to insert \$2.2 worth of reactivity in order to go to hot standby (675°F/630 K). Although it is not a stated objective, the reactivity worth of the nine control rods is considerably larger than the positive sodium void worth. With all nine control rods inserted, the SAFR reactor would be in a subcritical state even if there were no sodium in the reactor.

4.6.2 Scope of Review

The staff discussed the SAFR reactor shutdown systems at length with RI, DOE, and ANL, although the reactivity worths of the control rods were not independently verified. Areas concentrated on in the review were

- uncertainties in control rod worth
- reliability of jet pump during LOF
- maximum rate of control rod withdrawal
- testing of the SASS

Two questions were also raised in the review:

- Have chemical shutdown systems been considered?
- Should the active shutdown systems be safety grade?

4.6.3 Design Criteria

The need for two diverse, independent means of reactor shutdown is discussed in Section 3.3 of this SER. This criterion is equivalent to GDC 26 and SAFR criterion PDC 3.7.

4.6.4 Research and Development Program

Westinghouse has tested the SASS, measuring jet pump efficiencies at various flow rates. The tests show that the jet pump efficiency remains high, even at flows as low as 15 percent of nominal. Thus, in a LOF event, the jet pump should function as intended long enough to ensure SASS actuation. Programs to measure radiation damage to the Curie point magnet are planned or are being planned. A program needs to be developed to show that the jet pump will be able to extract exiting sodium from adjacent fuel and blanket assemblies. Also, a method needs to be developed to test the SASS at regular intervals before SASS can be used with confidence in a reactor system. Such tests, which should be performed on the initial SAFR test module, will be difficult because such high temperatures are required for SASS actuation; they are, however, crucial to establishing the reliability of the SASS.

4.6.5 Safety Issues

Use of Passive Shutdown Characteristics

RI argues that the reactivity feedback and the inherent safety features reduce the need for a scram, thus allowing use of non-safety-grade shutdown systems. However, the large positive sodium void worth increases the risk for some events with a failure to scram, such as for a fast reduction in flow to the reactor (see also Section 4.3). The staff believes that SAFR needs at least one reliable, safety-grade scram system to reduce the potential for ATWS events. If the passive response of SAFR under ATWS conditions cannot eliminate sodium boiling and subsequent fuel melting from the positive void coefficient, a second diverse safety-grade shutdown system may be necessary.

Reliability of the SASS

The SASS provides excellent safety advantages for the SAFR plant, but some questions remain. First, the jet pump must be able to extract enough hot sodium from neighboring fuel and blanket assemblies. Second, the jet pumps become less efficient below 15 percent of full flow, and after that, the hottest sodium would not be directed along the magnet. Third, at low flows, the buoyancy of the hot sodium becomes a significant force. During a flow coastdown event (both primary pumps lose power at time zero), the flow reaches 15 percent at 34 seconds. If the SASS does not actuate by that time, it may not function for a long period (while the control assemblies slowly heat up). Finally, the HT-9 cladding and other reactor system materials are ferritic, so particulate material removed by the sodium will tend to collect at the SASS magnets. RI has claimed that such an accumulation would only degrade magnet performance and cause spurious scrams. RI considers it incredible that ferritic materials could build up sufficient to block flow.

Testability of the SASS

A significant problem with the SASS is that there is no routine means of testing the system Curie point in place. The closest concept discussed to date is to provide for electrical heating of the diverted sodium to expose the magnet to artificially heated sodium. However, no definitive proposal has been made yet.

4.6.6 Conclusions

The acceptability of the reactivity and control system design for SAFR remains an open item. In light of the potential reactivity insertion available from sodium voiding, the reliability of this design is of great importance. It is the staff's view that because of the large positive sodium void worth, the SAFR design needs highly reliable scram systems. The APTS or the SASS or both should be safety grade. However, the staff is not convinced of the potential for the SASS to be qualified as a safety-grade shutdown system.

4.7 Passive Safety Characteristics

4.7.1 Design Description and Safety Objectives

"Passive safety" refers to the reactor design features that will cause the reactor to reduce its power to a lower level during overheated conditions due to a negative temperature coefficient of reactivity. When the core exceeds its normal operating temperature (e.g., the steam generator fails to remove heat), the primary sodium system gradually heats up, which results in the reactor power level decreasing without any input from control systems or operator actions. (Such a reactor response is sometimes referred to as "inherent shutdown," but this term can cause confusion. The reactor does not move to the subcritical condition, it simply moves to a much reduced power level. It appears to be shut down in terms of heat production, but it is still in the critical state.) This passive feature is

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composed of several reactivity feedback characteristics. The main components of this feedback are discussed next.

Doppler: As the fuel temperature increases, more neutrons are parasitically absorbed in the resonance energy range. For metal fuel, Doppler feedback is a smaller negative factor than it is for oxide fuel because of the harder energy spectrum and lower fuel temperature. Doppler feedback still adds negative reactivity on a power increase, but its effect is less in metal fuel. This allows the temperature defect to be small (\$1.70), which means the reactor can be controlled by other natural feedbacks (i.e., axial and radial expansion).

Sodium Density/Void: For a small liquid-metal-cooled reactor such as EBR-II, this is a negative feedback, and is helpful. For the larger SAFR reactor, this will be a positive feedback. As long as the sodium is subcooled, the positive reactivity contribution is small. If the sodium boils, this feedback becomes large and prompt (around \$5 of reactivity within a few seconds).

Axial Fuel Expansion: Metal fuel expands significantly when it heats up. Axial expansion within the cladding increases the core size and decreases the effective density of the core materials. This increases the probability that neutrons will escape from the core, creating a significant negative reactivity feedback. The size of this feedback changes after about 2-percent burnup, when the fuel swells into contact with the cladding. The axial expansion is then controlled by the expansion rate of the cladding, since metal fuel has little strength.

Radial Expansion: The radial dimension of the core is determined largely by the 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, they spread the reactor and reduce the core density, which increases leakage and thereby reduces the net reactivity.

Bowing: When a fuel or blanket assembly is heated more on one side than the other, the heated side will expand more than the other, and the center of the assembly will bow to the hotter direction. This type of behavior will occur in the SAFR fuel and blanket assemblies. It has some reactivity contribution, but it is difficult to accurately calculate. The SAFR design uses a limited free bow-restraint system, which limits the importance of bowing and makes the contribution negative under conditions of interest.

Control Rod Drive Line Expansion: The control rod drive lines, which are in the upper internal structure, expand when they are heated. This will insert the control rods farther into the core and will add negative reactivity.

Reactor Vessel Expansion: Since the control rod drives are attached to the top of the vessel and the reactor attaches to a point much lower along the vessel wall, the expansion of the vessel wall as it heats up will pull the control rods out. This is a positive feedback, but it is not a major safety factor because it is quite slow to act.

4.7.2 Scope of Review

4.7.2.1 The Reactivity Feedbacks as Specified by RI

Independent analysis of the reactivity feedbacks has been limited to scoping calculations, and comparison against values for similar design. A simple calculation gave an estimate of the radial expansion feedback within 5 percent of that cited by RI, and extrapolation to axial expansion is not difficult. All other feedbacks are clearly within reasonable ranges. As a result of this review, the staff considers the feedbacks cited by RI to be achievable, although they contain uncertainties which at this time appear to be in the 10-25 percent range.

The key reactivity feedback parameters estimated by RI for the SAFR design have been compared to the equivalent feedbacks for PRISM, Super Phenix, EBR-II, and FFTF in Table 4.2 (The EBR-II and FFTF data were measured, that is, estimated from experimental data as shown in the table.) PRISM and SAFR have strong similarities, EBR-II is much smaller, and FFTF and Super Phenix use oxide fuel. According to Hummel and Okrent (Reactivity Coefficients in Large Fast Power Reactors, American Nuclear Society, 1978), the Doppler feedback for an oxide core should be about three times larger than that for a metal core. Sodium density worth depends largely on core geometry (leakage). Regarding radial and axial expansion, there are strong similarities for all five reactors, as shown in the table.

Because of the consistency in the various feedback calculations, it appears likely that the values cited by RI are approximately correct. However, the fact that EBR-II is obviously quite different from the other cores decreases one's confidence in extrapolating from the EBR-II test series. Analyses consistently indicate that the "passive shutdown" will work as designed in the SAFR, but a series of safety tests using a prototype reactor is needed for confirmation.

4.7.2.2 The Transient Calculations

As discussed in Appendix C to this SER, the RI calculations for TOP, LOF, and loss of heat sink (LOHS) have now been independently assessed. The staff believes the RI calculations are a best estimate of the likely SAFR response to the postulated challenges. However, significant uncertainties exist, particularly for the fast-flow-reduction events (see also Appendix C).

4.7.2.3 Failure of the Passive Shutdown Characteristics

For some postulated events in EC-III, the passive shutdown characteristics may fail to prevent fuel damage and sodium boiling. These events generally involve fast reductions in sodium flow to the reactor (i.e., reduction or loss of pump coastdown), as described in Chapter 15 of the PSID.

4.7.3 Design Criteria

RI designed the SAFR to passively achieve reactor power runback (shutdown) in response to reductions in both heat removal and reactor sodium flow rate. Further, RI also planned to design the SAFR to accommodate the withdrawal of all the control rods. (This will cause a power increase, until the reactivity feedbacks caused the power to level off.)

4.7.4 Research and Development Program

The DOE and RI approach for the SAFR is to work with an operating utility to build and license an initial SAFR plant. A series of transients would be executed to test the passive response of the reactor. This is discussed in Chapter 14 of the PSID (Ref. 4.1). Given the uncertainties in the reactivity feedbacks and the degree to which these feedbacks are dependent on the design of the reactor, this is not the preferred approach. The staff may require that responses to the unscrammed events be tested in a prototype facility.

4.7.5 Safety Issues

Is It a Shutdown System?

Because the power runback that generally develops when the SAFR overheats leaves the reactor in a critical state, it is not a true shutdown. However, the reduced power does maintain the reactor in a coolable condition with little or no core damage. Therefore, the staff cannot accept passive shutdown characteristics as a diverse, independent means of reactor shutdown for SAFR in the long term (see Section 7.2 of this SER).

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Is the Response Predictably Safe?

For the SAFR, the response to loss of flow, loss of heat sink, and transient overpower events appears to be acceptable, although the safety tests will be needed for confirmation. However, for events in which a loss of pump coastdown causes a reduction in flow, sodium boiling could occur. Prevention or mitigation of this event (or both) needs further study.

The Positive Void Worth

The positive sodium void worth is a concern in the passive safety argument. Because of it, one must qualify any characterization of the SAFR response as "passively safe" by pointing out that this is conditional on the sodium remaining below the boiling temperature. Should sodium boiling begin on a corewide basis under failure-to-scrum conditions, the reactor would be likely to experience a severe power excursion. Note, however, that sodium boiling appears to be extremely unlikely. Certain events analyzed for the SAFR have the potential to lead to sodium boiling and need further study before the acceptability of the SAFR design can be determined. These are discussed in Chapter 15 of this SER.

4.7.6 Conclusions

The passive response of the SAFR reactor is not a true reactor shutdown mechanism, but it does accomplish the essential function by reducing the power generation to a level at which heat removal with little or no core damage is possible. If it can predictably reduce the power to manageable levels in response to all EC-I through III events, the passive shutdown could be accepted as a diverse means of reactor shutdown, in the short term though a need to provide a positive means of shutdown for the longer term may still be required.

Analysis of the SAFR reactor response indicates that a safe response can be anticipated for most events. Proof would come from the safety tests performed in a prototype reactor.

4.8 References

- 4.1 DOE, "SAFR - Sodium Advanced Fast Reactor," Preliminary Safety Information Document, AI-DOE-13527, Initial Issue - October 1985, with amendments. (DOE Applied Technology.)
- 4.2 NRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," NUREG-0800, July 1982.

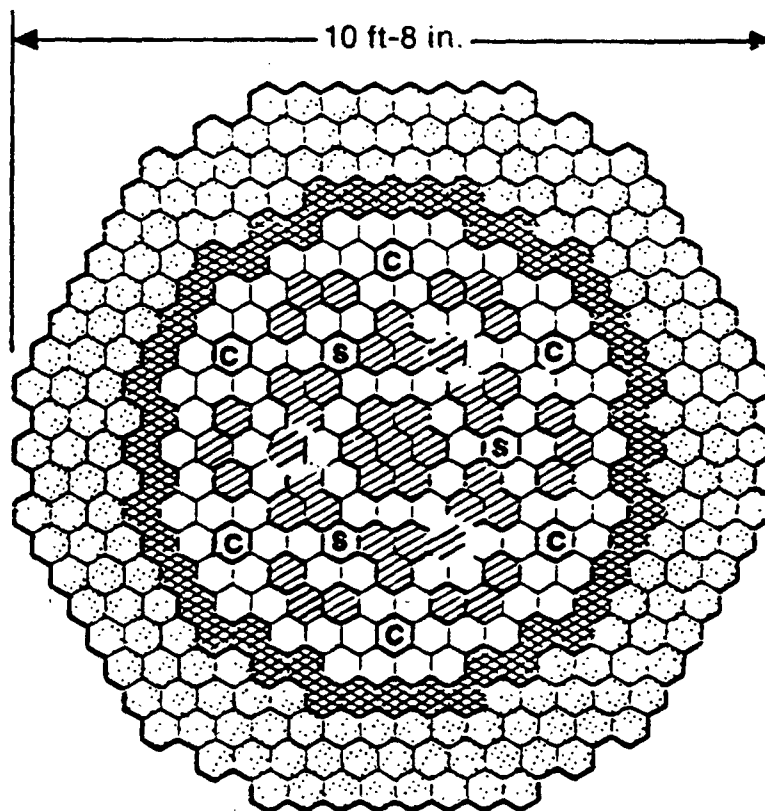
Table 4.1 Reactor design parameters

Item	Measurement
Core thermal power	900 MWt
Reactor mixed mean outlet temperature	950 °F
Reactor ΔT	275 °F
Core height	38 in.
Number of core enrichment zones	1
Refueling interval	365 days

Table 4.2 Reactivity feedbacks [$\Delta k/\Delta T(K)$], $\times 10^{-6}$

(Referenced to nominal conditions)

Feedback	PRISM	SAFR	EBR-II	FFTF	SuPhx
Doppler	-6.1	-4.2	-0.4	-14.6	-12.0
Sodium density	6.7	5.9	-8.7	-0.7	6.0
Radial expansion	-6.9	-9.7	-9.3	-22.0	-10.0
Axial expansion	-2.7	-2.9	-4.8	-1.8	-2.0



	DRIVER	96
	INTERNAL BLANKET	46
	RADIAL BLANKET	48
	CONTROL	6
	SAFETY	3
	REMOVABLE SHIELDS	
	STEEL	54
	B ₄ C	126
		379

Figure 4.1 SAFR core layout.

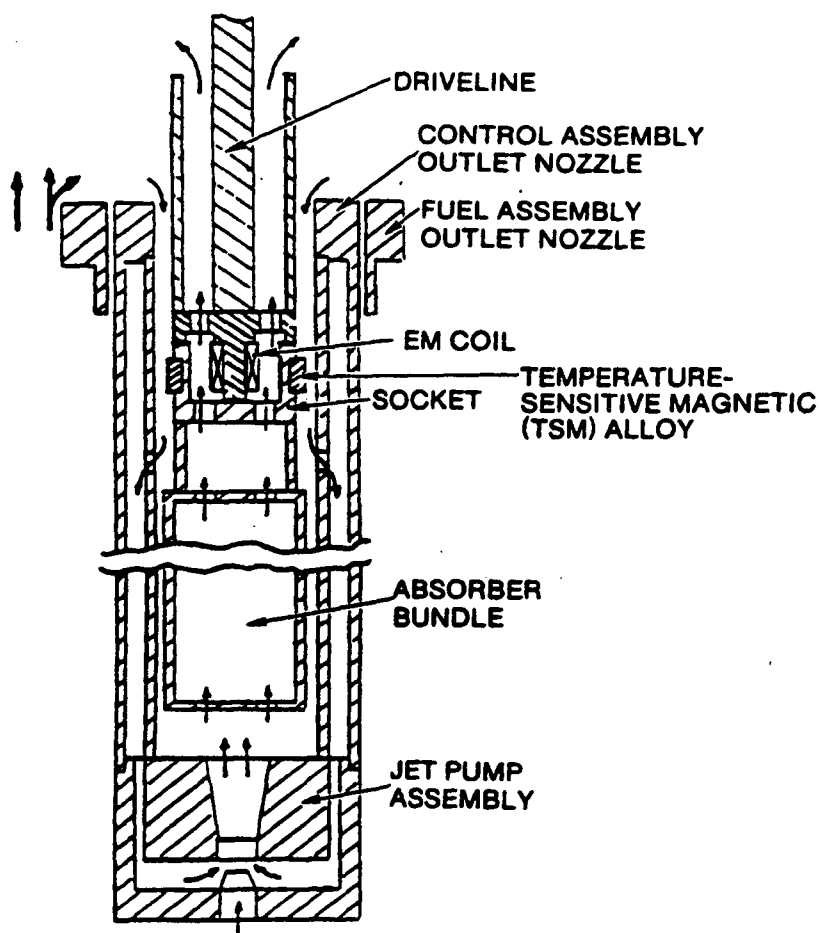


Figure 4.2 Temperature-sensitivity electromagnetic jet pump.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

In Section 5.1, the reactor enclosure system, the heat transport and steam generator systems, and the residual heat removal systems are described. The key aspects of these systems and the staff's review are discussed in Sections 5.2 through 5.6.

5.1 Design Description and Safety Objectives

5.1.1 Reactor Enclosure System

The reactor enclosure system will provide support and serve also as the primary containment for the reactor system (core assemblies and control rod drives), the primary heat transport system (PHTS), system components (primary pumps and intermediate heat exchangers (IHXs)), and the primary coolant (sodium) and inert cover gas system volume. The major elements of the system are the reactor vessel (RV) and the containment vessel, the reactor closure (consisting of a fixed annular deck and rotatable plug), and the reactor internal elements that include nonremovable components within the reactor vessel.

5.1.2 Power Pak Heat Transport System

The overall heat transport and steam generator systems are integrated into an independent power-generating unit called a "power pak." A SAFR plant will consist of one to four power paks. The heart of each power pak will be a pool-type, sodium-cooled reactor. Two primary pumps will circulate primary sodium within the reactor vessel and four IHXs will transfer reactor thermal energy to two independent IHTS loops. Superheated steam will be generated by two steam generators (one per intermediate loop). Heat will be removed from the reactor by the PHTS, transferred to the nonradioactive sodium intermediate heat transport system (IHTS) via the IHX, and then transferred to the steam/feedwater system via the steam generators.

5.1.3 Residual Heat Removal

Following normal plant shutdown, residual heat will be removed from the reactor by means of the steam generators. If the normal heat removal path is unavailable, the residual heat will be removed by the shutdown heat removal system (SHRS). The SHRS will consist of two independent systems: the non-safety-grade direct reactor auxiliary cooling system (DRACS) and the safety-grade reactor air cooling system (RACS).

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Design Description and Safety Objectives

In this section, the SAFR coolant pressure boundary is described and the criteria and methods used to ensure structural integrity and leak-tightness over the plant service life are discussed.

Reactor Coolant Boundary

The reactor coolant boundary comprises those components that form a leak-tight barrier against the release or leakage of radioactive primary coolant. For the SAFR plant, those components will be

- the sodium-containing surfaces of the reactor vessel
- the passive barrier within the IHXs that separate primary (radioactive) sodium and intermediate (nonradioactive) sodium

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- the passive barrier within the direct reactor heat exchanger (DRHX) that separates primary sodium and the nonradioactive sodium in the DRACS loop
- the passive barrier of the in-vessel primary cold trap that separates primary sodium from nonradioactive helium coolant (primary sodium connections to the cold trap that extend above the reactor closure are also considered part of the primary coolant boundary up to the first positive mechanical seal (blind flange or weld cap) or the second of two normally closed (or automatically isolable) valves.

Compliance With Codes

All components of the reactor coolant boundary will be constructed in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code. Consistent with its function of preventing loss of coolant inventory, the reactor vessel will be constructed in accordance with the requirements for Code Class 1 components. RI indicated that a coolant boundary failure within the IHXs, DRHX, or primary cold trap will not affect decay heat removal capability. Therefore, these components will be constructed as Code Class 2 components.

Overpressure Protection

The reactor vessel and primary side of the balance of the reactor coolant boundary are designed for internal pressure caused by the static head of the contained sodium, with a design cover-gas pressure of 14.9 psia. The cover-gas pressure is maintained essentially constant at 6 inches of water during steady-state operation by the inert gas system, and varies within acceptable limits during temperature transients caused by normal and off-normal operating events. The cover gas system includes pressure-relief devices and rupture disks, as appropriate, to protect against both overpressurization and depressurization of the cover gas. The pressure-relief provisions will be designed to satisfy the requirements of Section III of the ASME Code for Class 1 components.

Those portions of the heat transport system that contain nonradioactive sodium will be protected from overpressure by pressure-relief devices on the expansion tanks in the IHTS, DRACS, and cold-trap cooling circuits. The principal challenge to the primary coolant boundary from external sources will be the pressure pulse within the IHX that could result from tube ruptures in the steam generator.

Reactor Coolant Boundary Materials

The reactor coolant boundary materials will be

- | | |
|---------------------|-------------------------------|
| • Reactor vessel | Type 316 stainless steel (SS) |
| • IHX | modified 9Cr-1Mo |
| • Primary pump | 300-series SS |
| • DRHX | 300-series SS |
| • Primary cold trap | 300-series SS |

In-Service Inspection of Reactor Coolant Boundary

An in-service inspection (ISI) program for the reactor coolant boundary and its support structure will be developed in accordance with the requirements of Section XI, Division 3 of the ASME Code. The principal features of the program involve (1) continuous monitoring of the sealed, helium-filled space between the reactor vessel and its surrounding containment vessel by the sodium-to-gas leak-detection system and (2) visual inspection of vessel welds by a remote track-mounted television camera that will travel within the reactor vessel/containment vessel annular space. ISI of the primary coolant boundary within the heat exchangers and primary cold trap will consist of monitoring levels in the expansion tanks of the external sodium circuits for abnormal levels or level trends.

Detection of Leakage Through the Reactor Coolant Boundary

Any leak in the reactor coolant boundary will be detected by the sodium-to-gas leak-detection system. This system will continuously monitor the helium atmosphere within the reactor vessel/containment vessel annulus for the presence of sodium aerosol. Abnormal expansion tank levels that would result from in-leakage of nonradioactive sodium from the intermediate loop into the reactor vessel (by way of the IHX) will also be monitored.

5.2.2 Scope of Review

The review focused on two major issues associated with the reactor coolant pressure boundary:

- Would a leak (or larger failure) in the reactor vessel lead to unacceptable consequences?
- Is the IHX vulnerable to sodium-water reactions resulting from possible steam generator tube ruptures?

5.2.3 Design Criteria

SAFR criteria PDC 2.5 and 2.6, which are consistent with GDC 14 and 15, apply. In particular, GDC 15 states that the primary system boundary must be protected from challenge. This includes the IHX, which must serve as the primary coolant boundary.

5.2.4 Research and Development Program

In-service inspection (ISI) is likely to require development, particularly in the area of inspecting the reactor vessel for leaks. The small cameras needed for some of this work are available, but some development work is required in order to track these units in the gap between the vessels. Such inspection could be complicated in the SAFR. The reactor vessel and containment vessel will be constructed of steel plates welded together, thus requiring camera transport in different directions.

5.2.5 Safety Issues

Two of the reactor coolant pressure boundary surfaces have associated safety issues that are considered significant:

- Is a reactor vessel failure a major concern?
- Are major IHX tube failures likely?

In the initial SAFR design, the 1-foot gap between the reactor and containment vessel facilitated ISI. Because this gap was so large, a leak in the reactor vessel could result in the sodium level (within the reactor vessel) dropping below the IHX inlet ports, leaving the DRACS pathway (through the gas valve which must first open) as the only direct means of transporting hot pool sodium to the cold pool. When this was pointed out as a potential problem and the need for such a large gap was questioned, RI determined that a gap of approximately 7 inches would suffice for ISI. With the smaller gap, a reactor vessel leak would result in a smaller drop in sodium level, which would leave the IHX inlet ports covered in the sodium pool.

The IHX vulnerability appears to be a more difficult problem. Intermediate sodium would pass on the shell side of the IHX tubes, exposing the tube support plates to any pressure pulses coming from the steam generator. RI stated that the IHX is designed to 300 psi (but may survive pulses in the 540 to 880-psi range), and that this should be sufficient to withstand challenge from steam generator tube rupture events. However, the SAFR steam system will operate at nearly 2800 psi, and will utilize single-wall tubes. This presents the potential for damaging sodium-water reactions.

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Protection against such events will be provided by the sodium-water reaction pressure relief system (SWRPRS) and rupture disks on the intermediate loop side. The RI analysis (using the computer code SWAAM) of severe steam generator tube rupture events indicates that pressure pulses at the IHX are acceptable, that is, they will be in the 300-psi range. For this review, no audit calculations were performed to confirm this response. However, if future LMR designs contain apparent vulnerabilities to this class of events, then independent staff analyses will be necessary. In addition, it is suggested that consideration be given to redesigning the IHX to have intermediate sodium on the tube side to provide better structural capability to accommodate sodium-water reaction pressure pulses.

Catastrophic Reactor Vessel Failures

Despite the ISI and monitoring programs outlined by RI for the SAFR, there is at least some possibility that the reactor vessel will leak. However, because the containment vessel appears to serve as an adequate guard vessel, the concerns over this unlikely class of events are probably reduced to a level at which they can be excluded from Event Category EC-I through III (see Section 1.5 for description of event categories).

5.2.6 Conclusions

In general, RI has adequately addressed many areas affecting reactor coolant pressure boundary integrity. However, the combination of high-pressure steam system, single-wall steam generator tubes, and intermediate sodium on the shell side of the IHX (exposed IHX tube support plates) leads to significant concerns about sodium-water reactions, the magnitude of any resultant pressure pulses, and IHX integrity.

5.3 Reactor Vessel and Internal Components/Closure/Containment Vessel

5.3.1 Design Description

Reactor Vessel and Material

The reactor vessel (see Figure 1.6) is 39 feet in diameter, 47 feet, 6 inches long, and will be fabricated from Type 316 stainless steel to meet Section III, Class 1 requirements of the ASME Boiler and Pressure Vessel Code. The reactor vessel will consist of a 2-inch-thick cylindrical upper section and a 3-inch-thick lower head comprising a toroidal knuckle, a conical frustum, and a spherical cap. These shapes and thicknesses were chosen primarily to limit core accelerations and deflections to permissible levels during design-basis seismic events.

The middle section of the reactor vessel wall is designed to operate at close to the reactor outlet temperature. It will be protected from thermal shocks by a stainless steel liner. The temperature profile in the upper section of the reactor vessel between the hot pool and cold reactor support structure elevation will be controlled by reflective insulation located along the inner periphery of the reactor vessel, in combination with the natural-convection heat removal from the deck, reactor support, and reactor cavity. In this way, thermal stress intensities caused by axial and radial temperature gradients in the upper vessel will be maintained at acceptable levels. The pressure-retaining capability of the reactor vessel shell will be more than 10 times the pressure loadings at normal operating temperature.

Vessel Support Structure

The reactor vessel will be supported at the top through the reactor support structure, which is integral to the reactor vessel. The reactor support will consist primarily of a bi-metallic cone-shaped structure; the upper portion will be fabricated of Type 316 stainless steel and will be attached to the lower portion of the cone, which will be fabricated from 2-1/4Cr-1Mo, via a compressive bi-metallic weld.

The containment vessel, also constructed of 2-1/4Cr-1Mo, will be attached to the underside of the lower cone section.

Lower Internal Structures

The lower internal structures will consist of components internal to the reactor vessel directly around and below the reactor core. The principal components will be the shield barrel, the core barrel, the inlet and distribution plena, the core-support cylinders, the inlet modules, and the core former plate. The material specified for all of these is Type 316 stainless steel, with Inconel 718 piston rings in the inlet modules.

Upper Internal Structure (UIS)

The UIS is a hardware package suspended from the rotatable plug to a distance about 2 inches above, and centered on the tops of, the core assemblies. It will include

- shroud tubes containing and protecting the primary and secondary control rod drivelines
- structural features to position these drivelines
- core assembly outlet thermocouples and other reactor instrumentation
- baffles and flow-direction devices to promote mixing of core discharge and coolant

The outer boundary of the UIS is a 6-foot, 8-inch-diameter shroud which will extend down about 21 feet axially from the deck upper surface to a point about 2 feet above the reactor core outlet. Within the outer shroud will be nine control rod shroud tubes and an instrumentation tube at the core centerline. The shroud tubes and conduit will be perforated to allow the release of coolant to the hot pool at an elevation below the free surface. Horizontal guide plates welded to the UIS outer shroud will serve to locate and support the control rod drivelines. The bottom plate will be located about 2 feet above the core top and a second plate will be located about 6 feet above the first. The intermediate shroud tubes will be welded between the lower plate upper surface and the upper plate lower surface to distribute normal and seismic loads throughout a stiffened lower UIS structure.

Redan Assembly

The redan assembly will provide thermal and hydraulic separation between the hot and cold pools and will also form the lateral seismic connection between the top of the shield barrel and the reactor vessel. Hydraulic separation and seismic connection will be provided by a 3-inch-thick annular plate attached between the shield barrel skirt extension and the reactor vessel liner. Thermal separation will be provided by 0.75-inch-thick, segmented baffle plates supported 30 inches above and below the central 3-inch plate by flex rod supports. These baffle plates will enclose a stagnant sodium buffer zone between the open hot and cold pools. The downward heat flow between the pools will promote stratification in the buffer zone, while convection will be minimized by labyrinth seal features at the baffle segment peripheries. This stratification will minimize the thermal gradients in adjacent cylindrical structures. The thermal inertia of the buffer zone will also protect the critical structural plate from hot pool downshocks and cold pool upshocks.

Component penetrations of the redan assembly will be accommodated in three different ways. The two PHTS pumps and the cold trap assembly will pass through wells that attach to the redan structural plate and extend above the surface of the hot pool, thereby gaining access to the cold pool without breaching the hydraulic boundary. The upper portions of these penetration wells will be double wall to withstand the hot-pool-to-cold-pool temperature gradient.

The four IHXs and the DRHX will penetrate the hydraulic boundary but will be sealed against their penetration wells to minimize leakage and maintain the pressure difference. The extended penetration wells for the IHXs will allow these seals to be placed at the small-diameter lower IHX sections. The fuel storage basket and fuel transfer station will extend below the level of the structural plate but will

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be completely contained within well extensions of the structural plate, and thus will not penetrate the hydraulic boundary. Labyrinth seal features will also be employed (where the baffle plate meets the penetration well) to minimize convection and maintain stratification in the buffer zone.

The redan assembly will be constructed entirely of Types 304 and 316 stainless steel. Sealing surfaces of the heat exchanger penetration wells will be hardfaced for maximum wear resistance. The redan assembly will also provide support and positioning for localized canned B₄C shielding deployed around the inner peripheries of the IHXs and the fuel storage basket.

Pump Discharge Piping

The pump discharge piping will consist of four runs of 14-inch-diameter piping per pump, which provides the hydraulic connection between the pump discharge plena and the distribution plenum and which will be attached between nozzles fitted to the pump tanks and the shield barrel. The use of multiple pipe runs provides redundancy in the event of a pipe break and also permits the use of more flexible, smaller diameter piping fabricated from Type 304 stainless steel.

Permanent In-Vessel Shielding

Permanent radiation shielding will be used in the annular gap between the core and shield barrels and along the inner peripheries of the IHX fuel storage basket redan penetration wells. The ex-core shielding arrangement will employ two rows of rectangular stainless steel cans filled with granulated B₄C shielding material. Cylindrical posts attached to their end caps will fit into recesses in the upper surface of the plenum upper plate and in the underside of the core former, thereby providing support and positioning to the shielding cans.

Reactor Vessel Closure

The vessel closure will consist of a fixed annular deck supported from and sealed to the inner circumference of the reactor vessel flange. Inside this annular deck will be an eccentrically located rotatable plug assembly that is an essential feature of the under-the-head approach for in-vessel fuel transfer.

The fixed annular deck will have the following penetrations:

- four 84-inch penetrations for the IHXs
- two 48-inch penetrations for the primary pumps
- a 62-inch penetration for a primary cold trap
- a 34-inch penetration for a SHRS DRHX heat exchanger
- a rectangular 24-inch by 12-inch penetration for a slated fuel transfer track
- a 24-inch penetration for the in-vessel fuel storage basket support and drive shaft mechanism

In addition, there will be instrumentation penetrations.

Deck Structure

The deck will include the following elements or features:

- 18 inches of stainless steel multiple-plate reflective insulation on its underside for sodium pool slosh load protection plus reflective plates to provide multiple resistances to radiative heat transfer and almost total suppression of convective heat transfer (The insulation will be within the reactor cover gas containment boundary.)
- a 60-inch-deep, all-welded structure that will be a stressed-skin circular boxbeam constructed of

carbon steel (The bottom face plate and all the vertical elements, except the shear panels and the pump well cylinders, will also function as the containment for the reactor cover gas. The design of the structure is based on the limit of core-rotatable plug vertical displacement (rod withdrawal) during the safe-shutdown earthquake (SSE) seismic event.)

- shielding, primarily to attenuate the pool Na^{24} gamma radiation (Shielding is provided by filling the inside of the deck structure with iron-ore pellets (approximately 65% iron content) to such a depth that the shielding effectiveness of all material between the sodium pool surface and the head access area above the deck is equivalent to 27 inches of steel.)
- cooling by natural-convection (passive) circulation of head access area air within the structure, with exhaust chimneys providing the draft

Rotatable Plug

The rotatable plug will be supported by a bearing-seal assembly, that will allow the plug to rotate eccentrically relative to the deck centerline. For refueling, rotation of the plug will reposition the UIS to partially uncover the core assembly lattice to the extent that an offset arm in-vessel handling machine (IVHM) mounted through the plug can be rotated inward to index its grapple anywhere in the core assembly. Combined orientation of the IVHM arm and the plug will then provide indexing of the IVHM grapple over all core lattice positions.

The plug will be of essentially the same construction as the deck structure, except for the eccentricity of the central penetration in the plug. The plug will also incorporate iron-ore pellet shielding and the same natural-convection air-cooling approach.

Containment Vessel

The containment vessel will surround the reactor vessel with an annulus sufficient to maintain a nominal coolant level within the reactor so that adequate cooling of the reactor core is ensured in the unlikely event of a reactor vessel leak. The vessel will be fabricated from 2-1/4Cr-1Mo steel and will be top supported by the vessel support structure.

The containment vessel will serve as the primary reactor containment and will contain a fill-gas atmosphere of helium. Air circulation over the outer surface of the vessel will be the key feature of the RACS discussed in Section 5.6 of this SER.

5.3.2 Scope of Review

The review focused on the acceptability of the design of key components, including their materials, for safe, normal operation of the SAFR. Items related to transient performance are discussed in Chapter 15 and Appendix A of this SER.

5.3.3 Design Criteria

The reactor vessel and many of its internal structures will serve as the primary coolant pressure boundary. Therefore, both they and the outer containment vessel will require high integrity to satisfy general design criteria that the reactor coolant pressure boundary be designed and constructed to minimize the probability of leakage. SAFR criterion PDC 2.7, consistent with GDC 16, applies.

5.3.4 Research and Development Program

The testing priorities for safety and licensing needs for all materials are reviewed in Section 4.5 of this SER. In general, they include the separate and combined effects of neutron and thermal aging

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on stainless steel welds, the use of Inconel 718 and hardfacing for sliding seals, and methods for inspecting 100 percent of the welds of all components critical to safety during and, for closure welds, after assembly of the reactor vessel, containment vessel, and internal structures.

The design, including complex penetrations with seals that are expected to operate within a sodium environment, contains a number of features the operability of which may not be adequately demonstrated. Development studies and detailed engineering evaluations of many of these features need to be done to ensure that the reactor vessel, containment vessel, internal structures, and redan assembly maintain this integrity under the thermal and radiation environments to which they will be exposed.

5.3.5 Safety Issues

In view of the combined effects of radiation and thermal cycles on welds of stainless steel, it may be difficult to prove the integrity of the pressure vessel, the redan, and internal structures. Since these form part of the reactor coolant pressure boundary (and the thermal mixing barriers), their long-term integrity and inspectability must be demonstrated.

The reliability of the rotating, hard-faced seals in a sodium environment needs to be adequately demonstrated, as these form part of such safety-related equipment as control rod drives and refueling mechanisms.

5.3.6 Conclusions

The SAFR vessel internals design is unique and requires demonstration that the specific components will stand up to the operating environment of an LMR.

5.4 Primary Heat Transport System

5.4.1 Design Description and Safety Objectives

The PHTS is a "two-loop" pool system, comprising the coolant passages in the core region, the hot pool, the tube side of the four IHXs, the cold pool, the two centrifugal pumps, the pump discharge piping, and the reactor inlet plenum. The entire primary coolant (radioactive sodium) flow path will be contained within the reactor vessel. The PHTS sodium will flow from the cold plenum through the pumps to the reactor inlet plenum, which will distribute the flow to the fuel, blanket, reflector and control assemblies. The sodium will then exit to the hot pool, enter the IHXs, flow downward inside the IHX tubes, and then return to the cold pool.

The safety objective of the PHTS is to remove the heat generated in the core region and transport it to the IHTS through the IHXs in order to maintain temperature and pressure distribution within the reactor vessel below the design limits, under either normal or off-normal operating conditions. In the case of such events as reactor vessel break, pump failure, pump discharge pipe break, or loss of the IHX cooling, the reactor must be cooled effectively.

5.4.2 Scope of Review

The review focused on the identification and acceptability of key safety issues given in Section 5.4 of the PSID (Ref. 5.1), either based on analyses provided by RI or on independent staff analysis. Items regarding transient performance were largely deferred to Chapter 15 and Appendix A of this SER.

IHX Performance and Strength

As the IHX effectively forms both the primary coolant and the containment boundaries, its integrity must be assured. This is particularly true with respect to challenges resulting from steam generator tube ruptures and the sodium-water reactions that would follow. In the SAFR IHX, the intermediate loop sodium will pass on the shell side, thus exposing the tube support plates to possible pressure pulses coming from the steam generator. This was discussed in Section 5.2.5 of this SER.

Stratification in Pools

For a pool-type reactor, one must consider the possibility of thermal stratification and other issues regarding mixing. RI and ANL are performing experiments (Plexiglas water tanks) and calculations (COMMIX) to investigate this possibility. The staff reviewed these tests and calculations briefly, but performed no independent tests or analyses.

Pump Performance

At issue was whether, given that one pump seizes instantaneously, the second pump will trip and coast down in a fashion sufficient to preclude sodium boiling and/or fuel melting, even under a failure-to-scrum condition.

5.4.3 Design Criteria

SAFR criteria PDC 2.5 and 2.6, which are consistent with GDC 14 and 15, apply. In particular GDC 15 states that the primary system boundary must be protected from challenging conditions. This includes the IHX, which serves as part of the primary coolant boundary.

5.4.4 Research and Development Program

R&D programs are planned, or are in progress, to test the pump and IHX performance as well as flow distributions in pools. To some degree, these tests will focus more on performance and transient response than on severe challenges. The testing priorities for safety and licensing needs include

- pump performance during coastdown and during flow surges such as those expected when the other pump seizes (This area provides the most difficult challenge to the reactor passive shutdown capability and the pump behavior must be well understood.)
- IHX response to large pressure pulses on the shell side (The steam supply system operates at 2800 psi.)
- flow stratification under conditions of natural circulation (This must be evaluated for the SAFR design, partly because the reactor vessel component layout is not symmetrical.)

5.4.5 Safety Issues

Pump and Pump Discharge Piping

In the event of one pump seizure or one primary loop pipe break, the core flow will reduce quickly to 37 percent and 54 percent, respectively. If there is a failure to scram, it is imperative that the remaining pump will continue to run for around a minute before tripping (so the reactor power can decrease before there are further reductions in flow). The staff would need additional information and review effort to be convinced that the design provides adequate assurance that rapid flow reductions that can lead to sodium boiling are extremely improbable.

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PHTS to IHTS Pressure Boundary

By using a 2800-psi steam system, single-wall steam generator tubes, and an IHX with the intermediate fluid on the shell side, RI's design has raised concerns about the IHX vulnerability (see Section 5.2.5 of this SER).

5.4.6 Conclusions

The SAFR is a pool-type reactor with the entire primary coolant flow path contained within the reactor vessel. The SAFR PHTS has the potential to provide excellent capability for removing heat via forced or natural convection in the reactor vessel from the reactor if the system is scrammed. For unscrammed transients, a fast reduction in primary flow needs to be precluded. The SAFR centrifugal pumps need to ensure coastdown flow which provides time for the reactivity feedback mechanisms to work. Although pump seizures have been rare in U.S. LWR history (four events, all during coastdown), a pump seizure coupled with a failure to scram would be a major threat to a SAFR. Events in the "fast LOF" category are going to continue to be important and raise concerns as long as there is a large positive sodium void reactivity worth.

The IHX design in the SAFR raises questions about the ability to maintain primary system integrity during sodium-water reactions. Further assessments will be required prior to licensing.

5.5 Intermediate Heat Transport System

5.5.1 Design Description and Safety Objectives

The IHTS is composed of two independent and physically separated loops (see Figures 1.3-1.5). It is designed to circulate nonradioactive sodium in a continuous loop to transport reactor heat to the steam generator under all operating conditions. Each of the IHTS loops consists of the shell side of the IHXs, an intermediate pump, an expansion tank, associated piping, the shell side of the steam generator, and a steam generator, sodium-water reaction, pressure-relief system. The nonradioactive intermediate sodium is separated from radioactive primary sodium by a passive boundary (IHX tubes).

Intermediate Heat Exchanger

The IHX is a vertical straight-tube, flexible downcomer design. It comprises an enclosure shell, a tube bundle with replaceable bellows, and a cylindrical hanging support. The safety objective of the IHX is to isolate radioactive primary sodium from intermediate sodium and serve as a mechanical barrier to the transport of radioactive sodium out of the containment vessel. Thus, it is part of the primary containment system.

IHTS Piping

The IHTS piping routes intermediate sodium in a closed loop to transport reactor heat to the steam generator. Each IHTS piping run contains the necessary elbows, tees, and reducers to provide adequate loops for thermal expansion.

Intermediate Coolant Pump

The intermediate coolant pump is a free-surface, single-stage, vertically mounted centrifugal pump similar to the primary pump.

Steam Generator System

The steam generator system includes the steam generator, the sodium-water reaction, pressure-relief subsystem (SWRPRS), and the water-dump subsystem. The steam generator is a once-through

Benson-cycle unit converting feedwater at 495°F to superheated steam at 855°F and 2800 psig. Flow is countercurrent, with sodium flow down through the shell side and water-steam flow up in the tube side. A key safety objective of the steam generator system is to prevent the pressure generated by a sodium-water reaction within the steam generator from endangering the IHX coolant boundary integrity. A sodium-water reaction protection system is provided in the SAFR design similar to the one designed for the Clinch River Breeder Reactor Project (CRBRP). The present design-basis sodium-water reaction is a three-tube rupture in which the failures of the second and third tubes follow at 0.1-second intervals. The effects of this assumption are the same as that for a single-tube-rupture model. In the very closely-packed arrangement of steam generator tubes, with one tube surrounded by six adjacent tubes, if the first steam tube ruptures, the six adjacent tubes could be involved. However, the in-service inspection program to identify unexpected tube degradation combined with the leak-detection program for the steam generator is designed to make the possibility of instantaneous rupture of six adjacent tubes highly unlikely. Further PRA work is needed to validate this conclusion.

5.5.2 Scope of Review

The staff reviewed Section 5.5 of the PSID (Ref. 5.1), "Intermediate Heat Transport System." The review focused on design, construction, testing, and performance requirements for structures, systems, and components important to safety.

The review also focused on possible threats to the IHX coming from the intermediate loop or the steam generator system or both, with the most likely source being sodium-water reactions. As a result, the sodium-water-reaction pressure-relief system (SWRPRS), the rupture disks, and SWAAM computer code simulations were discussed at some length (see Questions 5.23, 5.24, 5.28, and 5.41 in PSID Appendix G, Ref. 5.1).

5.5.3 Design Criteria

GDC 15 (SAFR criterion PDC 2.6) specifies that the primary coolant system must be protected from challenging conditions. Most of the protection of the IHX tubes, which form part of the primary coolant system boundary, comes from the SWRPRS and the rupture disks. GDC 16 (SAFR PDC 2.7) also applies, as the IHX serves as a containment boundary as well.

5.5.4 Research and Development Program

RI cites R&D programs in the materials testing and seismic response areas for work with the intermediate loop. However, a key issue is how many steam generator tubes are likely to fail during a worst-case event and how well will the protection systems respond to the challenge.

5.5.5 Safety Issues

The most important safety issue in the IHTS design is the protection against IHX coolant boundary failure. Potential IHX coolant boundary failures may be caused by IHX materials failures or overpressure in the event of a large-scale sodium-water reaction (see Section 5.2.5 of this SER).

Small sodium-water reactions will be detected by the leak detection system and action will be taken (steam-water dump) before large-scale reactions occur. The ability to detect very small leaks is important to the prevention of large leaks. In the event of a large sodium-water reaction, the sodium and water inventories of the affected heat exchangers must be evacuated quickly to prevent major challenges to the IHX. The sodium-water reaction pressure-relief system is designed to accomplish this function; however, because of its key safety function it should be safety grade. Further data on the improbability of cascading failures of all seven steam generator tubes in a group needs to be developed before the three-tube-failure design basis is considered acceptable.

5.5.6 Conclusions

With resolution of the issues described above, the SAFR IHTS has the potential to be acceptable.

5.6 Shutdown Heat Removal Systems

5.6.1 Design Description and Safety Objectives

Three shutdown heat removal systems (SHRSs) are available in the SAFR, one normal and two backup systems. The normal SHRS operates through the main heat transport system which carries primary system heat through two parallel intermediate heat transport loops and then into the steam generation system. The first backup SHRS is the direct reactor auxiliary cooling system (DRACS). The emergency SHRS is the reactor auxiliary cooling system (RACS), which is described in detail in Section 6.3 of the SAFR PSID (Ref. 5.1) and reviewed in Section 6.3 of this SER. Of the three SHRSs in the SAFR, only the RACS is currently classified as safety grade. Figure 5.1 illustrates both the DRACS and RACS heat-removal systems.

Section 5.6 of the PSID (Ref. 5.1) focuses mostly on the DRACS. This system consists of the direct reactor heat exchanger (DRHX), a secondary sodium loop, a natural-draft heat exchanger (NDHX), a helium vent/supply valve that opens to activate the system, and dampers on the NDHX which are opened to allow air to cool the secondary sodium. When the helium vent/supply valve is activated, the helium pressure is reduced to near reactor cover gas pressure by either fully opening one of the two partially opened helium vent valves or closing one of several helium supply valves; the primary sodium from the hot pool is allowed to pass downward through the inside of the DRHX tubes and out into the cold pool. DRACS secondary system sodium will circulate under natural conditions in an upward direction on the shell side of the DRHX and on the tube side of the NDHX. The helium vent/supply valve will maintain a pressurized zone of helium that will keep the primary sodium level below intake lines to the DRHX. As long as helium is continuously pumped into this zone and the two escape valves remain closed, the sodium level will remain too low for the DRACS to function. The helium vent/supply valve and NDHX airflow dampers are designed to fail open. Conditions within the DRACS are continuously monitored to ensure its availability.

In Section 5.6 in the PSID (Ref. 5.1) the reliability of the overall SHRS is discussed. (Also see the appropriate section in the PRA.) This analysis includes repairability considerations since RI estimates that a heatup transient can progress for 30 hours without SHRS operation before ASME Service Condition D temperatures are exceeded. The 30-hour heatup period is estimated on the basis of heat capacity of the reactor vessel and its internals, about 2×10^6 Btu/°F. Data included in the PSID (Ref. 5.1) show that RI predicts that repairs can be made faster than the heatup rate of the reactor.

5.6.2 Scope of Review

The review focused on the safety-grade RACS (see Section 6.3 of this SER) rather than on the non-safety-grade DRACS. The DRACS is similar to systems in other LMR designs, and thus the review of this system drew upon experience gained at those plants.

5.6.3 Design Criteria

The need for two independent means of decay heat removal is addressed in Chapter 3 of this SER.

5.6.4 Research and Development

In the DRACS, only the helium vent/supply valve would be considered non-standard LMR technology. The helium vent/supply valve must be tested in order to establish DRACS reliability.

5.6.5 Safety Issues

The SAFR has diversity in the SHRS, but only the RACS is safety grade. The DRACS is largely passive and could be made safety grade if two issues were resolved. These involve operation of the helium vent/supply valve and the NDHX dampers. RI is studying the possibility of making the helium vent/supply valve safety grade to ensure an additional safety-grade flow path to remove energy from the hot sodium. The second problem is that the NDHX is not tornado-hardened. The dampers on the NDHX will be kept closed during normal operations to prevent sodium freezing. However, the dampers will fail open, and a provision is being made to install an optional hand crank to close them during emergencies. Thus, although the DRACS should be very reliable, it is not a safety-grade system.

5.6.6 Conclusions

RI has designed DRACS as the first backup decay heat removal system, but has chosen not to make it a safety-grade system. RI's claims that the system is largely passive and highly reliable appear to be reasonable. The major obstacle to making this system safety grade is the NDHX, and the cost associated with tornado-hardening. However, given the response of the SAFR to certain of the bounding events [particularly BE-3 (Section 6.1.4) loss of RACS], a second safety-grade decay-heat-removal system should be considered to provide increased confidence that a safety-grade RACS backup is available. (Note: upgrading DRACS to a safety-grade system would affect the definition of BE3.)

5.7 Reference

- 5.1 DOE, "SAFR -- Sodium Advanced Fast Reactor," Preliminary Safety Information Document, AI-DOE-13527, Initial Issue -- October 1985, with amendments. (DOE Applied Technology.)

Reactor Coolant System and Connected Systems

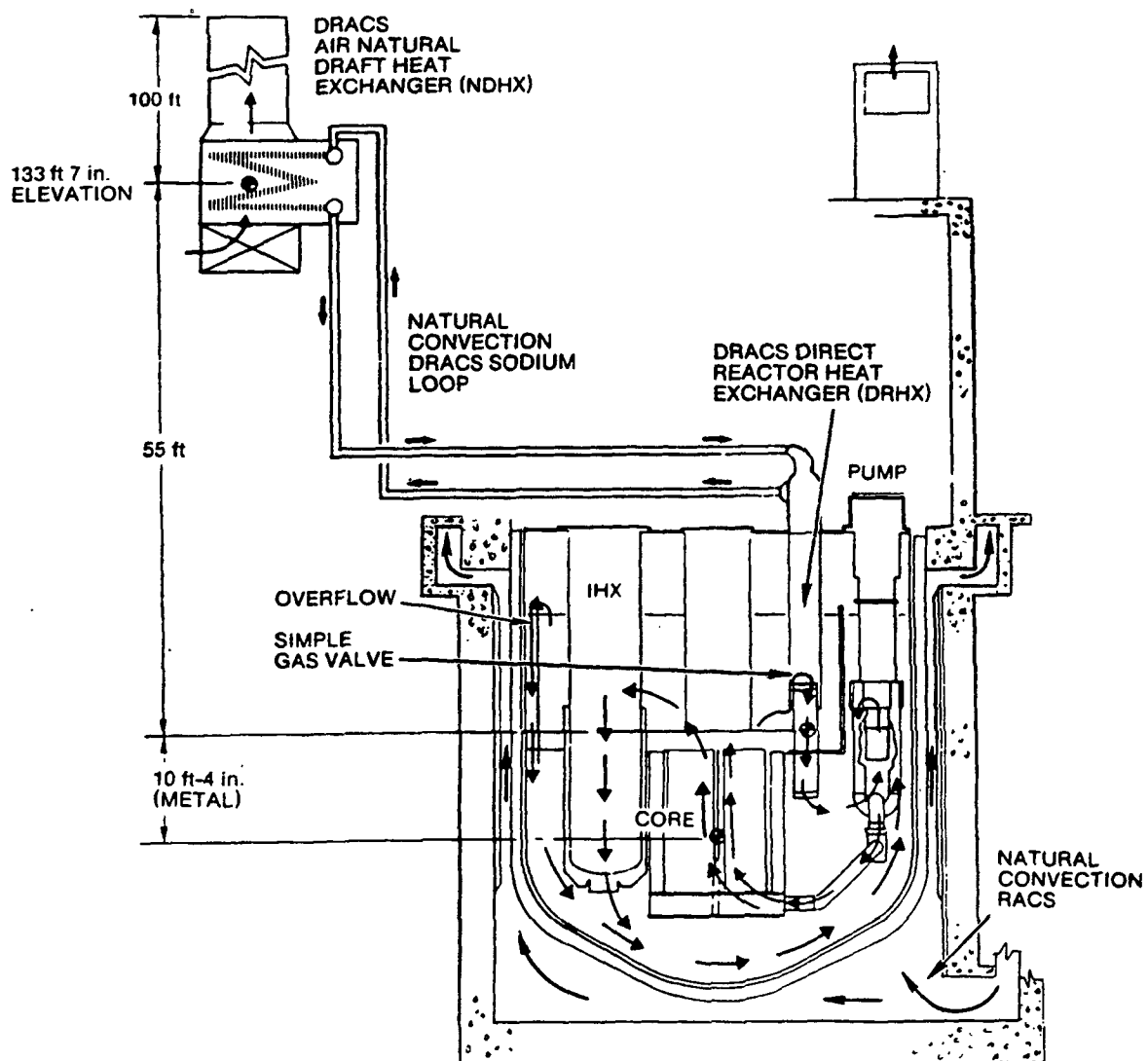


Figure 5.1 Natural-convection loop elevation.

6 ENGINEERED SAFETY FEATURES

Engineered safety features mitigate the consequences of accidents. The SAFR engineered safety features addressed here include the containment and residual heat removal systems. Component in-service inspection (ISI) is also addressed.

The containment system is discussed in Section 6.1. The design limits the leakage of radioactive material to the dose guidelines in 10 CFR Part 100. The system includes a primary containment, consisting of the containment vessel and the reactor cover-gas boundary, and a secondary containment barrier surrounding the reactor cover-gas boundary, consisting of a rectangular concrete structure designed to withstand low pressure. The containment system is designed for seismic Category I and is tornado hardened.

The RACS is the nuclear-safety-related portion of the residual heat removal system and is discussed in Section 6.2. Detail is given on the system design bases, design description, performance evaluation, tests and inspections, and instrumentation requirements.

Section 6.3 addresses the components subject to ISI evaluation, accessibility, and examination techniques and procedures. Inspection requirements are given for both safety-related and non-safety-related systems.

6.1 Containment System

6.1.1 Design Description and Safety Objectives

The containment system is designed to ensure that the 10 CFR Part 100 dose guidelines are not exceeded over a wide range of postulated accidents, including those initiated by external natural phenomena.

The containment system consists of portions of the reactor containment building (RCB) and systems and components within it. The primary containment consists of a containment vessel and the cover-gas and primary-to-secondary sodium boundaries in the reactor closure. The upper part of the RCB performs the secondary containment function and is designed to withstand seismic and tornado forces. The two containment systems are illustrated in Figure 6.1.

Reactor Containment Building (RCB)

The RCB is a seismic Category I structure designed to protect the plant against the effects of the design-basis tornado. The lower portion of the RCB consists of a cylindrical reactor shield wall (at least 6-feet thick) that surrounds the reactor vessel and containment vessel.

Primary Containment

The containment vessel forms the lower portion of the primary containment boundary and surrounds the reactor vessel with a 7-inch annulus to maintain a coolant level within the reactor that will keep the core covered in the unlikely event of a reactor vessel leak. The containment vessel is classified as a Safety Class 2 (SC-2), seismic Category I structure designed to the ASME Code, Section III, Division 1, Subsection NE, "Metal Containment."

The cover-gas and primary-to-secondary sodium boundaries, which serve as the upper portion of primary containment, are SC-2, seismic Category I structures designed to the ASME Code, Section III, Division 1, Subsection NC, Class 2 components. The upper portion of the primary containment boundary consists of components and reactor vessel enclosure head penetrations that are sealed with at least a primary O-ring seal and a backup seal (see Figure 6.2). Criteria will be developed to meet 10 CFR Part 100 requirements for seal leakage. The primary containment has a 10-psig design limit

Engineered Safety Features

to withstand an IHX tube leak and reactor vessel leak into the containment vessel. The leak rate at 10 psig is limited to 0.1 percent per day. Radiation monitors and isolation valves are used for automatic primary containment isolation of piping systems.

Secondary Containment

The secondary containment, which is above the top of the reactor vessel, is the upper portion of the RCB. It is a rectangular-shaped, concrete structure with a flat concrete roof. The secondary containment is designed to sustain a pressure of 1.7 psig, which results from tornado external depressurization. The secondary containment is designed to limit leakage to 100 percent of its contained volume per day under a pressure of 1.0 psig. The same signal that isolates the primary containment isolates the open systems of the secondary containment.

RI evaluated the effectiveness of the primary and secondary containment using a proposed siting source term equivalent to that source resulting from the meltdown of a single fuel assembly at power (see Table 6.1). The entire fission-product inventory of the fuel assembly was assumed to be released instantaneously to the sodium pool; however, credit was taken for retention of some of the fission products in the sodium (see Table 6.1). RI's calculated offsite doses using this source term are shown in Table 6.2.

6.1.2 Scope of Review

The staff reviewed the proposed containment system and corresponding design-basis accidents described in the PSID (Ref. 6.1) and responses to staff questions. For the review, the staff has developed proposed criteria for the containment system. These criteria are discussed in SECY 88-203 (Ref. 6.2).

The staff reviewed the design to assess its potential to meet the proposed criteria. Protection of safety-related systems, structures, and components from sabotage is discussed in Section 13.4 of this SER, "Safeguards and Security."

6.1.3 Research and Development Program

The acceptance of the proposed containment system depends on the ability of the plant to meet certain dose guidelines for a wide range of accidents. Proposed R&D to demonstrate this ability is reviewed in Chapter 14 and Appendix A of this SER.

6.1.4 Conclusion

The staff does not believe the proposed SAFR containment design is acceptable. The reasons for this are

- BE-1: Inadvertent Withdrawal of All Control Rods With Failure To Scram

Most fuel pins in the inner rings will likely fail during the event in which the RACS is the sole operating heat removal system. This results in a potential for fuel motion and reactivity insertion.

- BE-3: Loss of All Decay Heat Removal for 36 Hours

Assuming that some heat escapes to the earth (a safe assumption), fuel damage and cladding penetration can be expected near the end of the first day. Fuel movement in an LMR core must be regarded as hazardous, although for this scrammed event, some fuel could move without reactivity-related consequences (i.e., power transients). If the system geometry is somehow maintained, large-scale sodium boiling would develop about 12 hours later and major failures leading to fuel melt and vessel melt-through could result.

- **BE-4: Instantaneous Loss of Flow From One Primary Pump With Failure to Scram**

Loss of one pump leads to a rapid reduction of core sodium flow to 37 percent of full-power flow. If the other pump continues to operate, the reactor power will drop quickly due to reactivity feedbacks (assume scram fails), and there should be no damage. However, if the remaining pump trips soon after the first pump fails, sodium boiling could develop and fuel damage would result.

- **BE-7: Flow Blockage of a Single Fuel Assembly**

Since, with the present SAFR design, flow blockage cannot be detected until the fuel has been damaged, this event can lead to local sodium boiling and local loss of coolable geometry, which could lead to a large reactivity addition.

Since these four events have the potential to lead to early core melt (and possibly reactor vessel and containment vessel penetration) or positive reactivity feedback accidents (which could breach the containment) they represent a fundamental concern with the SAFR design. These events are discussed further in Chapter 15 of this SER. Resolution of these concerns remains an open item.

6.2 Shutdown Heat Removal Systems

6.2.1 Design Description and Safety Objectives

The SAFR emergency shutdown heat removal systems include the reactor auxiliary cooling system (RACS) and the direct reactor auxiliary cooling system (DRACS). DRACS is covered in Section 5.6 of the PSID (Ref. 6.1) and in Section 5.6 of this SER. The RACS is the only safety-grade SHRS in the SAFR.

In the RACS, the cooler atmospheric air will flow inward through ducts located about 40 feet above grade and will pass through the reactor building to the bottom of the containment vessel. The air flow then will pass upward, drawing heat from the outside of the containment vessel and from both sides of a finned collector surface. Hot air will then be exhausted about 15 feet higher than the intake ducts. The air will flow continuously by natural circulation. Figure 6.3 illustrates an elevation view of the heat transfer air flow paths and Figure 6.4 provides a plan view of the heat-transfer paths.

Under full-power operation, little heat will transfer between the reactor and the containment vessel. Heat will be conducted through the reactor vessel wall and then will be transferred primarily by radiation (through a helium-filled annulus) to the containment vessel. The heat transferred under normal full-power operations is low because the cover-gas inside the reactor vessel resides along the upper (hot pool area) vessel wall, so the vessel wall remains at the cover-gas temperature. Under accident conditions, the reactor vessel will heat up considerably because of the higher sodium temperature and the fact that heating of the reactor vessel wall from within is augmented by hot pool sodium spilling over the reactor vessel liner. This will occur naturally as the sodium expands with temperature (i.e., level swelling), and spills down along the reactor vessel wall to the cold pool. As a result, the thermal radiation from the reactor vessel to the containment vessel will increase and the ability of the RACS to reject heat will increase by a factor of three as compared to steady-state operation.

Because the SAFR sodium pools will be quite large and the component configurations will be asymmetric, multidimensional analysis of the sodium flow patterns during RACS operation may be necessary. In Section 6.3 of the PSID (Ref. 6.1), plots showing TEMPEST and COMMIX-1A multidimensional model nodalizations are provided. (Note: COMMIX-1A is a special version of COMMIX with a model for radiative heat transfer added for simulating the RACS.)

6.2.2 Scope of Review

By independent calculations, the staff verified RACS performance (see Table 6.3). The PASCOL computer code was used to analyze the air-cooled vessel system. In these calculations the reactor vessel temperatures were specified and PASCOL calculated the containment vessel temperatures along with the air flow rate and temperatures. Using reasonably conservative assumptions for emissivity and air flow path form losses (pressure drops), it was possible to confirm the RI-predicted response of RACS.

Sensitivity calculations also indicated that major flow blockages could likely be tolerated without serious degradation in RACS performance. Calculations using reasonable judgments from the literature have confirmed the potential of the RACS to perform as the designers claim.

6.2.3 Research and Development Program

DOE is currently sponsoring a large-scale test of RACS-type panels at ANL. Early experimental results have been consistent with predictions from analysis.

The RACS should be tested as part of the safety test program for the first SAFR module.

6.2.4 Safety Issues

Because the designers have chosen to make the RACS the only safety-grade, decay-heat-removal system in the SAFR, this system has been scrutinized carefully. Its performance under nominal conditions must be further verified, although independent analysis indicates that the RACS will perform as expected. Partial blockage of the air pathways, resulting from a seismic event or sabotage, has been postulated. Time would be available to remove blockages during a heatup event before design limits would be challenged. However, the RACS is not capable of preventing sodium boiling or fuel melting, or both, under all conditions postulated in EC-III. As mentioned in Section 6.1.4 of this SER, the SAFR response to BE-1 may lead to sodium boiling and fuel melt.

Leakage of sodium from the reactor vessel into the containment vessel changes RACS performance in two ways. First, if enough sodium leaks out, the sodium level drops below the height needed for overflow of the vessel liner, and the reactor vessel will heat up. However, having sodium between the two vessels improves the vessel-to-vessel heat transfer. On balance, RACS performance is probably improved by such a leak, as the RI analysis suggests. This is not an easy calculation to verify and should eventually be independently evaluated. Some fouling of the heat-transfer surface is acceptable, since large performance margins exist.

6.2.5 Conclusions

The RACS appears to have the potential to be a satisfactory decay-heat-removal system for many events, assuming its performance lives up to the analytical predictions. Its strengths appear to be that it is highly fault tolerant, runs continuously without human intervention, is self-activating, and time is available to restore it to working order during an accident. The RACS has the potential to be a good, passive-shutdown, heat-removal system, provided it can be demonstrated that it will perform satisfactorily for all events in EC-I through EC-III.

6.3 In-Service Inspection

ISI presents a greater challenge in a sodium system than in a water system, since sodium is opaque. Major difficulties are anticipated, particularly with respect to components within the reactor vessel. However, the SAFR will provide a degree of fault-tolerance that may be sufficient to make the ISI requirements manageable.

There will be R&D associated with ISI, particularly for inspecting the sodium-filled vessel, but this has not been discussed extensively.

6.4 References

- 6.1 DOE, "SAFR - Sodium Advanced Fast Reactor," Preliminary Safety Information Document, AI-DOE-13527, Initial Issue - October 1985, with amendments. (DOE Applied Technology.)
- 6.2 NRC, "Key Licensing Issues Associated With DOE Sponsored Advanced Reactor Designs," SECY 88-203, July 15, 1988.
- 6.3 NRC, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," Regulatory Guide 1.4, Rev. 2, June 1974.

Table 6.1 SSST for the SAFR
(Resulting from single fuel assembly meltdown)

Element	Single fuel assembly inventory (grams)	Amount released to cover gas (grams)	Percent of total inventory in all fuel assemblies
Noble gases (Xe, Kr)	60.1 1100	60.1 1100	1 1
Volatiles (Cesium)	986	9.86×10^{-4}	1.0×10^{-4}
Halogens	87.9	2.60×10^{-4}	1.0×10^{-6}
Plutonium	13,272	1.05×10^{-4}	7.8×10^{-9}

Table 6.2 Doses resulting from SSST

Case	Time period/location	Primary containment deck leakage (%/day)	Secondary containment leakage (%/day)	Doses (Rem) Regulatory Guide 1.4 (Ref. 6.3)	
				Thyroid	Whole body
1	2 hours at SB ^a	0.1	100	6.2×10^{-2}	6.2×10^{-4}
2	2 hours at SB	0.1	No hold up	7.4×10^{-5}	7.4×10^{-3}
3	30 days at LPZ ^b	0.1	100	4.5×10^{-4}	2.5×10^{-2}
4	30 days at LPZ	0.1	No hold up	4.5×10^{-4}	2.5×10^{-2}
10 CFR 100 criteria at construction permit stage				150	25

^a SB = Site boundary (0.4 mile)

^b LPZ = Low population zone (3.0 miles)

Table 6.3 SAFR RACS performance during decay heat removal

Item	SAFR-RI	SAFR-BNL
Emissivity	0.65	0.65
Surface effectiveness	--	1.5 ^a
$K_{\text{air}}(\text{in}) + K_{\text{air}}(\text{out})$	1.	10.
W_{air}	39.	37.2
Q(MW)	3.90	3.96
$\Delta\theta$ (C), air	99.4	102.5

^aSurface effectiveness is likely much higher than 1.5.
 At a best-estimate value of 2.2, Q is estimated to be 4.85 MW.

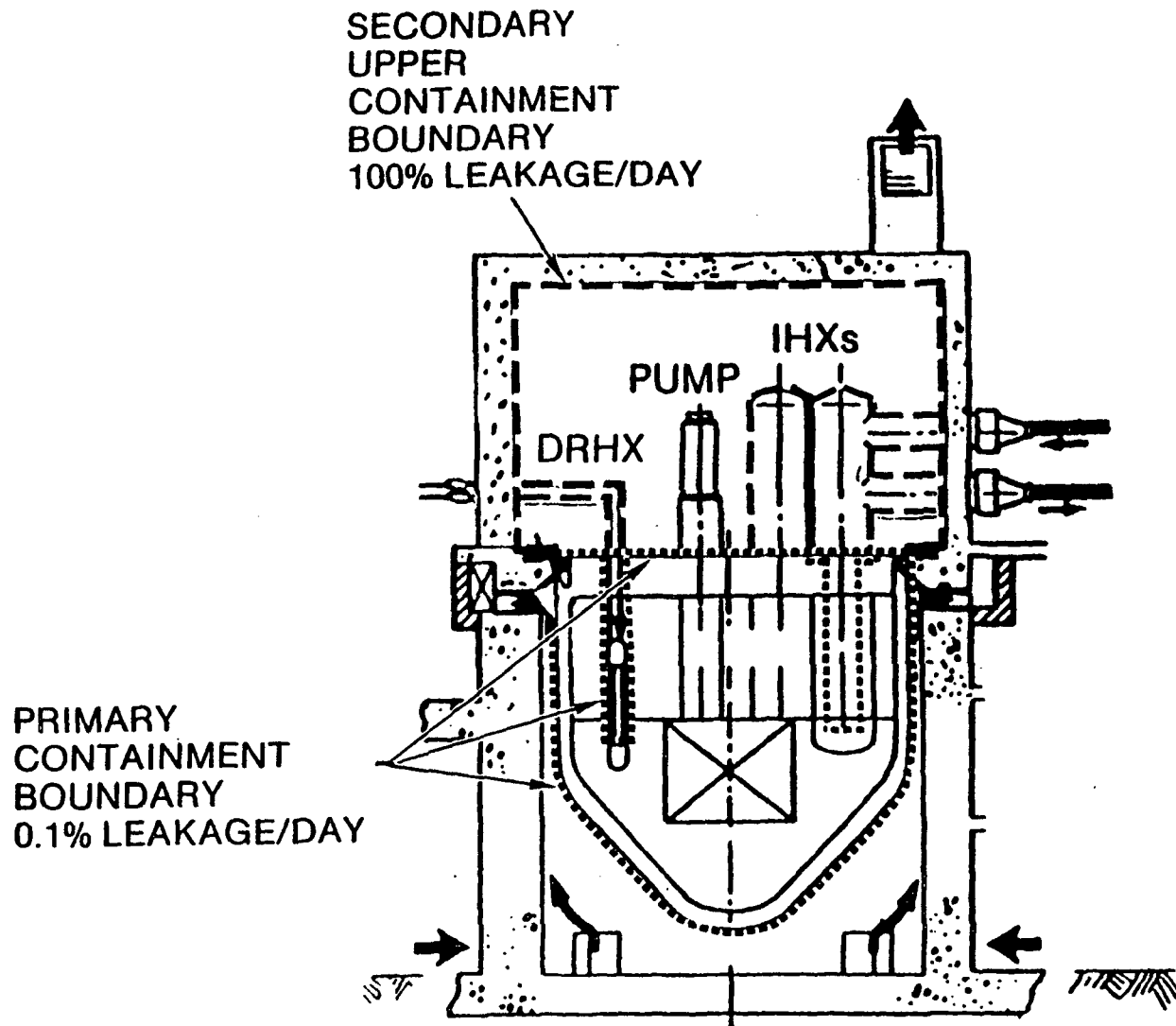


Figure 6.1 SAFR simple containment concept.

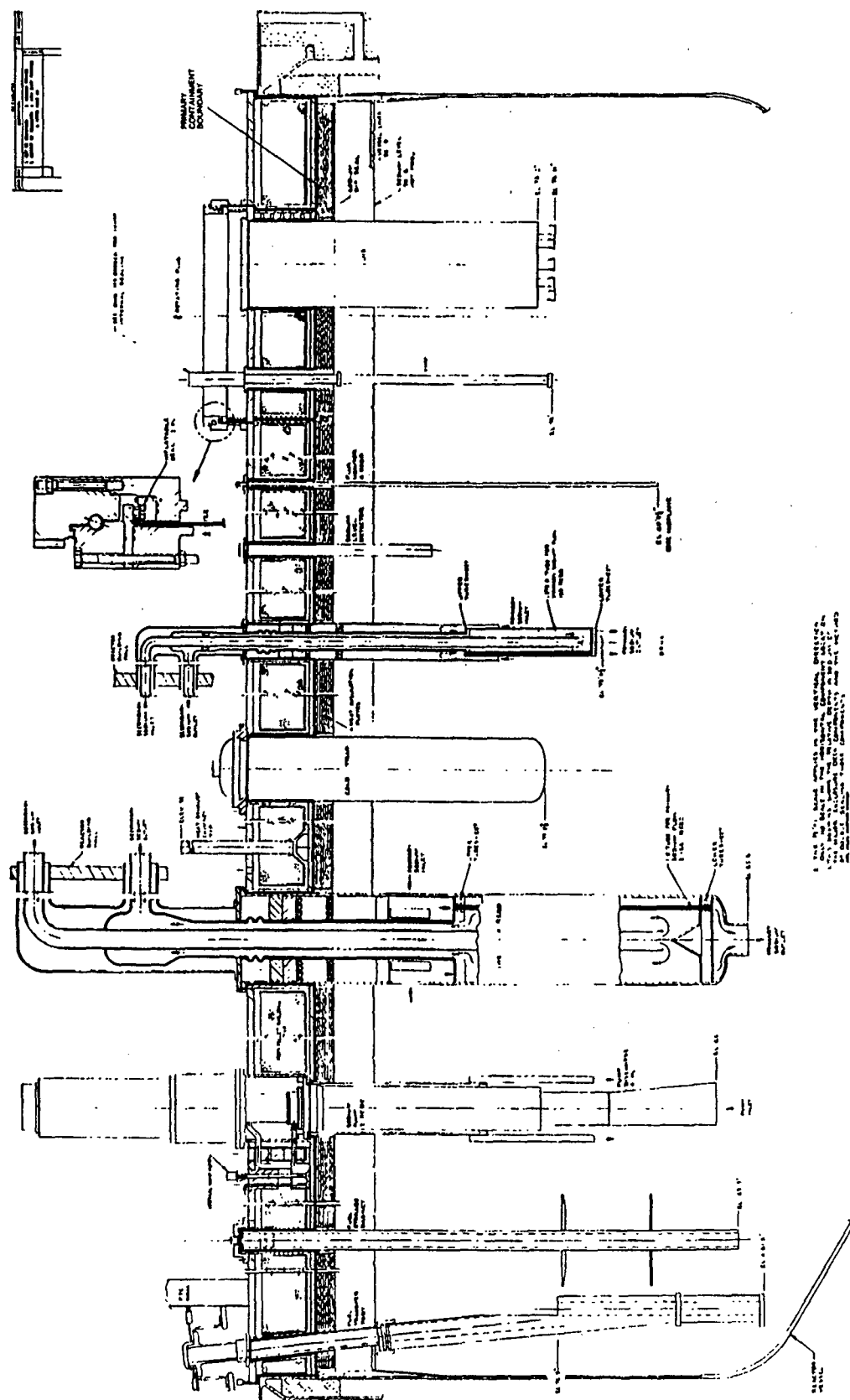


Figure 6.2 Closure head and attached components.

Engineered Safety Features

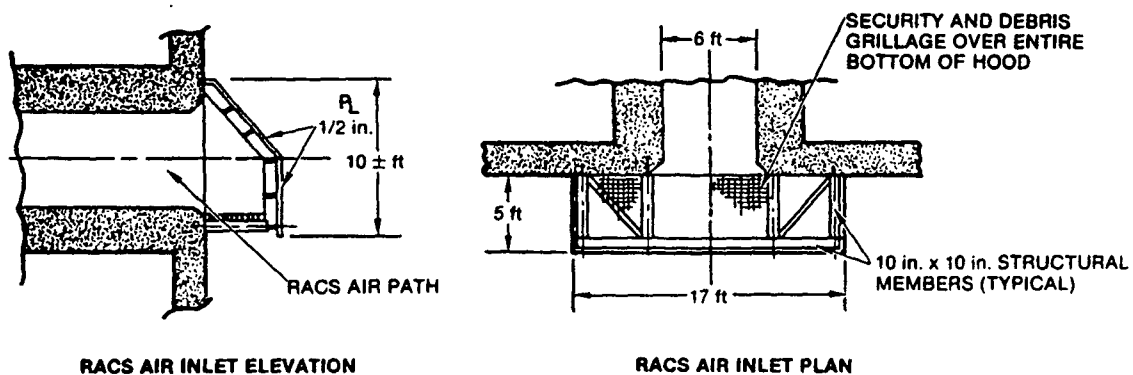
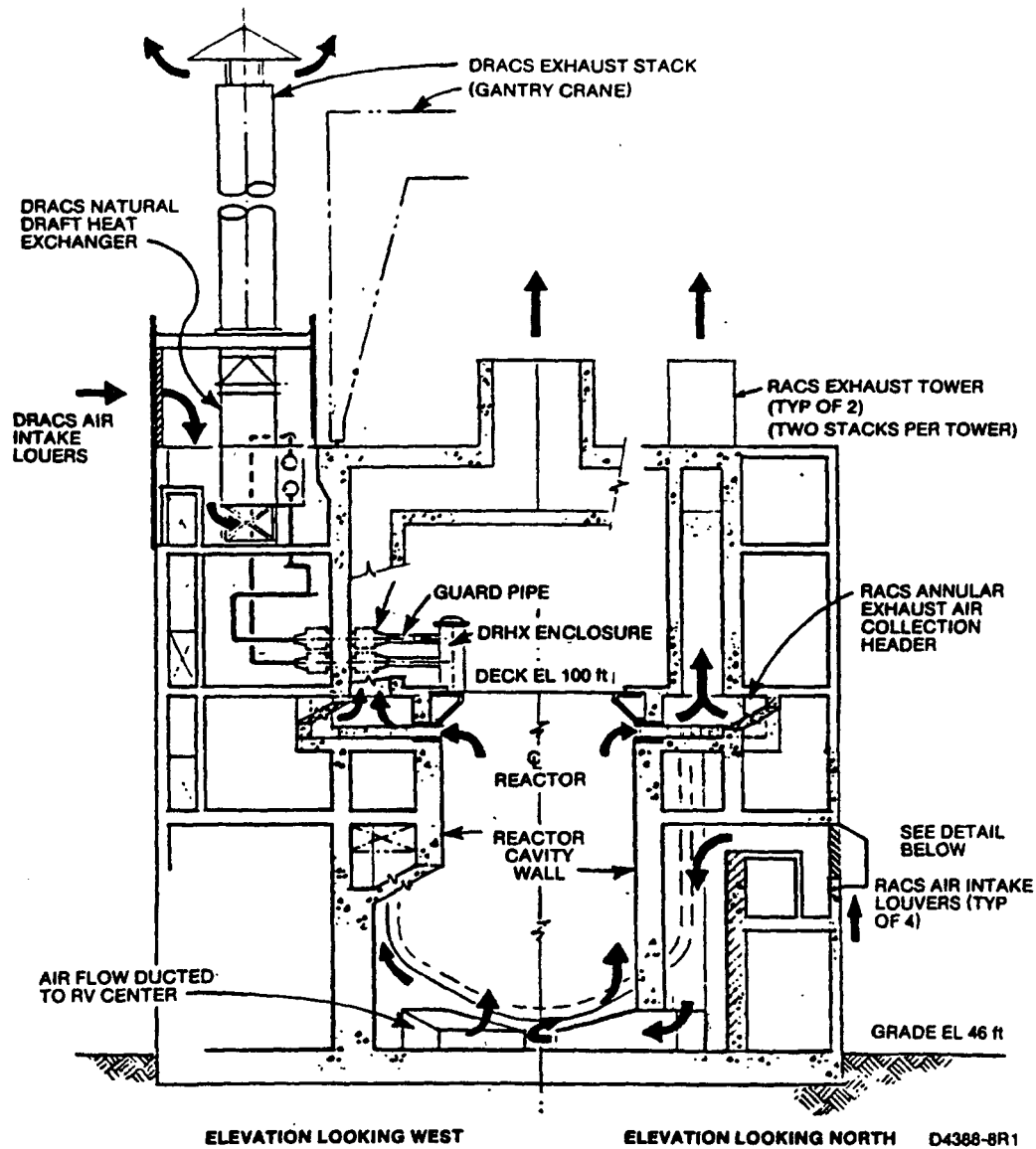


Figure 6.3 RACS airflow paths.

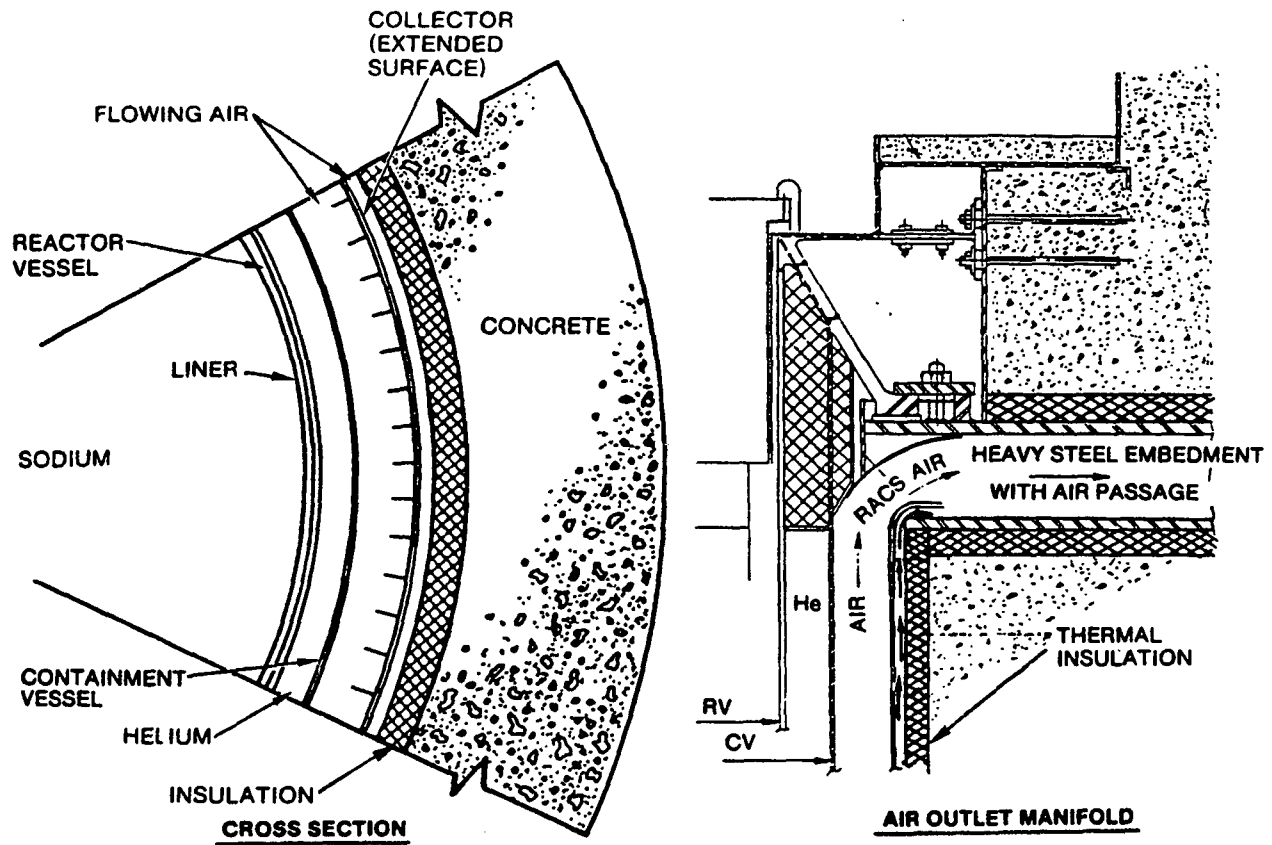


Figure 6.4 RACS/cavity cooling configuration.

7 INSTRUMENTATION AND CONTROLS

7.1 Description

The SAFR design approach places little safety demand on the instrumentation and controls (I&C) systems because of the designers' assertions that it is passively safe.

The only safety-related I&C system is that required for the containment isolation system (CIS). The I&C portion of the scram system is not classified as safety related. The post-accident monitoring instrumentation is designed to meet safety criteria; however, it is not classified as a safety-related system.

7.1.1 Scope of Review

The staff concentrated its review on the containment isolation function, the automatic plant trip system (APTS), and the post-accident monitoring functions.

7.1.2 Safety Criteria

The SAFR design identifies a number of IEEE standards and NRC regulatory guides applicable to safety-related I&C systems.

The IEEE standards include

- IEEE-279-1971, "Standard Criteria for Protection Systems for Nuclear Power Generating Stations"
- IEEE-308-1980, "Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations"
- IEEE-317-1983, "Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations"
- IEEE-323-1974, "Qualifying Class 1E Electric Equipment for Nuclear Power Generating Stations"
- IEEE-336-1985, "Power, Installation, Inspection, and Testing Requirements for Power, Instrumentation, and Control Equipment at Nuclear Facilities"
- IEEE-338-1977, "Standard Criteria for Periodic Testing of Nuclear Power Generating Station Safety Systems"
- IEEE-344-1975, "Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations"
- IEEE-352-1975, "General Principles for Reliability Analysis of Nuclear Power Generating Station Protection Systems"
- IEEE-379-1977, "Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Class 1E Systems"
- IEEE-382-1985, "Standard for Qualification of Actuators for Power-Operated Valve Assemblies With Safety-Related Functions for Nuclear Power Plants"
- IEEE-383-1974, "Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations"

Instrumentation and Controls

- IEEE-384-1981, "Standard Criteria for Independence of Class 1E Equipment and Circuits"
- IEEE-420-1982, "Standard Design and Qualification of Class 1E Control Boards, Panels, and Racks Used in Nuclear Power Generating Stations"
- IEEE-494-1974, "Standard Method for Identification of Documents Related to Class 1E Equipment and Systems for Nuclear Power Generating Station" (replacing IEEE-279)
- IEEE-603-1980, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations"
- IEEE-627-1980, "Design Qualification of Safety System Equipment Used in Nuclear Power Generating Stations"

The regulatory guides include:

- RG 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems"
- RG 1.11, "Instrument Lines Penetrating Primary Reactor Containment"
- RG 1.22, "Periodic Testing of Protection System Actuation Functions"
- RG 1.28, "Quality Assurance Program Requirements (Design and Construction)"
- RG 1.29, "Seismic Design Classification"
- RG 1.30, "Quality Assurance Program Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment"
- RG 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants"
- RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems"
- RG 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems"
- RG 1.62, "Manual Initiation of Protective Actions"
- RG 1.63, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants"
- RG 1.64, "Quality Assurance Requirements for the Design of Nuclear Power Plants"
- RG 1.73, "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants"
- RG 1.75, "Physical Independence of Electric System"
- RG 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants"
- RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident"
- RG 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants"

- RG 1.105, "Instrument Setpoints for Safety-Related Systems"
- RG 1.118, "Periodic Testing of Electric Power and Protection Systems"
- RG 1.131, "Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants"
- RG 1.151, "Instrument Sensing Lines"
- RG 1.153, "Criteria for Power, Instrumentation, and Control Portions of Safety Systems"

In addition, the staff has proposed criteria (Section 3.2) that would require the SAFR to have at least two diverse, independent means of reactor shutdown.

7.2 Reactor Shutdown System

7.2.1 Description

The SAFR safety-related shutdown or trip function is accomplished through passive means (reactivity feedbacks), as described in Chapter 4 of this SER. There is no reliance on active I&C systems.

The SAFR does include an APTS but it is not classified as safety related. See Section 7.7.3 of this SER for more information. The SAFR design includes a self-actuating shutdown system (SASS) which consists of an electromagnetic latch located between the rod drive and the shutdown rod. The electromagnet uses a temperature-sensitive (Curie point) magnetic material that loses its magnetic properties at a higher-than-normal operating temperature. Therefore, the safety rod releases automatically on reactor overtemperature. The electromagnetic latch is also actuated by the signals from the APTS which can interrupt power to the magnet.

7.2.2 Scope of Review

The staff concentrated its review on the acceptability of the APTS and SASS designs as well as their classification as not safety related.

7.2.3 Safety Issues

7.2.3.1 Safety Classification of Shutdown System

The reliance on a single passive shutdown feature does not meet present proposed staff criteria. Although such a feature may prove to be highly reliable, it does not respond to all events that may require shutdown (e.g., seismic events), nor does it provide the operator with an active safety-grade means of shutting down the reactors in response to situations which, in the operator's judgment, require fast shutdown. Since the passive shutdown characteristics only reduce the reactor power level (and do not achieve subcriticality), an active means of reactivity addition is needed to recover from inherent shutdown events.

7.2.3.2 Curie Point Magnetic Release

The self-actuating shutdown system cannot be tested on-line. This significant drawback has led the staff to conclude that SASS may be an excellent backup reactor shutdown system but that it cannot be a substitute for the automatic plant trip system, which should be classified as safety related.

7.2.3.3 Manual Scram

The SAFR designer does not propose to include a "safety related" manual scram capability. The staff believes that such a scram feature should be included and that upgrading of the APTS would make such a feature readily feasible via safety-related equipment.

7.2.4 Conclusions

Having reviewed the conceptual design presented in the PSID (Ref. 7.1), the staff believes that the SAFR designer should classify the APTS as a safety-related, automatic (with manual capability), reactor-shutdown system.

7.3 Containment Isolation System

The only engineered safety feature system with proposed safety-grade instrumentation and control features will be the containment isolation system (CIS).

The CIS will consist of radiation sensors and the supporting instrumentation, including the logic -- manual trip switches and trip breakers.

There will be four radiation sensors. The sensors will be physically separated and located on the periphery of the cover-gas piping penetrating the primary containment.

The CIS will monitor the radiation level of the primary containment cover-gas effluent, and activate the isolation system if the radiation level exceeds the trip set point. A non-CIS radiation detection system will monitor the radiation level in the secondary containment atmosphere and ventilation system exhaust ducts. The CIS monitors will be powered by the four Class 1E divisional power supplies.

Isolated radiation signals will be sent to the plant control system. Each radiation signal will also be hardwired to the post-accident monitoring system. If two of the four radiation sensors detect high radioactivity, the isolation logic will deenergize the trip coils, thereby opening circuit breakers and removing power from the containment isolation valve solenoids. When an individual pilot solenoid is deenergized, the valve will actuate, closing the isolation valve. If maintenance activities are being performed on a radiation sensor and monitor, then two-out-of-three logic can be implemented. Manual actuation to trip all containment isolation valves will be initiated at the safety-grade trip switch located at or near the CIS panels. Alternate, non-safety-grade trip switches, electrically isolated from the Class 1E CIS trip switches, will be located in the main control room (MCR) and in a remote-shutdown panel in the computer room in the control building. Individual valves can be closed through the plant control system (PCS).

The power for the monitors and to the trip breaker logic is Class 1E. Power to the pilot solenoids is not Class 1E. By design, loss of all non-Class 1E power will cause the CIS valves to close. Selected valves would only be opened for reasons of plant investment protection (e.g., to cool the secondary containment) once the power is restored and radiation levels in the primary and secondary containment are acceptable. Although power to the remote valve controls and the wiring to the containment isolation valves are not Class 1E, the valves and valve actuators are safety grade. Valve actuator pilot solenoids will be qualified to Class 1E. The provisions for on-line testing of electrical and mechanical equipment are included in the design, and comply with applicable portions of IEEE Standards 336 and 338 for inspection and testing.

Each detector will have a check source used for testing. In addition to the comparators and electronics, buffers and power supplies will be provided for each channel. Single-channel bypass will be provided. The test source will be used to test the instrument channel through the comparator. Signals will be inserted to test the logic and the opening of the respective division logic circuit. Since

two-out-of-four isolation signals are required to isolate the containment, the CIS valves will not close during the momentary test.

Channel output indications will give the operators early indication of anomalous instrumentation performance. This equipment is not safety related.

7.3.1 Safety Issues

7.3.1.1 Use of Non-Class 1E Power

The electrical power to the pilot solenoid valves of the CIS is classified as non-Class 1E power. In response to staff questions, RI has indicated that modes of power failure have been evaluated, including complete and partial loss of power. The assured safe failure position is the closed position.

The staff believes that the use of non-Class 1E power can compromise the safety function. Therefore, at a later design stage, the staff may require that further failure mode and effect analyses be performed and that the potential for failure as a result of overvoltage conditions be explored.

7.3.1.2 Lack of Diversity in Sensed Parameters

RI has stated that the CIS design will use only radiation as the sensed parameter. RI justified this approach on the basis of work at the Clinch River Breeder Reactor. The staff agrees that there are no precursor signals faster than radiation and that, therefore, signal diversity is not practical.

7.3.2 Conclusions

On the basis of its review of the conceptual design, the staff concludes that the containment isolation system can be implemented in an acceptable manner, subject to resolution of the concern on the use of non-Class 1E power.

7.4 Systems Required for Safe Shutdown

The SAFR design concept does not include any active component for safe shutdown. The staff does not concur with this position, as is discussed in Section 7.2.3.1 of this SER.

7.5 Safety-Related Display Instrumentation

Regulatory Guide 1.97 (Ref. 7.2) presents the definitions and the design and qualification requirements for instrumentation used to assess plant and environmental conditions during and after an LWR accident. Because SAFR accident scenarios differ significantly from accident scenarios for LWR plants, the specific variables used for post-accident monitoring (PAM) are not addressed by the guide.

The SAFR PAM system follows the functional definitions and the design and qualification requirements given in the guide. The specific selection of variables to be monitored follows the intent of the guide but is based specifically on the design attributes of SAFR during and following an accident.

Variables with instrumentation designed and qualified to Category 1 requirements are representative of those variables that directly indicate the accomplishment of the safety function. Reactor flux indicates that the reactor is safely shut down and is no longer generating power above decay heat values. The reactor sodium outlet temperature is the most direct means of indicating that decay heat is being removed and core temperatures are being maintained below prescribed limits.

Containment isolation status ensures that all barriers against release of radiation outside containment

have been secured. Containment area radiation measurements provide indication that barriers are maintained. This variable also serves to provide the operator with the indication to manually isolate the containment in the event the automatic isolation system fails.

Variables associated with the Category 2 requirements serve as backup information to the above variables just discussed. These variables serve to verify that the safety-grade reactor air cooling system (RACS) is functioning properly and that Class 1E power sources are available and functioning correctly. Variables associated with the Category 3 requirements are used mostly for diagnostic purposes.

7.5.1 Conclusions

On the basis of its review of the PSID (Ref. 7.1), the staff believes that the SAFR provisions for safety-related display instrumentation can be implemented in an acceptable manner.

7.6 Other Instrumentation and Control Systems Required for Safety

There are no other safety-related I&C systems presented in the PSID (Ref. 7.1). The staff believes that at least a safety-grade remote-shutdown area and a control room habitability system should be provided for SAFR. These could be reviewed in detail at a later design stage.

7.7 Instrumentation and Control Systems Not Required for Safety

The SAFR designer states that the plant control system (PCS) and auxiliary I&C systems are not safety related because their failure does not affect the operation or ability of safety systems to terminate or mitigate the results of design-basis events (DBEs).

7.7.1 Plant Control System

The PCS will provide the overall control and coordination of each power pak during plant startup, power generation, standby, and shutdown modes. During both manual and automatic operation, the PCS will provide the major operator interface with the plant. The PCS is designed to maintain power pak parameters within bounds and to minimize rates-of-temperature change in system components during both power pak steady-state and load change periods. The PCS, along with the data-handling and transmission system and the automatic plant trip system (APTS), defines what is referred to as "the total plant control and trip system."

The foundation of the PCS is a plantwide, fault-tolerant, distributed-control system. Power pak sensors and actuators are connected to panels located in the field, which are in turn addressable through the power pak data buses. Specific commands and instructions will be issued by the supervisory control system to the local control systems through the same plant data buses. Top-level commands will be issued to the supervisory control system from the control console located in the main control room.

Each of the local control systems has limited capability for local operations. This feature is primarily intended to facilitate maintenance and operational checkout. Local operations will be carried out through a hand-carried computer with the appropriate hardware and software to interface with the local input-output cabinet. Local operations will be confined to the immediate local system. At all times, the supervisory control system will maintain the capability to override local operations.

The PCS is designed so that a single failure, and often multiple failures, will not result in reduced power generation or equipment damage. This fault-tolerant aspect is designed in both levels of the PCS. At the local system level, redundant sensors are employed based on the critical importance of the measured parameter and economic considerations. Sensor signals are processed independently and then verified through a cross-check system that ensures accuracy before distribution for controls application.

7.7.2 Auxiliary Instrumentation and Control Systems

The auxiliary I&C systems include

- Fuel Handling System -- with equipment and controls for refueling and moving fuel between the power paks and the fuel cycle facility
- Primary Heat Transport -- with equipment for evaluating such reactor status as pump discharge pressures and core inlet and outlet temperatures
- Failed Fuel Element Monitoring -- with equipment to monitor the primary coolant and cover-gas for activity that indicates fuel failures
- Liquid-Metal to Gas Leak Detection -- with equipment to detect leaks from liquid-metal components into air or inert gas spaces
- Sodium-Water Reaction Detection -- with equipment to detect sodium/water reactions (Two diverse means are to be provided -- high-frequency acoustic leak-detection and sodium chemical composition changes.)

7.7.3 Automatic Plant Trip System

The SAFR design includes a non-safety-grade reactor-shutdown system to protect the reactor investment.

Although the automatic plant trip system (APTS) is classified as not safety related, it includes the design principles (such as redundancy and independence) of a typical safety-related trip system.

The APTS consists of four identical microprocessor-based subsystems. These four subsystems will be powered by redundant non-Class 1E battery-backed power. Each parameter will be measured by four identical sensors located as close to the same physical location as possible. Each sensor will be connected to one of the divisional subsystems. Each APTS division will have three signals as inputs:

- reactor flux
- reactor flow (one for each pump)
- core inlet temperature.

Each APTS division will develop five trips:

- high reactor flux
- reactor flux to delayed flux ratio
- flux to flow ratio
- high core inlet temperature
- low core inlet temperature

Sensor signals will be hard-wired to the input-output modules of the data processing equipment. Measurements will be time tagged and written into buffered memory in real time, using directed-memory access techniques. Measurements from each sensor will be validated, any required calculations will be performed, and the results will be compared with a trip-point setting which, if exceeded, will set a trip flag for that limit. This trip signal will then be transmitted to each of the other divisions and voting will take place. The logic will be programmed to produce a trip demand signal if it detects trip signals on two-out-of-four inputs for that trip limit. Each logic unit will provide input to dual-auctioneered power supplies for the trip breakers. The output of each trip breaker power supply will pass through the last element in each logic train -- the manual trip switch. The manual trip switches will mechanically break and latch open the power to all trip breakers.

Instrumentation and Controls

These switches will be located at the operator's console in the main control room and in the computer room. The switches are designed so that individual contacts (one in each trip breaker power supply output) can be opened independently for testing.

Eight primary and eight secondary trip breakers will be arranged in a two-out-of-four logic configuration. These breakers will be closed when receiving current from the dual power supplies and will hold closed spring-loaded contacts that pass electric power to the primary and secondary control rods. If power is cut off to the primary and/or secondary control rods, the affected rods will fall into the core.

A signal will be sent to the plant control system (PCS) on a reactor trip. The PCS will take action to shut down the sodium pumps, steam generators, and turbines when the reactor is tripped.

7.7.4 Safety Issues

APTS A very low unreliability of 8.1×10^{-6} per demand is claimed for the APTS. Although the system will have attributes of a typical reactor trip system, the staff does not believe that such unreliability is attainable with a non-safety-related system. This fact, coupled with the inability to test the SASS on line (see Section 7.2 of this SER) and the need for a safety-related manual scram (see Section 7.2 of this SER) has led the staff to believe that the APTS should be upgraded to a safety-related system.

Auxiliary I&C Systems The staff finds that the failed fuel monitoring, liquid-metal to gas leak detection system and sodium/water-reaction detection system should be considered safety-grade systems because they are so important in ensuring certain key assumptions in the design are, in fact, maintained. Specifically,

- The failed fuel monitoring system is a line of defense against fuel failure propagation leading to a large-scale, core-damage accident by rapidly detecting fission products and allowing operators to shut down the reactor.
- The liquid-metal to gas leak-detection system is key to the staff's acceptance of the leak-before-break (in lieu of designing for a double-ended pipe rupture) position on the SAFR, which allows designing for more-benign sodium leakage.
- The sodium/water reaction detection system is key to detecting small steam generator tube leaks and allowing action (e.g., steam/water dump) before a larger leak develops.

7.7.5 Conclusions

Changes are needed in the SAFR I&C area to result in an acceptable design. The staff has not completed its review of the non-safety-related control systems because of its position that the APTS and certain other systems should be upgraded to safety related. Such changes could have an impact on the overall review of this area.

7.8 References

- 7.1 DOE, "SAFR - Sodium Advanced Fast Reactor," Preliminary Safety Information Document, AI-DOE-13527, Initial Issue - October 1985, with amendments. (DOE Applied Technology.)
- 7.2 NRC, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Regulatory Guide 1.97, Rev. 3, May 1983.

8 ELECTRICAL POWER SYSTEMS

8.1 Overall Electrical Systems

8.1.1 Description

The electrical power system includes onsite and offsite sources of electrical power. The system supplies power during all modes of operation, including normal operation and emergency shutdown.

The offsite power supplies are physically independent to minimize the probability of simultaneous loss of the sources.

Onsite electrical power sources include the main turbine generators, non-Class 1E auxiliary generators, and both Class 1E and non-Class 1E station batteries. The Class 1E station batteries supply power to Class 1E systems to isolate the containment and to facilitate post-accident monitoring (PAM).

8.1.2 Scope of Review

The review focuses on the Class 1E power systems. Non Class-1E systems are reviewed to verify that they will not degrade the Class 1E system.

8.1.3 Design Criteria

The SAFR designer identified a number of industry standards and regulatory guides which are to be applied to the Class 1E power systems. The regulatory guides include

- RG 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Their Distribution Systems"
- RG 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants"
- RG 1.47, "Bypassed and Inoperable Status Indications for Nuclear Power Plant Safety Systems"
- RG 1.63, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants"
- RG 1.75, "Physical Independence of Electrical Systems"
- RG 1.118, "Periodic Testing of Electric Power and Protection Systems"
- RG 1.128, "Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants"
- RG 1.129, "Maintenance, Testing, Replacement of Large Lead Storage Batteries for Nuclear Power Plants"

The industry standards include

- National Electrical Manufacturers Association (NEMA) PB-1 for vital ac instrument panels
- IEEE-450 and NEMA IB-4 for Class 1E batteries, and ANSI Standard C34.2 for Class 1E battery chargers, inverters, and voltage regulators
- IEEE-308-1980, "Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations"

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- IEEE-317-1983, "Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations"
- IEEE-323-1974, "Qualifying Class 1E Electric Equipment for Nuclear Power Generating Stations"
- IEEE-336-1985, "Installation, Inspection, and Testing Requirements for Power, Instrumentation, and Control Equipment at Nuclear Facilities"
- IEEE-338-1977, "Criteria for the Periodic Testing of Nuclear Power Generating Station Safety Systems"
- IEEE-344-1975, "Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations"
- IEEE-382-1985, "Standard for Qualification of Actuators for Power Operated Valve Assemblies with Safety-Related Functions for Nuclear Power Plants"
- IEEE-383-1974, "Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Station"
- IEEE-420-1982, "Standard Design and Qualification of Class 1E Control Boards, Panels and Racks Used in Nuclear Power Generating Stations"
- IEEE-603-1980, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations"
- IEEE-627-1980, "Design Qualification of Safety System Equipment Used in Nuclear Power Generating Stations"

8.1.4 Conclusions

The staff believes that the PSID (Ref. 8.1) conceptual description of the electrical system design and its associated design criteria are acceptable. Its capacity and safety-grade loads will need to be reevaluated at a later design stage in view of the fact that additional systems may be designated safety grade (e.g., see Section 7.7.4 of this SER).

8.2 Offsite Power System

8.2.1 Description

The plant will be connected to the utility grid through physically separated and electrically independent offsite power circuits. The offsite circuits will be capable of supplying the plant auxiliary loads to facilitate all modes of operation.

Physical and electrical separation of transmission lines, transmission towers, transformers, switchyard components, and controls will minimize the possibility of simultaneous failure of the offsite power supplies.

The offsite power system will have no safety-related function. Failure of this system will not compromise any safety-related system or component, nor will it prevent a safe shutdown of the plant.

8.3 Onsite Power Systems

8.3.1 Description

The onsite electrical power system will include the following subsystems:

- ac power distribution system (non-Class 1E)
- Class 1E uninterruptible power supply (UPS) system
- Class 1E dc system
- non-Class 1E dc and instrument ac systems
- grounding and lightning system

8.3.1.1 ac Power Distribution System (non-Class 1E)

For each power pak, power will normally be supplied to the unit auxiliaries from the turbine generator through the auxiliary power transformers. When the generator breaker is opened, auxiliary power will be supplied from an offsite source through the generator stepup transformer and unit auxiliary transformers. If the primary source is not available, power will be supplied to the unit auxiliaries from a secondary offsite power source through the common station service transformers.

The ac power distribution system will be composed of a 7.2-kV, medium-voltage, three-phase system and a 480-V, three-phase, low-voltage distribution system. Loads greater than 300 hp will be served from the 7.2-kV system and loads less than 250 hp will be served from the 480-V system. Small single-phase loads will be supplied from 277-V or 120-V ac systems. The control rod drive system will be supplied from a reliable power system, which will be supported from the 480-V ac power distribution system and includes both primary and backup power supply feeders.

Auxiliary ac power will be available from two onsite electrically independent and physically separated auxiliary generators. These generators will furnish power to selected loads important to plant investment protection. Electrical separation and independence will be maintained within the auxiliary power system to ensure high availability and reliability.

8.3.1.2 Class 1E Uninterruptible Power System

The Class 1E uninterruptible power system (UPS) will provide 120-V ac, single-phase power to vital Class 1E control and instrumentation loads for each power pak. Each UPS will include four independent 120-V ac instrument buses and associated equipment. The UPS will supply power to the four channels (channels A, B, C, and D) of the primary heat transfer system (PHTS) instrumentation, flux monitoring system, and containment isolation system (CIS) radiation monitors.

The only active emergency safety feature function included in these onsite power systems is isolation of the containment. The remaining requirements for Class 1E dc power will all be to satisfy post-accident monitoring (PAM) requirements of Regulatory Guide 1.97 (Ref. 8.2).

Each 120-V ac bus is designed in accordance with the requirements of IEEE Standard 308. Power will be provided for each bus from a Class 1E, 125-V dc system through a dedicated inverter. If the inverter supply is not available, a static transfer switch will automatically transfer the vital loads to an alternate power supply. This alternate power supply will provide regulated ac power from the 480-V ac power system through a dedicated regulating 480/120-V ac transformer. A manual bypass switch will allow the static transfer switch to be isolated.

8.3.1.3 Class 1E dc System

The Class 1E dc system will provide 125-V dc power for safety-related I&C circuits and will serve as the normal power source for the Class 1E 120-V UPS for each power pak. The system will include four independent 125-V dc busses and associated equipment per power pak.

Each 125-V dc bus is designed to meet the requirements of IEEE Standard 308. Each bus (channel) will have adequate capacity to start and operate the loads. The system is a two-wire, ungrounded system. Each bus will be connected to a dedicated battery supply and to a dedicated battery charger. Each full battery will be capable of supplying full load for at least 8 hours. Each battery charger will have adequate capacity to supply the steady-state loads while recharging its associated battery to its required load capacity within 12 hours. A standby battery charger will also be provided for each bus. The battery chargers will be interlocked to ensure that only one charger is connected to the bus at any time. Monitoring and other devices will protect the system from faults. A permanently installed, redundant, Class 1E wiring system will connect the Class 1E dc system to a location from which a portable ac generator can be connected to recharge the batteries.

8.3.1.4 Non-Class 1E dc and ac Instrument Systems

A non-Class 1E 125-V dc system will be provided for each power pak to supply power to non-Class 1E control circuits and turbine dc oil pumps. The system will consist of two separate 125-V dc busses and associated equipment. Each bus will be connected to a dedicated battery supply and a battery charger. A standby battery charger will also be provided. The battery chargers will be interlocked to ensure that only one charger is connected to the bus at any time.

A 120-V ac instrument system will be provided for each power pak. This system will supply regulated power to instrumentation and control circuits. The system will consist of two separate 120-V ac busses and associated equipment.

A common non-Class 1E UPS will supply reliable power for I&C circuits that are important to the continued operation of the total plant. The system will include two separate 120-V ac instrument busses and associated equipment.

8.3.1.5 Grounding and Lightning System

A grounding system will ground equipment and serve as a return path for fault currents. This system will ensure personal safety in that all parts of metal structures, switchgear enclosures, motor frames, and raceways will be at ground potential.

A lightning protection system will protect the plant against the effects of lightning strikes. The lightning protection system will be connected to the plant grounding grid.

8.3.2 Safety Issues

8.3.2.1 Non-Class 1E Loads on Class 1E Power Supplies

In response to staff questions regarding loads on Class 1E busses, RI clarified that no non-Class 1E loads will be allowed on the Class 1E supplies, nor will they be connected in any way to the vital ac distribution system. The Class 1E power system loads will be restricted to the CIS and PAM loads.

8.3.2.2 Pump Coastdown

The staff noted that some of the Chapter 15 events in the PSID appeared to take credit for "pump coastdown," yet there was no safety-related electrical supply requirement to ensure coastdown.

In response, RI noted that the solid-state speed controllers are so designed that fluctuations in both frequency or voltage at the input to the controller will not affect pump coastdown. The incoming ac power will be converted to dc before it is conditioned for powering the pump motor. The speed controller will shut off the voltage output if the ac input voltage falls below levels required to support the required internal dc voltages. The ac frequency fluctuations will not affect the dc conversion or speed controller performance in general. RI agreed to analyze the effect of internal failures within the speed controller using failure modes and effects analyses (FMEA) techniques. RI also stated that design features will be added as required to eliminate potential impacts to the pump coastdown. The staff finds this acceptable for the conceptual design phase; the more-detailed FMEA will be reviewed at a later design phase.

8.3.2.3 Capacity for Class 1E Loads

At a later design stage, the capacity of the Class 1E power supply system should be reevaluated based upon final safety classification of systems in the plant.

8.4 Conclusions

On the basis of its review, the staff concludes that RI has described a conceptual electrical system that can be transformed into a design which will meet NRC requirements.

8.5 References

- 8.1 DOE, "SAFR - Sodium Advanced Fast Reactor," Preliminary Safety Information Document, AI-DOE-13527, Initial Issue, October 1985, with amendments. (DOE Applied Technology.)
- 8.2 NRC, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident," Regulatory Guide 1.97, Rev. 3, May 1983.

9 AUXILIARY SYSTEMS

Chapter 9 of the PSID (Ref. 9.1) provides preliminary information on the conceptual design of the following auxiliary systems:

- fuel storage and handling system
- water systems
- process auxiliary systems
- heating, ventilation, and air conditioning (HVAC) system
- sodium purification system
- fire protection system
- communication systems
- lighting systems
- gas turbine-generator auxiliary systems
- overhead load-handling system
- habitability system

Some of these systems are not well defined at this point. For example, the designer has deferred defining the auxiliary systems for the auxiliary generators. Other systems (communication, lighting) are only briefly described.

The staff has selected systems unique to LMR technology for consideration in its review at this stage. The other systems will be reviewed at a later design stage. The staff will review

- fuel storage and handling system
- primary sodium processing system
- intermediate sodium processing system
- fire-protection system
- overhead heavy-load-handling system

9.1 Fuel Storage and Handling System

9.1.1 Design Description and Safety Objectives

The fuel-handling system includes the systems and equipment necessary to receive, store, and transport fuel, and blanket, control, and removable radial shield assemblies.

The systems are

- the in-vessel handling machine (IVHM)
- the in-vessel fuel storage basket to store assemblies within the vessel
- the fuel-handling cell (FHC) to provide space for shielding and temporarily storing new and spent fuel
- the fuel-transfer cell (FTC) to provide space for transferring assemblies between the reactor vessel and the fuel-handling system
- the fuel-handling machine (FHM) to move fuel within the FHC
- the fuel transporter to move assemblies -- in sodium-filled core component pots (CCPs) between the fuel cycle facility (FCF) and the FHC

The last item in the list identifies the fuel cycle facility as a component of the fuel-handling system. The SAFR design is based on the integral fast reactor (IFR) metal-fuel cycle that can incorporate onsite fuel reprocessing and fabrication. The FCF would incorporate the reprocessing and fabrication equipment. This aspect of the design was not submitted for NRC review. The PSID (Ref. 9.1) noted, however, that all core assemblies will be processed through the FCF before being moved (via the rail transporter) to a power pak.

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A simplified sketch of the refueling operation is shown in Figure 9.1. The figure shows the general arrangement of the major fuel-handling and storage equipment in the reactor containment building (RCB). All core assemblies will be moved in sodium-filled CCPs via the transporter. The transporter will be sealed to the bottom of the FHC. The FHM will remove the new core assembly, placing it in interim storage, and will set a spent core assembly in the transporter. The new core assemblies will be stored for reactor refueling. After reactor shutdown, the spent fuel assemblies, blanket assemblies, and control assemblies, having been held in the in-vessel storage basket from the refueling function conducted during the previous year, will be exchanged for new core assemblies. This will be done by using the IVHM, the FTC A-frame, and the FHC FHM during control rod disconnect before the rotatable plug can be moved. The reactor will be refueled by exchanging new and spent core assemblies between the in-vessel storage basket and the core on a one-for-one basis with the IVHM.

Interim Storage

The new fuel assemblies and spent fuel assemblies will both be stored in their sodium-filled CCPs in the same thimble array within the interim storage (IS) area within the FHC. The thimbles will be mounted in an enclosed steel structure designed to ensure a subcritical configuration and to allow natural circulation cooling in and around the thimbles to remove decay heat.

In-Vessel Storage

All spent core assemblies (fuel, blanket, and control) will be stored and allowed to decay in the in-vessel fuel storage basket for 1 year following their removal from the reactor core before being moved to the FHC interim dry storage and the FCF for ultimate processing. One-fourth of the core will be refueled during each annual outage; the actual refueling period is estimated at 6.4 days.

The in-vessel storage basket will be located in a hot-pool storage well, adjacent to the reactor fuel transfer position. The fuel will be stored at the same elevation as in the reactor. The spent fuel will be cooled by natural circulation of primary sodium. The storage thimbles will allow reactor sodium to flow around the spent assembly and then to circulate upward through the assembly, discharging to the upper hot pool.

In-Vessel Handling Machine

The in-vessel handling machine (IVHM) is an under-the-shield refueling machine that will be mounted on the rotatable plug of the reactor head. The primary function of the IVHM is to move core assemblies between their normal positions in the reactor core, the in-vessel storage basket, and the transfer position outside the core, but within the reactor vessel. In the transfer position, the core assemblies will be accessible for transfer into or out of the reactor vessel by the FTC A-frame bucket. The IVHM will execute vertical movements, while horizontal movements are being performed by combined rotation of the rotatable plug in the reactor head and the IVHM arm. The machine is depicted in Figure 9.2. The IVHM is based on a concept that has been tested and used at the FFTF for several years. Major subassemblies (e.g., grapple and core assembly identification system) are very similar to comparable subassemblies of the FFTF.

Fuel-Transfer Cell A-Frame -- Fuel-Transfer Machine

The fuel-transfer cell (FTC) A-frame is the fuel-transfer machine that will move new or spent core assemblies from the reactor transfer position to the FHC transfer position. The assembly is basically a two-position bucket and bail on wheels with two sets of fixed guide tracks and one movable or swing guide track between the two fixed sets. The bucket will be hoisted from the reactor transfer position to the apex of the swing track; the track will be moved to the second position and the bucket will be lowered to the FHC transfer position. The general arrangement of the A-frame is shown in Figure 9.3. The bucket will contain two CCPs filled with sodium. The bucket will be attached to

the bail that will be used to assist in righting the bucket. The assembly will be hoisted by redundant hoist chains attached to both the bail and hoist drum. To elevate the bucket, the chains will wrap around the drum in reverse spirals. Sodium in the bucket will remove decay heat from a spent fuel core assembly. Core assembly shielding, cell leaktightness, and inert-gas processing will be provided by other systems.

The fuel-transfer machine is based on the fuel-handling concept being used and developed in the United States and in other countries (e.g., France, Germany, Russia, England, and Japan). The U.S. concept for the large-scale prototype breeder has been under development in the liquid-metal fast-breeder reactor (LMFBR) base program for several years. The tests have included determination of g-loads on a dummy fuel assembly to find optimum operating speeds for the FTM to protect the core assemblies from mechanical damage during transfer from the reactor transfer position to the FHC transfer position.

Thermal analysis of a fuel assembly in the bucket is based on the analyses and tests performed on the CRBRP fuel assembly in the CCP when the assembly is being transported in the ex-vessel transfer machine. Testing, including a full-scale FFTF/CRBRP thermal test at Hanford Engineering Development Laboratory (HEDL), resulted in the analytical thermal model used to predict thermal performance of the SAFR system.

Fuel-Transporter/Fuel-Handling Machine

The fuel-transporter/fuel-handling machine will move the fuel from the FHC to the FCF. The transporter is a top-loading, single-barrel, shielded, and sealable handling device. It will be mounted on rails for ease of handling. The transporter concept is illustrated in Figure 9.4.

Axial and radial shielding will be provided in the transporter to limit the dose rate to less than the criteria given in PSID Sections 12.1.1 and 12.1.2 at the surface of the cask body. Approximately 10 inches of lead and 4 inches of borated polyethylene will serve as shielding. The pressure boundary of the transporter will be sealed by elastomer seals. The cask top will be adequately sealed to prevent excessive radioactive emissions from escaping into the environment. Radioactivity released from the transporter will not exceed the limits given in Section 12.1 of this SER.

The decay heat of the spent core assemblies transported will be approximately 2 kW. At this value, based on the FFTF full-scale tests, the majority of the sodium in the CCP will be frozen. Cooling will be achieved by conduction through the sodium to the cask. These solidified sodium heaps prevent the core assemblies from incurring mechanical damage.

Thermal analysis indicates an adequate thermal dissipation capacity for safely transporting the spent core assemblies. Assessment of the wheel footprint of the transporter during an OBE and SSE indicates the structure will not overturn. When mated to the FHC or FCF, stabilizers will prevent mechanical damage during a seismic event.

Refueling Procedure

For refueling, the reactor will be shut down and the sodium temperature will be reduced to the refueling temperature of 450°F. The control rod drivelines will be disconnected from the control assemblies that will be fully inserted in the core. The drivelines will then be raised to clear the parting plane to allow reactor head plug rotation. Concurrently, the reactor cover-gas will be purged and purified to reduce radioactivity levels in the gas to a very low level.

The sequence for handling fuel will begin with the one-at-a-time receipt from FCF of new core assemblies in sodium-filled CCPs in the transporter. The transporter will be mated to the underside of the FHC. After the doors sealing the transporter and FHC are opened, the FHM will grapple the CCP, remove the new core assembly in the CCP, and store that CCP in a vacant thimble storage

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position in the FHC interim storage area. The FHM will then grapple a CCP containing a spent core assembly and reload the transporter with the spent assembly. The transporter and FHC will then be released and the transporter will be de-mated from the FHC. This cycle will continue until all spent fuel from the previous refueling is exchanged with new fuel. The next refueling will be able to proceed independent of reactor operating status.

9.1.2 Scope of Review

The staff has compared the description of the conceptual design in the PSID (Ref. 9.1) with the acceptance criteria given in SRP Section 4.3 "Nuclear Design" and SRP Sections 9.1.1 through 9.1.5, "Auxiliary Systems (for Fuel)" (Ref. 9.2) and Regulatory Guide 3.15, (Ref. 9.3). The staff focused its review on the acceptability of the conceptual design with regard to maintaining fuel element integrity, fuel element cooling, and subcriticality. These functions related to meeting the dose limits per 10 CFR Part 20. No independent calculations were performed.

9.1.3 Review and Design Criteria

The applicable review and design criteria are the acceptance criteria of SRP Sections 9.1.1 through 9.1.5. These sections of the SRP specify acceptance criteria for new-fuel storage, spent-fuel storage, spent-fuel cooling, the handling of light loads, and the handling of heavy loads. The design descriptions in a PSAR should be written in a form consistent with the SRP.

Criticality is prevented in stored fuel by the use of physical systems or processes utilizing geometrically safe configurations. The effectiveness of such physical systems, processes, and configurations in preventing criticality will be based on the use of demonstrated techniques and validated analysis in accordance with the guidance of Regulatory Guide 3.15 (Ref. 9.3) and of ANSI/ANS-8.1, "Nuclear Criticality Safety in Operations With Fissionable Materials Outside Reactors."

The DOE has committed to meet the intent of GDC 62 ("Prevention of criticality in fuel handling and storage") and Regulatory Guide 1.25, (Ref. 9.4).

9.1.4 Research and Development Needs

Analytical methods used for criticality safety studies and design need to be validated against experimental data as prescribed in Regulatory Guide 3.15 (Ref. 9.3) and ANSI/ANS-8.1. Additionally, experimental confirmation of the accuracy and uncertainty of projected decay heating rates of the LMR fuel should be addressed together with validation of the heat-removal capabilities for the various conditions and configurations associated with handling of the core assemblies.

9.1.5 Safety Issues

The next design stage should provide additional design description and performance analysis of the fuel-handling and storage system. A detailed engineering design description is expected that would demonstrate the integrity and leaktightness of the systems. Experimental confirmation is required for the analysis of subcriticality and decay heat loads.

9.1.6 Conclusions

The PSID (Ref. 9.1) indicates that the proposed design will be based on proven technology. The design of the fuel-handling and storage system is conditionally acceptable. Additional detailed design descriptions and safety analyses would have to be documented in the next design stage with experimental confirmation where appropriate.

9.2 Sodium Auxiliary System

9.2.1 Design Description and Safety Objectives

The sodium auxiliary system comprises three subsystems: the sodium-receiving system, and the primary and intermediate sodium-processing systems.

Sodium-Receiving System

This system will provide the capability to receive and melt fresh, solidified sodium, delivered to the site in tank cars or drums, and to transfer the sodium to the reactor vessel and the intermediate sodium storage vessel. System capacity is based on the meltout and draining of an 80,000-lb-capacity tank car in 16 hours.

This system will consist of a tank car, oil-heating station for meltout of sodium tank cars, a clamshell heater for melting drums of sodium, transfer piping and valves, and filters for cleaning up fresh sodium. All liquid-metal piping, valves, and filters of the sodium-receiving system will be constructed of Type 304 stainless steel.

Primary Sodium Processing System

This system will purify primary sodium (cold trapping). The primary sodium processing system will limit the oxygen content to 2 ppm (by forming sodium oxide in the cold trap) and the hydrogen content to 0.2 ppm (by forming sodium hydride), and will maintain the tritium content within limits that satisfy plant radiological release criteria.

The principal component of this system is the primary sodium cold trap. The trap is rated at 50 gpm at the normal sodium system operating temperature. This size is sufficient to limit oxygen and hydrogen to the specified limits.

The primary cold-trap assembly will operate submerged in the reactor vessel. The unit will be suspended in a standpipe filled with cold pool sodium, as shown in Figure 9.5. Inlet sodium will be supplied by a line from the inlet plenum below the reactor core; pressure will be provided by operation of the main primary heat transport system (PHTS) pumps. The cold trap will be located in the gas-filled support shell. Sodium will be pumped to the top of the cold trap (economizer). Here, the main stream will flow down through the economizer. A side stream will flow down through the plugging temperature indicator and will discharge to the cold pool inside the standpipe. The main stream will flow through the mesh-filled outer annulus of the crystallizer where most of the trapping will be accomplished, and then back up through the mesh-filled center region. Sodium will be cooled to about 250°F at the bottom of the trap. Sodium will flow through the economizer into a pipe that will discharge to the cold pool in the standpipe. The crystallizer will be cooled by helium which will flow through the head down to the economizer level where a side stream will cool the plugging temperature indicator. The main helium stream will flow to the bottom of the crystallizer where the gas will feed into a cooling annulus outside the crystallizer. Helium will be collected at the top of the crystallizer where another pipe leads up to the head.

Analysis of Loss of Cold-Trap Cooling

If cooling is interrupted on the PHTS cold trap, and sodium flow continues, the sodium temperature in the crystallizer will rise rapidly and approach the inlet temperature, which will have risen to 675°F. The consequence is the dissolution of the solid sodium oxide (Na_2O) and solid sodium hydride (NaH). A high sodium temperature alarm will alert the operator to a loss of cooling and will also automatically stop sodium inlet flow by closing the sodium inlet control valve. This action will immediately and automatically prevent a release of redissolved impurities into the primary sodium.

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The following analysis discusses the potential for such a release in the event that automatic isolation fails and no operator action is taken.

If the automatic cold-trap sodium isolation scheme fails to function, the time available to take corrective action (e.g., to shut off sodium flow into the cold trap) will depend on the rate of dissolution. Calculations have previously been made of the time required to dissolve the entire contents of the PHTS cold trap as a function of the sodium temperature in the crystallizer. The results indicate that, following loss of cold-trap cooling and failure of automatic cold-trap sodium isolation, several hours will be available for the operators to take corrective action before enough oxide or hydride can be dissolved from the cold trap to raise their concentrations in the PHTS to saturation levels. Assuming that the cold-trap flow is stopped within hours after a loss of cooling, no potential for plugging any part of the PHTS exists. Consequently, operation of the cold-trap helium-cooling system is not a safety function and the system is classified as non-safety grade. However, the primary cold traps will be connected to the reactor coolant boundary and will be part of the containment boundary. Therefore, the cold-trap sodium boundary will be classified, designed, and constructed as a Safety Class 2 component.

Intermediate Sodium-Processing System

The intermediate sodium-processing system will purify the sodium in each of the IHTS loops. Each of the IHTS loops will have a separate purification system consisting of a pump, two cold traps (one shared), and the necessary valves and piping. A single trap per IHTS loop is sufficient to remove anticipated oxygen and hydrogen in-leakage and to limit these impurities to a maximum of 2 and 0.2 ppm, respectively. The IHTS cold trap will serve for about 2 years before a regeneration process is required.

The intermediate sodium-processing system is a nonradioactive system. A leakage in a purification circuit is considered the most significant event, because it ultimately causes loss of the IHTS loop to which it is connected. The loss of fluid would be signaled by the level indicators on the IHTS loop expansion tank, and operators could immediately isolate the purification circuit (remote-controlled isolation valves). This event would cause loss of one IHTS loop because of the inability to maintain required purity levels, but the plant could continue to operate on the other loop at a reduced power level. Power outage to all pumps, or loss of cooling to all cold traps, may cause plant shutdown, but does not constitute a safety problem.

9.2.2 Scope of Review

The staff focused its review on the acceptability of the conceptual design with regard to meeting the radionuclide control criteria of 10 CFR Part 100 and the dose limits of 10 CFR Part 20 and 10 CFR Part 50, Appendix I. No independent calculations were performed.

The staff has attempted to compare the brief description of the sodium auxiliary system given in Section 9.5 of the PSID (Ref. 9.1) with acceptance criteria for systems that might be considered to be functionally equivalent in LWRs as described in appropriate sections of the Standard Review Plan (SRP, Ref. 9.2). SRP Section 9.3.4, "Chemical and Volume Control System (CVCS)," is applicable and SRP Sections 9.2.2, "Reactor Auxiliary Cooling Water Systems"; 6.5.3, "Fission Product Control System and Structures"; and 5.4.8, "Reactor Water Cleanup System (BWR)," should be considered for applicability.

9.2.3 Review and Design Criteria

The staff has not performed, at this time, a detailed assessment of the proposed sodium auxiliary system against the set of acceptance criteria provided in the SRP for potentially equivalent functions in light-water reactors. The design descriptions in the next design stage should be developed in a

form consistent with the functional intent of the SRP as it applies to primary coolant cleanup systems and the primary coolant boundary integrity.

At a later review stage, the staff must determine if criteria in SRP Sections 9.3.4, 9.2.2, 6.5.3, and 5.4.8 are functionally applicable to the SAFR auxiliary sodium system.

9.2.4 Research and Development Needs

Additional analytical, and possibly experimental, work is needed to confirm the behavior of contaminants (dissolved solids) held in the cold trap following cold-trap malfunctions.

The staff is aware of ongoing research and cold trapping using materials that are not temperature dependent in their trapping properties. Further information should be developed regarding the potential for system blockage (including subassembly blockage) from cold-trap cooling malfunctions.

Information regarding tritium holdup in the cold trap should be developed. The effect on tritium conditions from malfunctioning valves in the cold trap should be addressed.

9.2.5 Conclusions

The PSID (Ref. 9.1) indicates that the proposed design will be based on proven technology. The next design stage of the SAFR plant's sodium auxiliary system requires a more detailed safety analysis.

9.3 Fire-Protection System

9.3.1 Description and Safety Objectives

The SAFR design concept separates the fire-protection systems into two categories: (1) non-sodium fire protection and (2) sodium fire protection.

Non-Sodium, Fire-Protection System

The non-sodium fire-protection system (NSFPS) will offer fire protection for the nuclear islands, the various balance-of-plant (BOP) buildings, and the common control and service areas of the plant. The scope of protection includes the means to detect, locate, contain, extinguish, and mitigate the effects of non-sodium fires within the plant. In addition to avoiding property damage, the NSFPS will maintain the functional capability of safety-related systems or components, provide for personnel safety, and prevent interruptions to the generation of power.

Two 300,000-gallon redundant storage tanks, heated by electric-immersion-type heating elements for freeze protection, will supply the water needed for the NSFPS. The fire pumps will take suction from the storage tanks through suction lines that will be cross-connected within the pump house and will discharge through separate lines to the underground yard distribution system. An alternate water source will be provided by a 4-inch connection to the plant service water system supply main. The water will be supplied by two centrifugal fire pumps, each rated at approximately 2500 gpm and 125-psig total head. One pump will be driven by an electric motor and the other will be driven by a diesel engine. A 30-gpm pressure maintenance pump will maintain the system pressure at 140 psig. These pumps will be located in the fire pump house.

The yard distribution system will be a 12-inch underground piping loop surrounding the power block. Branch lines with individual curb box service valves will supply water to two-way yard hydrants located at approximately 250-foot intervals. Fully equipped hose houses enclosing yard hydrants will be strategically placed.

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The water supply system will deliver water to sprinkler, water spray, and wet and dry standpipe systems located throughout the plant in areas that do not contain liquid metals. The RCB and SGB will not be served by automatic fire-suppression systems that use water.

Hydraulically balanced water spray systems will protect equipment and systems from high fire challenge, including transformers located outdoors, lube-oil reservoirs in the turbine generator building (TGB), and hydraulically (oil) operated steam valves.

Halon 1301 will be used for fire protection in computer rooms and electronic panel rooms. These automatic total flooding systems will have pressurized storage cylinders located outside the protected rooms and cable ways.

The standby generator drivers and generators will be protected by carbon dioxide stored in a refrigerated tank outside the TGB. Portable chemical fire extinguishers will be located throughout the plant for suppressing small conventional, electrical, and liquid-metal fires.

The bearings and exciter for each main turbine will be protected by carbon dioxide or by water spray, depending on the preference of the turbine manufacturer.

Passive oil-collection pans and drain tanks and/or dry chemical suppression systems will be provided for the lube-oil systems serving the primary sodium pumps and the IHTS pumps.

Sodium-Fire-Protection System

The sodium-fire-protection system (SFPS) provides the means of detecting, locating, alarming, and containing sodium fires. The system will consist of fire-detection and alarm instrumentation, aerosol-release-limiting instrumentation, a leak-collection and catch-pan system, portable fire extinguishers, and protective clothing and equipment. The leak-collection and catch-pan system is designed to mitigate the consequences of a design-basis sodium leak.

To prevent the uncontrolled release of a moderate energy fluid system (MEFS)-size leak with its attendant potential sodium spray and fire, the plant design includes leak-mitigation provisions consisting of three principal features: an insulation confinement and collection system, a system of piping to carry away sodium leakage, and catch-pans to safely collect accumulated leakage. The insulation design is not part of the SFPS per se; however, its design must be integrated into the overall leak-collection systems and, in this context, it is described here. The design as described is applicable, in principle, to both the main IHTS sodium piping and to the DRACS sodium piping. Details of the collection piping and catch-pans for the DRACS have not yet been developed. The description for the IHTS piping follows.

All components, including piping, will be insulated to leave an annular space of approximately 1 inch between the component surface and the inner surface of the insulation. This space will be maintained by standoffs and will enclose the electrical heaters used for component preheating. The space will also serve as a means for detecting sodium leakage, by sampling the enclosed atmosphere, and will also serve to confine and collect potential sodium leakage. The piping insulation design is shown in Figure 9.6. The inner surface of the insulation will be covered with a stainless steel sheath to withstand jet impingement loading from a MEFS leak. Longitudinal seals will be suitably overlapped and crimped or riveted. Circumferential seams will be seal welded at appropriate locations along horizontal lengths of the IHTS piping. Drain connections will carry away sodium leakage. The drain connections will be 4 inches for all IHTS pipes, based on discharging flows up to 400 gpm without significant (1 psig) pressure buildup in the annular space. On the basis of the testing program performed on the CRBRP insulation system, 95 percent of the MEFS design-basis leak (DBL) will be directed to the collection piping by the leak jacketing arrangement. Only 5 percent of the MEFS DBL need be considered as a spray fire source.

The drain connections will discharge into a piping system that will carry leakage away by gravity drainage. The piping configuration (typical for each IHTS loop) is shown on Figure 9.7. Three 8-inch horizontal headers will service all the insulation drain connections in the loop and will run to a common 10-inch vertical header. Each insulation drain line will connect to the pipe header at a tee with a loose cover. The drain line and cover will not be connected; enough play will be provided to permit the differential movement between the process piping and the drain headers. Blind flanged connections will be located within the piping system for purposes of inspection and cleanout.

The 10-inch sodium-collection header will be routed to a catch-pan that will cover the floor of the SWRPRS/drain tank vault located below grade adjacent to the SGB. The catch-pan will be an open steel-lined, seam-welded container that will cover the floor and walls up to a height sufficient to contain all of the largest potential spill. Thermal protection will be afforded the structural concrete by thermal insulation under the wall liner and a layer of insulating concrete below the floor liner. A similar catch-pan will be provided over the floor of the SGB to collect the 5 percent of sodium that escapes from the insulation barrier around the IHTS piping. This catch-pan will slope to a low point or sump which, in turn, will drain to the catch-pan for the drain tank vault. Vent lines to the atmosphere will be provided in the vault structure to relieve vault overpressure. The lines will be equipped with dampers or check valves to prevent air intake following overpressure relief.

Portable sodium carbonate (NaX) fire extinguishers, hand-held and wheeled-cart types, will be provided and stored in locations convenient to spaces in which there is sodium equipment in the RSB and SGB. The extinguishers and their storage locations will be compatible with building space allocations and passageways. These commercially available extinguishers will be distributed and labeled in accordance with NFPA Code 10.

On the basis of ready availability for rescue or repair operations, personnel protection equipment, protective clothing, self-contained breathing apparatus, and other equipment essential to personnel safety in the event of a sodium fire will be stored at selected locations.

9.3.2 Scope of Review

The staff reviewed PSID Section 9.6.1 (Ref. 9.1) within the framework of GDC 3 (Fire protection), Branch Technical Position (BTP) APCSB 9.5.1., "Guidelines for Fire Protection for Nuclear Power Plants," and Standard Review Plan (SRP) Section 9.5-1. The conceptual nature of the design would not allow a detailed review at the component level. Additional reviews will be needed as the design evolves, especially regarding the design and arrangement of redundant safety trains.

9.3.3 Review and Design Criteria

GDC 3 focuses primarily on the probability and effects of fires and explosions on plant systems. After the fire at Browns Ferry, the NRC staff issued specific guidance for implementing GDC 3; this guidance is contained in BTP APCSB 9.5.1.

The non-sodium fire-protection subsystems and components will be designed to meet all applicable National Fire Protection Association (NFPA) and American Nuclear Insurers (or Nuclear Mutual Liability) standards. During the construction of additional power paks, provisions will be made to preserve the integrity and operability of the fire-protection system serving existing plants.

Fire-detection and fire suppression systems for the NSFPS will minimize fire damage to safety-related structures, systems, and components (SSCs) in accordance with PDC 1.3. Potential fire hazards in safety-related areas throughout the plant will be analyzed, and the effect of postulated fires on reactor shutdown and radioactive releases will be evaluated. Defense-in-depth protection against these hazards will be furnished, using BTP CMEB 9.5.1 as guidance, except that a seismic Category I water supply system will not be provided for the following reasons:

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- A design-basis event (DBE) followed by a fire in all four RSB electrical rooms housing safety-related equipment is not considered credible.
- A conventional fire could not prevent the safe shutdown of the reactor(s) because the control rod insertion drive is self-actuated and totally immersed in the reactor vessel sodium pool, and decay heat is removed by very reliable, passive, natural convection to the outside air system (RACS).

The NSFPS is designed so that neither system piping failure nor inadvertent operation caused by a seismic event will result in the loss of function of safety-related equipment.

The specific design bases for the sodium fire-protection system are as follows:

- The design-basis sodium piping leak is a moderate energy fluid system (MEFS) leak that is defined as leakage from a sharp-edged circular orifice whose area is equal to one-quarter of the pipe wall thickness multiplied by the pipe inside diameter.
- The integrated design of the insulation and leak-collection system shall collect a minimum of 95 percent of the sodium leakage. A maximum of 5 percent of the sodium leakage shall be allowed to escape in an uncontrolled manner to the room atmosphere.
- In the event of a DBL, the system shall be designed to limit the temperature imposed on the structural concrete to a level sufficient to ensure the structural integrity of the building.
- The system shall be designed to contain the maximum potential spillable volume from a full-flow piping leak in a manner to preclude chemical reaction between the liquid pool and the structural concrete.

9.3.4 Research and Development Required

Tests performed on the CRBRP IHTS pipe insulation have demonstrated the practicality of designing an insulation-confinement system capable of collecting 95 percent of a MEFS-sized leak. Similar tests will be performed for the SAFR when the design details are complete.

Inspection and Testing Requirements

The SFPS is designed to permit inspection of all instrumentation and portable fire extinguishers in accordance with NFPA codes. The catch-pans and fire-suppression decks will be inspected periodically, as appropriate.

Sodium carbonate, NaX, used in fire extinguishers, has been tested by Underwriters Laboratory and is listed for use on potassium, sodium, and sodium-potassium fires for temperatures up to 1400°F. The Underwriters Laboratory listing includes NaX in 50-lb pails and in hand-held and wheel-type fire extinguishers of 30-, 150-, and 350-lb capacities. NaX has been widely accepted in the United States as the preferred fire-extinguishing agent for sodium fires. The use of NaX does not harm materials used in the plant's heat transport systems.

9.3.5 Safety Issues

The NSFPS, while not an emergency safety feature, is designed to minimize the adverse effects of fires on nuclear-safety-related structures, systems, and components (SSCs) and personnel in the control building.

Smoke (ionization, photoelectric) or flame (ultraviolet) detectors or both will be located in all safety-related areas. Signals from these detectors will activate local fire control panels and a microprocessor

that communicates with the central station equipment in the technical support center (TSC). Appropriate fire-suppression equipment will automatically respond to fires in hazard areas.

There will be no fire-protection water pipes in the RCB, the SGB, or the cells containing sodium. This is to protect against sodium reactions with water. Oil-collection systems or chemical-suppression systems or both will be provided to meet the identified hazards (lube-oil reservoirs) in the RCB and SGB.

The design-basis sodium leak for the piping is conservatively assumed to be the MEFS leak. For the IHTS loops, MEFS leaks at various locations in the piping are summarized in Figure 9.8. The consequences of the MEFS leak are evaluated in Chapter 15.

The staff believes that, at this stage of development of LMR systems, special care should be taken to ensure that sodium piping systems will maintain their integrity over long periods of time. In addition to the full volumetric inspection of sodium piping during the construction stage (which the DOE is committed to), the staff recommends a periodic reinspection of selected welds. The period and specific locations would be determined at a later design stage and should be based on the likelihood of potential consequences of a failure. The BTP APCSB 9.5.1 and SRP 9.5.1 both have a defense-in-depth philosophy not otherwise found in the broad GDC 3 guidelines. On the basis of previous LWR experience, GDC 3 alone will not provide adequate fire protection. From a defense-in-depth perspective, justification for deviation from regulatory criteria must be based on showing an equivalence in the level of protection.

Backup fire-suppression capability is defined as fire hose stations, portable fire extinguishers, and yard hydrants. In order to justify adequate backup fire-suppression capability however, the DOE should make an appropriate commitment to manual fire-fighting procedures and training.

On the basis of previous LWR experience, emphasis on automatic rather than manual fire suppression raises special concerns with regard to inadvertent actuation. At a later review stage, the designer should address the potential impact of inadvertent fire-suppression system actuation on safety systems and components. In addition to the consequences, the mechanisms by which actuation is initiated (e.g., excessive room temperature, smoke, steam, dust, maintenance activity) should be considered.

The SAFR conceptual design does not allow evaluation of fire-induced system interactions. As the design matures, the consequences of fire-induced system interactions on multiple module control and shutdown systems will have to be evaluated.

A quality assurance (QA) program for fire protection should be part of the overall plant QA program. Specific criteria, such as those found in BTP APCSB 9.5.1 and SRP 9.5.1, should be met.

9.3.6 Conclusions

Just meeting the intent of GDC 3 will not provide adequate fire protection for the SAFR plant. The design should meet 10 CFR 50.48 and should show an equivalent level of protection to that found in BTP APCSB 9.5.1. Equivalence cannot be shown at this time because of the conceptual nature of the design.

The adequacy of the design with regard to fire and fire-mitigating activity, including inadvertent actuation, will play a substantial role in subsequent evaluations of the fire-protection system. Through proper engineering design, safety system separation, redundancy, and protection, a large portion of BTP APCSB 9.5.1 criteria can be met.

A detailed fire-hazard analysis will help identify potential fire hazards and their effect on plant systems. A PRA regarding fire should be performed early in the SAFR design stage and should be

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revised periodically as the design progresses. In addition, specific QA criteria should be met during design and construction of the fire-protection system.

The potential impact of fire-induced system interaction on the plant will have to be evaluated. The need for, and adequacy of, remote shutdown during postulated fires involving multiple trains of safety equipment (e.g., control room fires) will also have to be determined.

The staff should evaluate the role of sodium piping integrity assurance through the QA program and inspections during construction and during long-term operation.

9.4 Other Auxiliary Systems

As mentioned at the beginning of this chapter, the staff chose to focus on the auxiliary system unique to the SAFR LMR at this time. These systems were discussed above. The remaining auxiliary systems are judged to be relatively conventional, and detailed review is deferred to a later stage. It should be noted that a significant review effort will be required to provide a basis for accepting the DOE's position that these (and other) BOP systems have no impact on plant safety.

9.4.1 Water Systems

Plant Service Water System

The plant service water system (PSWS) is designed to remove heat from the compressed-air system and the plant HVAC system and to transfer this heat to the PSWS cooling tower.

The PSWS will include two full-capacity service water pumps that will take suction from two PSWS cooling towers. Cooling water will be pumped through the air compressor water-jackets, aftercoolers, and the water-chiller condensers, and will then be returned to the operating PSWS cooling tower. A branch connection will also be made to the fire-protection water-supply system. The PSWS will be connected to the non-Class 1E standby emergency generators to support the HVAC system during a loss of offsite power.

The PSWS has no safety-related function. It is not designed as seismic Category I. Failure of the PSWS will not compromise any safety-related system or component, nor will such failure prevent a safe shutdown of the plant.

Auxiliary Cooling-Water System

The auxiliary cooling-water system for the SAFR is the normal chilled-water system (NCWS). The NCWS is a closed-loop system that removes heat from non-safety-related space air-cooling units in the power block.

The NCWS will be designed to supply chilled water for all non-safety-related air conditioning cooling coils located in the power block. The administrative complex will have its own heat-removal system. The NCWS will remove heat from the following spaces during normal plant operation:

- reactor containment building(s) (RCB)
- reactor service building(s) (RSB)
- turbine-generator building(s) electrical equipment areas
- balance-of-plant (BOP) plant service building
- control building
- personnel service buildings
- nuclear island guardhouse
- nuclear island maintenance building

The NCWS is a non-safety-related system that will not be required to remove heat from any safety-related equipment or structure during accident conditions. Passive post-accident heat-removal systems for the containment are described in Section 6.2.2 of the PSID (Ref. 9.1). Features will be incorporated in the NCWS design to prevent a sodium-water reaction.

Demineralized Water Makeup System

The demineralized water makeup system is designed to provide a supply of treated water of the required quality for the condensate storage tank and for other demineralized water users.

The demineralized water makeup system will consist of an activated carbon filter, two demineralizer trains in parallel, two caustic feed pumps, two acid feed-pumps, a caustic storage tank, an acid storage tank, a demineralized-water storage tank, three demineralized-water transfer pumps, a chemical-waste sump, two chemical-waste sump pumps, and a neutralizing-tank sample pump.

Potable and Sanitary Water Systems

The potable and sanitary water systems incorporate distinct systems:

- the potable water system (SWS), which provides plumbing fixtures with cold and hot water of a quality acceptable for human consumption
- the sanitary water system (SWS), which treats and disposes of waste from plumbing fixtures

Potable water will be supplied in sufficient quantities to satisfy the demand of all plumbing fixtures for the entire plant. The SWS is designed to treat and dispose of the sanitary waste in accordance with the applicable standards for sewage effluent. The potable and sanitary water systems are designed to preclude contamination by radioactive systems.

9.4.2 Process Auxiliaries

Compressed Gas Systems

The compressed gas systems supply instrument air, service air, hydrogen for generator cooling, and carbon dioxide for generator purging. The compressed-air system (CAS) is used for instrument air and service air. The hydrogen and carbon dioxide for generator purging are covered in the miscellaneous gas supply system (MGSS).

The CAS will consist of the instrument air subsystem and the service air subsystem. The CAS will satisfy all of the plant's compressed air requirements for pneumatic instruments and valves, and for service air outlets located throughout the plant for operation of pneumatic tools and for personnel breathing.

DOE and RI claim the CAS has no safety-related function other than the integrity of the piping in the secondary containment and secondary containment penetrations. Failure of this system will not compromise any safety-related system or component, nor will it prevent a safe shutdown of the plant. All air-operated containment isolation valves will fail closed on loss of air supply. The instrument air line penetrating the secondary containment will be closed to the containment atmosphere. The service air line to the secondary containment will normally not be used during reactor operation. It will be isolated manually, and the position will be monitored in the control room.

Portions of the system located within secondary containment are designed to ANSI B31.1 and seismic Category II. Other portions outside the secondary containment, including the compressors and filters, are industrial-grade quality, meeting applicable standards and regulations, and designated seismic Category III.

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Miscellaneous Gas Supply System

The MGSS consists of the hydrogen subsystem and the carbon dioxide subsystem. The hydrogen subsystem is designed to provide the storage and supply of hydrogen required for generator cooling during operation. The carbon dioxide subsystem is designed to provide liquid carbon dioxide storage and the carbon dioxide gas supply required for purging the generator when the unit is not in operation.

The hydrogen will be stored in high-pressure cylinders located outdoors, away from the plant structures. The cylinders will be stored horizontally in a modular steel frame to which they are fastened. The frame will be mounted on concrete pads. The liquid carbon dioxide will be stored in a tank located outdoors. For generator purging, the liquid will be converted to gas by an electric vaporizer (heater). Both sources are common to all power pak turbine generators. However, an option considered is to locate a limited bottle supply of hydrogen at each turbine generator to provide an added capability for short-term demands.

DOE and RI claim the MGSS has no safety-related function and it is designed as seismic Category III. Failure of this system will not compromise any safety-related system or component, nor will failure prevent a safe shutdown of the plant. It will not penetrate the secondary containment. The MGSS is industrial-grade quality, meeting applicable standards and regulations for this type of high-pressure gas system.

Process Sampling System

The process sampling system will have provisions for obtaining representative samples from all nuclear island and BOP normal process systems and principal components. The process sampling system is designed to help maintain radiation exposures as low as is reasonably achievable (ALARA) in accordance with 10 CFR 20.1(c) and Regulatory Guide 8.8 (Ref. 9.5). The functional design of the process sampling system follows the applicable guidelines of Regulatory Guide 1.21 (Ref. 9.6) and ANSI Standard N13.1.

The post-accident sampling system (PASS) will allow the prompt removal of samples from the secondary containment atmosphere, under accident conditions, for chemical and radiological analysis. The PASS will meet the intent of Item II.B.3 in NUREG-0737 (Ref. 9.7) and of Regulatory Guide 1.97 (Ref. 9.8).

Operation of the PASS for the secondary containment will not be safety related and, when the piping is open, it will serve as a structural extension of secondary containment. Therefore, blowers and valves will be supplied with non-Class 1E normal and auxiliary power, and structurally the components will meet ANSI Standard B31.1 and seismic Category II.

All other sampling systems will be non-safety-related, seismic Category III, and will be designed and constructed to industrial-grade quality in accordance with applicable Federal, State, and local regulations.

Equipment and Floor Drainage System

The plant equipment and floor drainage system (EFDS) is designed to collect the drainage from all such plant equipment as pumps, tanks, and heat exchangers, as well as from the floor. It will contain separate drains for potentially radioactive and nonradioactive liquid waste.

Under normal operating conditions, the floor drains in the plant will perform a housekeeping function. The EFDS will be sized to accommodate the maximum postulated flooding event such as a pipe

rupture, tank rupture, or sprinkler discharge, while limiting water accumulation on the floor. All safety-related equipment in the plant will be mounted on pads at a level above the maximum flood level for each area. This maximum flood level considers a postulated complete blockage of the drainage system. In some cases, the gaps under doors will be sized to provide an alternative passive means for drainage to egress from a cell with safety-related equipment.

There will be separate EFDS drains or sumps or both for potentially radioactive and nonradioactive areas of the plant.

The EFDS is designed so that it is not reasonably possible for any significant radioactive drainage in the plant to be discharged out of the plant without undergoing the required processing.

No EFDS drains will be in enclosures and cells where piping or equipment containing sodium are located. Sodium will not be able to leak into the EFDS.

The EFDS will not be safety related, except for the penetration and valve required for secondary containment isolation. This valve will be normally closed and will be designed to ANSI Standard B31.1 and seismic Category II.

Inert-Gas System

The inert-gas system is part of the inert-gas receiving and processing system. This section covers the receipt and distribution of helium, argon, and nitrogen to cover sodium gas spaces and areas requiring inert gas. Processing of potentially radioactive gases is discussed in Section 11.2 of this SER.

There will be three separate subsystems for inert gas:

- helium storage and distribution
- argon storage and distribution
- nitrogen storage and distribution

Normal operating pressure control will be maintained by feed-and-bleed valve arrangements. Overpressure of sodium cover-gases (e.g., the reactor vessel cover-gas) will be prevented by relief valves. These features will be provided at the connections to each cover-gas or atmosphere serviced.

The helium subsystem will provide helium for primary sodium cover-gas (reactor vessel), reactor head components, the fuel-transfer cell (FTC) atmosphere, primary sodium auxiliaries, and the annulus between the reactor vessel and the containment vessel.

The argon subsystem will be used primarily as a cover-gas in secondary sodium systems (e.g., the IHTS and DRACS, SWRPRS, and secondary sodium sampling equipment). The IHTS and DRACS sodium piping enclosures within the RCB will be supplied with argon for the manual flooding of each enclosure if a sodium leak occurs.

Nitrogen is normally used for LMR primary sodium cell atmospheres and as the gas used for pneumatic valve operators within these cells. However, these uses will be eliminated or restricted on the SAFR. The present use is intermittent and related to providing SWRPRS stack/cell inerting, steam generator (steam side) inerting, and as a cover-gas for the decontamination and sodium-removal system.

Except for portions of the helium-distribution system that interface with the primary containment (reactor vessel cover-gas and containment vessel) and the secondary containment boundary, the inert gas system is not safety related. Double isolation valves (ASME Code, Section III, Class 2) will be used to ensure primary containment integrity. The piping and valving will be seismic Category I.

Auxiliary Systems

Although not safety related, special pressure and purity control will be provided for the FTC during operations in order to minimize the possibility of radioactivity release. At standby, the FTC will be maintained slightly above atmospheric pressure to minimize air in-leakage and maintain gas purity. When fuel is being transferred (or significant radioactivity is detected in the cell), the FTC will be maintained at a slightly negative pressure to preclude releasing radioactivity. Increased FTC purging and a special helium purification unit will minimize the effect of increased air in-leakage during this time period.

9.4.3 Heating, Ventilation, and Air Conditioning System

The HVAC system is a non-safety-related system that provides space cooling, heating, ventilation, filtration, and humidity control for the plant enclosures. There are four HVAC subsystems. These are the nuclear island, the control building (CB), the turbine-generator building (TGB), and the auxiliary generator building (AGB) subsystems.

- The nuclear island HVAC subsystem provides space cooling, heating, ventilation, filtration, and humidity control for the
 - RCB (secondary containment)
 - RSB
 - SGB

The nuclear island HVAC system also includes a behind-the-liner cooling subsystem to cool the helium atmospheres in the FTC.

- The RCB HVAC subsystem is a non-safety-grade system designed to provide air cooling, ventilation, and purging for the secondary containment in order to allow personnel access and equipment operation during normal conditions. Two of the four unit coolers located within the secondary containment are connected to non-Class 1E standby power to protect the non-safety-related motors and instruments from overheating following a loss of normal power. The safety-related secondary containment isolation valves located in the ventilation supply and exhaust ducts are part of the containment isolation system.
- The RSB HVAC subsystem is a non-safety-related system designed for the following functions:
 - Provide air conditioning for personnel access and equipment operation during normal periods.
 - Provide ventilation or air purging for potentially radioactive rooms to protect plant personnel.
 - Provide airborne particle filtration for contaminated exhaust air to protect site personnel and the public during normal operation.

Safety-related Electrical Equipment Room (EER) HVAC

EERs A, B, C, and D (which contain the containment isolation system, APTS, and reactor monitoring panels as well as supporting Class 1E batteries, battery chargers, and inverters) will be located in the east and west wings of each RSB. Each of these EERs will be provided with redundant cooling equipment.

Safety-related equipment in the EERs will be shielded from sodium aerosols by physical separation, firewalls, and by normally closed dampers at the outdoor-air openings.

The SGB Heating and Ventilation (H&V) System

This is a non-safety-related system designed to provide ventilation and space heating for equipment operation and personnel access. The system design meets the following requirements:

- During normal operation, maintain the steam generator space temperature between 60°F and 110°F and the IHTS pump room temperature between 60°F and 104°F.
- Supply sufficient outdoor air to meet Occupational Safety and Health Administration (OSHA) standards for ventilation.
- In conjunction with the IHTS, furnish vent paths for relieving building pressure during a sodium reaction.

Since there are no steam lines or hot- or chilled-water pipes in the SGB, the likelihood of a sodium-water reaction is reduced. The SAFR steam generator system is comparable to the steam generators and primary pumps in a pressurized-water reactor (PWR) in that they do not require ambient air cooling for safe reactor shutdown.

The CB HVAC system consists of three non-safety-related HVAC subsystems serving the main control room (MCR) and technical support center (TSC) room, the other occupied rooms, and the unoccupied electrical and HVAC equipment rooms. The system is designed to meet the following criteria:

- Maintain the normally occupied spaces at approximately 75°F and 50-percent relative humidity during normal operation.
- Maintain the temperature range in the equipment rooms between 60°F and 104°F during normal operation.
- Supply sufficient outdoor air to meet OSHA standards for ventilation during normal operation.
- Permit purging of the MCR, TSC, computer room, and switchgear room following a fire. The MCR and TSC smoke exhaust rate is 10-room-volume changes per hour. The computer room and switchgear room smoke exhaust rate is 1.5 cfm/per square foot.

The HVAC subsystem serving the MCR, supervisor's room, MCR toilet, and the TSC rooms (the MCR/TSC area) is not safety related. The MCR/TSC area HVAC subsystem is a minimum outside-air recirculation system composed of air conditioning and filtration equipment with redundant rotating components.

The MCR/TSC HVAC subsystem is not required to operate following a design-basis accident (DBA), because there are no Class 1E or safety-grade controls located in the control complex. However, non-safety-grade habitability equipment similar to the equipment used in LWRs is provided. The habitability equipment is described below.

The HVAC subsystem serving the computer room, the secondary alarm station (SAS), the office space, and the other occupied CB rooms (except the MCR/TSC areas) is not safety related. This HVAC subsystem includes an air-handling unit with two 100-percent-capacity supply fans, four packaged air conditioners, and exhaust fans (normal and smoke). The central station air-handling unit and ductwork have sufficient capacity to provide backup capability should any local packaged air conditioner fail during normal operation.

The HVAC subsystem serving the CB electrical and HVAC equipment rooms is also not safety related. This subsystem includes an air-handling unit with two 100-percent-capacity fans, two battery room exhaust fans, and a smoke exhaust fan. The redundant air-handling unit fans and the battery room exhaust fans are switched to non-Class 1E auxiliary generator power automatically following a loss of normal power, to provide cooling and hydrogen removal for the electrical and HVAC equipment supplying the MCR/TSC areas and SAS rooms.

Auxiliary Systems

The control complex does not contain any safety-grade equipment or controls. PAM stations are located in the MCR and computer room, as well as in each RSB. The reliability of the control complex HVAC system is enhanced by redundant rotating equipment and backup ac power.

The control complex HVAC equipment is similar to that provided for the LWR plants. However, because of the reduced safety functions performed in the SAFR control complex, the HVAC system differs from LWR systems as follows:

- Passive HVAC components such as coils, filters, and ducts are not redundant.
- Fire separation is not provided for HVAC subsystems.
- Alternate outdoor-air intakes are not provided for the MCR/TSC HVAC system.
- Class 1E electrical power is not connected to the MCR/TSC HVAC system. Instead, non-Class 1E auxiliary generators supply backup power to the plant service water system, normal water chillers, pumps, fans, and HVAC controls serving the MCR/TSC, the SAS, and the electrical and HVAC equipment rooms.
- The HVAC equipment will not undergo IEEE Standard 323 qualification testing. Instead, high-quality industrial equipment will be used if previously qualified models are not available.
- The HVAC system for the MCR area is extended to include the TSC to meet the TSC habitability requirements of NUREG-0737 (Ref. 9.7).

During a loss of normal power, one of two normal water chillers and one of two MCR supply fans will cool the MCR using power from the (non-safety-related) standby generator. During a station blackout, MCR cooling is not required because the small heat load from the monitoring displays, emergency lighting, and personnel would be removed by passive transmission to the concrete boundary, which forms a heat sink. Likewise, a postulated event that prevents mechanical cooling without disrupting the MCR heat load could be accommodated by shutting down unnecessary heat-generating equipment and lights if the room temperature became excessive. Alternately, the MCR personnel could move to the EERs in the RSB(s) for post accident monitoring PAM.

The DBA release of airborne radioactive contaminants does not cause the dose level for personnel in the control complex to exceed 5 rem (whole body), or its equivalent to any part of the body, for the duration of the accident. However, a non-safety-related high-efficiency particulate air (HEPA) absorber filter train is included in the design to reduce radioactivity to ALARA. Duct-mounted radiation detectors supply the signals that automatically close the area isolation valves, provide annunciation, and start a filtration train fan.

MCR isolation equipment and features similar to those found in LWRs are provided to protect against the entry of toxic gas. The SAFR MCR isolation equipment is not safety class because the reactor(s) can be shut down automatically, and PAM can be delayed until the toxic gas subsides, without jeopardizing the passive removal of decay heat from the reactor(s).

The TGB HVAC system is a non-safety-related system designed to provide space cooling, ventilation, and heating for equipment operation and personnel access. The system design meets the following requirements:

- During normal operation, maintains electrical equipment ambient temperature between 60°F and 104°F. Maintains other spaces in the TGB between 60°F and 110°F.

- Supplies sufficient outdoor air to meet OSHA standards for ventilation.
- Provides dehumidification of electrical switchgear and control panels.

The TGB HVAC subsystem is not safety related and is not required to operate following an accident, because no Class 1E electrical equipment or other safety-related equipment is in the TGB.

The auxiliary generator building (AGB) H&V system is a non-safety-related system designed to provide heating and cooling for the AGB during normal and loss-of-offsite-power (LOOP) conditions. The system is designed to meet the following specific criteria:

- Maintain the temperature of the standby generator cells between 50°F and 120°F during generator testing and emergency operation so that the ambient temperature range required by the generators is not exceeded.
- Maintain the temperature of the standby generator cells between 50°F and 110°F when the turbines are not running, for personnel access and to facilitate generator starts.
- Limit the concentrations of combustible vapors that leak from the fuel storage components.

Each of the two non-Class 1E standby generator cells has a 100-percent-capacity H&V subsystem. The H&V components provided for the AGB are the same type as those provided for LWR emergency generator cells, except that they are non-nuclear safety grade. The H&V system has the same high level of reliability as the generators that it serves.

9.4.4 Conclusions

The design of the auxiliary systems is consistent with the accepted practice used in the design of LWRs. At a later design stage, the systems will be reviewed in detail, including applicability to LMR technology and requirements.

9.5 References

- 9.1 DOE, "SAFR - Sodium Advanced Fast Reactor," Preliminary Safety Information Document, AI-DOE-13527, Initial Issue - October 1985, with amendments. (DOE Applied Technology.)
- 9.2 NRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," NUREG-0800, July 1982.
- 9.3 NRC, "Standard Format and Content of License Applications for Storage Only of Unirradiated Power Reactor Fuel and Associated Radioactive Material," Regulatory Guide 3.15, Rev. 1, April 1983.
- 9.4 NRC, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," Regulatory Guide 1.25, March 1972.
- 9.5 NRC, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," Regulatory Guide 8.8, Rev. 3, June 1978.

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- 9.6 NRC, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Release of Radioactive Materials in Liquid and Gaseous Effluents From Light-Water-Cooled Nuclear Power Plants," Regulatory Guide 1.21, Rev. 1, June 1974.
- 9.7 NRC, "Clarification of TMI-2 Action Plan Requirements," NUREG-0737, November 1980.
- 9.8 NRC, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident," Regulatory Guide 1.97, Rev. 3, May 1983.

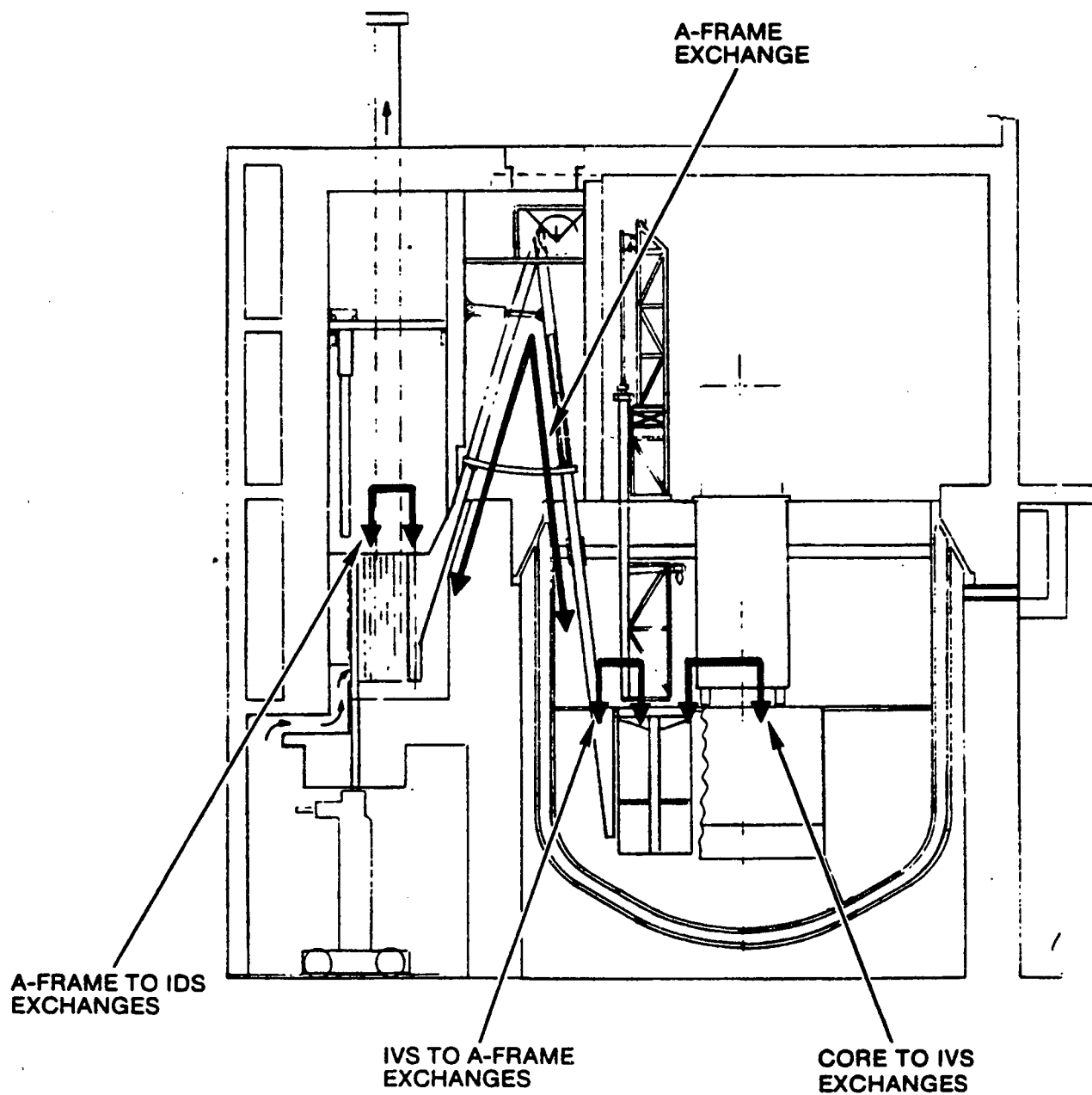


Figure 9.1 SAFR refueling operations.

Auxiliary Systems

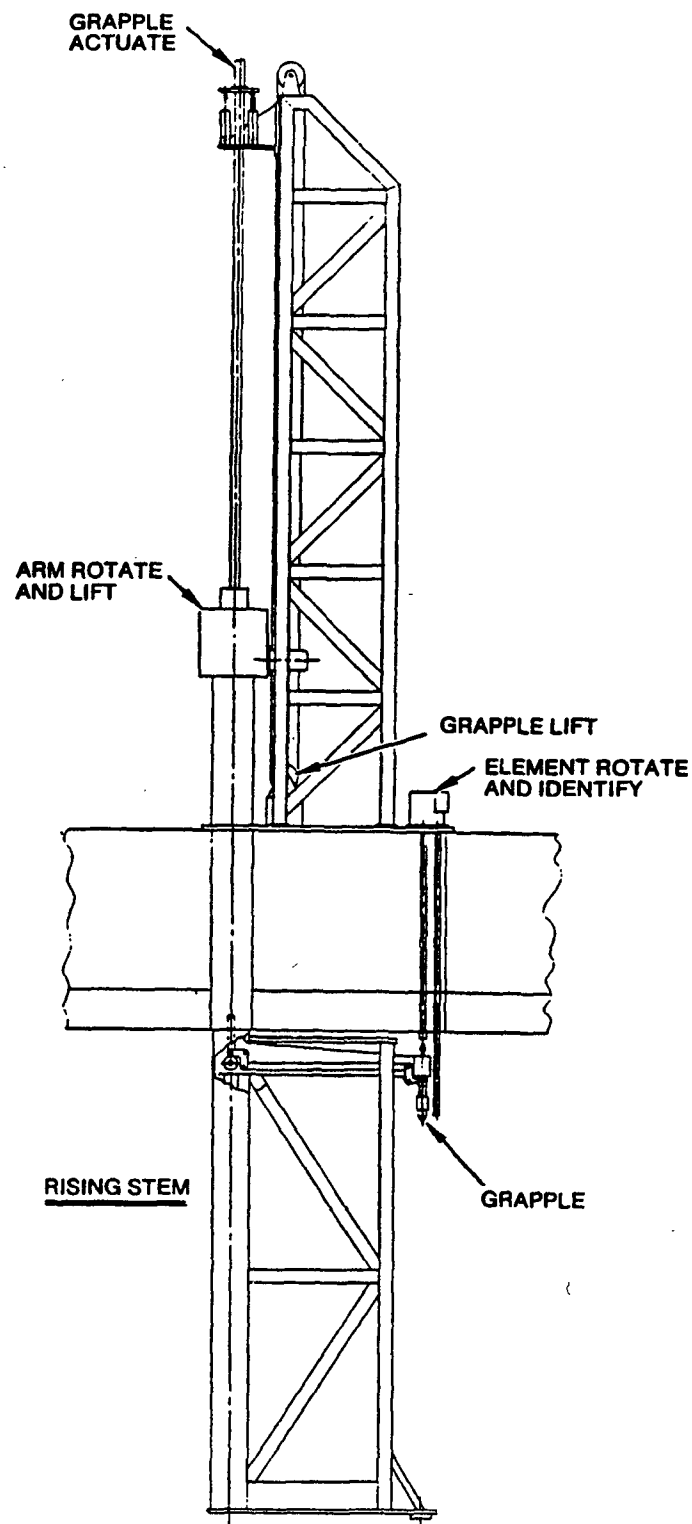


Figure 9.2 In-vessel handling machine (IVHM).

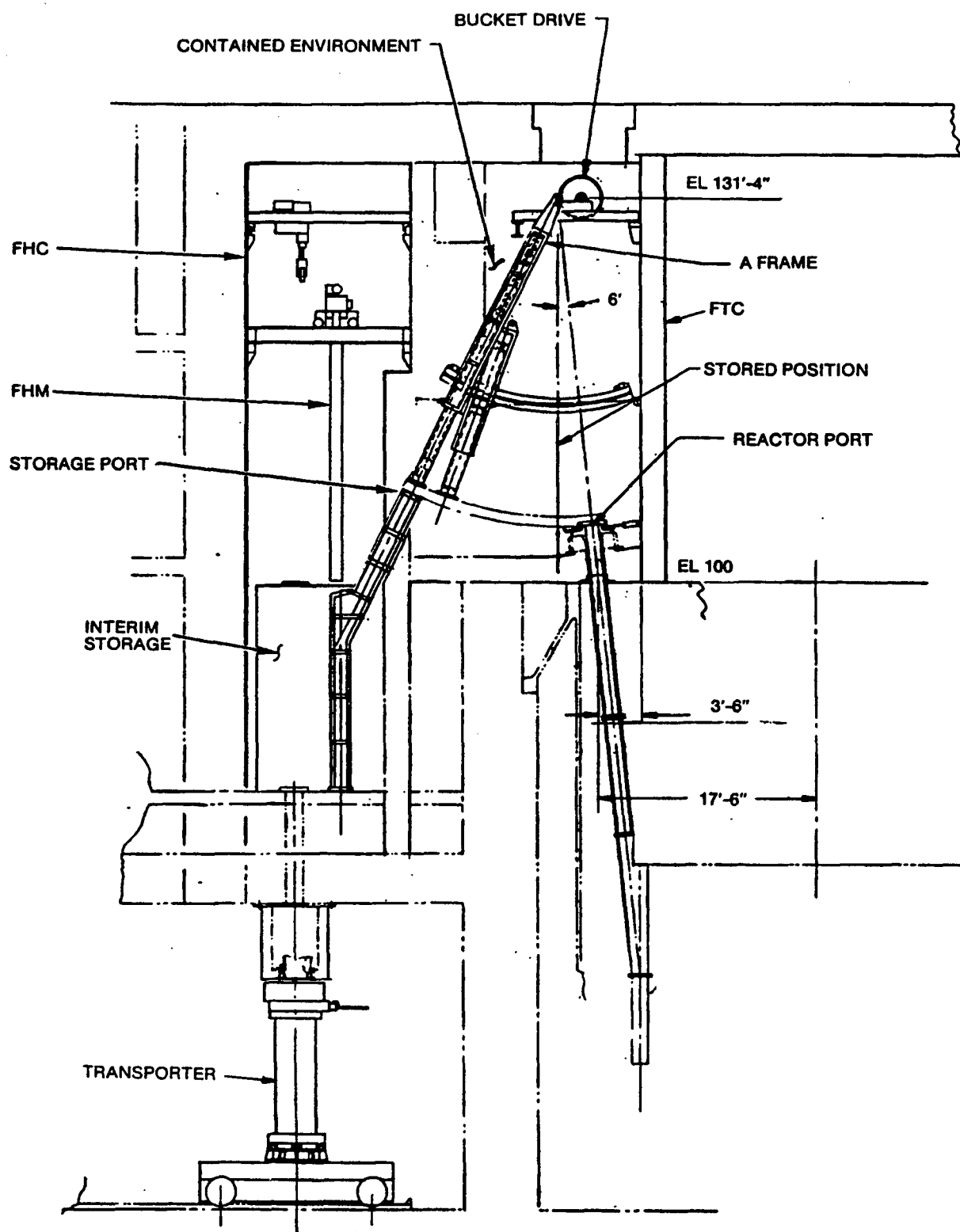


Figure 9.3 Ex-vessel fuel-handling equipment.

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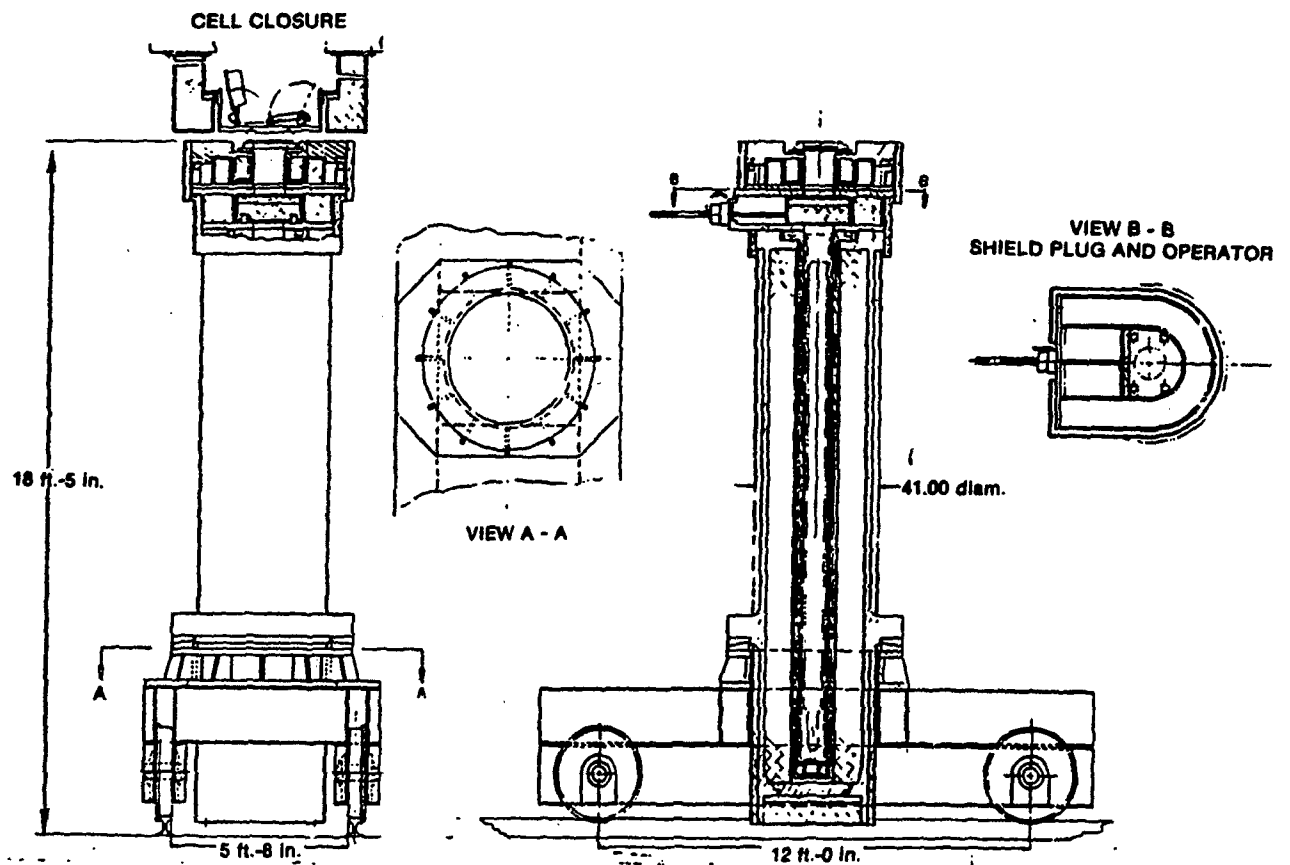


Figure 9.4 Spent-fuel transporter.

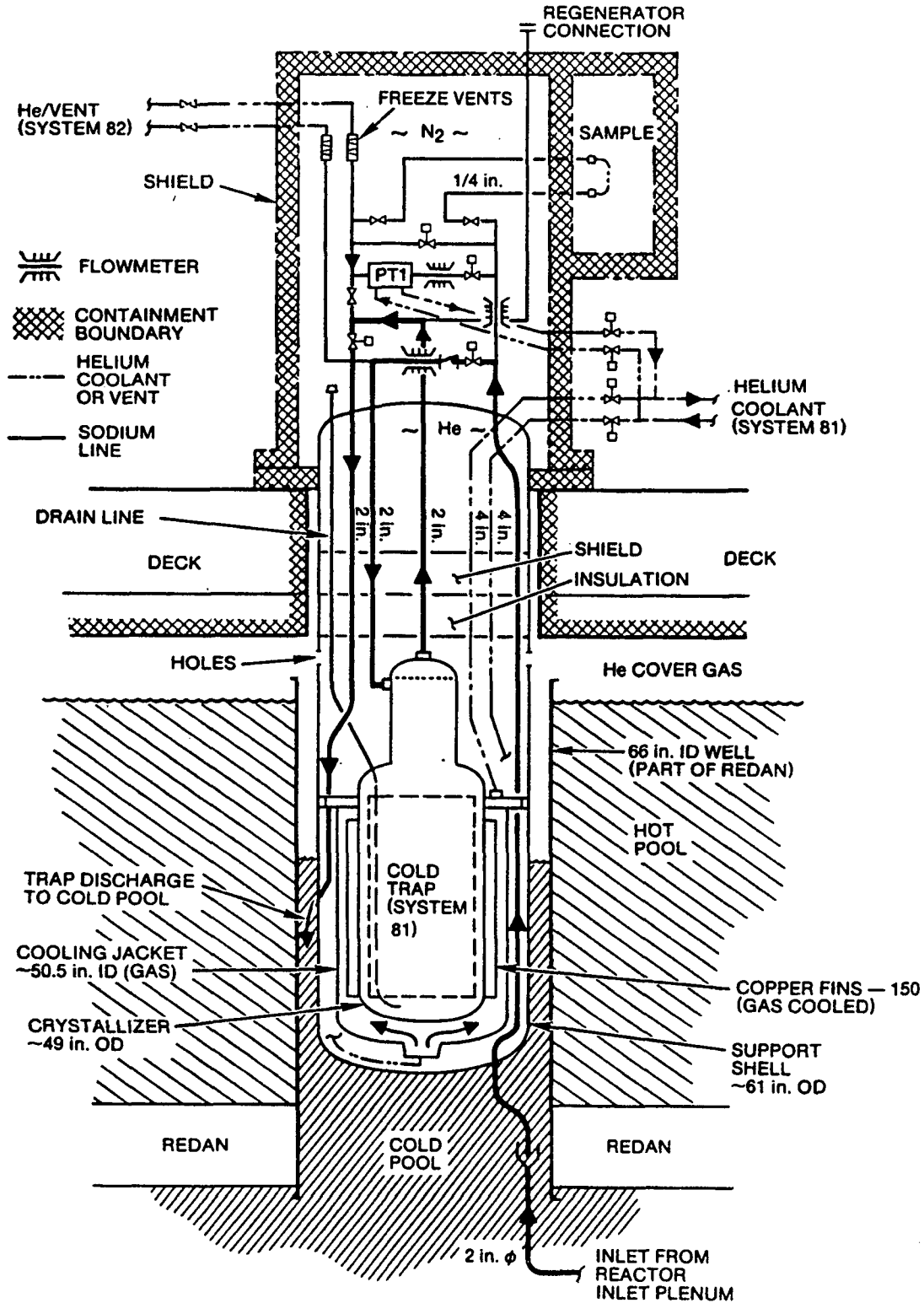


Figure 9.5 In-vessel cold-trap assembly.

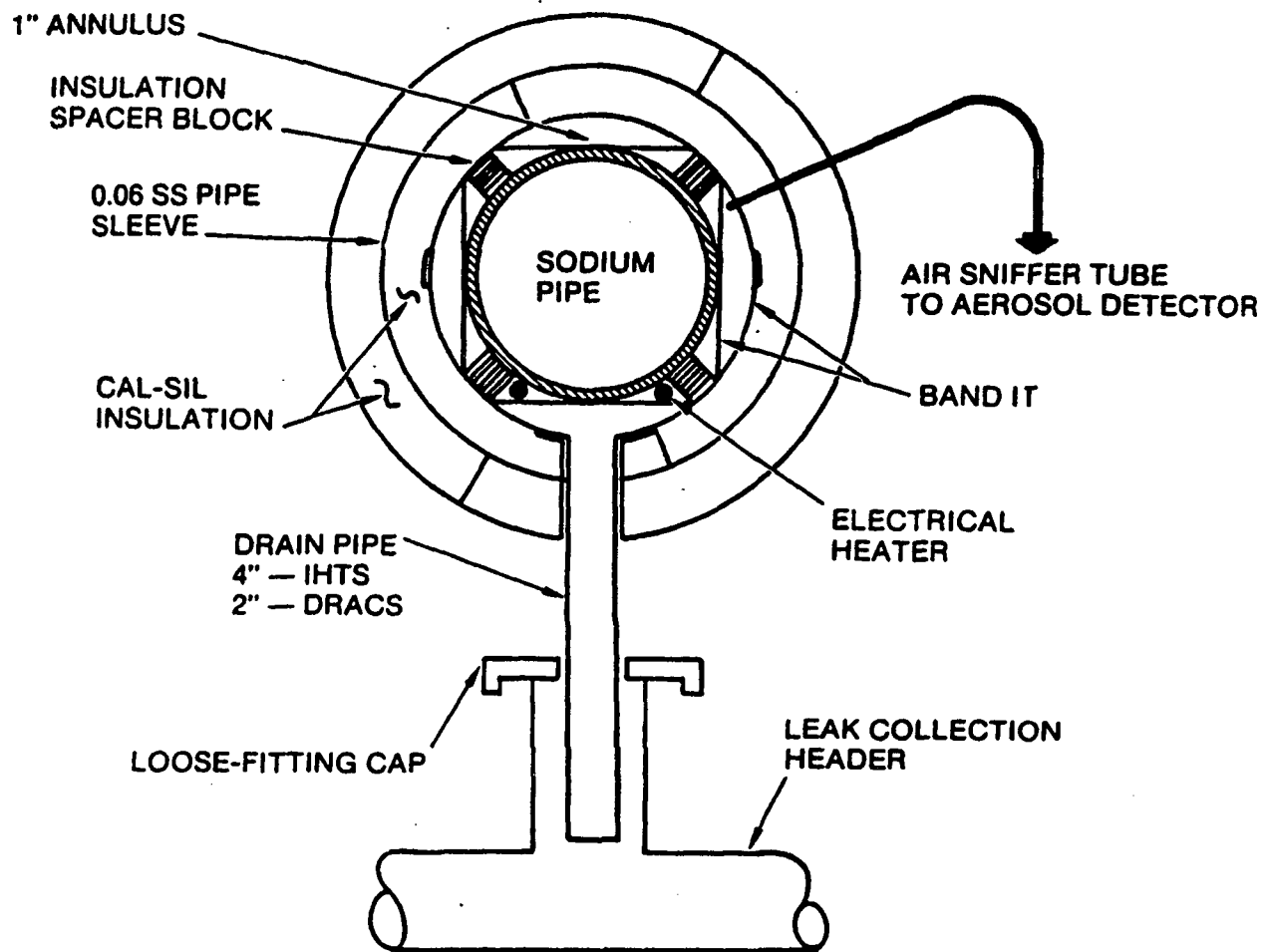


Figure 9.6 IHTS insulation, leak detection, and collection provisions.

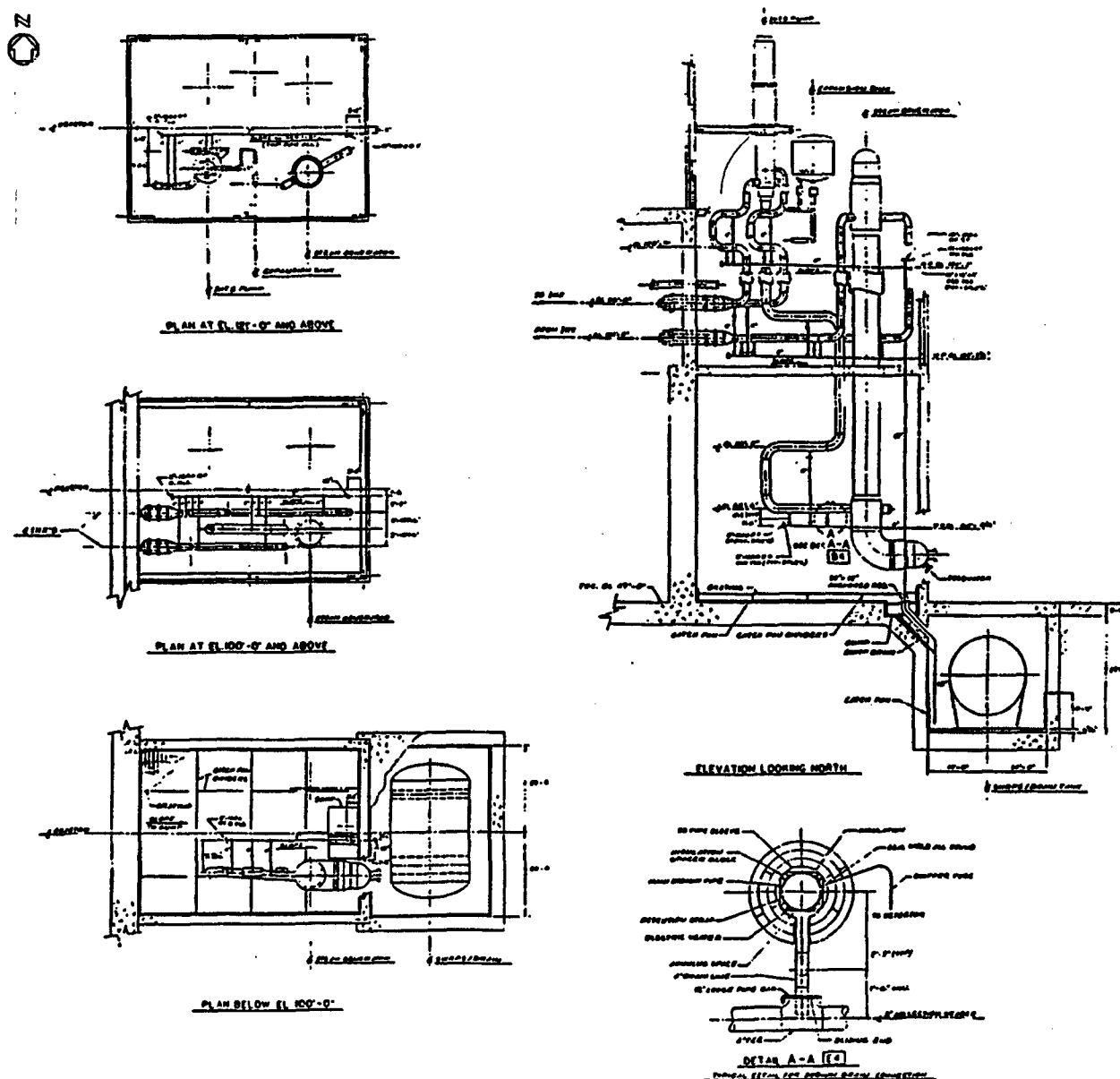
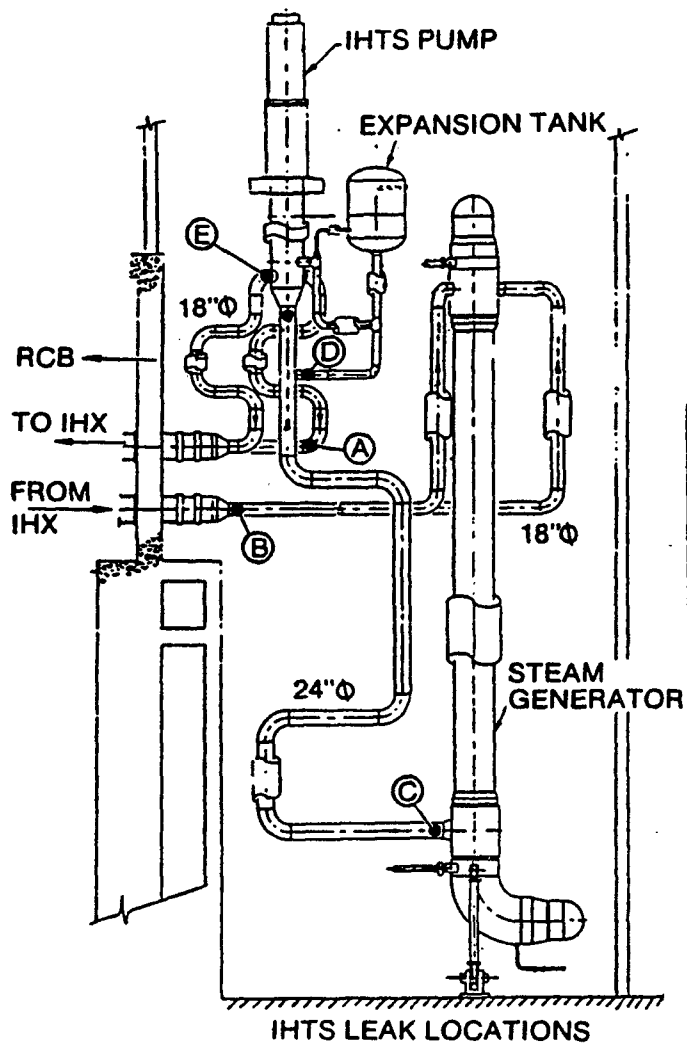


Figure 9.7 Leak collection piping.

Auxiliary Systems



LEAK POINT	EQUIV. HOLE SIZE (IN.)	INITIAL LEAK RATE (GPM)	TOTAL SPILL VOL. (GAL)	Na PIPE SIZE (IN.)
A	1.44	374	8326	18
B	1.44	367	10135	18
C	1.67	335	18969	24
D	1.08	60	7417	10
E	1.44	364	4397	18

Figure 9.8 IHTS sodium leak data (based on MEFS leak size).

10 POWER CONVERSION SYSTEMS

The power conversion systems include the main steam supply system (MSSS), the turbine generator with its related auxiliary systems, the main condenser, the feedwater and condensate system, and the circulating water system. These systems provide the normal heat removal path under routine operation and anticipated transient conditions. It is DOE's position that operation of these systems is not required for plant safety. The staff believes further studies are needed to ensure the balance of the plant is separated from safety functions.

10.1 Summary Description and Safety Objectives

The MSSS receives thermal energy from the intermediate heat transfer system (IHTS) and transports it to the steam turbine. In the turbine, the available thermal energy contained in the steam is converted to mechanical energy. The steam turbine drives the generator which converts the mechanical energy into electrical energy. The main condenser returns the condensate to the steam generator by way of the feedwater system. These systems are described in detail below.

The power conversion system (PCS) equipment and components are not safety related; they are classified as seismic Category III and are designed to either (1) ANSI Standard B31.1 (piping), (2) ASME Code, Section VIII, Division 1 (tanks), or to (3) other acceptable high-quality industrial standards representative of their function in a commercial power-generation facility.

10.1.1 Main Steam Supply System

The MSSS, shown in Figure 10.1, begins at the main steamline connection to the steam generator and extends up to the turbine stop valves. One common line, which receives steam from each of the two steam generators, conveys steam to the turbine. At the turbine, the common header branches into each of the valve chests, consisting of two stop valves and two control valves.

The MSSS is designed to convey superheated steam from each of the two steam generators to the high-pressure turbine. Steam conditions at the steam generator outlet nozzle are 2800 psia and 850°F. The system also serves as a backup steam supply to the auxiliary steam system.

A main steamline break does not jeopardize any safety-related equipment because such equipment is not located in the same areas as the main steam line. Main steam lines do not enter the steam generator building.

10.1.2 Turbine Generator and Auxiliaries

The turbine-generator unit is comprised of a turbine, a generator, an exciter, a moisture separator and reheater, and auxiliary systems.

As described here, the turbine-generator characteristics apply to a typical unit only and do not refer to the particular design of any vendor. At the final design stage, the design must satisfy system interface requirements.

The turbine generator and related systems are designed in accordance with the steam conditions, flow requirements, and electrical power output, as specified in the heat balance data shown in Figure 10.2.

The turbine rotors will be machined from "monoblock" forgings, thus eliminating the need for shrunk-on disks. This rotor design will eliminate the existence of highly stressed "hole" and "key-slots," that make the disk susceptible to stress-corrosion cracking and consequent loss of integrity. The turbine is expected to operate free of cracks and the "disk integrity" should not pose a safety problem. The

Power Conversion Systems

NRC recognized the "significantly lower" potential for missile generation in "turbines without shrunk-on disks" in NUREG-0933 (Ref. 10.1). More details about this subject will be developed when a vendor is identified.

The turbine bypass system is designed to accommodate a turbine trip without a reactor trip and to dump excess steam during startup. The turbine bypass system will consist of four automatically operated control valve stations that will discharge steam to the condenser. Interlocks will keep the valves from opening when the condenser is not available.

10.1.3 Main Condenser

The condenser will consist of a single shell with divided water boxes. It will be mounted beneath the low-pressure turbine exhaust. An energy dissipating device will be furnished to allow the turbine bypass flow to enter the condenser without any detrimental effect to the condenser or turbine. The divided water boxes will permit the leaking condenser tubes to be plugged while the unit is operating at reduced power.

The main condenser is designed to condense the maximum turbine exhaust flow. It is also designed to accept up to 50 percent of main steam flow during the turbine bypass mode of operation. The condenser hotwell is designed for a 3-minute storage capacity at maximum load and to provide the capability of deaeration to maintain a dissolved oxygen concentration of less than 0.0005 cm³/l.

The design of the condenser complies with the Heat Exchanger Institute's "Standards for Steam Surface Condensers." It is not regarded as safety-related equipment because emergency decay heat is removed by the reactor auxiliary cooling system (RACS) (see Section 6.3) of this SER.

10.1.4 Condensate and Feedwater Systems

The condensate and feedwater systems are designed to provide high-purity feedwater to the steam generators during both normal operation and anticipated transient conditions. The feedwater will be heated by six feedwater heaters, including the deaerator. These systems will be isolated from the steam generators when

- A chemistry upset occurs that could have an adverse effect on the steam generators.
- The feedwater system is rendered inoperable.

Because plant safety functions can be accomplished without using these systems, they are not specifically designed to accommodate tornados, missiles, or earthquakes; nor will they require emergency standby power to operate. The design of the closed feedwater heaters and the deaerator storage tank complies with Section VIII of the ASME Code and the pertinent Heat Exchanger Institute standards.

The condensate and feedwater systems will be used for normal, non-safety-related decay heat removal through the steam bypass system and the main condenser. When it is not available during loss of outside power or other malfunctions, the heat removal function will be provided by the direct reactor auxiliary cooling system (DRACS) and/or the safety-related RACS.

10.1.5 Condensate Cleanup System

The condensate cleanup system (i.e., condensate polishing unit) is designed to maintain the condensate purity by removing the following contaminants:

- corrosion products that result from the corrosion that occurs in the main steam and turbine extraction piping, feedwater heater shells, drains, and condenser

- suspended and dissolved solids that may be introduced by small leakages of circulating water through the condenser tubes
- solids carried in by the makeup water and miscellaneous drains

The condensate cleanup system polishes 100 percent of the condensate and is designed for continuous performance. Conductivity will be continuously monitored at points within the condensate system, feedwater system, and the steam generator. High-conductivity alarms will alert the station operators to an abnormal condition.

There is no safety consequence from failure of any part of the system. There is sufficient redundancy in the system to negate the possibility of any difficulties in handling the condensate from the condenser when the system is operating at normal influent concentrations.

10.1.6 Circulating Water System

The circulating water system will supply cooling water to condense all steam exhausted to the condenser, while it maintains the turbine exhaust pressure within specified design limits. Heat gained by the circulating water will be dissipated to the atmosphere by means of mechanical draft wet-type cooling towers. The circulating water system will not be required to be operational during or following an accident.

The cooling tower will be constructed of nonflammable materials to minimize the potential for fire. Because of the location of cooling towers, normal drift from the towers will not affect the electrical switchgear and transformers or other outdoor equipment.

As designed, the circulating water system is a non-safety-related system and is not required during an emergency plant shutdown. If available, it is used for normal decay heat removal.

10.2 Scope of Review

The staff has not reviewed the MSSS at the conceptual design stage. This system will be reviewed at the next design stage. At that time, the staff will compare the design of the MSSS with the acceptance criteria in SRP Section 10.3 (Ref. 10.2) for LWRs and the appropriate regulatory guides.

As noted, the turbine generator described in the PSID (Ref. 10.3) is a "typical" unit, and a specific design does not yet exist. A detailed review will be done at the PSAR stage using acceptance criteria in SRP Sections 10.2, 10.2.3, and 10.4.3 (Ref. 10.2) and the appropriate regulatory guides. DOE and its contractor maintain that the turbine generator and its auxiliary systems are not required to function for plant safety purposes. However, these systems will affect the number of transients experienced by the plant and will, therefore, determine the duty cycle. The designer will update the duty cycle periodically as the plant design progresses. The current duty cycle is 120 turbine trips without a reactor trip and 4 with a reactor trip (Ref. 10.3, p. 3.9-8).

The staff will review these values and their significance when a final design is submitted. The staff will also review the turbine disk design at that time.

10.3 Design Criteria

All of the power conversion systems are non-safety related, and classified seismic Category III. The piping is designed to ANSI Standard B31-1, ASME Code Section VIII (tanks use Division 1), or to high-quality industrial standards, for example, the Heat Exchanger Institute standards. Many of the systems will be tested and inspected in accordance with ANSI Standard B31-1.

Power Conversion Systems

The staff will review the standards for these systems at a later stage of the design. At that time, the staff will compare the criteria with the acceptance criteria in SRP Chapter 10 (Ref. 10.2).

10.4 Research and Development Program

No research and development efforts have been identified for the power conversion systems. The staff notes that thermal stresses from startup-shutdown cycles with the high-pressure/high-temperature steam that will be used in the SAFR may require some review of the appropriate duty cycle to be assumed for this plant. This will be reviewed at a later design stage.

10.5 Safety Issues

No significant radioactive contamination of any kind is expected in the PCS. The only possible contaminant will be tritium. Tritium is generated in the reactor and a portion diffuses into the IHTS through the intermediate heat exchanger (IHX). The IHTS sodium cold traps, however, will effectively remove tritium as they remove sodium hydrides. A further buffer to diffusion into the PCS will be provided by the steam generators and the high operating pressure of the steam and feedwater systems. The tritium level will be monitored in the PCS and in the non-radioactive turbine building drains to ensure that unacceptable levels of tritium are not allowed to accumulate in the PCS or be released. The PCS tritium monitor will be installed on the main steam sample line since tritium is expected to concentrate in the steam phase if it is present in the PCS. A high-level reading could be indicative of reduced IHTS cold-trap efficiency. Radiation shielding is not required for any components of the PCS. Access to all components will be provided for maintenance purposes.

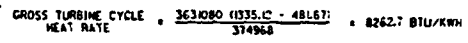
The designer and DOE maintain that operation of the power conversion systems is not required for plant safety. Although the turbine and condenser systems will provide a means for heat removal during normal operation and anticipated operational transients, they are not required, since the passive reactor air cooling system (RACS) is always available to remove decay heat loads. The staff has not identified any direct safety issue at this stage of its review. As noted, the appropriate duty cycle for high-temperature and high-pressure operation of the turbine will be reviewed at the PSAR stage. The staff believes that a thorough evaluation must be made at the PSAR stage of the basis for the judgment that the BOP (in this case the power conversion systems) has no effect on plant safety. Possible interfaces between such systems and the NSSS would be identified and potential interactions would be considered at that point. A better understanding of the capabilities and response characteristics of the NSSS and the integrated NSSS-BOP configuration is needed before a definitive finding can be made regarding the degree to which the BOP and NSSS safety performance are separated.

10.6 Conclusions

Subject to a detailed review of the appropriate acceptance criteria (including the appropriate duty cycle), and evaluation of separability of the NSSS safety performance from the BOP, the staff believes the design of the power conversion systems described in the PSID (Ref. 10.3) can meet staff requirements for safe plant operation.

10.7 References

- 10.1 NRC, "A Prioritization of Generic Safety Issues," NUREG-0933 including Supplements 1-6, August 1987.
- 10.2 NRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," NUREG-0800, July 1982.
- 10.3 DOE, "SAFR - Sodium Advanced Fast Reactor," Preliminary Safety Information Document, AI-DOE-13527, Initial Issue - October 1985, with amendments. (DOE Applied Technology.)



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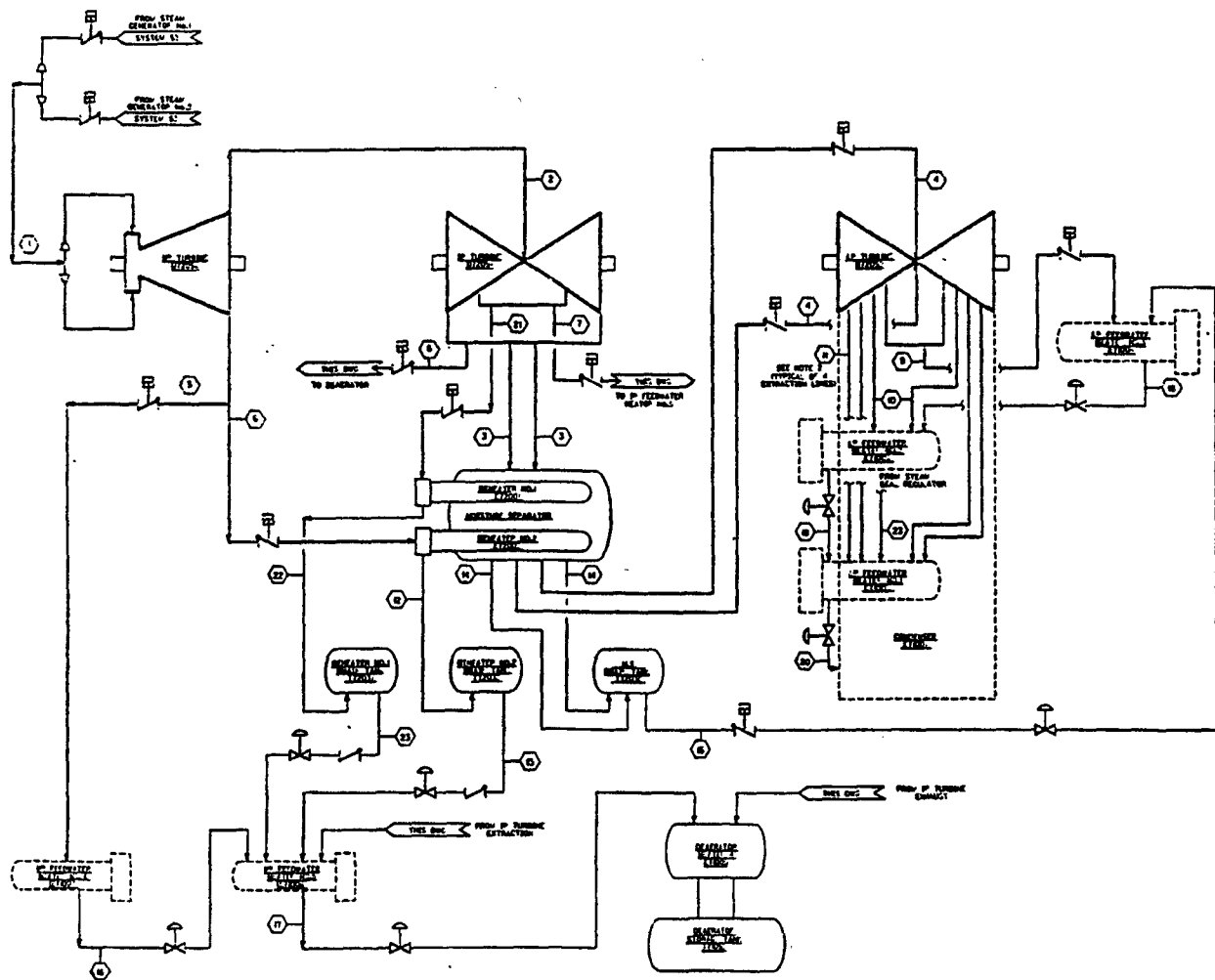


Figure 10.2 Heat balance.

11 RADIOACTIVE-WASTE MANAGEMENT

11.1 Design Description and Safety Objectives

11.1.1 Liquid-Waste Management Systems

The liquid-waste management system (LWMS) will provide the equipment and facilities for collecting, processing, monitoring, storing, and disposing of liquid radioactive waste (liquid radwaste). The LWMS is designed to ensure compliance with Federal regulatory standards and safety rules described in 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 100.

The LWMS will collect, process, monitor, store, and provide load-out capability for liquid radwaste being released by (1) personnel decontamination, (2) LWMS sampling operations, (3) radioactive maintenance operations (primary sodium removal and decontamination), (4) radioactive liquid spills, (5) heating, ventilation, and air conditioning (HVAC) filtering (potentially radioactive condensate), (6) intermediate heat transport system (IHTS) sodium purification cold-trap regeneration (potential for tritium-contaminated water), and (7) the main steam system (monitor and isolate only for tritium-contaminated water).

The LWMS is designed to

- Control the release of radioactive materials in liquid effluents with sufficient holdup capacity for retention until the liquid radioactive waste can be loaded out of the plant for mobile processing on site.
- Ensure adequate safety during normal and abnormal operation, suitable shielding, and appropriate containment of the liquid radwaste.
- Monitor liquid radwaste for excessive radiation levels and effluent discharge paths for radioactivity that may be released from normal and abnormal operation.

The LWMS will consist of a small (500 gal) collection tank for each power pak and the health physics operations in the administration building, a central collection tank (5000 gal), and a small liquid-radwaste processing unit for potentially radioactive liquid waste from the power paks and health physics operations. The LWMS will also provide a means of loading out those liquid wastes that cannot be processed by the central waste processing unit. The yearly volume is considered low, since the only major source of waste is the sodium cold trap and decontamination activities during maintenance operations.

The power pak (module) collection tank will be sized in direct proportion to the volume of liquid flow from the most rapid and largest source, the personnel shower. A transfer pump will be provided within the collection tank to transfer collected liquid to the larger central collection tank or to the central processing unit.

The central collection tank will accept flow from the modules and nuclear island maintenance building (NIMB) radioactive maintenance operations. The tank will be equipped with an agitator and an internal pump. The pump will circulate tank waste to obtain a representative sample and periodically to transfer collected waste to a processing unit.

Sodium removal activities and decontamination of large and small radioactive components is the largest source of liquid radwaste. Steel activation products and solid fission products are removed when radioactive components are decontaminated, and these are presumed to be present in the liquid transferred to the waste system. The volume of the waste will vary according to the component and the degree of decontamination necessary. About 100 gallons of waste liquid will be drained to the radwaste system (RWS) per maintenance washdown. The activity levels of those liquid radwastes are

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system specific and will vary depending on the physical location of the component within the plant. In general, the radioactivity levels of liquid wastes from decontamination activities will be higher than radioactivity levels in other liquid wastes. The central processing system is not designed to process this higher activity waste. That waste will be sent directly to the central waste collection tank for temporary storage until more intensive load-out processing is contracted, including final solidification. This will probably take place about twice a year, once all four power paks have reached stable power operation.

Analyzed LWMS effluent will be discharged to the environment on a monitored batch basis only and will be discharged with the cooling tower blowdown flow(s), via the nonradioactive liquid-waste system. This will reduce the plant effluent radioactivity concentrations to below the values allowed by 10 CFR Part 20.

The major equipment of the central liquid-waste processing unit will consist of a cartridge filter, a disposable demineralizer, a monitor tank, and a monitor tank pump. Filter and resin cartridges will be replaced when the pre-set differential pressure or radiation limits or both are reached. Processed water will be held in the monitor tank for radiological analysis before it is discharged to the environment.

The main RWS facilities will be located in the NIMB. Equipment in the NIMB will be located and arranged to isolate the operating and maintenance areas from operating equipment so that plant personnel will be exposed to as little radiation as possible.

Nonradioactive auxiliary equipment and components, that accommodate the operational and safeguard requirements of LWMS facilities will be isolated from the radioactive components and piping in the buildings. Shield walls and limited-access cells have been used extensively to protect plant personnel against radiation exposure. Radioactive piping is designed, where feasible, to eliminate dead legs and low points in which radioactive materials may accumulate, as well as to segregate radioactive from nonradioactive lines. The piping design restricts the routing of radioactive lines through nonradioactive areas. Components and equipment will be arranged to permit maintenance to be performed by means of flushing and decontamination connections. Areas in which radioactive spills can contaminate the floor will be fitted with decontamination facilities, washdown facilities, and radionuclide-collection facilities. Additionally, the floors in these areas will be treated to prevent radioactive materials from seeping into the concrete. The floor below the components and pipes that contain radioactive material has been designed to withstand seismic loads and still retain the spilled fluid. The floor is designed as a basin to prevent spilled fluid from leaking into other areas.

At present, no radioactive laundry facility is included in the plan as only a minimal quantity of contaminated laundry is anticipated. Should the fuel cycle facility (FCF) be in operation, laundry from the power paks will be sent to the FCF laundering facility. Until that time, contaminated laundry will be sent to an offsite laundering facility or will be treated as solid waste.

The LWMS is not safety related because of the low activity level of the liquid collected and the relatively slow release of any liquid leaked from the system. Although system components are seismic Category III, the LWMS is located in the seismic Category I reactor service building (RSB) and the seismic Category II NIMB. Cells containing LWMS equipment are designed to retain leakage under all conditions. LWMS components are designed consistent with commercial practice and standards and conform to applicable regulations. The design uses stainless steel piping and components and all-welded construction to restrict leakage. Further, sump pumps will be located beneath the two major tanks and in collection areas, so that leakage will be returned to the collection system. All components will be tested and inspected as separate components and as integrated subsystems. Tests will be performed in accordance with applicable regulations.

11.1.2 Gaseous-Waste Management Systems

Gaseous waste from the SAFR will be managed by subsystems of the inert-gas receiving and processing system (IGRPS). These are the helium recycling subsystem, the inert-gas monitoring subsystem, and the radioactive inert-gas processing subsystem.

The gaseous-waste management system is designed to meet the following requirements:

- Process both the reactor and the primary component effluent cover-gases before they are vented to the atmosphere.
- Maintain the negative pressure in the primary process subsystems interfacing the reactor vessel (RV) at a higher pressure than the vessel's negative design pressure.
- Monitor and process, if necessary, the gas vented from cells and enclosures that is potentially radioactive.
- Monitor and recycle for continued processing the processed gas discharge activity that temporarily exceeds the discharge limits.
- Maintain radioactive releases to the environment within 10 CFR Part 20 and 10 CFR Part 50 (Appendix I), limits for normal releases, and 10 CFR Part 100 for accidents.

The gaseous waste management system is designed to

- Provide sufficient holdup capacity and control releases of radioactive materials in gaseous effluents.
- Provide adequate safety during all phases of operation, including shielding, and adequate containment of radioactivity.
- Monitor gaseous radwaste for excessive radiation levels; and monitor effluent discharge paths for radioactivity that may be released from all phases of operation.

The helium recycling system will use two parallel, alternating, charcoal delay columns to separate the krypton and xenon fission gases from the helium cover-gas of the RV. To reduce the chances for nitrogen contamination of the helium from air in-leakage, recycling will be limited to the RV cover-gas and the negative pressure of certain portions of the system will be kept to a minimum. Helium will be vented through the RV pressure maintenance valve at a normal flow rate of about 1.7 to 2.7 scfm. The vent line and vacuum tank will be maintained at a negative pressure (10.7 psia) by two parallel compressors. The compressors will exhaust to a surge tank at 30 psig with enough capacity to maintain a steady feed to the charcoal delay column.

The two delay columns will alternate in stripping heavier molecules of xenon and krypton from the recycling stream by means of molecular transport delay in activated charcoal columns. While the second column is stripping krypton and xenon from the helium, the first column will be allowed to increase in temperature and to outgas to the radioactive inert-gas processing system.

On leaving a column, the helium will be collected in the recycle tank. Since the pressure drop through the column(s) will be low, the tank pressure will be slightly lower than that of the surge tank (about 25 psig). Fresh helium will be added to the recycle tank, as necessary, to replace losses from leakage and outgassing to the processing subsystem; also, helium will be added, as necessary, to control the nitrogen impurity level in the recycled helium. During scrams and other shutdowns, when the RV sodium temperature is lowered, fresh helium will be added to offset sodium shrinkage and cover-gas cooling. During reactor startup, when the RV gas is heated and its volume is

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decreased, excess helium will be vented from the recycle subsystem to the radioactive inert-gas processing subsystem.

The inert-gas monitoring subsystem accepts inert gas from the following sources: fuel transfer cell (FTC) (helium and air); RV-containment vessel annulus (helium and air); and RV head components (helium).

The inert gas or air (after maintenance operations) is normally nonradioactive or very low in activity. After monitoring to ensure that the radioactivity level is low, the gas/air mixture can be mixed with the air exhausted from other portions of the nuclear island and released to the environment through a high-efficiency particulate air (HEPA) filter.

The gas entering this subsystem will be collected and monitored for radioactivity before release, consistent with the requirements and limits of 10 CFR Part 20 and 10 CFR Part 50 (Appendix I). If the most stringent limits of either requirement would be exceeded by direct release, the gas will be exhausted automatically to the radioactive inert-gas processing subsystem.

The radioactive inert-gas processing subsystem will be centrally located, will serve all SAFR modules, and, except for piping connection to the modules, will be located in the NIMB. Considering each module separately, sources of radioactive gas will be

- outgassing from the recycle columns
- purge or once-through operation of the cover-gas system
- sampling of cover-gas system (chromatograph)
- radioactive gas from the inert-gas monitoring system
- pressure control of the primary component cover-gases
- effluent gas from the IHTS cold-trap regeneration (optional)

The main components of the subsystem will be a vacuum tank, two vacuum blowers, a surge tank, and six charcoal delay beds holding a total of about 17 tons of charcoal. The vacuum blowers will maintain the vacuum tank at a negative pressure. An adequate differential pressure exists to accommodate any abnormal surge flows of gas. The function of the surge tank is to hold the gas and release it at a fixed discharge pressure to the charcoal delay tanks. After passing through the tanks, the gas is monitored and either released to the atmosphere through the NIMB exhaust or is reprocessed by reentering the vacuum tank and undergoing processing again. The charcoal delay columns will be operated at ambient temperature (approximately 80°F). The major inventory of radioactive materials will be in the charcoal tanks. The entire inventory from four power paks in a bypass mode (inoperable helium purification subsystems) is less than that requiring safety-related charcoal delay tanks.

The maximum normal operating pressure will be about 10 psig in the surge tank and outlet of the blowers. The subsystem capacity will be of such a size to contain the total normal flow rate of four power paks (modules) with once-through cover-gas systems (without recycling) and one inert-gas monitoring subsystem. The total flow is expected to be about 10.0 scfm. Given this flow rate, the total atmospheric release of this subsystem and the inert-gas monitoring subsystems of the individual modules will not be allowed to exceed the more stringent limits of either 10 CFR Part 20 or 10 CFR Part 50 (Appendix I). The processing subsystem effluent will be monitored and if the radioactivity level of the gas leaving the delay beds is too high, the effluent will be reprocessed automatically.

The gaseous-radwaste processing system is not safety related, except for the containment penetrations and the standard motor-operated containment isolation valves used by the helium recycling subsystem (primary containment) and the inert-gas monitoring subsystem (secondary containment). These portions will be kept at a negative pressure by the vacuum blowers and compressors and could be a path for radioactive leakage from the containment.

Portions of the piping containing radioactive gas will use all-welded construction and bellows-sealed valving to restrict leakage. Blowers will have low leakage requirements or will be enclosed to contain leakage.

The low pressure in the processing charcoal delay beds will not only reduce the radioactive inventory of xenon and krypton gases in the beds but will also reduce the rate of release from the beds in the event of an inadvertent tank depressurization. Release of radioactivity caused by a charcoal bed fire may not be credible because of the inert gases being processed. The result is a postulated design-basis accident (DBA) release that is determined by the applicant to result in offsite doses of less than 0.5 rem. Calculated offsite doses are dependent on site-specific factors and, therefore, will be verified when these factors are quantified.

The processing subsystem sizing assumes all four power paks have to use once-through operation with the additional flow from one inert-gas monitoring system, without exceeding release limits specified in 10 CFR Part 20. The allowable rate of nonaccidental release from this subsystem with once-through cover-gas system, assuming no reactor cover gas recycling, is highly dependent on the amount of radioactive argon-41 in the processed gas released to the environment. Argon-41 is expected to be the dominant released isotope and will be continuously monitored. Additional delay or reprocessing will be added if needed.

The processing subsystem will be located in the NIMB and will be connected to the power paks by a buried pipe in a trench shared with the liquid-radwaste collection line. The pipe will be embedded in concrete within the trench. The NIMB will be a seismic Category II building. The gas-processing subsystem will be seismic Category II, designed to good commercial practice and applicable regulations.

All components will be tested and inspected as separate components and as integrated subsystems. The key component variables are the blower and compressor flow rates, their resulting inlet and exit pressures, and the efficiency of the charcoal delay columns and beds. The latter will be checked during operation of the first power pak to ensure that they will be viable and can be extended to all four power paks.

Containment isolation valve operators are qualified and tested against IEEE Standard 323.

All subsystems will be continuously monitored, as necessary, during all phases of operation (pressure, temperature, flow, and radiation). The design values of the piping and tanks are well within the maximum values expected for any of the variables. The inert gas precludes any chemical action that would lead to deterioration of the subsystem components.

11.1.3 Solid-Waste Management Systems

The solid-waste management system (SWMS) comprises the equipment and facilities for collecting, compacting, monitoring, storing, and preparing wastes for shipping and disposal.

The SWMS is designed to receive such wastes as

- compatible solids (e.g., clothing, shoe coverings)
- low-activity, noncompatible solids (e.g., tools, components)
- high-activity, noncompressible solids (e.g., non-fuel-core components)
- metallic sodium in containers
- sodium-bearing solids
- optional liquid-waste-solidification residue

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The SWMS is designed to package solid wastes for shipment to a Federally licensed or State-licensed burial site. The surface dose rates for the packaged shipment containers will comply with Nuclear Regulatory Commission (NRC) and Department of Transportation (DOT) regulations.

The system is designed to process or handle four types of radioactive waste (i.e., compatible solids, noncompatible solids, metallic sodium (received by the SWMS in drums that meet DOT specifications), and sodium-bearing components).

The solid-radwaste disposal management system will control the safe packaging and handling of radioactive wastes, including sodium-contaminated radioactive wastes, and nonradioactive wastes contaminated with sodium. Remote-handling equipment will transfer such solid wastes into storage areas and from the storage areas to transportation vehicles.

Approximately one or two control rods will be replaced in each power pak every year. This will be the major source of high-activity, noncompressible waste. Plans call for reusing the upper portion of the rod and sectioning the lower portion for disposal. The method of shielding is yet to be determined.

The processing and disposal of metallic sodium and sodium-bearing solids are being studied and the preferred methods are yet to be determined. In response to the results of these studies at other DOE liquid-metal reactors and at the SAFR, applicable methods of processing, transportation, and disposal will be developed.

A compactor reduces the volume of compressible low-activity wastes. Although they are not compressible, small tools and small portions of components that are considered contaminated (low activity) may be included.

A facility to clean contaminated clothing and other washable items is not included in the current SAFR design, but the need for such a facility is being evaluated. Laundry service will be contracted if it is possible, especially for the operation of the first power pak. The service will be added later when the increased need from additional power paks makes onsite laundering of contaminated materials economical.

The SWMS is not safety related and is designed as seismic Category III. Components are designed to good commercial practice in accordance with applicable regulations. Health physics (HP) personnel will monitor wastes and determine the activity of wastes prepared for offsite disposal. Components and support system design precludes radioactive plant releases from collection, compaction, storage, and the preparation of solid wastes for shipping. All components will be tested and inspected by HP personnel for operability whenever they are to be used.

11.1.4 Process and Effluent Radiological Monitoring and Sampling Systems

Process systems that could, or do, contain radioactivity during normal operation, or that serve as the first barrier to release from such systems, will be monitored to detect, measure, and record radiation levels. This monitoring will protect plant personnel and the public from exposure to radiation exceeding regulatory limits. The process monitors will perform the following functions:

- Monitor the cover-gas effluent from the primary containment and transmit a containment isolation system (CIS) signal to isolate both the primary and secondary containments when excessive radioactivity is detected in the cover-gas effluent piping. These monitors are safety related.
- Monitor the reactor containment building (RCB) ventilation exhaust, transmit a signal to the control room, and emit a high-level radiation alarm. These monitors are used for post-accident monitoring (PAM) and are safety related.

- Monitor to ensure that the main control room and technical support center (MCR/TSC) areas are supplied with clean air following a release of airborne radioisotopes outside the control complex.
- Monitor the primary cover-gas and sodium for fuel failure.
- Monitor helium recycling of the cover-gas to activate the standby purification column on radiation increase.
- Monitor inert gases vented from the radioactive sodium and fuel-handling equipment cells and enclosures and transmit a signal to divert these gases to a cleanup system if excessive radioactivity is detected.
- Monitor the radioactive-waste systems and the exhaust air in the discharge stacks for radiation levels acceptable for discharge.
- Permit post-accident sampling of the secondary containment.

The process radiation-monitoring group will detect and measure radiation levels in the associated process systems throughout the plant. Annunciation of abnormal events of each process radiation level will give audible and visual alarms on the local panel and also in the main control room for each process radiation level.

Process monitoring involves monitoring

- gross cover-gas activity (CIS) (safety related)
- air exhausted from the RCB (PAM) (safety related)
- air in the MCR/TSC areas
- reactor sodium and cover-gas for fuel failures
- helium recycling of reactor cover-gas
- normally nonradioactive inerted atmospheres (both collectively and individually)
- radioactive-waste systems (both gas and liquid processing)
- air exhausted from potentially contaminated air atmosphere cells in the RSB and the NIMB
- post-accident sampling of the RCB

Only the CIS and PAM monitors are safety related. The CIS monitors will have Class 1E electrical power. These monitors (four channels) will be separated and installed on the cover-gas effluent piping and will be located between the reactor and the primary containment isolation valves. A high-radiation signal, indicating that 0.1 percent of the core fuel is releasing fission gas, from any two of the four monitors will be transmitted locally to the CIS. The two-out-of-four logic will ensure that periodic maintenance, or failure of any one monitor and a possible false signal, does not activate the CIS. During the short repair period, two-out-of-three logic will be used. An isolation signal will isolate both primary and secondary containment.

The RCB exhaust monitors (PAM) (four channels) are normally supplied with Class 1E power, used to monitor the activity within the RCB, and can also be used for PAM. They will be physically separated and located on the seismically hardened RCB air exhaust duct, upstream of the RCB isolation valves. They also will use two-out-of-four and two-out-of-three logic, as described above, to limit false alarms. RCB isolation will require operator action.

The HVAC process radiation monitors for the MCR/TSC air will perform monitoring functions similar to corresponding LWR functions. Two monitors (each with the ability to detect particulates, radioiodines, and radiogases) will be provided by the process monitoring subsystem to continuously verify that the ventilation air, taken into the MCR/TSC during normal and abnormal conditions, is suitable for habitability. One monitor will be located in the outside-air intake for the control room. The second monitor will be in the MCR/TSC supply-air duct. The output of each monitor will be

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indicated by a rate meter mounted on the radiation monitoring panel display in the control room. The ventilation air monitors will initiate the transfer from low-efficiency, outdoor-air filtration to high-efficiency filtration and adsorption by supplying actuation signals to the air-intake valves and filtration fans when the "high" radiation set point is exceeded. The outdoor-air intake valves could also be closed by toxic gas, smoke, and aerosol detectors. The second monitor in the control room supply duct, along with local room monitors, will enable the operating personnel to decide if anything more than nuclear filtration is necessary.

The fission-gas monitor will be located downstream of the bleed pressure control valve of the reactor cover-gas. The process gas stream monitored can be either recycled helium or once-through helium. This process monitor will be a multichannel sodium iodide (NaI) detector. Radioactivity data will be analyzed by a computer spectrometer and can be monitored in the control room. The NaI detector and spectrometer-computer attachment can identify a complete spectrum of cover-gas isotopes, can provide data to operating personnel, and can detect gross fuel failures.

The helium-recycling columns monitor will allow the plant to reclaim most of the helium used by the primary cover-gas and to recycle this gas back to the reactor components and cover-gas. The helium-recycling columns will separate radioactive fission gases of xenon and krypton isotopes from the same cover-gas flow that passes the fission-gas monitor. The NaI process monitor, located at the exit of the separating columns, will detect activity that begins to pass through the column in use. Signals from the monitor will isolate the column from the recycle cover-gas flow, open the column so it may outgas to the radioactive-gas processing subsystem in the NIMB, and open the second column to recycling flow. Therefore, the monitor will alternate the columns and the respective separation and outgassing cycles depending on the radiation level acceptable for recycled helium.

The inert-gas monitoring subsystem will accept potentially contaminated gas from the inerted fuel-transfer cell (FTC) and any inerted enclosures or primary seals within the power pak. The gas will be monitored for radioactivity. If radioactivity is at an acceptable concentration, it will be released with RSB HVAC exhaust air. If excessive radiation is detected in the gas, the monitor will automatically close the valve to the RSB exhaust and open the valve to the central gas processing facility. All the inert atmosphere in the RCB and the FTC will be continuously sampled and the gases will be sequentially monitored for radiation and impurities. The source of contamination will be identified and subsequently isolated.

Gaseous radwaste received by the inert-gas processing subsystem is considered to be radioactively contaminated by fission gases, tritium, and argon. These gases will be pumped to a series of tanks filled with charcoal. Enough charcoal is required to provide a 40-day delay (minimum) of xenon. The monitor will measure the magnitude of released isotopes in the gas leaving the delay beds to ensure that the beds are adequately cleaning up the gas before release. If any one isotope exceeds the limits established for release, the monitor will emit signals that isolate the vent and open the bypass line so that the exiting gas is sent back to the inlet of the processing system and is not allowed to enter the NIMB air exhaust.

Liquid radwaste will be monitored both at the nuclear islands and at the central collection tank in the NIMB. Both a liquid composite sample monitor and a scintillation monitor will be used on the central collection tank. Liquid samples can be taken at all tanks.

Post-accident monitors will allow radiation sampling of the secondary containment atmosphere during and after an accident at the discretion of control room personnel. Redundant sample lines will draw gas from the secondary containment, permit monitoring, and return the sample gas to the secondary containment. Since these lines will normally be closed and kept closed by the secondary containment isolation valves if radioactivity in the containment air spaces causes secondary containment isolation, the operator will have to override the isolation signal manually to initiate system actuation. The two redundant lines will ensure that the loss of one line does not preclude operation of the other sampling line. The detectors will be wide range, portable, and normally used in other areas of the plant.

Two methods of sampling the secondary containment atmosphere will be available. The first method will use the mobile particulate, radioiodine, and radiogas containment atmosphere monitors (CAMs) to continuously withdraw and analyze the containment atmosphere. Two lines, appropriately shielded, will connect a CAM to a pair of penetrations. The air sample will be withdrawn through one line, analyzed in shielded chambers, and returned to the containment. The second method of determining the radioactivity in the containment atmosphere is to withdraw an air sample into a portable shielded sample chamber and perform the analysis in the counting room. This method will allow a more complete and accurate determination of radionuclide concentrations but must be time calibrated.

The CIS and PAM radiation monitors will be safety-related, will conform to Class 1E requirements, will be supplied with Class 1E emergency power, will be seismic Category I, and will be installed on the cover-gas piping that penetrates the primary containment and on the RCB exhaust. Piping, blowers, and valves in the post-accident sampling subsystem, outside the containment, will be an extension of the secondary containment boundary when the subsystem is in operation. The subsystem boundary, including the normally closed containment isolation valve, will neither be safety related nor designed to ANSI Standard B31.1 specifications; it will be seismic Category II. The portable monitors will not be safety related, since operational monitors will be available, calibrated, and preferred. Non-Class 1E emergency power will be available for the portable monitors, blowers, and valves. Radioactivity release will be through the effluent monitors.

All other monitors will be used on non-safety-related process systems and will be seismic Category III. Components are designed to good commercial practice, in accordance with applicable regulations.

Monitors will be checked regularly for proper operation by both HP and instrumentation personnel. Monitors will be equipped with test channels and will also be periodically calibrated in the counting room. Failures will be indicated both locally and remotely.

Effluent radiation monitoring subsystems are designed to meet the following requirements:

- Effluent from the plant is continuously monitored to detect, measure, record, and annunciate, as required, concentrations of radionuclides that are released to the environment. Dose assessments are made to verify that plant releases are below the limits of 10 CFR Part 20 and 10 CFR Part 50 (Appendix I), and reports are issued.
- Each effluent monitor output is continuous and can be displayed in the control room on video displays. Annunciation occurs when radioactivity levels in the effluent exceed a monitor's set point. Effluent output and radioactivity levels in the effluent are permanently recorded.
- The monitoring process complies with NRC Regulatory Guide 1.97 (Ref. 11.2) to monitor fission products released from the primary system.

The effluent radiation monitoring subsystem will provide continuous monitoring or sampling or both for all discharge points that serve radioactive processes or areas (ventilation) and the continuous monitoring that would be required if an accident were to occur. Gross radioactivity concentrations will be monitored continuously in discharge lines and ventilation exhausts that could experience significant changes in effluent radioactivity level. At all potentially radioactive effluent discharge points, samples will be taken to measure specific radionuclide concentrations in the discharge to the environment. Results of the counting-room analysis of collected samples and of the continuous monitoring will be permanently recorded and reported to the NRC in a manner consistent with the guidelines of NRC Regulatory Guide 1.21 (Ref. 11.3).

The following equipment is included in this equipment group:

- stationary particulate, radiogas, and radioiodine CAMs (with fixed-air sampler channel)
- stationary particulate and radiogas CAM (with fixed-air sampler channel)

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- stationary radiogas CAMs
- stationary tritium samplers
- stationary particulate air samplers
- composite liquid sampler
- stationary two-channel plutonium monitor

During an accident, high dose rates may prevent manual sampling at the discharge point. Equipment with an extended range will provide gross count rates that serve as a means of monitoring the progress of the event and the operation of the equipment serving to mitigate the consequences of the event. Environmental sampling techniques will be employed to obtain more definite information on releases. When defined by the NRC, perimeter and environmental monitors will be installed.

CAMs (radiogas, particulate-radiogas, and particulate-radioiodine), tritium samplers, and particulate samplers will be installed to monitor and indicate gross radioactivity concentrations in gas analyzers and gaseous discharges.

CAMs (particulate-radioiodine and particulate-radiogas) will be installed for monitoring the postulated DBA. A monitor, qualified to NRC Regulatory Guide 1.89 (Ref. 11.4) on the nuclear island HVAC effluent line, will furnish information on the plutonium activity released to the environment. A plutonium monitor, also on the effluent line, will furnish information on the plutonium released to the environment, and will provide backup to the units monitoring filter performance.

Each abnormal condition will annunciate locally on the monitor housing, and will indicate on the radiation monitoring video display that corresponds to the specific abnormal event.

All gaseous effluent discharge points, except the accident release points, will be periodically sampled. Sampling stations will consist of a regulated air pump (to maintain desired sample flow) and an appropriate collection device. Sampling stations will also be installed at locations where continuous monitoring is performed to allow an isotopic identification of the gaseous effluent. Samples will be prepared and analyzed in the counting room located next to the HP room in the administration building. The radioisotopic analysis (including tritium analysis) will be performed using the counting-room instruments. Permanent records will be maintained for all analysis runs.

A 24-hour composite sample will be taken of the liquid effluent leaving the plant through the plant discharge pipe. The sample will be analyzed in the counting room for radioisotopic and tritium content. Permanent records will be maintained for all analyses.

Monitors will be checked regularly for proper operation by both HP and instrumentation personnel. Monitors will be equipped with test channels and will also be calibrated periodically in the counting room. Failures will be indicated locally and in the control room. Monitors will be tested in conformance with applicable regulatory requirements.

Those effluent monitors that have the potential for detecting a release that could result in an offsite (boundary) dose exceeding 0.5 rem will be safety related, supplied with Class 1E emergency power, and seismic Category I. At the present time, none of these monitors have been identified by DBA dose studies. The other process monitors will be seismic Category III and designed to good commercial practice in accordance with applicable regulations.

11.2 Scope of Review

The staff reviewed the radioactive-waste management systems used in the SAFR conceptual design for consistency with existing NRC requirements currently applied to light-water reactors. These requirements are defined primarily by 10 CFR Part 20, 10 CFR Part 50 (Appendix I), and SRP Sections 11.1 through 11.6 (Ref. 11.5).

11.3 Design Criteria

The SWMS is designed to comply with applicable regulations on handling and shipping, including 10 CFR Part 20, 10 CFR Part 50, 10 CFR Part 61, 10 CFR Part 71, and 10 CFR Part 100, and the applicable regulatory guides. It is designed to

- Control the release of solid radioactive materials with sufficient storage capacity for retention until radwaste can be removed from the plant.
- Ensure adequate safety during storing and handling, including accident situations, suitable shielding, and appropriate containment of radioactive material.
- Monitor excessive radiation levels and effluent discharge paths for solid radwaste during storage, handling, and accidents.

The process monitors are designed to comply with all applicable regulatory requirements. Federal requirements are contained in 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 100. The process monitors are designed to

- Control releases of radioactive materials to the environment.
- Provide adequate radioactivity control in systems monitored with appropriate shielding, inspection, and testing of safety-related instrumentation, and containment of radioactivity.
- Monitor radioactive waste systems to avoid excessive radiation levels and to provide the means to monitor the primary and secondary containment atmospheres and effluent discharge paths.

11.4 Research and Development Program

No R&D programs were presented in the PSID (Ref. 11.1), nor did the staff find that any were necessary.

11.5 Safety Issues

The staff noted no significant safety issues. Additional information requirements are the only issues remaining as a result of the staff's reviews. These are

- Source-term information used as the design basis for expected releases should be provided to demonstrate that the applicable requirements of 10 CFR Part 20 and 10 CFR Part 50 (Appendix I) are met.
- Regulatory Guide 1.143 (Ref. 11.6) provides design guidance which may be applicable to the LWMS. It is suggested that this guidance be considered and described as appropriate in future SAFR safety information documents.
- Information should be provided on postulated releases should tanks containing liquid wastes fail. Acceptance criteria for these postulated releases are the radionuclide concentrations at the nearest potable water supply, not in excess of 10 CFR Part 20 (Appendix B, Table II, Column 2).
- Regulatory Guide 1.143 (Ref. 11.6) provides design guidance which may be applicable to the IGRPS. It is suggested that this guidance be considered and described as appropriate in future SAFR safety information documents.

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- Additional information should be provided on postulated releases if a waste-gas system leaks or fails. The acceptance criterion for these postulated releases is a total body exposure to an individual at the nearest exclusion area boundary not in excess of 0.5 rem, which is consistent with the guidelines of 10 CFR Part 20 and is substantially below the guidelines of 10 CFR Part 100.
- Regulatory Guide 1.143 (Ref. 11.6) provides design guidance which may be applicable to the SWMS. It is suggested that this guidance be considered and described as appropriate in future SAFR safety information documents.

11.6 Conclusions

The radioactive waste management systems utilized in the SAFR design appear to meet the intent of the regulatory requirements for such systems, subject to satisfactory consideration of the additional guidance suggested above.

11.7 References

- 11.1 DOE, "SAFR - Sodium Advanced Fast Reactor," Preliminary Safety Information Document, AI-DOE-13527, Initial Issue - October 1985, with amendments. (DOE Applied Technology.)
- 11.2 NRC, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident," Regulatory Guide 1.97, Rev. 3, May 1983.
- 11.3 NRC, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents From Light-Water-Cooled Nuclear Power Plants," Regulatory Guide 1.21, Rev. 1, June 1974.
- 11.4 NRC, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," Regulatory Guide 1.89, Rev. 1, June 1984.
- 11.5 NRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," NUREG-0800, July 1982.
- 11.6 NRC, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," Regulatory Guide 1.143, Rev. 1, October 1979.

12 RADIATION PROTECTION

12.1 Summary Description and Safety Objectives

Chapter 12 of the PSID (Ref. 12.1) gives information on radiation protection methods and estimated occupational radiation exposures to operating and construction personnel during normal plant operations and anticipated operational occurrences (AOOs).

The radiation protection measures incorporated for the SAFR design are intended to ensure that internal and external occupational radiation exposure to plant operating personnel, contractors, administrators, visitors, and the general population as a result of station conditions, including AOOs, will be within the limits of 10 CFR Part 20, "Standards for Protection Against Radiation," and will be as low as is reasonably achievable (ALARA).

12.1.1 Design Considerations

The objectives of the radiation protection design are

- Minimize the necessity for going into radiation areas and amount of time spent there.
- Minimize radiation levels in routinely occupied areas and in the vicinity of plant equipment expected to require personnel attention.
- Limit occupational radiation exposure to 20 man-rem per year.
- Meet the requirements of 10 CFR Part 20 and 10 CFR Part 50 during plant operations, shutdown, and refueling.

Some of the design considerations used to meet the plant objectives include modularization of radioactive components for ease of disassembly and removal to lower radiation areas for repair, remote operation including use of special tools or equipment, use of labyrinth entrances to shielded cubicles, utilization of remote viewing devices, and provisions for venting, purging, and decontamination to reduce radiation levels in systems that may experience plateout. These design considerations conform to the guidelines of Regulatory Guide 8.8 (Ref. 12.2).

12.1.2 Operational Considerations

DOE's operational considerations include the development and implementation of operating plans and procedures for controlling exposure to radiation as discussed in Regulatory Guides 8.8 and 8.10 (Ref. 12.3). These operating plans and procedures will cover system operation, maintenance, surveillance, testing, fuel handling, emergencies, radiation protection, and administration. Procedures prepared for workers in radioactive areas should ensure that

- Adequate preparation and planning precedes the undertaking of applicable activities.
- Work that is performed adheres to appropriate radiation protection recommendations and support.
- Evaluations produced during post-work debriefings are used to improve future activities.

12.2 Scope of Review

The basis of the staff's review is that doses to personnel will be maintained within the applicable limits of 10 CFR Part 20, "Standards for Protection Against Radiation." The applicant's radiation protection design and program features must also be consistent with the guidelines of Regulatory Guide 8.8 (Rev. 3).

12.3 Design Criteria

The applicant has committed to ensure that the SAFR will be designed, constructed, and operated in a manner consistent with Regulatory Guides 1.8 (Rev. 1), 8.8, and 8.10.

The SAFR's ALARA program will be managed in two phases throughout the lifetime of the plant. The first phase will consist of the ALARA activities associated with design, fabrication, construction, and preoperational testing activities. For this phase, the ALARA program will involve the interaction of multiple engineering disciplines (i.e., radiation analysts, shielding designers, shielding analysts, system designers, and component designers). The elements in this stage are as follows:

- establishment and control of estimated radiation exposure levels
- design of the components and systems to achieve the estimated radiation exposure and shielding objectives
- reviews by a SAFR ALARA committee to evaluate and manage the achievement of the objectives for radiation exposure
- reviews by experienced health physicists to obtain current LWR and LMR information

The second phase will consist of ALARA program policies and procedures that will apply during plant operations through the decommissioning of the SAFR.

This ALARA philosophy was applied during the initial design of the plant. The applicant will continue to review, update, and modify the design based on ALARA policy consideration.

12.4 Research and Development Program

Additional R&D is neither planned nor needed.

12.5 Safety Issues

This section contains a description of the sources and airborne radioactivity used as inputs for the dose assessment and for the shielding and ventilation designs. Also included are the assumptions made by the DOE in arriving at quantitative values of the contained and airborne source terms. The basis for acceptance in this review is that all sources of radiation that necessitate designed shielding, special ventilation designs, or access control considerations should be described to the degree needed for the shielding codes used in the design process.

The major sources of radiation within the reactor building are the reactor vessel (containing the radiative primary heat transfer system), spent fuel transfer (reactor vessel and fuel transfer cell connection), cover-gas piping, and components and seals within the reactor vessel head. The major sources within the containment building are fuel transfer operations and cover-gas purification and monitoring. The major sources in the nuclear island maintenance building are radioactive maintenance and decontamination activities, radioactive-gas processing, and liquid-radwaste collection. The location of solid-waste collection, compaction, and temporary storage has not yet been determined.

The next design stage should contain the following information relative to radiation sources in normal operations, anticipated operational occurrences, and accident conditions affecting in-plant radiation protection:

- the description of radiation sources used as the basis for designing the radiation protection program and for shield design calculations

- the description of airborne radioactive material sources in the plant considered in the design of the ventilation systems, and used for design of personnel protective measures and for dose assessment

12.5.1 Shielding

Radiation shielding will protect operating personnel (both inside and outside the plant) and the general public (during normal operation, anticipated operational occurrences, and accidents) against radiation. The shielding is designed to meet the requirements of the radiation dose rate zone system discussed above.

Shielding thicknesses are based on the desired personnel radiation zones and the specific sources of radiation. Shielding permits operating and maintenance personnel access to required portions of the plant in order to operate the reactor and associated systems, refuel the reactor, limit neutron activation of secondary systems, and perform the required normal inspection and maintenance activities.

Radiation leakage from source areas is minimized by the proper design of embedded ducts, piping, and conduit; shield penetrations; access doors, hatches, and plugs; and labyrinth passages. The shields are designed to limit the radiation exposure of operating personnel and the general public to less than that required by the applicable sections of 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 100. The ALARA program and shield design permit the size of the operating crew to be established by work requirements and not by radiation exposure limitations.

The designer has provided six radiation zones as a basis for classifying occupancy and access restrictions for various areas within the plant. Radiation doses for each cell, enclosure, and area are established by defining the expected dose present from the enclosed systems and surrounding areas, during both operation and maintenance periods. Radiation zones for personnel access are coordinated with the ALARA design. Zones are established during operation and for 10 days after reactor shutdown. Preliminary radiation zones have been assigned to the nuclear island and to the nuclear island maintenance building (NIMB). Areas outside the buildings are denoted Zone I, with the exception of the roof of the nuclear island. The zones within the buildings will be periodically updated as more becomes known on the various sources. On this basis, maximum design dose rates are established for each zone and used as input for shielding of the respective zones. For example, design radiation levels in operating areas where personnel are expected to be working a 40-hour week will be less than 0.2 millirem per hour. The areas that will have to be occupied on a predictable basis during normal operations and anticipated occurrences are zoned so that exposures are below the limits of 10 CFR Part 20 and will be ALARA. The zoning system and access control features will also meet the posting entry requirements of 10 CFR §20.203 or the Standard NRC Technical Specifications.

The next design stage should contain the results of a design review of station shielding to ensure the accessibility of vital areas after an accident (in accordance with the criteria of Item II.B.2, NUREG-0737, Ref. 12.4). These results should include post-accident source terms, a listing of plant systems containing highly radioactive materials following an accident; a set of post-accident radiation zone maps depicting the radiation levels in various areas of the plant one hour after the accident, a list of the vital areas that will require continuous or frequent occupancy following an accident, and a summary of the integrated doses to personnel in the above-listed areas for the duration of the accident. This requirement is applicable to the SAFR design.

In addition to the above information, PSID Section 12.1.2.3 (Ref. 12.1) should contain a tabulation of the assumptions used to calculate shield thickness, the calculational methods used, and the boundaries chosen. The staff believes it is advantageous to use a good calculational procedure, since an effective shield design is essential to meeting the criterion that occupational radiation exposure will be ALARA.

Radiation Protection

The information on shielding contained in Section 12.1.2.3 of the PSID is acceptable for this stage of the design.

12.5.2 Ventilation Systems

The ventilation system for the SAFR will be designed to ensure that plant personnel are not inadvertently exposed to airborne contaminants exceeding those given in 10 CFR Part 20. The applicant intends to maintain personnel exposures ALARA by

- maintaining air flow from areas of potentially low airborne contamination to areas of higher potential concentrations
- ensuring negative or positive pressures to prevent exfiltration or infiltration of potential contaminants
- locating ventilation system intake components so that intake of potentially contaminated air from other building exhaust points is minimized (These design criteria are in accordance with the guidelines of Regulatory Guides 1.52 (Ref. 12.5) and 8.8. During the PSAR review stage, DOE should illustrate the features of the air cleaning system design that protect personnel from radiation.)

12.5.3 Area Radiation Monitoring System

The ARMS is designed to

- monitor the radiation levels in areas where radiation levels could become significant, and where personnel may be present
- alarm when the radiation levels exceed pre-set levels to warn of increased radiation levels
- provide a continuous record of radiation levels at key locations throughout the plant
- provide criticality warning for new and spent fuel storage areas

The area monitors will respond to incident gamma-radiation. The area radiation monitoring system will meet the criteria of Item II.F.1(3) of NUREG-0737 (Ref. 12.4) and Regulatory Guide 1.97 (Ref. 12.6) and together with the plant data handling and transmission system will accomplish the following specific functions in relation to the ALARA program:

- Provide the control room with a continuous determination of the radiation status of the plant.
- Provide an alarm for conditions of potential radiological hazard.
- Enable operator action to reduce the spread of airborne radioactivity concentrations within the plant via control of the heating, ventilation, and air conditioning (HVAC) systems.

The surveillance data obtained from the ARMS will be available to control room operators through the video displays of the data handling and transmission system, which also produces a printout of alarms every day.

12.5.4 Airborne Radiation Monitoring System

Fixed and mobile continuous air monitors (CAMs) will alarm audibly and visually if airborne concentrations of radioactive materials exceed the monitor set points. These instruments will be used to continuously monitor the atmosphere at

- any specific location where maintenance is being performed

- any specific location where a process system failure is suspected of causing an airborne radioactive materials leakage
- inerted-cell purging activities

High and low-volume portable air samplers will be used for representative sampling of breathing air at infrequently occupied locations. Samples will be analyzed for gross radioactivity and radioisotopic content as appropriate. Frequency of use will depend on operational need.

Continuous (fixed) monitors will serve the reactor containment building and reactor service building operating areas. Continuous (mobile) monitors will serve in frequently occupied operating areas adjacent to potential radioactive sources, for example, Zones I and II.

The SAFR design objectives and location criteria for area and airborne radiation monitoring systems are in conformance with 10 CFR Parts 20, 10 CFR Part 50, and Regulatory Guide 8.8, and are acceptable.

12.5.5 Dose Assessment

The applicant has not provided a dose assessment as described in Regulatory Guide 8.19 (Ref. 12.7).

12.5.6 Operational Radiation Protection Program

The PSID for the SAFR does not contain a Section 12.5. As mentioned in SRP Section 12.5 (Ref. 12.8), this PSID Section 12.5 should describe the applicant's health physics program with respect to organization, equipment, instrumentation, facilities, and procedures. Since the PSID is preliminary to the PSAR and since the PSID describes a standardized plant (rather than a plant at a specified site), most of the level of detail included in Section 12.5 of the SRP is not warranted at this stage of the review. However, during the next review stage, Chapter 12 of the PSID should include

- a description of the administrative organization of the health physics program, including the authority, responsibility, and training of each position identified
- the criteria for selecting portable and laboratory technical equipment and instrumentation for performing radiation and contamination surveys, area and airborne radioactivity monitoring, and personnel monitoring during normal plant operations and AOs
- a description (including location) of the health physics facilities, access control stations, laboratory facilities, decontamination facilities, and other contamination control equipment and facilities

12.6 Conclusions

The radiation protection program described in the SAFR PSID is acceptable at this conceptual design phase.

12.7 References

- 12.1 DOE, "SAFR - Sodium Advanced Fast Reactor," Preliminary Safety Information Document, AI-DOE-13527, Initial Issue - October 1985, with amendments. (DOE Applied Technology.)
- 12.2 NRC, "Information Relevant To Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," Regulatory Guide 8.8, Rev. 3, June 1978.

Radiation Protection

- 12.3 NRC, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonable Achievable," Regulatory Guide 8.10, Rev. 1, May 1977.
- 12.4 NRC, "Clarification of TMI-2 Action Plan Requirements," NUREG-0737, November 1980.
- 12.5 NRC, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Features, Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Regulatory Guide 1.52, Rev. 2, March 1978.
- 12.6 NRC, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident," Regulatory Guide 1.97, Rev. 3, May 1983.
- 12.7 NRC, "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants- Design Stage Man-Rem Estimates," Regulatory Guide 8.19, Rev. 1, June 1979.
- 12.8 NRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," NUREG-0800, July 1982.

13 CONDUCT OF OPERATIONS

13.1 Emergency Planning

13.1.1 Description

The proposed approach to emergency planning for a SAFR plant identified in PSID (Ref. 13.1) Appendix F, "Emergency Preparedness Planning Basis," differs significantly from that of previous light-water reactor (LWR) applications. Currently, to implement this approach, applicants would have to request exemptions to existing requirements. The staff does not anticipate that there will be any significant differences in onsite emergency planning for SAFR from the guidelines presented in NUREG-0654 (Ref. 13.2). Onsite accident management planning will be retained, and emergency operating procedures (EOPs) will be extended from prevention of core damage to management of core accidents. The SAFR designer is proposing to simplify offsite planning by eliminating the need for the rapid evacuation of people and the resulting extensive cooperative effort between the applicant and State and local governments in developing plans and implementing them. The SAFR designer proposes to accomplish this by limiting doses from design-basis events (DBEs) to the lower protective action guideline (PAG) dose, and by designing the plant so that severe accidents have long delay times to release, with the result of an extremely low probability of early health effects. For these reasons, RI proposes that planned evacuations are unnecessary, and such offsite protective actions as delayed evacuation or sheltering or both during plume passage, followed by early relocation, are both adequate and effective means for protecting health and safety during severe accidents. In summary, RI proposes that the plume exposure pathway emergency planning zone (EPZ) be encompassed by the plant exclusion area boundary and, therefore, offsite emergency planning requirements can be greatly reduced.

Because of this simplified approach to emergency preparedness, RI anticipates that no extensive coordinated effort in preparing onsite and offsite plans will be necessary. As proposed, the emergency plan would be developed in accordance with applicable requirements of NUREG-0654 (Ref. 13.2). It would not address offsite planning other than to identify notification and communication methods established for alerting responsible individuals off the site that a severe accident had occurred. The following specific items are required by 10 CFR Part 50 (Appendix E); SAFR proposes to address them later, at the PSAR stage:

- determination of onsite and offsite emergency organizations and notification means
- arrangements with local, State, and Federal government agencies
- protective measures to be taken within the site boundary and EPZ to protect health and safety in the event of an accident
- facility features provided for administering first aid and decontamination at the site and for emergency transportation of injured or contaminated onsite individuals to offsite facilities
- provisions for emergency treatment at offsite facilities
- provisions for a training program
- preliminary analysis that projects the time and means to be employed in the notification of State and local governments and the public in the event of an emergency

13.1.2 Scope of Review

The staff has reviewed the SAFR emergency planning philosophy and supporting analysis presented in Appendix F of the PSID (Ref. 13.1).

Conduct of Operations

RI presented probabilistic risk assessment (PRA) analyses, similar to those discussed in NUREG-0396 (Ref. 13.3), to show the probability of exceeding the Environmental Protection Agency (EPA) lower level PAGs of 1 rem (whole body) and 5 rem (thyroid) at the site boundary.

The offsite emergency planning requirements proposed by RI are consistent with the key licensing issues for advanced reactors identified by the staff in SECY 88-203 (Ref. 13.4). The staff reviewed these in a manner that was consistent with that document.

13.1.3 Design Criteria

Currently, offsite protective actions are based upon the EPA PAGs for projected dose which are 1 to 5 rem (whole-body) and 5 to 25 rem (thyroid). At the lower projected dose, protective actions should be considered. At the higher projected dose, protective actions are warranted. (Any dose that has already been accumulated before the decision on whether to take protective action is not considered as part of this planning decision.)

In the past, the Commission has limited offsite emergency activities to situations in which the lower level PAGs were expected to be exceeded. For example, emergency planning for research reactors is restricted to the area around the reactor where the lower PAGs are expected to be exceeded. This is usually within the owner-controlled area. For fuel-cycle facilities, the rule (10 CFR 30.32(i)(1)(i)) exempts those facilities at which the lower PAGs will not be reached outside the owner-controlled areas. Therefore, a precedent exists for not requiring offsite emergency planning, beyond simple notifications, where warranted by operation. Response of certain offsite agencies into the owner-controlled area (e.g., police, fire, medical) is traditionally considered part of the onsite planning.

The staff believes that emergency planning requirements for advanced reactors can be based upon the characteristics of those designs. This principle is similar to that in the emergency planning rule (10 CFR 50.47) which states that the size of the EPZ for high-temperature, gas-cooled reactors can be determined on a case-by-case basis.

In consideration of the potential for improved safety and greater safety margins in advanced reactor designs, particularly the prevention of core melt, credit in the emergency planning area can be given for the plant's ability to prevent significant releases of radioactive material and to provide long periods of time preceding releases for all but the most unlikely events. The emergency planning criteria provide such credit, consistent with evaluating a range of events similar to those evaluated for LWRs.

If the proposed criteria are met, the staff believes that ad hoc evacuation, aided by prompt notification of offsite authorities, will be sufficient. This proposal is subject to change pending Commission action on the key policy issues associated with such advanced reactor designs as the SAFR design.

13.1.4 Research and Development Program

R&D on the passive safety associated with the SAFR design is required to demonstrate the accuracy of the PRA estimates. This is discussed in Chapters 14 and 15 of this SER. With regard to emergency planning, R&D should be conducted to evaluate the event sequence and consequence analysis needed for a detailed Level III PRA.

13.1.5 Safety Issues

No safety issues are associated with emergency planning.

13.1.6 Conclusions

The staff has reviewed the emergency preparedness planning basis and philosophy presented in the PSID. Ad hoc offsite emergency planning may be considered sufficient if a long enough time is available before radiation releases, even for the highly unlikely events. However, as discussed in Chapters 6 and 15 of this SER, certain events that the staff believes should be considered in the SAFR design can lead to core melt and to possible energetic reactivity insertion events, which casts doubt on the ability of the SAFR design to meet the staff's proposed criteria for reduced offsite emergency planning (Ref. 13.4). Therefore, the staff cannot accept the proposed reduction in emergency planning for the SAFR design at this time.

13.2 Description of Operational Modes

At this stage, detailed procedures have not yet been developed. A general description of operation and primary plant parameters for major operational modes are provided.

The plant will comprise up to four individual power paks that will function independently of each other. The power paks will be controlled from a common control room with an individual control console for each plant. The plant will operate in a base-load mode. The power operation range of each unit will be 40 percent to 100 percent of rated power. Since each power pak will be functionally independent, the operational modes will be applicable to each unit. All operational modes will be performed automatically by the plant control system or operator command signals. Each mode is discussed in some additional detail in the PSID.

13.2.1 Conclusions

The staff considers that the procedures and details of the operational modes are important items that should be addressed early in the next review stage.

13.3 Role of the Operators

The staff views the safety role of the operator as

- to provide a line of defense in accident situations by serving as a backup to safety systems to verify they perform their functions and, if systems do not perform well, to take appropriate action
- to maintain communications with appropriate onsite and offsite personnel
- to initiate recovery actions following an event

The operators represent an important source of knowledge concerning plant status and design, and their behavior could prove extremely valuable in understanding, responding to, and recovering from an accident situation. In the staff's view, the operators should be protected from natural phenomena, accident environments, and potential intrusion. Accordingly, the staff believes that the operations center should be located inside the protected area and should be designed to LWR-equivalent safety-related standards for natural phenomena and habitability. This will also help protect the key plant documentation usually kept in the control room.

13.3.1 Conclusions

Although the staff agrees that much of the instrumentation and control inside the control room and at the alternate shutdown console need not be safety grade, at a minimum a safety-grade scram button for each reactor module should be located at each control console. The need for other safety-grade instrumentation at these consoles can be determined at a later review stage.

Conduct of Operations

The PSID did not address the size of the operating crew necessary for the SAFR design. The staff considers this an important item that should be addressed early in the next review stage.

13.4 Safeguards and Security

13.4.1 Design Description and Safety Objectives

This section addresses the physical and administrative protection and security provided for the plant facilities and operations and the safeguarding of nuclear materials. The safety objective is to prevent theft of nuclear materials and any other activities that could affect safe operation of the plant.

13.4.2 Scope of Review

The staff reviewed the potential of the design to meet existing requirements for protection against radiological sabotage contained in 10 CFR Part 73, including §73.1, §73.2, §73.55, and Appendices B and C, and interpretation of requirements given in NUREG-0908 (Ref. 13.5), in Regulatory Guide 5.65 (Ref. 13.6), and in SRP Section 13.6 (Ref. 13.7) and SRP references. The review also paid special attention to how SAFR would foster the objectives of the Commission's severe accident policy, which states:

The Commission also recognizes the importance of such potential contributors to severe accident risk as human performance and sabotage. The issues of both insider and outsider sabotage threats will be carefully analyzed and, to the extent practicable, will be emphasized in the design and in the operating procedures developed for new plants.

Also, Generic Issue A-29, "Nuclear Power Plant Design for the Reduction of Vulnerability to Sabotage", is one of the medium-priority generic safety issues for which that policy expects new designs to demonstrate technical resolution.

This review covers only protecting the reactor facility from sabotage. It addresses neither protection against theft of nuclear material nor protection against radiological sabotage of the co-located fuel cycle facility also discussed in the submitted documents. Specifically, Sections 2.8 ("Plant Description - Fuel Cycle Facility"), 3.5.4 ("Fuel Cycle Facility Protected Area"), and 4.0 ("Plant Safeguards Nuclear Material Control Plan") of Reference 13.8 are not covered by this SER. Such review would be necessary before licensing since the physical protection and nuclear material control requirements for strategic special nuclear material apply to SAFR fuel.

13.4.3 Design Description Criteria and Evaluation

Two separate but adjacent security areas will be in the owner-controlled area.

The balance-of-plant (BOP) area will contain non-safety-related power conversion structures and equipment, including up to four turbine buildings, condensate storage tanks, auxiliary steam generator buildings, cooling towers and circulating water pump-houses, and the BOP portion of the warehouse complex. Balance-of-plant area security will provide ordinary industrial-level security, with unalarmed physical barriers to channel cooperative individuals to access points.

Reactor systems and equipment containing radionuclides will be located in the nuclear island. Each of up to four reactor modules will be housed in its own reinforced-concrete structure within the nuclear island.

Steam piping, feedwater piping, and electrical cabling will connect the nuclear island and the BOP area. The nuclear island guard-house and the BOP guard-house will form part of the boundary between the protected area and the BOP area. Security access control points and the central alarm station (CAS) will be located within the nuclear island guard-house. Housed in the protected area

in the same structure as the nuclear island guard-house will be the control building and nuclear island plant service building. The secondary alarm station (SAS) will be located in the control building.

The nuclear island security program will consist of a nuclear-level physical security organization, a protected area, one or more vital areas within the protected area, physical barriers, controlled access points, detection aids, communication capabilities, a testing and maintenance program, and an armed response force.

In this review, no credit is given to balance-of-plant area security or to plant equipment located outside the nuclear island, under the conservative assumption that those would be vulnerable to a threat with the capabilities defined in 10 CFR 73.1.

13.4.3.1 Physical Security Organization

The description of the security force organization in Reference 13.8 is conditionally acceptable considering the conceptual stage of the design. It provides for a dedicated and trained security force supervised by a security manager who will report to the plant superintendent. It commits to written procedures with approval by both the security manager and operations management. Future revisions will need to identify the onsite position ultimately responsible for security at all times. Additional commitments will need to ensure that

- At least one full-time member of the security organization is on site at all times who has the authority to direct the physical security activities of the security organization, and that this person coordinates with the individual who has final responsibility for plant operations on a shift.
- A clear chain of succession is provided in the event a key member of the physical security organization is disabled during an incident.
- Provisions are made for the onsite security organization to appeal conflicts between operations and security to a higher authority.

These need to be addressed before a site is licensed.

13.4.3.2 Physical Barriers for Protected Areas

In meeting the requirements of 10 CFR 73.55, Reference 13.8 describes a barrier to a protected area that appears to meet or exceed the definition of 10 CFR 73.2(f)(1). Two chain-link fences of adequate height and a vehicle barrier are described. The clear zone next to the outer fence and the isolation zone of 25 feet between the fences plus another 20 feet inside the inner fence exceed the acceptance criteria guidelines for isolation zones (Ref. 13.5). Exceptions to the isolation zone requirements at the nuclear island guard-house and at the warehouse facility are conditionally acceptable. Reference 13.8 notes that these are partially in the protected area and partially in the BOP area and provisions are expected to ensure that no penetration paths exist through interior areas or by way of roofs for personnel or prohibited material (Reference 13.8, Section 3.6.11, "Material Inspection and Control "; and Reference 13.9, Section 2, response 13.1(m)). One wall of each of the BOP auxiliary steam generator buildings forms part of the exterior fence of the nuclear island, and these buildings occupy portions of what would otherwise be a 20-foot clear zone outside the outer fence. Care will be needed to ensure that these do not offer a way to bridge the intrusion-detection systems in the isolation zones.

Illumination of the protected area is committed to be no less than 0.2 foot candle with an uninterruptible power supply (Reference 13.8, Section 3.6.7, "Power Sources," and Section 3.6.5, "Exterior Lighting"). Reference 13.8 states:

Conduct of Operations

Although no buildings included in the protected area barrier system are presently identifiable as accessible structures, the exterior illumination plan would accommodate requirements for illumination of the tops and sides of buildings if they should be identified as such.

Illumination of the tops and sides of the warehouse facility and nuclear island guard house will be required as necessary to support the alarm assessment system to be provided there.

13.4.3.3 Physical Barriers for Vital Areas

RI identified vital areas and vital equipment in Reference 13.8. Reference 13.8 presents analyses that indicate no other equipment would need to be protected to prevent sabotage from resulting in radioactivity releases in excess of the 10 CFR Part 100 guidelines. This is conditionally acceptable based on the review guidelines provided in Review Guideline 17, with exceptions noted in a closed meeting with RI on plant security. All of this equipment will be located within the nuclear island protected area.

Reference 13.8 notes that substantial barriers will be incorporated into the design of structures housing vital equipment to provide enough delay against determined adversaries so that the onsite response force can be alerted and can engage the adversaries outside the building rather than within. Building exterior walls and penetrations (e.g., portals, doors, ducts, vents) are committed to be hardened to yield a penetration delay comparable to penetration of at least 8 inches of reinforced concrete (Reference 13.8, Section 3.6.1.3, "Building Barriers"). Roof hatches and maintenance access ports of the reactor containment building are major pieces of equipment and special handling equipment is required to open them (Reference 13.9, Section 2.0, response 13.1(e)). During normal operation, access to the containment vessel and reactor vessel can be gained only through the reactor air cooling system (RACS) vents and inspection portals. A commitment has been made to include barriers with long delay times and intrusion-detection alarms on these vents and inspection portals (Reference 13.9, Section 3.0, "Response to Additional NRC Comments and Questions on the SAFR 'Safeguards and Security Plan and Assessment'").

The RACS air inlet and outlet openings represent vital equipment not protected by vital area barriers. Disablement of all decay heat removal systems for a long enough period (many hours) to cause significant fuel damage is considered by RI to be beyond the capabilities attributed to the design-basis threat from sabotage.

The CAS and the SAS will be in bullet-resistant structures. However, it is not clear that the CAS and onsite secondary power supplies for security equipment are designated as vital, as required by 10 CFR 73.55(e). These will need to be addressed in a later design stage.

13.4.3.4 Access Requirements

In accordance with 10 CFR 73.55(d), all points through which personnel and vehicles can gain access to the protected area will be controlled. A single security guard, who will be located in a bullet-resistant guard station, will be responsible for controlling the final point of both vehicle and personnel access into the protected area, as well as for monitoring and controlling equipment used to detect firearms, explosives, and incendiary devices. Although not addressed in Reference 13.8, additional guards or security personnel would be necessary outside the bullet-resistant station to conduct hands-on searches where indicated by the equipment searches. In order to inspect a vehicle (Reference 13.8, Section 3.6.2, "Personnel and Vehicle Access Control"), the guard must leave the guard-house through the access door; when the door is opened, adversaries could seize control of the guard station. This problem will be addressed in a later design stage.

Reference 13.8 provides for a search of 20 percent of screened personnel, and 100 percent of unscreened persons, as suggested by NUREG-0908 (Ref. 13.5). That provision is obsolete and the SAFR design requires walk-through equipment search of all persons, screened or otherwise, except

for on-duty law-enforcement officers (10 CFR 73.55(d)(1). (51 Federal Register, 27822). This is conditionally acceptable at this stage of the design, but needs to be addressed before a site can be licensed.

A photo-badge and key-card system, using encoded information will identify individuals who are authorized unescorted access to protected and vital areas and will be used to control access to these areas. Persons entering the protected area will be identified positively by such methods as biometrics-type personnel identification equipment.

13.4.3.5 Detection Aids

The plant designer recognizes that 10 CFR 73.55(e) requires perimeter intrusion-detection systems, but defers the selection of specific equipment on the grounds that selection of actual sensor types are dependent on such site-specific environmental conditions, as heavy precipitation, electrical storms, or high winds (Reference 13.8, Section 3.6.3, "Intrusion Detection System"). This is conditionally acceptable at this conceptual stage of the design, but will need to be addressed before licensing.

Reference 13.8 commits to locating an appropriate sensing system for coverage of buildings that form a segment of the protected area barrier system.

Reference 13.8 states that unattended vital areas will be locked and protected with intrusion-detection alarms. A commitment has also been made to include intrusion-detection alarms on RACS vents and inspection portals (Reference 13.9, Section 3.0, "Responses to Additional NRC Comments and Questions on the SAFR 'Safeguards and Security Plan and Assessment'").

The intrusion-detection systems transmission lines and associated alarms annunciation hardware will be line supervised and tamper-indicating. Alarm annunciators in two separate bullet-resistant alarm stations within the protected area will indicate the type of alarm and its location. Alarms will be assessed by closed-circuit television (CCTV). The alarm stations will be so located and designed that a single act cannot interdict the capability of calling for assistance or responding to alarms. A commitment has been made that activities that could interfere with the alarm functions will not be conducted at those alarm stations (Reference 13.8, Sections 3.6.8, "Central Alarm Station," and 3.6.9, "Secondary Alarm Station").

13.4.3.6 Communications

As required in 10 CFR 73.55(f), the designer has provided for continuous communications between the central and secondary alarm stations and with local law-enforcement authorities through the use of a conventional telephone system and a security radio system. A dedicated intercom system will also be provided between security stations. The security radio system will also permit two-way conversation with onsite security personnel.

13.4.3.7 Test and Maintenance Requirements

The physical security testing and maintenance was not described. This is conditionally acceptable at this conceptual stage of the design, but will need to be addressed before licensing.

13.4.3.8 Response Requirements and Vulnerability Analysis

In addressing the requirements of 10 CFR 73.55(h), the designer has analyzed the ability of the plant's armed response force to intercept an adversary before the adversary can penetrate the nuclear island building (Reference 13.8, Section 3.6.6, "Response Capabilities"). However, the site layout appears to put the site defenders who will be stationed at the nuclear island guard-house at a disadvantage, as they would have to go around (or through) the warehouse and nuclear island maintenance building complex and travel a total of about 400 feet to reach the closest of the four

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nuclear island buildings, and about another 400 feet to reach the farthest, while the adversaries would have only about 50 feet to cross to reach any of the nuclear island buildings after penetrating the alarm zone. Although the adversary would still have to penetrate the interior fence and building barriers in order to sabotage vital equipment, a site plan that placed the protected area perimeter further from the vital areas (or included other means of delay) and located members of the armed response force and their response weapons and equipment at or closer to the reactor buildings would help ensure timeliness and effectiveness of response.

The size of the armed response force deviates from the nominal force of 10 identified in 10 CFR 73.55. NUREG-0907 (Ref. 13.10) includes factors that cannot be evaluated until a site has been selected. It is premature at this conceptual design stage to confirm that the force size identified in Reference 13.8 would be acceptable.

13.4.3.9 Employee Screening Program

The nuclear island employee screening program to ensure trustworthiness of persons authorized unescorted access to the nuclear island and to vital equipment was not described. This is conditionally acceptable at this conceptual stage of the design, but will need to be addressed before licensing.

13.4.3.10 Severe-Accident Policy Considerations

Although the designer has not established SAFR design criteria for protection against radiological sabotage, the passive safety features of the SAFR design appear to offer advantages in protection against insiders and outsiders, as compared to a current-generation LWR. For protection against induced reactor transients, there are a number of redundant means of decay heat removal, one of which, RACS, is a passive system that would be difficult for saboteurs with capabilities assumed in 10 CFR 73.1 to totally disable. RACS, as well as the non-safety-related direct reactor auxiliary cooling system (DRACS), is designed to operate and remove decay heat using natural circulation in the event of a complete loss of both electrical power (station blackout) and instrumentation.

Design features in the gantry crane should prevent a single insider from using this crane as a tool of sabotage (Reference 13.8, Appendix C, "Gantry Crane Operation"). Additional protection against the insider threat is provided by invoking a two-man rule or use of CCTV monitoring during operations in vital areas (Reference 13.9, Section 2, response 13.1(n)).

13.4.4 Research and Development Program

No R&D program is needed.

13.4.5 Safety Issues

- A site plan that places the protected area perimeter further from the vital areas and locates members of the armed response force and their response weapons and equipment at or closer to the reactor buildings would help ensure timeliness of response.
- One wall of each of the BOP auxiliary steam generators buildings forms part of the nuclear island area exterior fence, and these buildings occupy portions of what would otherwise be a 20-foot clear zone. Care will be needed to ensure that these do not provide a means of bridging over the intrusion-detection systems in the isolation zones.
- Illumination of the tops and sides of the warehouse facility and nuclear island guard-house will be required as necessary to support the alarm assessment system to be installed there.

- Additional equipment may need to be designated as vital. Besides reactor equipment, CAS and onsite secondary power supplies for security equipment are required to be protected as vital by 10 CFR 73.55(e). These will need to be addressed as the design progresses.
- Care will be needed in selecting reliable door hardware for nuclear island buildings that will provide adequate delay while assuring provision for timely access and rapid exit for emergency situations.
- Since such considerations as the time it would take local law- enforcement authorities to arrive in force should enter into considerations of a reduction in the size of the onsite response force from the nominal force of 10 specified in 10 CFR 73.55, it is premature at this conceptual design stage to confirm that the smaller force size identified in Reference 13.8 would be acceptable.
- The security plan will have to commit to a search of all persons entering the protected area, not just a random sample. Additional security personnel may be needed for vehicle search and to support personnel search equipment.
- The description of the security organization will need to be expanded and descriptions of employee screening and security equipment test and maintenance programs will have to be provided at a future program stage.

13.4.6 Conclusions

The staff believes that the SAFR safeguards against radiological sabotage are in an acceptable stage of development for a conceptual review. The design appears to be less dependent than LWRs on proper functioning of security systems for protection against insider and outsider sabotage. There are, however, a number of open issues that will need attention at a later design stage.

13.5 References

- 13.1 DOE, "SAFR - Sodium Advanced Fast Reactor," Preliminary Safety Information Document, AI-DOE-13527, Initial Issue - October 1985, with amendments. (DOE Applied Technology.)
- 13.2 NRC, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants", NUREG-0654, November 1980 (FEMA-REP-1, Rev. 1).
- 13.3 NRC, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants", NUREG-0396, December 1978 (EPA 520/1-78-16).
- 13.4 NRC, "Key Licensing Issues Associated with DOE Sponsored Advanced Reactor Designs," SECY 88-203, July 15, 1988.
- 13.5 NRC, "Acceptance Criteria for the Evaluation of Nuclear Power Reactor Security Plans," NUREG-0908, August 1982.
- 13.6 NRC, "Vital Area Access Control, Protection of Physical Security Equipment, and Key and Lock Controls," Regulatory Guide 5.65, September 1986.
- 13.7 NRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," NUREG-0800, July 1982.

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- 13.8 DOE, "Safeguards and Security Plan and Assessment", 149SSP000001 Rev. C, September 25, 1987. (Safeguards/DOE Applied Technology.)
- 13.9 DOE, "Supplement No. 1 to the Safeguards and Security Plan and Assessment", 149SSP000002 Rev. A, October 22, 1987. (Safeguards/DOE Applied Technology.)
- 13.10 NRC, "Acceptance Criteria for Determining Armed Response Force Size at Nuclear Power Plants," NUREG-0907, February 1983.

14 DEMONSTRATION AND SAFETY TEST PROGRAMS

This section covers both the initial startup test program described in Chapter 14 of the PSID (Ref. 14.1) and the prototype safety test program planned as part of the design certification program for the initial SAFR plant. The prototype safety test program was presented in a separate document, the "Safety Test Plan" (Ref. 14.2). The SAFR prototype safety test program did not propose a dedicated prototype reactor be built at an isolated site to demonstrate the safety of the design before design certification. Rather, the SAFR program was proposed on the basis of building at least one module at a commercial site and using it to demonstrate the safety of the design before design certification. The staff review of this demonstration prototype program, along with other proposed research and development needs, appears in Section 14.2 below.

14.1 Initial Startup Test Program

Chapter 14 of the PSID provides, to the extent possible, information related to the period of initial operation and testing. The information pertains to the scope of the test program and summarizes the test program and its objectives. The initial test program will be consistent with the guidelines established in Regulatory Guide 1.68 (Ref. 14.3) as applicable to the SAFR.

14.1.1 Design Description and Safety Objectives

The SAFR operating licensee will be responsible for ensuring that a suitable initial (preoperational and startup) test program is conducted for the facility. The primary objectives of the program are to

- Provide additional assurance that the facility has been adequately designed and, to the extent practical, validate the analytical models and verify the accuracy or conservatism of assumptions used for predicting plant responses to anticipated transients and postulated accidents.
- Provide assurance that construction and installation of equipment in the facility have been accomplished in accordance with design.

Other key objectives are to familiarize the plant operating and technical staff with the operation of the facility and to verify by trial use, to the extent practical, that the facility operating and emergency procedures are adequate. Initial startup test programs satisfying these objectives should offer the necessary assurance that the facility can be operated in accordance with design requirements and in a manner that will ensure that the health and safety of the public is maintained.

The test program is required to include suitable testing of all structures, systems, and components (SSCs) important to safety. Both Appendices A and B to 10 CFR Part 50 recognize that some SSCs are more important to safety than others. For example, those SSCs that are designated as seismic Category I by Regulatory Guide 1.29 (Ref. 14.4) are considered more important to safety than some of the other SSCs that are identified as important to safety in the functional design criteria of Appendix A to 10 CFR Part 50. It is not intended that the same test requirements be established for all SSCs important to safety. A graded approach to testing has been established in order that adequate assurance, considering the importance to safety of the item, is provided and that the item will perform satisfactorily while, at the same time, the testing is accomplished in a cost-effective manner. Documentation associated with testing (such as procedures and records) are to be commensurate with the importance to safety of the item being tested.

14.1.2 Scope of Review

The staff has reviewed the startup test program presented in Section 14.1 of the PSID for consistency with existing regulatory requirements and to determine if the proposed program meets the intent of those requirements.

Demonstration and Safety Test Programs

14.1.3 Design Criteria

Section 50.34, "Contents of Applications: Technical Information," of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires, in part, that an applicant for a license to operate a production or utilization facility include the principal design criteria for the proposed facility in the safety analysis report (SAR). The introduction to Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 states that these principal design criteria are to establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety (i.e., SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public).

Section XI, "Test Control," of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 requires that a test program be established to ensure that SSCs will perform satisfactorily in service. Since all functions designated in the general design criteria (GDC) are important to safety, all SSCs required to perform these functions need to be tested to ensure that they will perform properly.

The SAFR initial startup test program will be designed to meet the intent of Regulatory Guide 1.68 (Ref. 14.3). This guide is written for light-water reactors (LWRs), but most of this guide also applies directly to SAFR. In cases of basic differences between LMRs and LWRs, the SAFR test program will substitute LMR-specific tests which meet the intent of the regulatory guide.

14.1.4 Research and Development Programs

R&D programs were not discussed in the PSID. Other than development of preoperational predictions and acceptance criteria in accordance with 10 CFR Part 52, the staff has identified no R&D requirements for the initial startup test program (as distinguished from the safety test program discussed in Section 14.2 below).

14.1.5 Safety Issues

The staff has identified no safety issues associated with the demonstration test program.

14.1.6 Conclusions

The demonstration test program insofar as it is described appears to meet the intent of the regulatory requirements for such programs.

14.2 Safety Test Program

The safety test plan material originally provided in Chapter 14 of the SAFR PSID was replaced by a separate volume, "SAFR Safety Test Plan" (RI Report 149TP000001, Rev. B, 10/21/87). This report was reviewed, and the following material applies to Revision B.

14.2.1 Design Description and Safety Objectives

The safety test plan document covers both R&D planned for the next 10 years and the safety tests planned for the prototype module. RI has explained that the R&D planned for the various facilities constitute the licensing basis for the first SAFR module and that safety tests at the first module, before commercial operation, will provide the final bases for a standard plant design certification. RI cites four major safety-related areas to be addressed in the programs:

- decay heat removal
- reactor shutdown mechanism

- operation with failed fuel element
- reactor passive response

Decay Heat Removal Systems

Decay heat removal systems utilized in the SAFR design include DRACS and RACS. DRACS is similar to systems planned or utilized in other LMR designs. It is not safety grade. RI plans to defer demonstration of DRACS until the first SAFR module is being tested, since RI considers this to be off-the-shelf technology. RACS is the safety-grade, decay heat removal system, and several preliminary tests are proposed. Air-flow pathway performance is to be tested at a large-scale facility at ANL along with a smaller scale facility to test RACS performance with aerosol fouling. Another ANL facility (water in a 1/11-scale Plexiglas model of the vessel and internals) will be used to verify in-vessel flow patterns. Sodium-side RACS performance is to be tested at the Energy Technology Engineering Center (ETEC) facility. Ultimately, the RACS performance of the first SAFR module is to be tested (including tests with partially blocked air pathways) on the first SAFR prototype facility. However, some limits may be placed on these tests to ensure that the module's useful lifetime of 60 years is not compromised.

Reactor Shutdown Mechanisms

The SAFR shutdown mechanisms include the automatic plant trip system (APTS), which is part of the reactor protection (scram) system and the self-actuated shutdown system (SASS). The SASS consists of three secondary rods held out of the core by a Curie point magnet which loses its magnetic properties when it overheats, causing it to drop the three secondary shutdown rods into the core. The jet pump design for the SASS (used to ensure core sodium flow past the magnets) has already been tested to some degree. The magnetic latch is to be tested at EBR-II to high integral radiation levels, at least as great as would be experienced in the SAFR applications. The SASS units in the first SAFR module would be tested at a reduced (25% normal) current to get a trip at 1020°F, 30°F below the normal trip temperature, primarily to keep the temperature below the code limits on the upper internal structure temperature.

Operation With Failed Fuel Element

Operation with failed fuel pins is primarily an integral fast reactor (IFR) metal-fuels program item. No specific tests are planned for the first SAFR module.

Reactor Passive Response

The reactor passive response is a major part of the SAFR safety test plan. RI compared the SAFR reactor to EBR-II, Rapsodie, and the FFTF, and argued that one can extrapolate quite a bit from these reactors. However, RI cited a need to perform engineering mockup critical tests to study the reactor physics of the metal fuels in order to minimize reactivity uncertainties and establish the control rod worths needed (keeping transient-overpower (TOP) initiators as small as possible). RI plans to study several critical states in the SAFR core life at the Zero Power Plutonium Reactor (ZPPR) facility in Idaho. RI also cited the need for further passive safety tests at the EBR-II facility, using the ternary metal fuel rather than uranium-fissium fuel. For this EBR-II work, RI prescribes increased utilization of the SASSYS computer code, since it is more suitable for analyzing the SAFR than the NTDEMO code, which is normally used for EBR-II.

The reactor passive response tests planned for the first SAFR module are patterned after the EBR-II test series, although RI cautions that some of the full-power, full-flow tests could threaten the plant investment (but not public safety). RI believes that a loss of flow from the 100-percent power conditions and a 35 percent TOP (also from full power) can be accommodated, but has concerns regarding the loss-of-heat-sink (LOHS) accident which may expose the structures to high temperatures. It is also possible RI may have to disable SASS to perform the LOHS accident test.

Demonstration and Safety Test Programs

RI indicates that APTS set points would also have to be raised to accommodate the test series. In the Safety Test Plan Report, RI displayed what it would consider "expected successful results for safety tests." This is the peak temperatures versus time where measurements fall within error bars around predictions and below both plant investment protection and safety limits.

14.2.2 Scope of Review

The staff discussed the SAFR test plan with RI and DOE, and discussed some components of the plan with experts from ANL and Westinghouse (SASS). The staff visited facilities at ANL for testing RACS and in-vessel flow distributions, and monitored the EBR-II program and facilities.

Decay Heat Removal

The staff visited the ANL facility for the RACS panels, and discussed the test plan with the RI and ANL staff.

Reactor Shutdown Mechanism

SASS is a Westinghouse development. Westinghouse described the ongoing test program in a recent paper and the staff discussed the paper with Westinghouse personnel.

Operation With Failed Fuel

The staff discussed experience with metal fuel at EBR-II with fuel experts at ANL. A visit to FFTF gave the staff an opportunity to view an HT-9 cladding assembly that had just been withdrawn from the core (there was no visible deformation).

Reactor Passive Response

In addition to discussions with the RI team, the staff met with experts at EBR-II and the FFTF regarding the passive, "inherent shutdown," and key reactivity feedbacks. Extensive discussions with personnel at EBR-II who managed and ran the EBR-II tests were particularly helpful.

14.2.3 Design Criteria

The requirements for testing new reactor systems are found in 10 CFR 52.47(b)(2).

14.2.4 Research and Development

The SAFR safety test plan describes the major portions of the SAFR R&D needs and the plans to fulfill them. A number of areas are identified and, for this stage of the design, it seems clear that a substantial effort has been made to identify the major items that must be addressed to achieve a successful program. The SAFR R&D needs are listed in Table 14.1.

14.2.5 Safety Issues

Two questions must be addressed:

- Do the described tests adequately confirm the necessary component or system performance?
- Are there other areas of concern?

Adequacy of Tests Described

RI has chosen to emphasize component tests rather than integral system testing of the first SAFR module. Although integral tests are generally preferred, component testing may be adequate to

demonstrate individual system performance. Such testing should not be relied on to confirm overall design requirements, but could validate certain key issues.

- Decay Heat Removal Systems: Testing of the first SAFR module's RACS is necessary in the long term, but the ANL component tests will resolve many of the issues. Similar tests may be needed for the DRACS given the lack of experience with systems similar to RACS.
- Reactor Shutdown Mechanisms: Tests in the first SAFR module are required to truly test SASS. However, component testing will be necessary to establish an acceptable confidence level before the staff would consider SASS as an acceptable alternate shutdown means.
- Operation With Failed Fuel: Questions here will be resolved as part of the IFR/EBR-I program.
- Passive Response: This is a most difficult system characteristic to confirm without a full SAFR core to test. Unless predicted safety margins are large, it will be difficult for RI to claim reliable performance of calculated feedbacks until the calculations are confirmed by prototype testing. Even then, reliability can only be assured by monitoring and testing over the plant life.

14.2.6 Conclusions

The following issues concerning RI's safety test plan need to be addressed at the design stage:

- additional component and integral system testing to provide a knowledge of margins sufficient to allow transient testing of the prototype.
- additional emphasis on integral testing of the initial SAFR module
- fuel behavior during fast overpower events
- development of a system for monitoring and testing the passive reactivity feedback response of the reactor (this may reveal as-yet-undefined R&D needs.)

On the basis of its review to date, the staff believes that a full-scale prototype of a SAFR module is needed to demonstrate the predicted performance of the passive reactivity control systems and the decay heat removal systems. Successful completion of the prototype tests and supporting R&D (Table 14.1) are necessary to support final design approval and design certification of the SAFR design.

14.3 References

- 14.1 DOE, "SAFR - Sodium Advanced Fast Reactor," Preliminary Safety Information Document, AI-DOE-13527, Initial Issue - October 1985, with amendments. (DOE Applied Technology.)
- 14.2 DOE, "SAFR - Sodium Advanced Fast Reactor," Safety Test Plan, 149TP000001, October 21, 1987. (DOE Applied Technology.)
- 14.3 NRC, "Initial Test Programs for Water-Cooled Nuclear Power Plants," Regulatory Guide 1.68, Rev. 2, August 1978.
- 14.4 NRC, "Seismic Design Classification," Regulatory Guide 1.29, Rev. 3, September 1978.

Demonstration and Safety Test Programs

Table 14.1 Major research and development required for the SAFR

Area	Required Development	National Laboratory Support	Status
Metal Fuel	- Fuel performance, HT-9 cladding tests - Reprocessing (ternary fuel)	- EBR-II^a, TREAT^b, FFTF^c - ANL^d-reprocessing	- Proceeding - Proceeding
Passive Safety	Confirm passive response and safety margins	- EBR-II tests (ATWS) - ANL-reactor flow tests - ANL-RACS heat-removal test - Prototype reactor test	1986 results - Proceeding - Proceeding - Proceeding
In-Vessel Refueling Machine	- Design definition (EBR-II, FFTF, PEC^e, and PFR^f technology) - Under-sodium performance	- ANL-design support - Prototype test-TBD^g	- Proceeding - Planned
Control Drive	Scram performance and reliability	Prototype test-TBD	Planned
Failed Fuel Detection	Sodium and cover gas monitoring with hermetically sealed system	- ANL-under-head delayed neutron monitor - HEDL^h-passive fission gas monitor	- Proceeding - Pending
Reactor Physics	- Validate analytical tools - Confirm rod worths, power distribution, etc.	ANL-ZPPRⁱ	Pending

^aEBR-II - Experimental Breeder Reactor-II

^bTREAT- Transient Reactor Test Facility (Idaho National Engineering Laboratory)

^cFFTF - Fast Flux Test Facility

^dANL - Argonne National Laboratory

^ePEC - Prova Elementi Combustibili

^fPFR - Prototype Fast Reactor

^gTBD - To be determined

^hHEDL - Hanford Engineering Development Laboratory

ⁱZPPR - Zero Power Plutonium Reactor

15 ACCIDENT ANALYSES

15.1 Design-Basis Accidents

Chapter 15 of the PSID (Ref. 15.1) describes how design-basis accidents (DBAs) are selected and how the SAFR responses to DBAs are evaluated. The concept of "defense in depth" is discussed, as are the acceptance criteria for the consequences of DBAs. Assumptions regarding the initial conditions, systems responses, and assumptions that lead to the DBA are also provided.

(1) Defense in Depth

The designer cited three levels of defense in depth for the SAFR plant. These levels correspond to (1) reliable operations (prevention), (2) safe responses to challenges (APTS and DRACS), and (3) safety systems that are largely passive (RACS and SASS). The reactor's passive safety features and the containment vessel are claimed to provide a line of defense similar to the LWR defense-in-depth concept. RI stated that there is at least one safety-grade system within each layer of defense. For reactor shutdown, the passive reactivity feedback features are identified as constituting the safety-grade system. For decay heat removal, the RACS constitutes the safety-grade system. For the containment barriers, the primary system boundary is identified as the fundamental safety-grade system.

(2) DBA Selection and Event Categorization

The SAFR DBAs are selected on the basis of their importance to the key safety functions of the plant; namely, reliable reactor shutdown, reliable decay heat removal, and containment of radiological release. Accidents that impair the ability of key safety systems that perform these functions are considered. The list of events is then compiled and supplemented with those considered for the FFTF, the CRBRP, and those events identified in the NRC Standard Review Plan (Ref. 15.2). Events considered for the SAFR concept are then grouped by category and frequency of occurrence. The accident categories include reactivity insertion events, undercooling events, perturbations in primary heat removal, local faults, miscellaneous events, and containment DBAs. The likelihood groupings include normal events (occurring more than once a year), anticipated events (at least 10^{-2} per year), unlikely events (10^{-2} to 10^{-4} per year), and extremely unlikely events (10^{-4} to 10^{-6} per year). Events occurring at a less than 10^{-6} per year range are considered beyond-design-basis accidents (BDBAs) by definition. Some of the BDBAs are discussed in Appendix A to the PSID and are reviewed in Appendix A to this SER.

For the purpose of this review, the staff is developing an approach for performing safety reviews of advanced reactors, as discussed in Section 3.3 of this SER. This approach, which will integrate considerations of DBAs and BDBAs, is still evolving. The currently proposed approach and its implementation are discussed in Section 15.2 below.

15.1.1 Design Description and Safety Objectives

Six categories of postulated enveloping DBAs were described for the SAFR concept, and the RI and ANL analyses of the various events are discussed. The categories include (1) reactivity insertion events, (2) undercooling events (reduction in reactor sodium flow rate), (3) change in heat removed from primary system (i.e., loss of heat sink), (4) local faults, (5) miscellaneous events, and (6) site suitability source term.

(1) Reactivity Insertion Events

Four events were classified as enveloping reactivity insertion events. For the control rod withdrawal event, all six primary rods were assumed to be withdrawn (adding 36¢ of reactivity). The system is scrammed by the APTS or the SASS or both. Analysis for the APTS-terminated

event indicates large safety margins. Analysis for the SASS-terminated event was for an 80¢ insertion which results in peak temperatures that are too high. The 80¢ initiator was retained from an early design and is now considered to be too high. In analyzing the safe-shutdown earthquake (SSE), RI assumed a reactivity insertion of 50¢. Analysis predicts an APTS scram at 1 second, and a SASS scram at 3 seconds.

RI classified a reactor trip with the loss of the primary heat sink, that results in the establishment of natural circulation cooling, as a "reactivity insertion" event. Uncontrolled rod withdrawal at startup is a variation on the first event in this category, except that the APTS detects the problem quickly and prevents temperatures from reaching full-power, steady-state levels.

(2) Undercooling Events

The events described by RI as "undercooling events" could also be described as reactor coolant flow reduction events. Loss of offsite power is accommodated, largely using passive safety. If APTS or SASS scrams the reactor, this is a design-basis event (DBE) that is accommodated as long as decay heat is removed via the IHTS, the DRACS, or RACS. If there is a failure to scram, the event becomes a BDBE, as discussed in Appendix A of this SER.

A leak in the primary piping, which runs from the pump outlets to the reactor inlet plenum, does not result in a loss of sodium inventory. Instead a broken pipe (one of eight) is like a short-circuit in the fluid flow network, and flow to the reactor is reduced by about half. The RI estimate of 42 percent flow to the core during a double-ended guillotine break is conservative and indicates that sodium boiling will not occur.

RI also analyzed the instantaneous seizure of one of the two SAFR pumps with the other pump remaining at full speed. The RI estimate put the core flow at 40 percent of normal. If combined with a failure to scram, this becomes a BDBE event for Appendix A analysis.

(3) Change in Heat Removal From Primary System

This category refers to loss-of-heat-sink (LOHS) events. Unscrammed LOHS events are BDBEs; they are discussed in Appendix A of this SER. Analyses for scrambled LOHS events are summarized in PSID Figure 15.2-18. Heat removal through the DRACS is calculated to be effective in keeping system temperatures below safety bounds. Should the RACS be the sole means of removing decay heat, the lengthy transient is estimated to result in temperatures high enough to require subsequent shutdown and inspection for damage. However, if some heat can be diverted to one or both intermediate heat transport system (IHTS) loops, this will augment the RACS heat removal function sufficiently to keep temperatures below ASME Code Limit C (i.e., 1250°F), according to RI calculations. Furthermore, the large inventories of sodium and structural materials will provide a large heat capacitance, so the SAFR primary system can survive several hours of adiabatic heatup before one of the decay heat removal systems becomes needed.

These various scenarios are analyzed at two levels, heat balance and detailed flow distribution. Using heat balances, comparing the decay heat levels to the design performance of heat removal systems, it is not difficult to establish the relations shown in PSID Figure 15.2-8, where the peaked average core outlet temperatures are plotted as a function of time. However, SAFR has a large asymmetric hot and cold pool which requires multidimensional modeling. RI showed some preliminary COMMIX and TEMPEST multidimensional flow analysis of these pools to examine thermal stratification and flow paths under natural circulation. Because this type of analysis is extremely costly, it is used sparingly, mostly to establish trends and to search for problem areas.

(4) Local Faults

The "local faults" category refers to problems localized within the reactor, and the potential for these problems to affect the rest of the core. Problems discussed include formation of small gas bubbles, fuel pin failures, and the prospects for larger gas bubbles and more extensive fuel failures. Small gas bubbles in an assembly could add reactivity, but this reactivity insertion would probably be small compared to the assumed transient overpower (TOP) initiators analyzed in Appendix A of this SER. Both gas bubbles and materials released from failed fuel could degrade cooling in a localized area, but the high thermal conductivity of both fuel and coolant would be expected to compensate for this partially. The experience with metal fuel in EBR-II indicates that fuel failure propagation is not likely to be a problem. This is consistent with physical arguments regarding interactions between the fuel, cladding, and coolant. Large gas bubbles introduced at the core inlet could be a problem, given the positive void worth, but RI asserted that it would be nearly impossible to pump such bubbles to the core inlet plenum.

(5) Miscellaneous Events

The "miscellaneous events" category includes blockage of the RACS air vents, fuel-handling accidents, sodium fires, leaks from the cover-gas, large sodium-water reactions (steam generator tube ruptures), intermediate heat exchanger (IHX) tube ruptures, reactor vessel leaks, and inadvertent release of oil from the pump. Blockage of the RACS vents is considered to be acceptable if it is partial (not more than 75%) or if the passage way is cleared within a day. (RI claimed that ASME Service Limit D would not be reached for 30 hours.) Fuel-handling events were considered and RI claimed they would pose no risk to the public health and safety. Sodium fires are considered possible only in the intermediate loop, which places them away from the reactor. The principal direct threat to SAFR from sodium is through fouling of the RACS airflow pathways, but RI claimed a continued acceptable performance even if all residues would be deposited there. The staff notes that the rupture of the IHTS piping in the steam generator compartment, and the resulting sodium fire, could have structural consequences for the reactor module concrete wall. Because the cover-gas cleanup system is continually cleaning the helium, a leak of the cover-gas is deemed relatively inconsequential. The designers claimed that a large sodium-water reaction is one in which a steam generator tube breaks and is followed closely by two subsequent tube breaks, 0.1 second apart. The SWAAM computer code analysis indicated that this event would not pose a major threat to the SAFR. IHX tube ruptures would result in intermediate loop sodium entering the primary system, because of the higher pressure in the IHTS, and because the IHTS is elevated with respect to the vessels. Reactor vessel leaks into the containment vessel would result in a small drop in hot pool sodium level, but both the IHX and DRACS should continue to function. Under these conditions, RACS performance should improve because radiation heat transfer is being replaced by conduction, which should improve overall heat rejection ability according to RI. Inadvertent release of oil from the primary pump has not yet been evaluated.

(6) Site Suitability Source Term (SSST)

Before examining a containment DBA, RI pointed out that the SAFR response to DBAs and to many BDBAs would not result in a significant release of fission products. Yet, 10 CFR Part 100 requires definition of a site-suitability source term (SSST):

The fission product release for these calculations should be based upon a major accident, hypothesized for purposes of site analyses or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

Accident Analyses

The RI solution to the dilemma of finding an appropriate source term is to assume the melting of one fuel assembly. This results in a very small source term due to (1) the small amount of fuel melted, (2) low metal-fuel melting temperature causes several isotopic contributors to be retained in fuel (not volatile at low temperatures), (3) holdup of fission products in sodium pools, (4) holdup of fission products in the cover-gas, and (5) delays in release of fission products from the primary and secondary containment. The radiological consequences are well within the regulatory guidelines prescribed in 10 CFR Part 100.

15.1.2 Scope of Review

By independent analysis, the staff evaluated the performance of various systems under accident conditions. These independent analyses focused on unscrammed events and are discussed in Appendix A of this SER. The staff also reviewed the analyses and assumptions given in Chapter 15 of the PSID for the DBA evaluations, but performed no independent analyses. The staff's conclusions are given in Section 15.1.6 below.

15.1.3 Design Criteria

Acceptance criteria for postulated events are given in Chapter 15 of the PSID. Governing damage criteria are given in PSID Table 15.1-3. Because of the time-dependent process of eutectic formation (fuel-cladding interaction), short-term (60 seconds) fuel-damage severity limits are necessary and are given in PSID Table 15.1-4. Allowable temperature limits for the reactor vessel and internal structures are given in PSID Table 15.1-5.

In analyzing the postulated SAFR DBAs, it is necessary to determine when the reactor will scram. SAFR trip signal set points and estimated response times are given in PSID Table 15.1-6. When a trip signal is generated by the automatic plant trip system (APTS), power is cut to the primary and secondary shutdown systems, dropping in all nine rods. When the APTS does not send a trip signal and the SASS magnets reach the Curie point temperature of 1050°F, the three secondary rods drop.

Should the APTS shut the reactor down in response to any of the postulated DBAs, then the damage criteria will be utilized according to the frequency of the events. In the unlikely event of an APTS failure, the event frequency is reduced significantly and the damage criteria applicable to the extremely unlikely events would then apply. Thus, if the SASS or the reactor's passive systems are needed to terminate the events, relatively high damage acceptance criteria are utilized.

15.1.4 Research and Development

Various R&D efforts are taking place that will help resolve uncertainties identified in Section 15.1 of this SER. Tests of the RACS panels at ANL will resolve the uncertainties associated with natural circulation processes. Water tests in a 1:11-scale Plexiglas test facility at ANL should clarify issues regarding flow stratification in pools. Continued efforts in the DOE metal-fuel program will help resolve remaining questions about fuel-failure propagation. Further evaluation of the SAFR IHX tube performance during sodium-water reactions may be needed. This area requires further study to ensure the reliable performance of this design (intermediate sodium on the shell side), given multiple steam generator tube failures and the high pressure on the steam side.

15.1.5 Safety Issues

(1) Reactivity Insertion Events

As long as the APTS shuts the reactor down automatically, all plausible reactivity insertion events can be accommodated safely in the SAFR plant. Failure of the APTS reduces the event

probability into the beyond-design-basis-event (BDBE) range. Actuation of the SASS is expected to be highly probable, which will take the reactor to a hot-standby condition unless the reactivity insertion is larger than $\$6$ (i.e., the total worth of the SASS rods). Response of the SAFR to reactivity insertions with failure of both the APTS and the SASS is covered in Appendix A of this SER. There are no apparent major safety issues in this category.

(2) Undercooling Events

Flow-reduction events ("undercooling events") can present a challenge if the reactor fails to scram. The three events in this DBE category are easily accommodated if there is a scram, but the analysis of pipe break and pump seizure is much more challenging for the equivalent BDBE in Appendix A of this SER. A fourth event should be included in this category: since pump seizures have historically (four times in U.S. LWR experience) occurred during coastdowns, RI should analyze loss of power to the primary pumps with one seizing early in the coastdown. If the APTS scrams the system, this event should be accommodated successfully for the SAFR plant, but that accommodation must be demonstrated.

(3) Change in Heat Removal From Primary System

Loss-of-heat-sink events can be accommodated in the SAFR design. A great deal of time is available, and the DRACS and RACS appear to be quite reliable. The safety issue here would be a leak in the reactor vessel and the resulting complications to DRACS or RACS performance, but this appears to be of acceptable consequence.

(4) Local Faults

Local faults are of interest in LMRs because fuel migration can cause large shifts in reactivity. In particular, the DOE metal-fuel program should be monitored to watch for fuel-failure propagation and other significant failure modes. However, the present experimental data do not point to this as being a problem area. This must be closely monitored because of its potential safety importance.

(5) Miscellaneous Events

Most of the postulated miscellaneous events do not appear to be major problems. However, the steam generator tube rupture event postulated by RI may not be conservative. The SAFR design uses a high-pressure (2800 psi) steam system and single-wall tubes. If the sodium side rupture disks work, then large sodium-water reactions are not major threats to the reactor. However, a large sodium-water reaction and a failure of the rupture disks would send a large pressure pulse toward the IHX, and could displace the shell-side tube support plates and damage the IHX tubes. This is a highly improbable scenario, but because of the high steam side pressure and potentially serious consequences (failure of the primary boundary), this event deserves considerable attention. Further, the number of failed tubes in a design-basis event requires further study; it may be that damage propagation from a single tube could endanger all seven tubes in a group.

(6) Site Suitability Source Term

Definition of the SSST for the SAFR presents unique considerations. It is very difficult to melt SAFR fuel because of the high coolant heat capacitance and excellent decay heat removal systems. In addition, analysis of the unscrammed LOHS indicates that the decay heat removal is fairly easy to accommodate inherently. A single fuel assembly meltdown event is considered by the DOE and RI to be the appropriate basis for the SSST for the SAFR. Extreme ATWS events were analyzed and found not to provide an appropriate basis for the SSST (no core damage or radiological release occurs). Therefore, to obtain a radiological source release, a postulated meltdown of a single subassembly has been proposed. The event is nonmechanistic in origin and

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is intended to bound all credible accidents. The radiological consequences are well within the regulatory guidelines of 10 CFR Part 100. The staff believes that more information is needed to support the postulated fission-product release and holdup factors.

There is a possibility that the SAFR can experience a power excursion, which could result in a much larger source term. If a power excursion were to occur, the fuel temperature could exceed the melting temperature and release some heavy fission products. The holdup in the sodium, the cover-gas, and the containment areas may decrease significantly. The impact of these assumptions could change the SSST by several orders of magnitude.

15.1.6 Conclusions

In the category of reactivity insertion events, independent calculations generally support RI analysis of a 20¢ TOP event (see Appendix A of this SER), suggesting that RI's analysis of this class of events is credible, although nonconservative.

The reduction in reactor coolant flow events are a particular concern, and independent calculations indicate that reactor flows are likely to be about 50 percent for pipe break and 35 percent for pump seizure, making the latter a greater concern if there is no scram.

For the loss-of-heat removal events, independent calculation has confirmed RACS performance analysis, and has shown this system to be highly tolerant to such faults as flow blockage and surface degradation.

Local faults were not considered in detail, but RI arguments seem to be correct since the experimental evidence (at EBR-II) indicates no failed fuel propagation. Independent calculations confirm that a reactor vessel leak can be acceptable even if all residues are deposited on the reactor vessel. Because the cover-gas cleanup system is continuously cleaning the helium, a leak of the cover-gas seems to be relatively inconsequential.

The designers claimed that a large sodium-water reaction is one in which a steam generator tube breaks and is followed closely by two subsequent tube breaks, 0.1 second apart. The SWAAM computer code analysis indicated that this event is not a major threat to the SAFR. IHX tube ruptures would result in intermediate loop sodium entering the primary system, because of the higher pressure in the IHTS, and because the IHTS is elevated with respect to the vessels. As a result of its proposed approach for safety reviews of advanced reactors, the staff has required a very conservative analysis involving rupture of many steam generator tubes. This is discussed in Section 15.2 below. Reactor vessel leaks into the containment vessel result in a small drop in hot pool sodium level, but both the IHX and DRACS should continue to function. Under these conditions, RACS performance would improve because radiation heat transfer is being replaced by conduction, which should improve overall heat rejection ability (according to RI). The staff and its consultants have reviewed these analyses and find no significant disagreement. The staff did not evaluate the inadvertent release of oil from the primary pump.

Because the SAFR is designed to accommodate DBEs, the events discussed in Section 15.1.2 (above) are less challenging than the BDBEs covered in Appendix A of this SER. As long as APTS scrams the reactor, short-term TOPs, LOFs, and LOHSs do not significantly challenge the SAFR. Local fuel failure and steam generator tube ruptures are not easily eliminated as serious threats, but the probability of either failure mode leading to a major accident appears to be acceptable.

The staff has reviewed the proposed source term resulting from a postulated meltdown of a single subassembly. RI selected such a source term arbitrarily to bound all credible accidents. On the basis of DOE and RI analyses, the results of the radiological consequences from such an event appear to be acceptable. However, the suitability of this source term remains an open issue.

15.2 NRC Staff's Approach to Accident Selection

The selection of a spectrum of accidents that includes both design-basis and beyond-design-basis accidents as a set of postulated events is considered necessary for advanced LMR designs. The SAFR plant should survive these with large margins. Consideration of such a spectrum of accidents are intended to

- Ensure advanced designs comply with the Commission's safety goals (Ref. 15.3) and severe accident policy statement (Ref. 15.4).
- Provide a sufficient test of the capability of the design to allow consideration of mechanistic source terms for siting determinations and for decisions regarding containment and emergency evacuation plans.
- Ensure the shift in emphasis from accident mitigation to accident prevention, as compared to LWRs, does in fact result in designs with safety at least equivalent to that of current-generation LWRs.

For the purpose of this review postulated transients and accidents were grouped into four event categories. The events to be included in each of these categories are selected deterministically and are supplemented by insights gained from a probabilistic risk assessment (PRA).

Event Categories

Event Category I (EC-I): This category of events is intended to be equivalent to the current anticipated operational occurrence (AOO) class of events considered in LWRs. The frequency range for these events decreases to approximately 10^{-2} per year, which corresponds to the frequency of events that may be expected to occur one or more times during the life of the plant. These events are to be analyzed similar to analysis for LWRs to demonstrate compliance with Appendix I to 10 CFR Part 50 and 40 CFR Part 190.

Event Category II (EC-II): This category of events is equivalent to the current DBA category for LWRs and events are selected consistent with the selection of an LWR DBA envelope. Specifically, events in EC-II are to

- Be selected using traditional engineering judgment, complemented by PRA methods which would include internal events down to a frequency of approximately 10^{-4} per year ($10^{-4}/\text{yr}$ is based upon ensuring that any event that is expected to occur over the lifetime of a population of reactors is included).
- Include a traditional selection of external events.
- Be subject to single-failure criteria and other traditional conservatism (no credit for non-safety-grade equipment, etc.). (Events within this category require conservative analysis as is presently done for LWRs.)

Event Category III (EC-III): This category of events corresponds to those events beyond the traditional DBA envelope which are considered in the design, consistent with the Commission's severe-accident policy statement. The events in this category should be selected using engineering judgment, complemented by PRA. This is consistent with the guidance provided in the Commission's safety goal statement and severe-accident policy which encourage the use of PRA methods to supplement engineering judgment and deterministic analyses.

Specifically, events in EC-III are to

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Specifically, events in EC-III were to

- Include internal events (less likely initiating events plus multiple failure events) down to a frequency of approximately 10^{-7} per year ($10^{-7}/\text{yr}$ is based upon ensuring that the cumulative effects of events below $10^{-6}/\text{yr}$ are considered in assessing the ability of the design to meet a less than a $10^{-6}/\text{yr}$ frequency of a large release of radioactive material to the environment). External events beyond those included in EC-II, and equivalent to those to be applied to future LWRs as part of the implementation of the Commission's severe-accident policy, would be included.
- Include, using engineering judgment, additional bounding events to account for plant-specific uncertainties. The staff has proposed a set of bounding events (Tables 15.1 and 15.2) which are to be analyzed along with other events selected for EC-III.

In selecting the events to be included in EC-III, the design was specifically reviewed to identify those events with the potential of a large release, core melt, or reactivity excursion to ensure adequate prevention or protection is provided for these events before they are excluded from this category. EC-III events would be analyzed on a best-estimate basis. RI and ANL have analyzed most, but not all, of these events. The results have been given a preliminary review and they appear promising.

Event Category IV (EC-IV): Because the SAFR was considered to be in the conceptual stages and many of the systems and components were still in the preliminary design phase, this category of events was to be used for assessing DOE's request to reduce the need for emergency planning requirements. It included events of the type considered in the basis for the emergency planning zones and requirements for LWRs described in NUREG-0396 (Ref. 15.6).

Event Frequencies

Each of the four event categories discussed above includes in its definition a frequency value. This frequency is intended to be used as a guideline only and is not to be considered a rigid limit.

Acceptance Limits for Event Categories

Tentative acceptance limits for the structures, radiation exposures, and core conditions are summarized in Tables 15.3 and 15.4 for each of the four event categories.

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In analyzing each event from the four event categories, a determination must be made as to whether it applies to all modules simultaneously or to one module only.

15.2.1 Research and Development Program

Existing R&D programs for various portions of the SAFR system are provided throughout the PSID and this SER. Accident selection and event categorization are difficult for a new design, and reliability data for use in a PRA is often nonexistent. R&D programs associated with the staff's proposed review approach for advanced reactors (which defines a new accident-selection process) will be identified as the approach evolves and more detailed reviews of the bounding events (and other events) are completed.

15.2.2 Safety Issues

Defense in Depth

The SAFR design includes defense in depth, but it is very different from LWR defense in depth, because the SAFR lacks a classic containment building. The most obvious vulnerability in the SAFR design is through the IHX tubes, which serve as both primary coolant boundary and containment.

Event Categorization Reliabilities

With a new design, reliabilities are not well understood. What may currently look like an insignificant event may have to be treated as significant in near-term licensing interactions.

Scram Systems

Neither the APTS nor the SASS is safety grade. Because the passive shutdown ("inherent shutdown") is ineffective against some very-low-probability events, RI should consider making either the APTS or the SASS safety grade.

Beyond DBAs

RI argued that only DBAs with failure-to-scram events need to be considered. There are, however, some pump-seizure events that could lead to sodium boiling which would lead to power excursions if unscrammed. Because of the potentially serious consequences, these events need to be considered.

15.2.3 Conclusions

Defense in Depth

The potential for improving the containment system without impairing the passive RACs capabilities, should be examined further. The IHX is a potential weak spot in this design.

Event Categorization and Reliabilities

Due to uncertainties, some events RI considers to have very low probabilities may have to be considered, at least for emergency planning, that is, Category IV.

Scram Systems

RI should consider making either the APTS or the SASS safety grade.

Beyond DBAs

Some events beyond the unscrammed LOF (with coastdown), LOHS, and TOP may have to be considered, depending on the probability and/or potential consequences of such events. For example, the plant response to an unscrammed pump seizure should be evaluated.

15.3 Bounding Events

15.3.1 Design Description and Safety Objectives

The SAFR design has been described as passively safe. On this basis, the designers contended that core melt and sodium boiling do not have to be considered in the design, and on the basis of mechanistic analysis, have concluded that a conventional containment structure and traditional offsite emergency planning are not necessary for the SAFR. As discussed in Section 15.2 (above), the staff

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has required analyses of a set of bounding events (Table 15.1) to ensure that a sufficiently challenging set of events is considered in assessing the acceptability of the design and in determining whether or not the SAFR design can be considered to have a level of safety at least equivalent to that of current-generation LWRs.

As described in Section 15.2 (above), the staff developed the bounding events (BEs) to take into consideration the uncertainties associated with the design, equipment reliability, and initial stages of supporting R&D. The BEs address the following categories of events: reactivity additions; reactivity-insertion failures; heat-removal failures; sodium-water reactions; sodium fires; and such external events as earthquake, flood, and fire. Table 15.2 provides an assessment of, and the basis for, the frequency of occurrence estimated by the staff of each of the BEs.

The RI response to the list of bounding events was provided in Appendix H to the SAFR PSID. The RI response includes both probabilistic and deterministic analyses. In certain cases, comments regarding the appropriateness of the bounding events were given. The staff also performed independent analyses of BEs 1, 3, and 4. RI and staff analyses are summarized below.

15.3.2 Scope of Review

Bounding Event 1 (BE-1): Extended Unprotected Transient-Overpower (UTOP) Event

RI chose not to analyze Bounding Events 1A and 1B as specified, but rather to provide an analysis of how the SAFR system would likely respond during a postulated UTOP event. Essentially, the RI scenarios are more realistic and may provide a "best estimate" of what would likely develop, but some of the conservative assumptions behind the definitions of the BE-1 scenario have been eliminated. (For example, there are instances of operator actions during the RI scenarios.)

The closest (to the original event definition) "case" considered by RI is the "margin analysis" described in PSID Section H-2.5 (p. H-28, Ref. 15.1). Even here, some of the assumptions carried in from the nominal case include variations from the original BE-1 scenarios. Pertinent assumptions are listed below:

- RI assumed that 36¢ of reactivity is inserted (based on maximum control rod insertion) at a rate of 0.1¢ per second, to initiate the event. No justification is provided for the rate of reactivity insertion which appears to conflict with the assumption of 2¢ per second used in Appendix A of this SER for the reference UTOP event.
- RI assumed that the primary pumps continue to run until the operator trims them back to 30-percent flow after 2 hours, because of control system logic (computer software, presumably) that prevents pump trips before scram.
- RI assumed that the normal cooling mode via the balance of plant (BOP) is gradually lost due to the increased power production, and does not cease until 1 hour into the event when the steam generators dry out.
- Once the normal cooling stops, RI assumed that cooling occurs via the RACS, as the DRACS is assumed to fail. Because the primary pumps add more heat than the RACS can remove, RI assumed that the operators will trim back the pumps 2 hours into the event (to 30 percent flow).
- RI made several implicit assumptions regarding fission-product retention within the fuel, within the sodium, and within the vessel and the containment. Consistent with these assumptions, RI projected some fuel damage in the 12 hottest fuel assemblies, and some limited fission-product releases. Doses predicted at the site boundary during the 36-hour event total about 2.0 rem (whole body), based on Regulatory Guide 1.4 (Ref. 15.7) methodology, and the 30-day dose would be about 6 rem.

Independent BNL analyses (Ref. 15.8) have supported the RI and ANL predictions of the SAFR responses to UTOP and scrambled LOHS events. Although BNL did not simulate this particular event, the staff believes that the analysis provided by RI is credible. However, the scenario and assumptions used by RI are unacceptable. It is clear that a considerable portion of the SAFR system is important during this transient (e.g., the BOP), that operator actions may be required, and that some fission product could be released during the first couple of hours. This is not consistent with passive safety.

Bounding Event 2 (BE-2): Station Blackout for "X" Hours.

This event was specified to last 36 hours, but RI assumed that 16 hours is more reasonable. Regardless, this is not a very challenging event for the SAFR plant, and RI projected peak core outlet and hot-pool temperatures below ASME Service Level B (1050°F). In this event, it is assumed that both the RACS and the DRACS are available to remove heat, as neither requires power to function. As a result, this event is less challenging than the one that acts as the design basis for the RACS performance. This scrambled LOHS event is covered in Chapter 15 of the PSID, and calculations have supported the RI and ANL analyses for this event. It appears that this event will not present a challenge to the SAFR plant.

Bounding Event 3 (BE-3): The RACS Blocked for 36 Hours, 25 Percent Unblocked Thereafter

In PSID (Ref. 15.1) Appendix H ("Bounding Events"), RI argued against the likelihood of the fully blocked case for 36 hours. Instead, RI assumed 25-percent partial unblockage after 10 hours, arguing that it is difficult to identify any plausible blockage that could not be removed within 10 hours. In the margin case (PSID Section H-4.4.2) it was assumed that 85-percent blockage occurred in the RACS air-flow pathways, and this was assumed to continue indefinitely (for at least 100 hours). For these cases, RI calculated some cladding failures (due to strain).

Eventually, in PSID Section H-4.4.3 ("Supplementary Margin Analyses"), RI analyzed the specified BE-3. In this case, the temperatures climb to nearly 1700°F (1200 K) by 36 hours, and large-scale fuel failures develop by 24 hours (i.e., fuel and cladding eutectic damage fraction in the peak assemblies reaches 1.0).

Given that much of the core is severely damaged or molten by the end of 36 hours, the dose estimates provided by RI and ANL appear to be low, 64 rem over 30 days at the site boundary using Regulatory Guide 1.4 (Ref. 15.7) methodology. The low dose estimates appear to result from assumptions regarding high retention of key fission products within the sodium pool.

An independent calculation using simple heat capacity analysis confirms the SAFR projection of temperatures as shown in Figure 15.1. It also is apparent from eutectic failure versus time charts that large-scale failures would develop around 24 hours into the event. Thus, the RI projection of system temperatures and reactor damage are assumed to be correct. However, assumptions regarding fission-product retention in sodium pools are based in part on data from oxide fuel, and metal fuel would behave somewhat differently. Thus, RI's projection of low doses at the site boundary may be difficult to confirm in the near term. Further, large-scale failures in an LMR core would result in the movement of fuel, which could lead to subsequent recriticalities. The outcome of this event, were it to occur, could be far more severe than is indicated by RI's dose estimates.

Bounding Event 4 (BE-4): Unscrammed Pump Seizure Plus Coastdown

The event was posed as an unscrammed pump seizure with the second pump tripping and coasting down. Although the likelihood of a pump seizing is small, pump seizures have occurred. The likelihood and timing of the second pump tripping are crucial. If the second pump were to trip at the same time that the first one seized, the reactor flow coastdown could be too fast, boiling sodium and triggering a power excursion. There are three combinations to consider:

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- coincident tripping of the second pump
- resultant tripping of the second pump
- simultaneous tripping of both pumps, with one seizing early in the coastdown phase

RI chose to assume the first combination for BE-4, and concluded that coincident tripping of the second pump was too unlikely to consider. The second combination (and possible delay times before the second pump trip) was discussed with RI engineers during the SAFR design review. This case is covered in Section 5.5 of PSID Appendix H. The third combination may be the most probable, but it was not considered when BE-4 was proposed.

In the independent staff analysis, the MINET code (Ref. 15.9) was used to model the flow network around the core, as indicated in Figure 15.2. The most important aspect of this analysis is the modeling associated with the behavior of the (1) operating and (2) seized pumps. This ultimately determines how much flow will continue through the core. The seizure causes a drop in system impedance, while the unfailed pump experiences a large flow increase to 128 percent of its rated condition, causing the pump to cavitate. (The designers are fully aware of this problem and believe that they can design a pump to operate in this regime.) The resistance through the locked rotor directly determines the amount of flow that will bypass the core and feed back into the cold pool. The MINET model accounted for all of the above effects and found that 65 percent of the system flow will bypass the core through the locked rotor. As shown in Figure 15.3, the remaining 35 percent of the system sodium flow will still continue to flow through the core.

The prescribed flow condition from MINET was simulated in the SUPERSYSTEM code (see Figure 15.4), and the results from that simulation are presented in Figures 15.5 and 15.6. The power in Figure 15.5 drops immediately to 50 percent and reestablishes a critical condition at about 52 percent of rated power. In Figure 15.6, the reactivity feedbacks that make up the passive response to a sudden loss of flow in a metal-fueled core are shown. The initial reduction comes from the Doppler and axial expansion feedback. This generates a negative feedback, as both give a negative feedback for a temperature increase. Within the first few seconds, Doppler and axial expansion insert -6% and -9% , respectively. The drop in sodium flow, coupled with the increase in fuel temperature, leads to increased sodium temperature, which generates a positive feedback. This added about $+17.5\%$ of reactivity at its peak, and then slowly returned to zero at around 200 seconds when the reactor reestablished criticality. Hotter sodium temperatures increase temperatures in the surrounding structures and the load pads, which activates the radial expansion. From Figure 15.6, it is clear that the radial expansion feedback is the dominant mitigating factor, and that this is the most significant feedback in terminating the event.

It should be noted that the Doppler reactivity ultimately inserts only about $+7\%$ of reactivity after the power is reestablished at 52 seconds, as shown in Figure 15.6. A negligible Doppler feedback, and the correspondingly reduced temperature defect, make it possible for the small reactivities generated in the metal-fuel core to force the power to a much lower level.

Although this event is challenging, the safety margins are still large. The SUPERSYSTEM code calculated the peak fuel center line temperature to be 1581°F (1134 K) which leaves a 419°F (232 K) margin to the solidus temperature. The peak assembly outlet temperature was calculated to be 1431°F (1050 K), which is well below the saturation temperature of 1740°F (1222 K). Furthermore, the feedbacks reduced the power level to a point at which the maximum fuel centerline temperature and the maximum sodium temperature in the core are low enough at the new quasi-static condition that the system could remain in this condition indefinitely without serious consequences.

The ANL prediction (discussed but not included in PSID Appendix H) for this event was only calculated out to 18 seconds. One difference between the two analyses was that ANL estimated that the core would receive 40 percent of the system flow even after the pump seized, while the MINET results indicate the value to be 35 percent. Both calculations, however, predicted that this event would be mitigated by the feedbacks in the core with no fuel damage or requirements for immediate

operator action, as long as the second pump continued to operate. The minimum margin to sodium boiling for seven postulated unscrammed SAFR events, as estimated by RI and ANL, and by BNL, are compared in Figure 15.7. Because the loss-of-flow events develop quickly (the power must drop quickly to avoid overheating) and because the loss of pumping reduces system pressures, the margins are smallest for those cases. However, as long as the second pump continues to run, the margin to sodium boiling is no less than for the more likely unscrammed loss-of-flow (LOF) case.

If one adds the trip and coastdown of the second pump, the safety margins would be reduced. RI describes their simulation in Section 5.5 of PSID Appendix H, except with a longer time scale. It appears that the minimum margin to sodium boiling is about 50°F, judging by Figure 5-6 on page H-78 of the PSID. Given the uncertainties associated with several of the reactivity feedbacks, a margin of 50°F may not be sufficient. (Before the staff analyzed this case with MINET and SUPERSYSTEM computations, DOE committed to the PRISM design for subsequent advanced LMR development. Thus, the SAFR analyses were cancelled in favor of making more PRISM simulations. However, it is very likely that simulation of BE-4 would have indicated significant sodium boiling.) The pump-seizure class of events would have required closer scrutiny in future SAFR design reviews. BE-4 could result in a very severe accident, and some design modifications may have been required to preclude or mitigate this event, even though it is a very-low-probability event.

Bounding Event 5 (BE-5): Steam Generator Tube Rupture

Any liquid-metal reactor using a steam cycle for power production will have a sodium-to-water heat exchanger and that will present the potential for sodium-water chemical reactions.

For the SAFR design, RI is relying largely on the sodium-water reaction protection-relief system (SWRPRS) to prevent steam generator tube rupture events from damaging the IHXs and thereby degrading reactor conditions. Key parts of the SWRPRS are largely passive (e.g., pressure-relief tanks and rupture disks), and some degree of event mitigation is probable. However, the choice of single-walled steam generator tubes, a 2800-psi steam generator system, an IHX with faulted stress levels in the range of 540 to 850 psig, and the design decision to pass the secondary system sodium on the shell side of the IHX (i.e., where tube support plates are exposed) offers little "inherent" capability for the SAFR plant to withstand large-scale steam generator tube rupture events. One could postulate high-pressure sodium-water reactions triggering large pressure pulses that could crush the IHX tubes, perhaps by dislodging the tube support plates.

In support of its system, RI provided results of SWAAM-II computer simulations of postulated 40-tube and 100-tube rupture events (one rupture followed by two subsequent ruptures per 0.1 second thereafter). The analyses indicate that the predicted IHX pressures remain well below IHX faulted stress level limits. There are no known problems with the calculations.

This event was not reviewed in detail as it is a generic LMR event. The staff believes that with appropriate design features, verified by a suitable testing program, this event could be accommodated.

Bounding Event 6 (BE-6): Large Sodium Leaks

The SAFR design is for a pool-type system with primary system piping (extending only from the pump outlets to the reactor inlet plenum) completely within the reactor vessel. A primary loop pipe break is not a leak in the conventional sense, but, rather, it is a flow short-circuit.

Leakage and breaking of an intermediate loop pipe could lead to both a loss of IHTS inventory and sodium fires. Assuming that the RACS performs the safety-grade, heat-removal function and does not require the intermediate loop to function, the loss of IHTS inventory is not crucial to decay heat removal. Sodium fires can be a problem, but the SAFR should employ state-of-the-art, sodium-fire mitigation systems. The most interesting question here is whether a major sodium fire could effectively foul the RACS heat-transfer surfaces and degrade performance. RI and ANL stated that

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they examined this question and determined that even if all of the combustion products plated-out on the RACS walls, the RACS heat removal system would still be adequate. This was not independently verified, although the staff notes that the designers have been conservative in estimating the thermal emissivity. Some degree of surface fouling could be compensated for by using this design conservatism.

A leak of sodium from the reactor vessel into the gap between the two vessels results in a drop in sodium level and better heat transfer between the vessels. The SAFR has been designed so that a leak of sodium into the gap leaves the reactor vessel sodium level above the IHX inlet ports, allowing sodium to continue circulating in the primary system. Since the improved heat transfer between the vessels would help RACS performance, it is likely that RI's claim is probably correct that peak system temperatures under RACS cooling only are lower with a reactor vessel leak.

A related question is how RACS performance depends on the spillover of sodium. Under steady-state conditions, the sodium level in the annulus just within the reactor vessel is lower than the level in the hot pool, due mostly to the pump head. Once the pumps trip off, the sodium rises along the reactor vessel wall. During a heatup (LOHS) event, sodium expansion causes a further raising of the sodium level. As a result, the sodium level rises along the vessel wall and improves RACS performance. Independent calculations indicate that when the RACS spillover develops, there is actually little direct effect on the sodium level, that is, it reaches its operating height regardless of spillover action.

Once the hot-pool sodium spills into the annular region, the reactor vessel is heated more effectively by the flowing hot sodium, so some improvement in RACS performance is assured. On the basis of COMMIX code calculations reported in an ANL paper, the staff estimated RACS heat removal and vessel temperatures during a modified (no spillover) LOHS scenario. The results are shown in Figures 15.8 and 15.9. It is noted that the RACS performance would increase by about 25°F (14 K). Although not insignificant, these changes in RACS performance are not expected to become a major safety concern.

Bounding Event 7 (BE-7): Blocked Fuel Assembly

The staff did not ask RI to specifically reply to this event, since it was added after the SAFR review had been completed. The SAFR has no in-core instruments for measuring individual assembly flow or outlet temperature, and a blocked fuel assembly (such as from a fabrication error) could be detected only if the fuel failed (i.e., fission gas activity increase, delayed neutron signal, or change in reactivity). Sodium boiling or fuel motion in the blocked assembly, or both, could already have occurred as a result of the blockage. Since this event has the potential to lead to a large reactivity insertion accident, it may make it difficult to accept a design without a conventional containment structure.

Bounding Event 8 (BE-8): External Events

For a reactor design utilizing a maximum degree of passive safety, such as the SAFR design, the likelihood of internally initiated events leading to core melts or large releases will probably not dominate the risk. It would be expected to find that external events ultimately dominate the risk associated with passive reactor system designs.

However, treatment of external events (e.g., earthquakes, tornados, floods) is currently being reassessed for LWRs. RI provided a largely qualitative response for this bounding event, and simply stated its intention to address any such issues at the proper time. There are no apparent vulnerabilities in the SAFR design to external events, although it seems likely that seismic events would ultimately dominate the SAFR risk because seismic events carry comparatively higher risk than most other event classes.

15.3.3 Research and Development

Exclusive of the PRA and analyses of the bounding events, there is little discussion of severe accidents and source terms in the SAFR PSID. However, for BEs-1B, -3, and -7, some fuel melting is shown to be possible, and BEs-4 and -7 could lead to energetic events. If these events cannot be eliminated from EC-III by design, many questions regarding fission-product retention in metal fuels and sodium, as well as the energetics of the metal fuel during fast transients, need to be answered through the R&D program.

15.3.4 Safety Issues

RI has taken a position on the SAFR that (1) a conventional containment structure is unnecessary and (2) ad hoc offsite emergency planning is acceptable. These positions are defended largely through probabilistic arguments. The reliability of new and unproven passive systems is very difficult to estimate, particularly when assumed failure rates fall in the 10^{-6} range. As a result, the staff has proposed a set of bounding events to bound the uncertainties and to help it assess the design. Four of these events have the potential to lead to the release of large amounts of fission products. For BE-1B, a combined UTOP and the failure of non-safety-grade cooling systems, an early release is possible. For BE-3, failure of the safety-grade heat-removal system (RACS), a large release would occur in the 24-to-36-hour time frame. BEs-4 and -7 could lead to sodium boiling and possibly energetic events.

15.3.5 Conclusions

Since some of the bounding events identified by the staff for inclusion in EC-III have the potential to lead to core melt or energetic reactivity accidents, or both, the acceptability of the SAFR design, particularly the proposals to have no containment and reduced emergency plans, remains an open item.

Accident Analyses

15.4 References

- 15.1 DOE, "SAFR - Sodium Advanced Fast Reactor," Preliminary Safety Information Document, AI-DOE-13527, Initial Issue - October 1985, with amendments. (DOE Applied Technology.)
- 15.2 NRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," NUREG-0800, July 1982.
- 15.3 NRC, "Safety Goals for the Operation of Nuclear Power Plants," 51 Federal Register 28044, August 4, 1986.
- 15.4 NRC, "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," 51 Federal Register 32138, August 8, 1985.
- 15.5 NRC, "Key Licensing Issues Associated With DOE Sponsored Advanced Reactor Designs," SECY 88-203, July 15, 1988.
- 15.6 NRC, "Planning Basis for the Development of State and Local Government Radiological Emergency Plans in Support of Light-Water Nuclear Power Plants," NUREG-0396, December 1978 (U.S. Environmental Protection Agency, EPA 520/1-78-16).
- 15.7 NRC, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," Regulatory Guide 1.4, Rev. 2, June 1974.
- 15.8 NRC, Van Tuyle, G.C. Slovic et al., "Summary of Advanced LMR Evaluations - PRISM and SAFR," NUREG/CR-5364, Brookhaven National Laboratory (BNL-NUREG-52197), October 1989.
- 15.9 NRC, G.J. Van Tuyle, T.C. Nepsee, and J.G. Guppy, "MINET Code Documentation," NUREG/CR-3668, Brookhaven National Laboratory (BNL-NUREG-51742), December 1989.

Table 15.1 Postulated bounding events for the SAFR

Event	Description
BE-1	Inadvertent withdrawal of all control rods, without scram for 36 hours (one module): A. with forced cooling B. with DRACS and RACS cooling only (primary pumps running)
BE-2	Station blackout (all modules) 36 hours.
BE-3	Loss of forced cooling (one module) plus loss of DRACS and RACS, with 25 percent unblocked after 36 hours.
BE-4	Instantaneous loss of flow from one primary pump with coast down flow for other pumps(s) (single module).
BE-5	Rupture-justifiable number of steam generator tubes with failure to isolate or dump water from steam generator: A. defined sequence of ruptures B. without forced cooling
BE-6	Large sodium leaks (single module): A. MEFS size break of IHTS pipe B. reactor vessel leak (critical leak)
BE-7	Blockage of flow to and/or from one fuel assembly
BE-8	Several external events consistent with those imposed on LWRs.
Note:	These events are intended to bound the LMR DBA and BDBA spectrum in order to account for PRA uncertainties and provide conservatism in selecting an SSST and assessing the adequacy of containment and offsite evacuation plans. These events are judged to be bounding for the following categories: ● Reactivity insertion ● Heat removal ● Loss of coolant ● Sodium-water reaction

Accident Analyses

Table 15.2 Rationale for postulated bounding events

Event	Rationale	Probability Range Estimate ^a
BE-1	Assumed worst-case failure of non-safety-grade control system (due to fire or other mechanism) which results in inadvertent withdrawal of all control rods combined with failure to scram	• Fire or control system failure - 10^{-1} to 10^{-4}/yr • Failure to scram - 10^{-5} to 10^{-7}/yr • Modules on site - 4 • Range = 10^{-5} to 10^{-10}/yr
BE-2	Two- to sixteen-hour station blackout is assumed for LWRs. Additional time added to compensate for lack of conventional containment.	• 2- to 16-hour blackout - 10^{-5}/yr for LWRs • Additional 20-hr loss - 10^{-2} to 10^{-3}/yr • Range = 10^{-7} to 10^{-8}/yr
BE-3	Severe external event could cause loss of offsite power and temporary loss of DRACS. RACS is not safety grade.	• External events cause loss of off-site power and blocks RACS - $<10^{-7}$/yr • Modules on site - 4 • Range = $<10^{-6}$/yr
BE-4	Loss of one primary pump is anticipated event (10^{-2} /yr) combined with ATWS.	• Instantaneous loss of one primary pump - 10^{-2}/yr • Failure to scram - 10^{-5} to 10^{-7}/yr • Modules on site - 4 • Range = 10^{-6} to 10^{-8}/yr
BE-5	SG and its water dump and isolation system are not safety grade. Experience with SG tubes indicates multiple failures have occurred. Exact number to be determined later but should be at least 40 based on PFR experiences.	• Multiple SG tube ruptures have occurred in the past. Such ruptures would leave plant on RACs cooling only.
BE-6	Consistent with CRBR	• IHTS or RV leak - 10^{-6} to 10^{-7}/yr (CRBR PRA) • Modules on site - 4 • Range = 10^{-5} to 10^{-6}/yr
BE-7	Fabrication error results in blocked assembly being inserted into the core.	• Fabrication errors have occurred. Experience shows fabrication and loading errors occur.
BE-8	Severe external events analyses	• To be determined. Under development for LWRs. Needed for equivalency.

^aProbability range estimates are for illustrative purposes only and intended only to show the potential for the bounding events to be in the EC-III range.

Table 15.3 Summary of tentative acceptance limits

Design-basis event category	Frequency ^a range (per reactor year)	Acceptance limits			
		Core conditions	Structural	Exposure to plant personnel	Exposure to offsite public
EC-I Normal Operation	$F \geq 10^{-2}$	Table 15.4	ASME Code Service Level A limits	10 CFR 20 limits	10 CFR 50 Appendix I limits
EC-II Anticipated Events	$10^{-2} > F \geq 10^{-4}$	Table 15.4	ASME Code Service Level B limits	10 CFR 50 limits	10 CFR 20 Appendix I limits
EC-III Unlikely Events	$10^{-4} > F \geq 10^{-7}$	Table 15.4	ASME Code Service Level C limits	10 CFR 20 limits	10% of 10 CFR 100 limits ^c
EC-IV Extremely Unlikely Events	less than 10^{-7}	Table 15.4	ASME Code Service Level D limits	Note b	10 CFR 100 limits ^c

a Event frequencies are nominal values.

b Radiation exposure to plant personnel in main control room not to exceed 5 rem (whole body), 30 rem (inhalation), and 75 rem (skin) from any one event.

c For relaxation of emergency planning requirements, lower dose must be met.

Table 15.4 Tentative acceptance criteria for ternary alloy metal fuel

Design-basis event category	Frequency ^a range (per reactor year)	Acceptance Limits			
		Peak transient temperature (°F)		Long term temperature (°F)	
		Bulk coolant	Cladding ^b	Bulk coolant	Cladding ^c
EC-I: Normal Operation	$F \geq 10^{-2}$	1200	1200	1200	1200
EC-II: Anticipated Events	$10^{-2} > F \geq 10^{-4}$	1200	1200	1200	1200
EC-III: Unlikely Events	$10^{-4} > F \geq 10^{-7}$	1300	1450	1300	1300
EC-IV: Extremely Unlikely Events	less than 10^{-7}	1300	1450	1300	1300

a Event frequencies are nominal values.

b Temperature at cladding centerline based on preventing cladding breach by stress rupture.

c Temperature at fuel-cladding interface based on preventing cladding breach by low-temperature point formation.

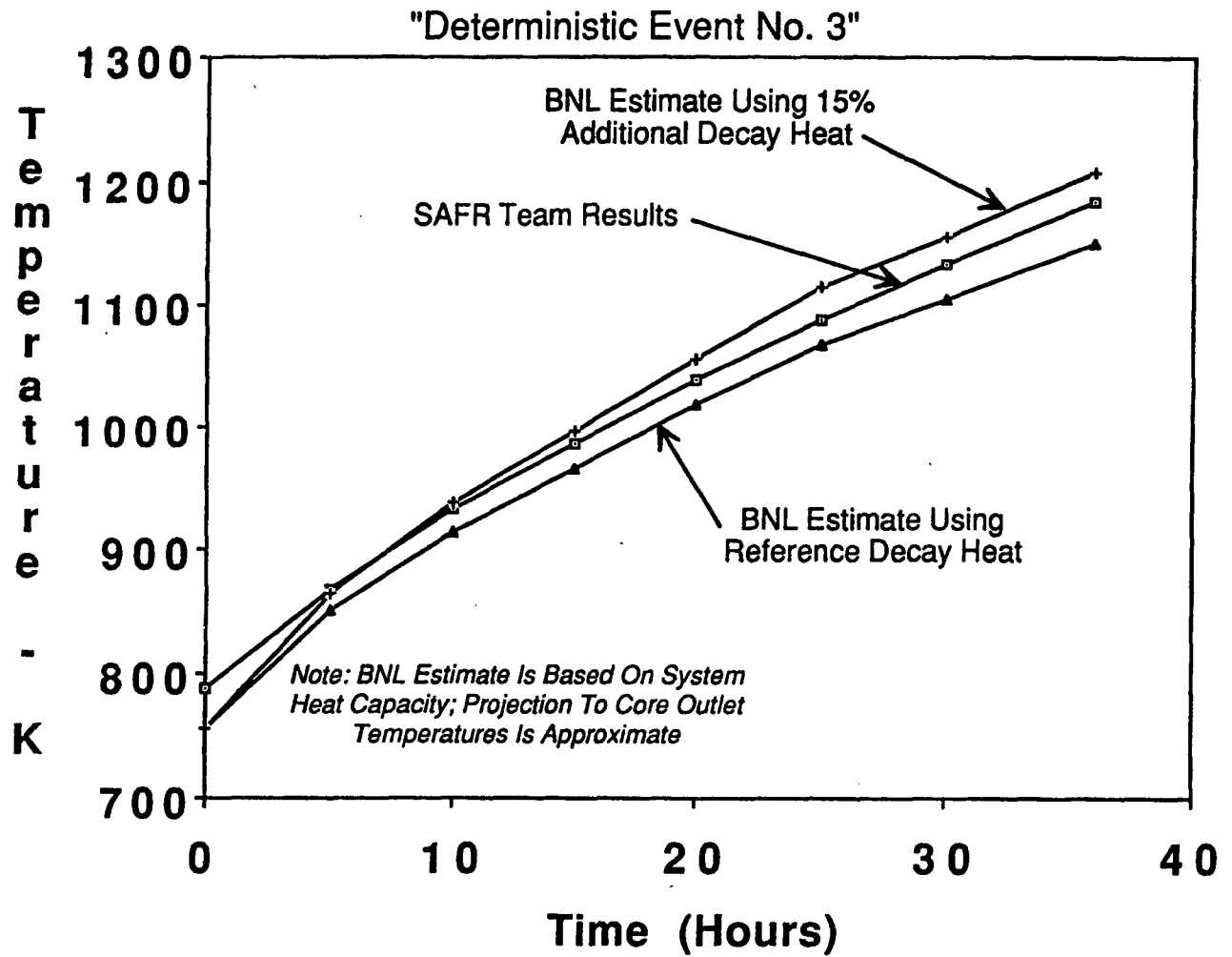


Figure 15.1 SAFR adiabatic heatup core outlet sodium temperature.

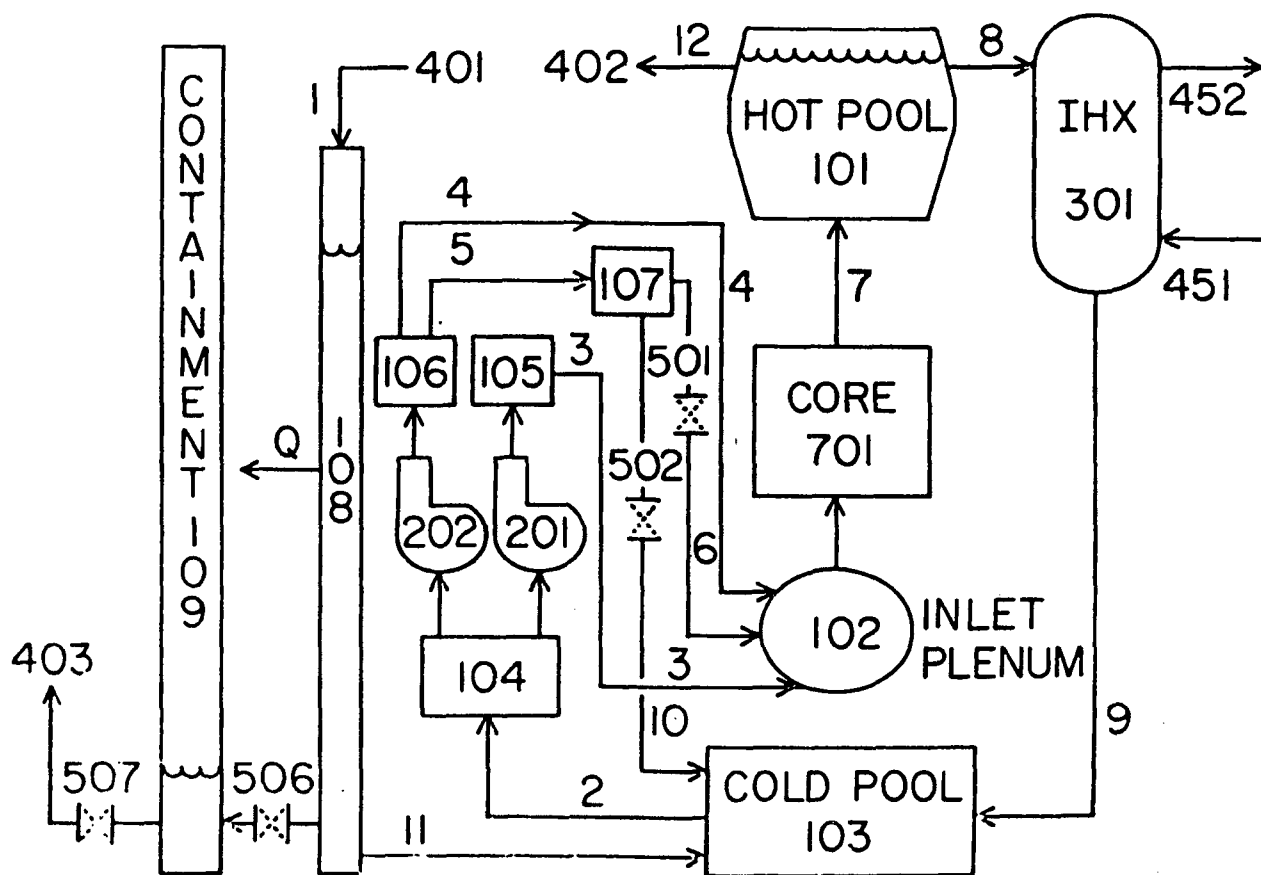


Figure 15.2 Schematic representation of LMR used in MINET pipe-break analysis.

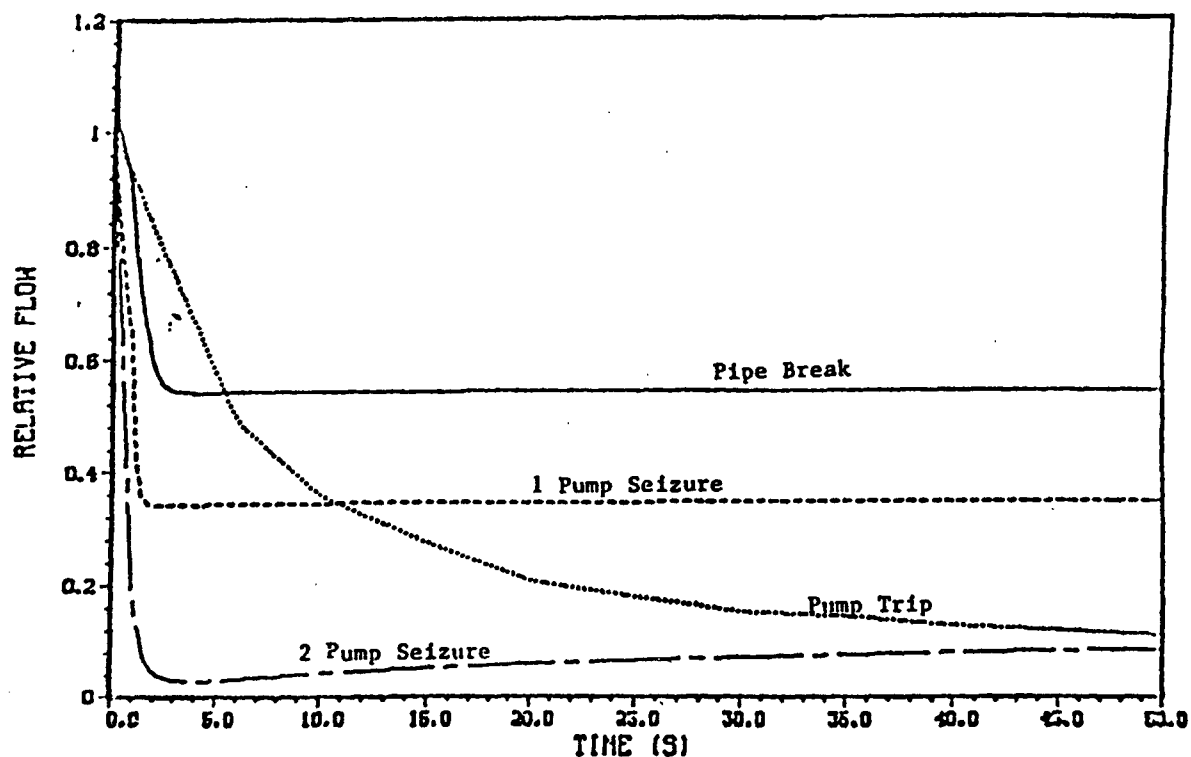


Figure 15.3 MINET prediction of SAFR core flow rate for various LOF cases.

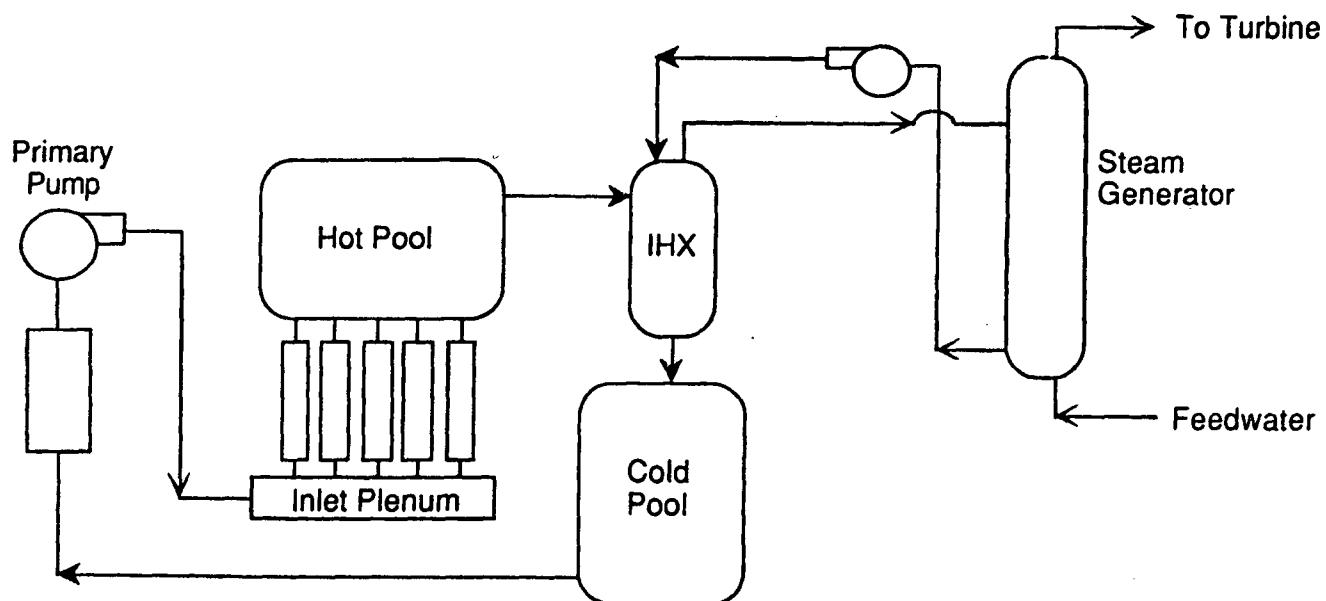


Figure 15.4 SUPERSYSTEM code representation of liquid-metal reactor systems.

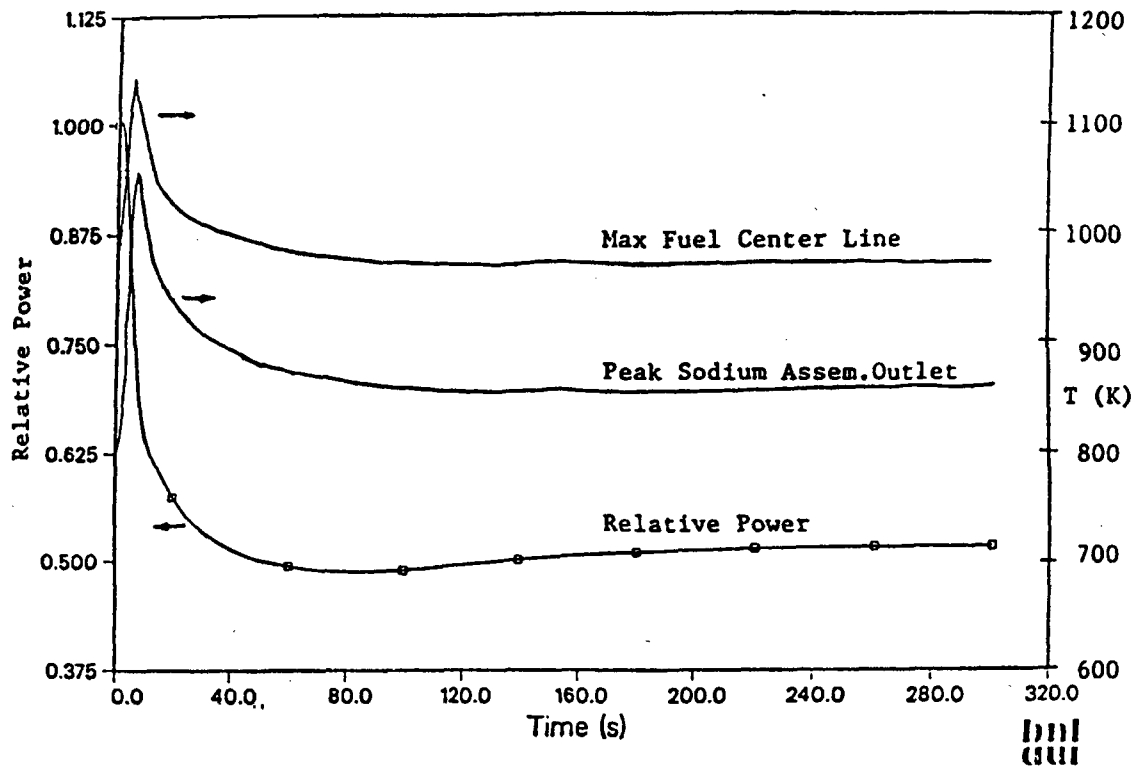


Figure 15.5 SUPERSYSTEM code prediction for SAFR pump seizure -- relative power and key temperatures

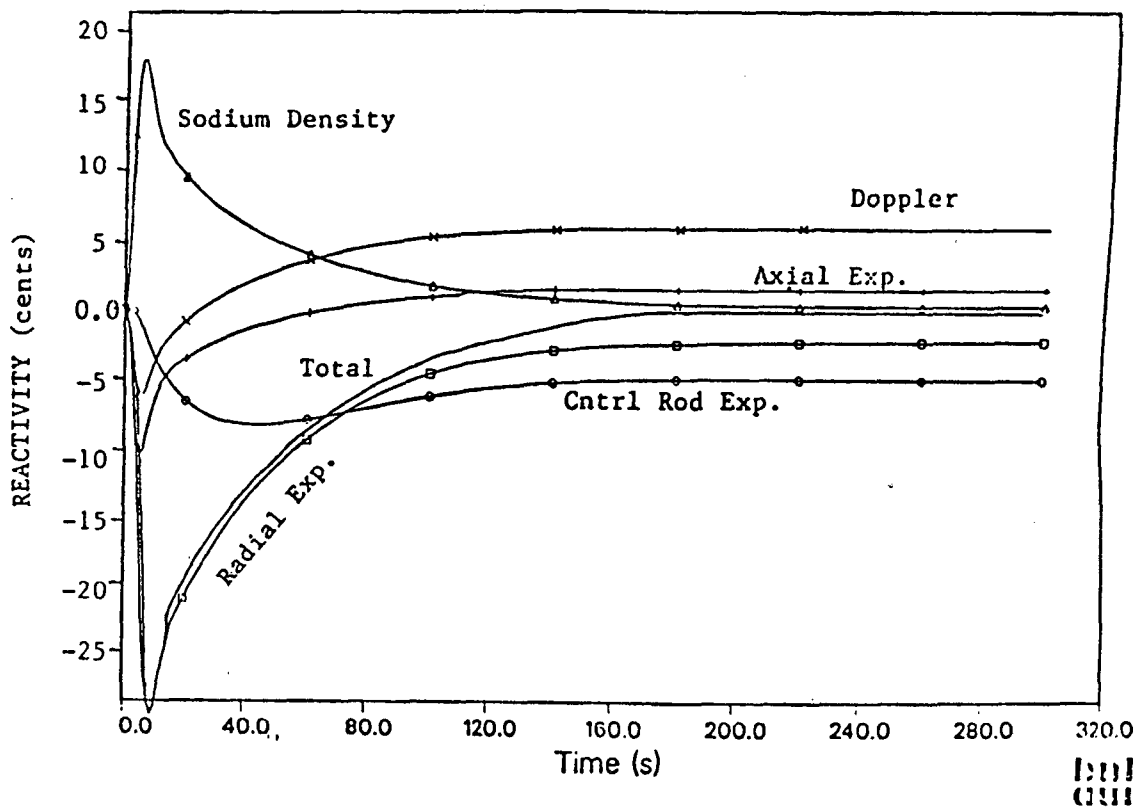


Figure 15.6 SUPERSYSTEM code prediction for SAFR pump seizure -- reactivity feedbacks.

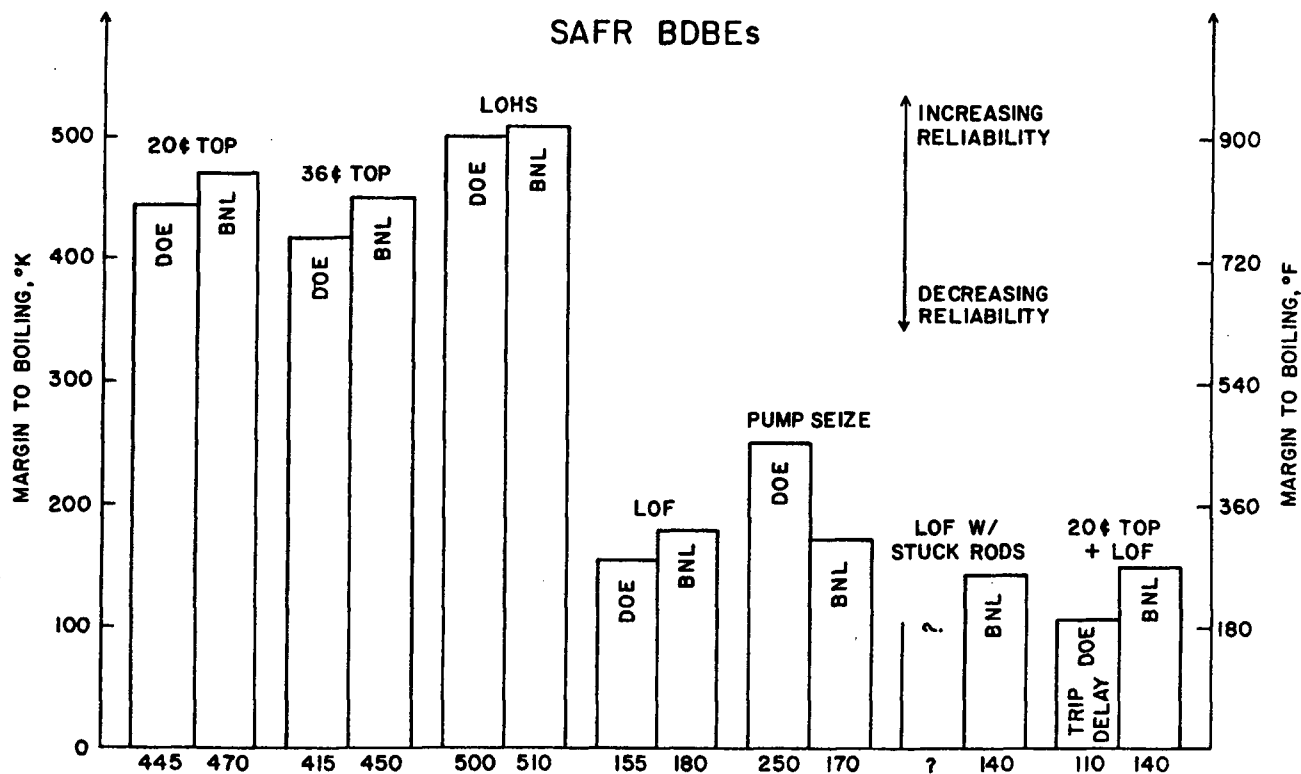


Figure 15.7 Minimum margin to sodium boiling for SAFR beyond-design-basis events (BDBEs).

Minimum margin to sodium boiling for 7 postulated unscrammed events, as estimated by the DOE team of RI and ANL, as well as BNL.

- 1 - 20 cent transient overpower (TOP)
- 2 - 36 cent TOP
- 3 - Loss of heat sink (LOHS)
- 4 - Loss of flow (LOF)
- 5 - pump seizure
- 6 - LOF with stuck rods
- 7 - 20 cent TOP + LOF

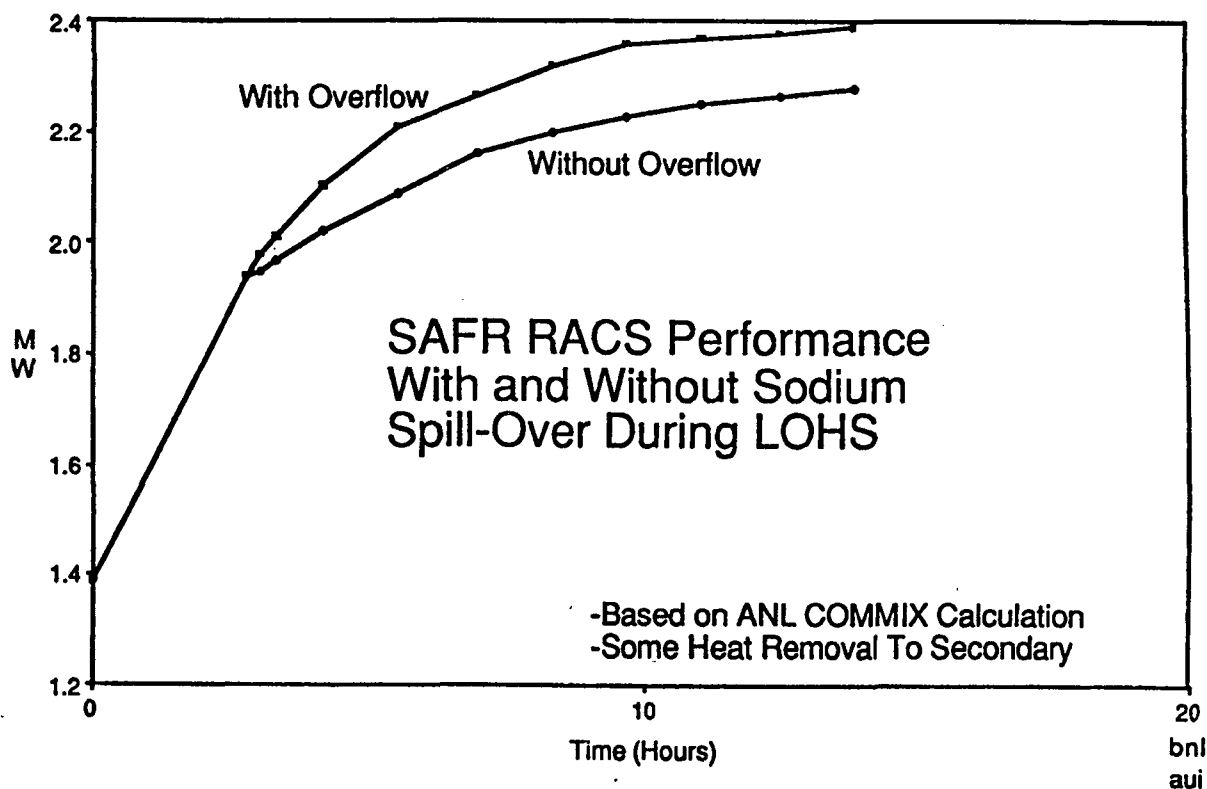


Figure 15.8 SAFR RACS performance with and without sodium spillover during LOHS.

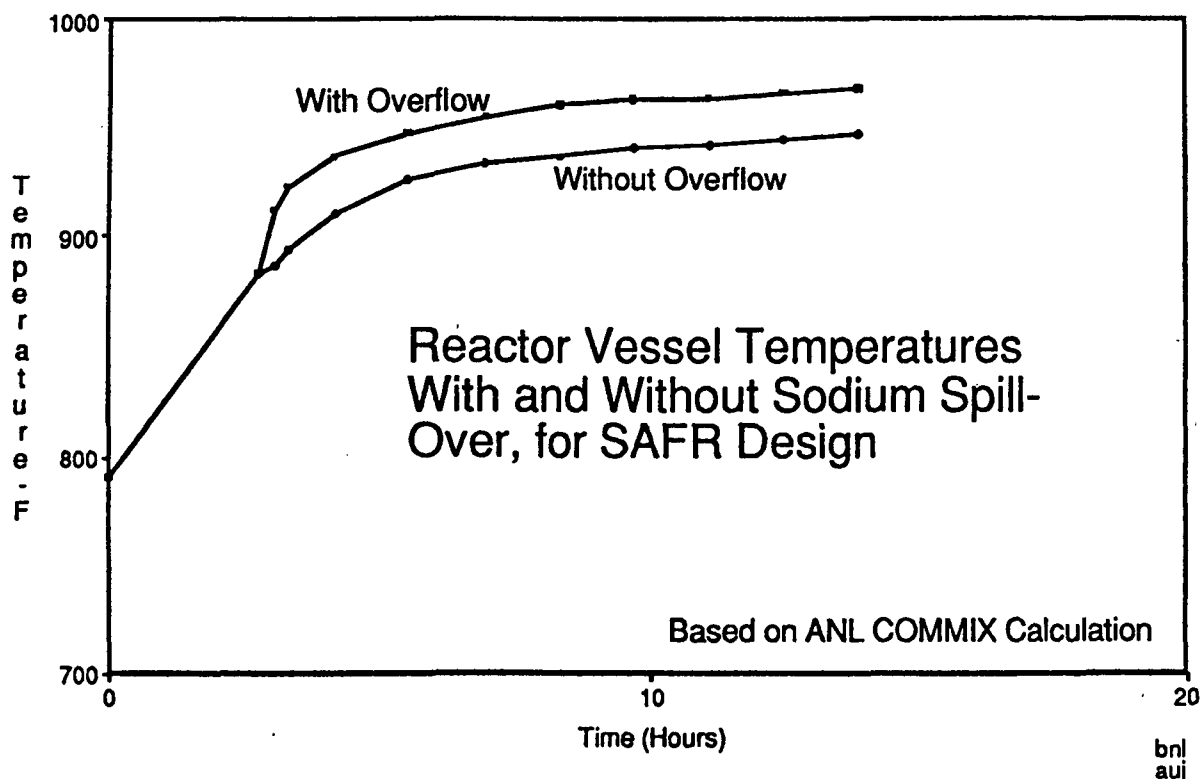


Figure 15.9 Reactor vessel temperatures with and without sodium spillover for SAFR design.

16 TECHNICAL SPECIFICATIONS

16.1 Summary Description

The technical specifications are not reviewed at this stage of the preapplication review.

17 QUALITY ASSURANCE PROGRAM

17.1 Summary Description

The requirement for a quality assurance (QA) program is imposed on all activities by DOE Order DOE 5700.6B (dated 9/23/86), "Quality Assurance." DOE Contract DE-AC0685NE37936, Subsection 5.1, "Quality Assurance," requires the Rocketdyne Division of Rockwell International (RI) to prepare a QA plan to ensure compliance with NRC licensing, and with ASME and ANSI code requirements. SAFR QA Plan 149QPP000001, Rev. 2 (dated 4/4/87), states that the QA plan applies to RI, Bechtel, and Combustion Engineering (CE). RI delegates QA execution responsibility to the associate contractors, reviews and accepts their plans after comments are resolved, and monitors compliance through audits of the approved QA programs.

17.2 Conceptual Design Review

During conceptual design, the eight criteria of NQA-1 1983 (listed below) were applied to the RI scope of work:

- organization
- quality assurance program
- design control
- instructions, procedures, and drawings
- document control
- corrective action
- quality assurance records
- audits

The staff reviewed the documents listed below to evaluate the degree to which the QA requirements in Chapter 17.1 of NUREG-0800 (Ref. 17.1), "Quality Assurance During the Design Control Phases," were satisfied for the eight areas addressed.

- 149QPP000001, dated 4/1/87, SAFR Quality Assurance Plan
- AI-DOE-13527, Rev. 13, (pgs. G 373-377), Responses to NRC Questions on the QA Program
- DOE 57000,6B, dated 9/23/86, Quality Assurance
- 149QPI000001, dated 3/3/87, SAFR Quality Assurance Program Matrix
- 149APQ000001, dated 1/19/87, SAFR Quality Audit Planning
- QAOP N1.02, dated 3/26/82, RI Procedure Quality Assurance Manual, Procedures, and Instructions
- QAOP N1.04, dated 12/16/83, RI Procedure Quality Assurance Audits
- QAOP N1.21, dated 10/10/80, Quality Assurance Plans
- Gavigan (DOE) to Morris (NRC) dated 6/18/87, transmitting Quality Assurance Program information
- EMP-3-63, dated 5/1/84, Documentation, Release and Control of Engineering Computer Programs
- A-500, dated 1/20/86, Rocketdyne Operating Policies and Procedures
- J-500, dated 1/10/86, Quality Program
- J-500.1, dated 8/27/86, Quality Program Audits

During its review of the documentation and responses to questions, the staff determined that design verification to establish the accuracy of technical activities be deferred until the next design stage. DOE and its consultants conducted programmatic design reviews. Because rigorous design verification was deferred in some areas, it may be necessary to repeat work or obtain other evidence of accuracy so that all designs and analyses relied upon in the safety analysis report have objective evidence of quality. Early project activities such as R&D and the development of computer codes and other design methods and procedures sometimes fall into this category. DOE

Quality Assurance Program

stated that Procedure EMP-3-63 was being implemented for the control of computer codes. This procedure has a separate requirement for verification which, if implemented, would avoid difficulties in that areas.

Corporate Procedure QAOP N1.04 is clear in establishing a requirement to "verify implementation of operating applicable to the quality assurance programs of ESG and its suppliers." The product integrity statement quoted in the SAFR QA Plan is also clear; "13) Conducting periodic audits ... to measure compliance with established operating procedures." The QA Plan section on audits is less clear: "Audits are conducted of systems and procedures of the design quality organization's processes. The procedures audited are first evaluated against code, standard, and contract requirements, and then the effectiveness of their implementation to the ongoing work effort is established during audits." A clarification provided in response to an NRC questions reads as follows: "The QA Program is composed of procedures and instructions necessary to describe how quality affecting activities are to be performed, and it is these procedures that are the subject of the audit."

17.3 Reference

- 17.1 NRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," NUREG-0800, July 1982.

APPENDIX A

INDEPENDENT ANALYSIS OF SELECTED EVENTS

A.1 DESIGN DESCRIPTION AND SAFETY OBJECTIVES

One of the objectives of the Sodium Advanced Fast Reactor (SAFR) design is to achieve a safe response to a design-basis event without scram (DBEWS). This safe response is achieved partly through the passive reactivity feedbacks, which can reduce reactor power during an accident without reliance on active systems or operator actions. Rockwell International (RI) considers the transients presented in this appendix to have very low probabilities of occurrence, as determined by probabilistic risk assessment (PRA) analyses. They are analyzed for three reasons. First, the anticipated transient without scram (ATWS) has historically been a major issue in the light-water reactor (LWR) designs, and thus has essentially become part of reactor safety analysis, although an ATWS is beyond the design basis. Second, RI would prefer to avoid having a safety-grade classification for either its automated plant trip system (APTS) or self-actuated shutdown system (SASS) (i.e., control rod systems). Although unlikely and extremely unlikely events without scram are perceived to contribute negligibly to radiological risk because of their low probability of occurrence, RI feels that demonstrating a passive response to such events is necessary to justify the non-safety-grade classification for both the SAFR APTS and SASS control system. (This position has not yet been accepted by the staff, and will be discussed later.) Third, because of the positive void worth, RI must demonstrate that the margins to sodium boiling are significant.

Argonne National Laboratory (ANL) performed much of the analysis using the SASSYS-1 code. Four channel types were used to represent the core. Channel 1 models the driver channel with the highest power-to-flow ratio. ANL used this channel to predict the peak subassembly behavior. Channel 2 represents the rest of the driver assemblies and Channel 3 models the internal blanket regions. A pseudo fourth channel was added to the SASSYS model to simulate the actual peak pin. A bypass channel was used to represent control rods and radial blankets. In the subsections that follow, the calculations performed by the designers will be discussed.

A.1.1 Loss-of-Flow Calculation

This transient scenario began with an instantaneous loss of power to the primary pump, intermediate loop pump, and steam generator feedwater pump. No scram was assumed, and the coastdown of the primary pump was characterized by an initial 6-second-flow halving time. By 140 seconds, the reactor vessel sodium flow was in natural circulation.

After the pumps initially tripped, the power began to drop almost immediately. The power level was predicted to drop to near decay heat levels by 200 seconds. This was due to the dominance of the negative reactivity feedbacks of the reactor. By 50 seconds, the passive feedbacks are estimated to have a net worth of -45¢ from the Doppler (-2¢), axial expansion (-4¢), control rod drive line expansion (-11¢), sodium density (+12¢), and radial expansion (-40¢). The initial feedback was shown to be negative, with Doppler and axial expansion being among the first feedbacks activated. The sodium temperature increased quickly, and this caused a positive feedback which countered the negative effects from the Doppler and axial expansion. Soon thereafter the heat deposited in the sodium was transported to the structure and led to the radial expansion which is the dominant negative feedback. The ANL calculations showed that the feedbacks brought the power level down to the 5-percent level by 200 seconds, and maintained it there for the rest of the 800-second transient.

The nearest approach to sodium boiling occurred at 40 seconds, at which time the margin dropped to 315°F when the peak assembly exit sodium temperature reached 1410°F (sodium saturation temperature is about 1725°F). At the same time, the maximum fuel center line and cladding temperatures were 1470°F and 1420°F, respectively. The long-term cladding temperature limit of 1340°F was surpassed during this transient for about 120 seconds. Since the stress rupture and eutectic penetration limits are 3 hours at this temperature, no fuel or cladding failure was expected.

Appendix A

A.1.2 Transient Overpower Events

Two different calculations were performed. The first analysis was for a 0.65¢-per-second reactivity ramp rate, up to a 20¢ total reactivity insertion. The second analysis was for a 5¢-per-second reactivity ramp rate, up to a 36¢ total reactivity insertion. In both sets of analyses, the primary pumps are assumed to continue running at 100-percent capacity, and nominal full-power heat rejection through the steam generators was assumed.

The first scenario, with the 0.65¢-per-second ramp to 20¢ insertion total, was intended to simulate a withdrawal of approximately three of the six primary control rods. RI presented the results showing that the power peaks at 130 percent of rated power around 30 seconds, and then drops to 104 percent by 800 seconds. The passive feedbacks, activated by the increase in temperatures, eventually brought the core power back to near-normal operating conditions. The dominant negative feedback was the radial expansion; the sodium density was the only positive feedback. The safety margins are large since the peak temperatures were 1540°F, 1175°F, and 1125°F for the fuel center line, cladding, and coolant, respectively.

The second calculation was a 5¢-per-second ramp to a total insertion of 36¢. This simulation represented the uncontrolled withdrawal of all six primary control rods. The worth of these rods came from the burnup swing that is estimated to be approximately 36¢ (i.e., the maximum worth of the rods in the core from the beginning to the end of the fuel cycle). RI predicted that the power would reach about 155-percent nominal power in 60 seconds and slowly return to around 107-percent nominal power by 700 seconds. The initial insertion of reactivity increased the power, which resulted in a primary system heatup. The temperature increase caused the radial expansion, Doppler, axial expansion, and control rod drive line feedbacks to initially insert negative reactivity. The sodium density is the only positive feedback for a heatup event. The net reactivity of the system became slightly negative by 175 seconds, and stayed negative for the rest of the simulation. The peak temperatures were predicted to be 1600°F, 1300°F, and 1200°F for the fuel centerline, mid-wall cladding, and coolant, respectively. The margin to fuel melting was 400°F; the margin to sodium boiling was 500°F. The mid-wall cladding temperature was very close to the long-term (i.e., 3 hours) cladding temperature limit of approximately 1340°F, based on stress-rupture and eutectic penetration, with a margin of only 40°F. This temperature level persisted through the 700-second transient and became the quasi-equilibrium temperature. Cladding wastage would be expected and this transient must be terminated within 3 hours or fuel failures would be expected. The life of the fuel could be shortened by this cycle even if the transient was terminated earlier. (Note: It is likely that a TOP event will evolve into a combined TOP and loss-of-heat-sink (LOHS) event after several minutes, as the heat production exceeds the ultimate heat dump capacity.)

A.1.3 Loss of Heat Sink

This event was initiated by the loss of all heat rejection capacity at the steam generator during the first 20 seconds. The primary and intermediate sodium pumps continued to operate at 100-percent flow capacity. After about 120 seconds, the primary sodium temperature increased sufficiently to cause sodium overflow into the annulus between the hot pool liner and the reactor vessel. This improved reactor air cooling system (RACS) performance significantly. RI predicted that the reactor power would be about 20 percent of nominal power by 800 seconds. The increase in temperature in the primary system caused such expansion of both the grid plate and above-core-load pads that the radial expansion became the dominant negative feedback. Furthermore, the overflow of sodium into the annulus caused a significant amount of vessel elongation in response to thermal expansion. This caused the control rods to be pulled from the core providing a positive feedback of +20¢ by 800 seconds. The sodium density effect is also a positive feedback effect with a worth of about +20¢ at 800 seconds. The net reactivity effect was negative throughout the transient, and the power level dropped to about 20 percent of nominal by 800 seconds.

The safety margins for this transient appear to be large. The fuel center-line peak temperature was estimated to be 1360°F, which is 640°F below the solidus temperature. The mid-wall cladding temperature remained below 1100°F (i.e., a margin of at least 240°F to long-term cladding limit); the margin to sodium boiling was about 700°F.

A.1.4 Pump Seizure

This transient was initiated by seizing one of the two primary reactor coolant pumps. RI acknowledged that these results are very sensitive to the unseized pump's behavior and the associated backflow resistance through the seized pump. The unseized pump must be capable of pumping while cavitating, since it would experience a 33-percent increase in flow in response to the drop in system impedance (flow resistance). The resistance in the locked rotor directly impacts on the amount of flow that will bypass the core during the event.

The results from RI's calculation indicate that 40 percent of nominal flow continued through the core while the remaining flow bypassed the core by flowing backwards through the failed pump to the cold pool. The initial effect was for the power level to increase a few percent in response to the positive sodium density feedback effect. However, the increase in temperatures caused radial expansion, control rod drive line expansion, axial expansion, and Doppler feedback to generate a net negative feedback by 2 seconds, which eventually reduced the power level to 65 percent of nominal by 18 seconds.

The peak temperatures predicted for the event were 1550°F for the fuel center line and 1375°F for the peak sodium temperature. The margins are 450°F and 325°F to fuel melting and sodium boiling, respectively. The mid-wall cladding temperature was not given, but has been estimated to be 1450°F at the peak and approximately 1300°F by 18 seconds when the transient calculation was ended. This exceeds the long-term cladding limit (for 3 hours) temperature of 1340°F, but for only 15 seconds, and should not be significant since the power will continue to drop because of the net negative reactivity of the core. The reactivity was predicted to be about -8¢ at the end of the 18-second calculation.

A.1.5 Combined LOF, TOP, and LOHS

The most significant matter influencing the severity of this accident scenario is the magnitude of the reactivity insertion for the TOP event. The case analyzed by RI assumed an insertion rate of 0.65¢ per second ramped to a total of 20¢. This ramp rate is assumed to represent the withdrawal of three primary control rods, and is expected to envelope the reactivity insertion associated with a safe shutdown earthquake (SSE). There are two possible sources for positive reactivity insertion during an SSE: relative core to control rod vertical movement and horizontal radial motion. The range of expected positive reactivity insertion during an SSE is 3¢ to 13¢, according to RI. The pump coastdown assumed was the standard 6-second halving time. The pumps (primary, intermediate, and feedwater) were assumed to trip when the power reaches 115 percent, which corresponds to the time when a trip signal would be sent by the control system, approximately 20 seconds.

The power level began to drop after pump trip in response to the increase in temperatures caused by the reduction in coolant flow. The two dominant feedbacks are the radial expansion feedback due to the above-core load pad thermal expansion and mechanical bowing due to thermal gradients, which together generated about -47¢ of reactivity during the first 50 seconds. The sodium density feedback inserted about +17¢ during the same period. The sum of the other feedbacks, that is, Doppler, axial expansion, and the control rod drive line expansion, gave a net negative feedback which reduced the power level to near the decay heat level by 200 seconds, and maintained that level throughout the rest of the 800-second simulation. The calculated peak temperatures were 1530°F, 1540°F, and 1530°F for the fuel center line, cladding, and sodium, respectively. The margin to fuel melting was 385°F; the margin to sodium boiling was 175°F. The long-term cladding failure

Appendix A

temperature of 1340°F (i.e., 3 hours) was surpassed for only a few hundred seconds; the short-term failure temperature of 1900°F was never approached.

Sensitivity Calculations

RI performed several sensitivity calculations to estimate the SAFR system response to the DBEWS with adverse variations in some of the key feedback parameters. The emphasis was placed on radial expansion reactivity feedback due to its dominance in mitigating several of the DBEWS events. For the LOF event, using a 40-percent reduction in radial expansion, RI estimated that the margin to sodium boiling dropped from 300°F to 190°F. The TOP, with a ramp of 5¢ per second to 36¢, was re-evaluated with a 40-percent reduction in radial expansion. The resulting margin to sodium boiling was reduced by 125°F. For the same reduction in radial expansion, the LOHS had a reduction in the sodium boiling margin from 800°F to 775°F. The final sensitivity calculation was performed on the combined LOF, TOP, and LOHS event. With the same 40-percent reduction in radial expansion, the margin to sodium boiling dropped from 175°F to 50°F. In this case, the long-term cladding failure temperature (i.e., 1340°F for 3 hours) was surpassed for the 750 seconds of the simulation, while the cladding temperature was at 1450°F. Thus, action would have to be taken to avoid fuel damage.

A.2 SCOPE OF REVIEW

The staff reviewed PSID Appendix A (Ref. A.1) in two steps. The first step involved reviewing the calculations presented in the PSID. This amounted to studying the transients analyzed by ANL (for RI) to determine whether a comprehensive package of transients was analyzed. The LOF transients were identified as the most challenging for this design with the single-pump-seizure event identified as a potentially challenging beyond-design-basis event (BDBE).

One important issue raised during the discussions was the accident at EBR-I in 1955, known to have been caused by fuel bowing. Bowing is one of the significant reactivity feedbacks in the SAFR design. The designers were asked to discuss how this event was factored into the SAFR design to avoid the fuel bowing. The solution to the EBR-I problem was found when Mark-III fuel was installed (a year or two after the accident). The fuel had wire-wrapped pins bundled in hexagonal ducts, reducing the positive reactivity feedback and power instability. The fuel planned for SAFR has similar wire-wrapped pins in hexagonal ducts. RI has positioned the restraint system so the thermal bowing of the ducts will generate a negative reactivity feedback under most conditions of interest. The fuel and restraint system in the SAFR reflects the experience gained in EBR-I, EBR-II, and the FFTF.

For the second part of the review, the NRC staff sponsored independent calculations by Brookhaven National Laboratory (BNL) (Ref. A.2). The SUPERSYSTEM and MINET codes were used to simulate the SAFR core, as well as the rest of the reactor vessel, intermediate loop, and steam generator. SUPERSYSTEM was modified to handle the reactivity feedbacks associated with the metal fuel. In most cases, design details were taken from the Preliminary Safety Information Document (PSID) (Ref. A.1).

The reactivity coefficients were taken from the data supplied by the designer. The coefficients supplied by the designers were reviewed and compared with previously published values for other liquid-metal reactors (LMRs). The staff found that the feedbacks are consistent in size and magnitude, and consistent with simple hand calculations. As these coefficients are crucial to the "inherent shutdown" arguments, ultimately they must be recalculated by an independent source. However, use of the applicant's reactivity coefficients is appropriate at this stage in the review process.

In the subsections that follow, the SUPERSYSTEM and MINET calculations for several DBEWSs, as performed by BNL, will be summarized after the SUPERSYSTEM model is described briefly.

A.2.1 SUPERSYSTEM Code Computer Model

SUPERSYSTEM was developed for LMR analysis at BNL while BNL was engaged in the Clinch River program. The code has been extensively validated against EBR-II and FFTF plant data in the past (although not for the EBR-II metal fuel). Recently, the code has been modified to model the characteristics of advanced LMRs.

The reactor module was represented in reasonable detail. Seven channels in the core were modeled, including the driver, internal and radial blankets, and control and shield assemblies. The volumes and flow areas for the reactor vessel and intermediate loop were taken from the PSID. The representation extended to the steam generator for which the inlet feedwater flow and temperatures were specified as boundary conditions. All major components were modeled, including the primary and intermediate pumps, the intermediate heat exchanger (IHX), the steam generator, and all appropriate piping.

The SUPERSYSTEM code was modified to account for the major inherent characteristics of the metal fuel. Modeling was added to track the fuel elongation in a condition when the fuel and cladding are locked up (i.e., when the fuel expands to contact the cladding). It has been determined that the cladding controls the fuel axial expansion of the fuel because of the structural weakness associated with metal fuel. The thermal expansion in the radial direction is also tracked during the analysis to account for the negative feedback expected during a thermal dilation and positive feedback from a radial contraction. This effect is evaluated by tracking the dimensions of the above-core load pads and the core-support plate to determine the radial dimensions of the core at any given moment. Control rod drive line expansion is accounted for by calculating its temperature and corresponding thermal expansion and contraction. When a rod which is suspended from above elongates, the control assemblies move deeper into the core causing a negative feedback. This feature is enhanced in the SAFR via flow collectors used to direct hot sodium from the surrounding driver assemblies over the drive line. However, the reactor vessel is also suspended from above, and during thermal expansion begins to withdraw the control assemblies from the core.

The thermal gradient bowing feedback is the only feature of the metal-fuel core that was not factored into the reactivity model of SUPERSYSTEM. This phenomenon occurs because the temperature gradients that are generated across an assembly, cause one side of the assembly to expand more than the other. The SAFR core restraint system uses the "limited free bowing" technique in which the load pads and restraint system are designed to give an outward bow in the center of the fuel. This restraint system is designed to make the bowing component smaller than the radial dilation and thus keep its effect negative over most regions of interest. This intricate phenomenon was not implemented in the SUPERSYSTEM modeling. It is noted that its 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. At this time, it does not appear that bowing can insert any positive reactivity during any portion of the accidents reviewed to date. Hence, neglecting it is a conservatism assumption. In subsequent reviews, the staff should attempt to confirm ANL and RI estimates of the bowing component.

A.2.2 Loss-of-Flow Event

The loss-of-flow (LOF) event is initiated by an instantaneous loss of power to the primary, intermediate, and steam generator pumps. No scram was assumed and the inertially controlled coastdown of the primary pump is characterized by an initial 6-second-flow halving time. Approximately 135 seconds after the loss of power, the pump rotor stops rotating and the coolant flows by natural circulation.

Key results are presented in Figures A.1 through A.3. In Figure A.1, it is shown that the power level drops off as the flow decreases. The fuel temperatures increase because of the reduced coolant flow in the core, activating the Doppler and axial expansion feedbacks. The reduction of coolant flow

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in the core also causes the sodium temperatures to increase. This inserts positive reactivity because of the hardening of the neutron spectrum. Once the sodium begins to heat up, it disperses the heat throughout the system and activates the radial dilation at the above-core load pads, which in turn activates the radial expansion feedback. In Figure A.2, the radial expansion feedback is shown to be dominant, causing a reduction in power. By 140 seconds, the power level reaches a quasi-static level at about 10 percent of rated power, where the reactivity feedbacks are balanced and the power remains steady. The peak channel sodium exit temperature is shown in Figure A.3 to be 1020 K (1377°F) and has a 180 K (324°F) margin to boiling. The peak fuel center line temperature is 1050 K (1431°F), which gives a margin of 316K (569°F) to the fuel melting point. The reduction in the solidus or fuel-melting temperature is not considered here since the effect of the zirconium migration was not simulated.

These results are quite similar to those predicted by ANL. The SUPERSYSTEM prediction for the peak coolant temperature was 1020 K (1377°F), while ANL calculated it at 1039 K (1410°F). SUPERSYSTEM predicted the peak fuel centerline temperature as 1050 K (1431°F), while ANL calculated it at 1071 K (1470°F). The reactivity feedbacks were predicted by both to reduce the power level to a much lower level and avoid damage to the fuel. Also, the margin to sodium boiling is large in both calculations.

The eventual result is that the reactor power level transitions to a much lower level. However, the feedbacks which initially started out strongly negative ultimately reestablish criticality after a few hundred seconds. The feedbacks were able to reduce the power level significantly because the temperature defect in a metal fuel core is small, about \$1.6. [The net loss of reactivity in going from room to operating temperature is defined as the temperature defect.] Hence, the negative feedbacks were able to overcome the Doppler feedback and decrease the power, while maintaining the core at elevated temperatures. The results indicate that the reactor could maintain this state for an extended period without danger and the operator would not have to make a fast decision or take quick action to avoid damage to the core for this event. However, the plant must be scrammed to shut the fission process down completely.

A.2.3 Transient Overpower Event

Doppler feedback in a metal-fuel LMR is significantly smaller than for an oxide core. This is because the metal core has a very hard spectrum with few neutrons in the Doppler resonance region. This characteristic works in a favorable manner for fuel loading and for minimizing the temperature defect, but during an event in which the power increases, the Doppler feedback is less effective in stopping the power rise. The designer must rely on other feedbacks to limit a power increase.

The SAFR TOP event is limited principally by the activation of the radial expansion feedback, although other negative feedbacks contribute. These feedback effects have time constants of a few seconds, unlike the prompt response for Doppler. The metal-fuel core has a small TOP initiator because the control rod worth is minimized. This is possible because the breeding ratio is high enough to place the burnup swing near zero, which allows the designer to minimize the rod worths. Only enough excess reactivity is added to the fuel cycle to overcome the temperature defect and the small expected burnup swing.

To analyze the response of the SAFR to the TOP event, two different initiators were used. The calculations that follow, using SUPERSYSTEM, were performed to independently verify those performed by ANL and discussed earlier in Section A.1 of this appendix.

20¢ TOP

The reactivity ramp rate for this event is 0.65¢ per second to a total insertion of 20¢. The plant is assumed to continue operating as normal except for the withdrawal of three of the six primary control rods. This also corresponds to worst-case, seismic-induced insertion. The ramp rate corresponds to

a withdrawal speed of 0.2286 meter per minute (9 inches/min). In the SAFR module, each control rod is operated individually, that is, rods are not operated in groups.

The results are shown in Figures A.4 through A.7. The power is shown in Figure A.4 to rise to 130 percent and return to a quasi-static power level of 115 percent by 160 seconds when criticality is reestablished (see Figure A.5). ANL predicted the power peak at the same level; but the power then drifted down to 104 percent. The difference comes from the control rod drive line expansion feedback, which ANL shows to initially go to -7¢ while the SUPERSYSTEM prediction goes to -4¢ and then approaches zero from vessel expansion. The other feedbacks match the ANL calculations very closely.

The sodium and peak fuel centerline temperature are comparable in both calculations. In Figure A.6, the peak sodium temperature in the core is shown to reach 870 K (1107°F); the ANL results go to 880 K (1125°F). The margin to boiling is predicted by both codes to be about 470 K (846°F). The peak fuel centerline temperature is shown in Figure A.7 to be 1138 K (1589°F), which compares well with the 1111 K (1540°F) predicted by ANL. The margin to fuel melting (not including the migration effects of zirconium) is 228 K (411°F). All margins are thus predicted to be adequate.

The SAFR module appears to be capable of withstanding a 20¢ TOP indefinitely since the elevated power and temperature pose little challenge to core margins. It should be noted (see Figure A.5) that the Doppler feedback inserted about 6¢ for this event while the radial expansion feedback supplied about -15¢ by 300 seconds and is clearly the dominant negative feedback mechanism. This suggests that TOP events can be mitigated as long as the rate of reactivity insertion is within the time constant for the radial feedback.

36¢ TOP

The second SAFR TOP event assumes a ramp rate of 5¢ per second to a total of 36¢. This corresponds to the withdrawal of all six of the primary control rods at a rate of 1.02 meters per minute (40 inches/min). Again, the rest of the plant is assumed to function as usual.

The SUPERSYSTEM results are shown in Figures A.8 through A.11. The power, as shown in Figure A.8, is predicted to reach 156 percent of nominal and slowly reduce to nominal conditions. The results closely match those of ANL in which power peaked at 154 percent and was at 120 percent by 350 seconds. For the same times, SUPERSYSTEM predicted the power to be at 156 percent and 125 percent, respectively. The transient simulation was terminated at 350 seconds since the peak fuel and cladding temperatures had already been exceeded, and the power level was continuing to decrease. The reactivities plotted in Figure A.9 also match within a few cents those predicted by ANL, except for the control rod drive expansion feedback. The SUPERSYSTEM predictions indicated that the reactor vessel was elongating and withdrawing the control rods from the core; the ANL plots indicate no vessel expansion by the constant control rod feedback. This effect caused the different rates of power reduction between the calculations. However, the effect is small, and both codes show the same trends.

In Figure A.9, the total reactivity is the sum of all of the components. The radial feedback is the largest of the negative feedback effects, while the second largest is the sodium density effect, the only major positive feedback. The Doppler feedback generated about -7¢, while the radial expansion was worth -25¢ by 350 seconds. Thus, the power history of the TOP event is basically controlled by the radial expansion feedback.

The margins to sodium boiling and fuel melting were substantial in this calculation. The peak sodium temperature in the core is shown in Figure A.10, where it is plotted against the saturation temperature. The results show that the temperature is about 895 K (1152°F), which gives a margin of 445 K (801°F). ANL predicted peak sodium temperature as 921 K (1200°F). The peak fuel center line temperature was calculated to be 1221 K (1738°F) using SUPERSYSTEM, which is higher

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than the 1144 K (1600°F) estimated by ANL. However, neither code predicted more than a 145 K (261°F) margin to the fuel melting temperature of 1366 K (2000°F). The peak fuel-cladding interface temperature momentarily reached the eutectic threshold of 972 K (1290°F) at around 20 seconds, but it settled back to about 944 K (1240°F) and will continue to decrease as the power descends. Consequently, no fuel damage is anticipated during this event.

A.2.4 Loss of Heat Sink

The pumps providing feedwater to both steam generators are assumed to lose power, causing the steam generators to dry out in 20 seconds. Heat rejection is lost, except for the small amount that is leaving through the RACS (about 2.5 MWt). The rest of the SAFR module continues to operate as normal.

The SUPERSYSTEM predictions are shown in Figures A.12 through A.15. The power, as shown in Figure A.12, drops from rated conditions to about 6 percent by 400 seconds. The reactivity feedbacks are shown in Figure A.13. The positive feedback from the sodium density is initially nullified by the negative feedback from the radial expansion. This is expected, since heat rejection is lost at the steam generators, allowing the heat to be dissipated throughout the system. Having such a large thermal dump dampens the thermal front.

By 40 seconds, the combined effects of all the negative feedbacks outweigh the positive feedbacks, and the power decreases. The radial expansion feedback is the dominant feedback reducing the power level.

From Figure A.13 it can be seen that the control rod drive line feedback begins a few cents negative. The time constant for this is 28 seconds. The reactor vessel also begins to thermally expand (because of increased temperatures) at a rate corresponding to its 750-second time constant, withdrawing the control assemblies from the core and causing a positive feedback. SUPERSYSTEM predicts this to be worth +10¢ by 400 seconds.

The peak temperatures for this event do not challenge the system. Figure A.14 shows the peak sodium temperature plotted against the saturation temperature, and shows that the temperature increase during the event is small. The margin to boiling is 507 K (913°F). The resultant peak fuel temperature history during this event is presented in Figure A.15 which shows that fuel center line (as well as the rest of the fuel) temperature decreases as the power drops.

The SUPERSYSTEM results for the LOHS differ somewhat from those predicted by ANL. The reactivity trends are similar, but the SUPERSYSTEM timing is compressed when compared to the ANL calculation. The discrepancy has been traced to the fact that the ANL model has user-specified, core-inlet sodium temperature (as a function of time), while the SUPERSYSTEM model calculated the inlet temperature (it increases with time). The SUPERSYSTEM model (which modeled the IHX, cold pool, and intermediate loops) predicted higher core-inlet sodium temperatures, which resulted in the power dropping faster. ANL predicted the power to be 19 percent at 800 seconds; SUPERSYSTEM calculated it to be 6 percent at 400 seconds. Both codes predicted the same general outcome. The power will change to a lower level and will pose no threat to the reactor system.

A.2.5 Pump Seizure

One of the two centrifugal pumps is assumed to seize during full-power operation. The other pump continues to operate, and the plant protection system fails to scram. A pump seizure at full speed is considered to be a very unlikely event. The only known full-speed seizure in the United States was a pump in the Sodium Pump Test Facility at Energy Technology Engineering Center (ETEC). All other seizures have occurred during the coastdown phase.

The MINET code was used to model the flow network around the core. The most important aspect of this analysis is the modeling associated with the behavior of the operating pump and the behavior of the seized pump. This ultimately determines how much flow will continue through the core. The seizure causes a drop in system impedance, with the operating pump experiencing a flow increase to 128 percent of its rated condition. The pump will cavitate in this mode. The designers are aware of this problem and will design a pump to operate in cavitation. The resistance through the locked rotor directly determines the amount of flow that will bypass the core and feed back into the cold pool. The MINET model accounted for all of these effects and found that 65 percent of the system flow will bypass the core through the locked rotor. As shown in Figure A.16, the remaining 35 percent of the sodium flow in the system will still continue to flow through the core.

The prescribed flow condition from MINET was simulated in SUPERSYSTEM and the results are presented in Figures A.17 and A.18. The power in Figure A.17 drops immediately to 50 percent and reestablishes criticality at about 52 percent of rated power. The reactivity feedbacks that make up the passive response to a sudden loss of flow in a metal-fueled core are shown in Figure A.18. The initial effect is from the Doppler and axial expansion feedback. The reduction in coolant flow causes heatup. Within the first few seconds, Doppler and axial expansion insert -6% and -9% respectively. The drop in sodium flow, coupled with the increase in fuel temperatures, leads to increased sodium temperatures, generating a positive feedback. This added about $+17.5\%$ of reactivity at its peak, and slowly returned to zero around 200 seconds when the reactor reestablished criticality. Higher sodium temperatures increase temperatures in the surrounding structures and the load pads, activating the radial expansion. From Figure A.18, it is clear that the radial expansion feedback is the dominant mitigating factor.

The Doppler reactivity only inserts about $+7\%$ of reactivity after the power is reestablished at 52 percent of nominal, as shown in Figure A.18. A negligible Doppler, and the correspondingly reduced temperature defect, make it possible for the small reactivities generated in the metal-fuel core to force the power to a much lower level.

Despite the challenge of this event, safety margins are adequate. SUPERSYSTEM calculated the peak fuel center line temperature (see Figure A.17) to be 1134 K (1581°F), which leaves a 232 K (419°F) margin to the solidus temperature. The peak assembly outlet temperature was calculated to be 1050 K (1431°F), which is well below the sodium saturation temperature of 1222 K (1740°F). Furthermore, the feedbacks reduced the power level to a point at which the maximum fuel center-line temperature and the maximum sodium temperature in the core are low enough at the new quasi-static condition that the system could remain at this condition indefinitely without serious consequences.

The ANL prediction for this event was terminated at 18 seconds and a comparison is not practical. A difference between the two analyses is that ANL estimated that the core would receive 40 percent of the system flow even after the pump seized, while the MINET results indicate the value to be 35 percent. Both calculations, however, predicted that this event would be mitigated by the feedback effects in the core with no fuel damage or requirements for immediate operator action.

A.2.6 Combined TOP and LOF

The 20% TOP with a ramp rate of 0.65% per second (9 inches/min or 13.72 m/sec) was rerun with a LOF added. The purpose of this simulation was to envelope the reactivity insertion that could accompany a safe-shutdown earthquake (SSE). The 20% insertion corresponds to the worth of three primary rods, and should be conservative for the SSE since RI estimated that the reactivity insertion from assembly movement would be a maximum of 13%. The event is initiated by tripping the pumps.

The SUPERSYSTEM results are shown in Figures A.19 through A.21. The power and flow history are shown in Figure A.19. The power increases a few percent and then drops off, by 200 seconds, to 14 percent of nominal power, and will continue to drop because the total reactivity remains

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negative (see Figure A.20). The plot of the reactivities demonstrates that the radial expansion feedback dominates the initial response of the core and, therefore, controls the drop in power. The control rod expansion and the radial expansion are the most significant negative feedbacks by the end of the 200-second transient. The power decreases because the average system temperature increases (which activates the radial expansion, control rod drive line expansion, and axial expansion for the negative feedbacks, while causing a positive response only from the sodium density), generating a net negative feedback.

The margins for this event are large. The peak sodium temperature was predicted to be 1060 K (1449°F), leaving a 150 K (270°F) margin to sodium boiling (see Figure A.21). The peak fuel centerline temperature was calculated to be below 1089 K (1500°F), which makes the margin to fuel melting greater than 277 K (499°F). No cladding damage would be expected from eutectic penetration.

A.2.7 Conclusions Regarding Independent Analyses

The results of the independent analyses indicate that the reactivity feedbacks inherent to the metal fuel core can provide a net negative feedback that will generally result from a temperature heatup event, and a power level that will usually decrease quickly enough to prevent core damage. This is possible because the small temperature defect (i.e., 1.6) offers little resistance to a transition to lower power. These findings are contingent upon the following:

- Metal fuel can be loaded with only about 36¢ of excess reactivity (minimize TOP initiators).
- The worth of the reactivity feedbacks reported in the PSID are known within the stated uncertainties.
- The halving time of SAFR pumps never drops below the 6-second reference value during the lifetime of the pump.
- The worth of the reactivities can change during a fuel cycle by only a limited amount.
- Properties of the metal fuel are as assumed in the PSID.

A.3 SAFETY ISSUES

Several key safety questions arise:

(1) Can the RI and ANL analyses for SAFR be replicated?

Independent calculations lead to similar predictions regarding the SAFR response to postulated unscrammed events. This is encouraging, as the two independent calculations are indicating almost identical behavior, particularly when minor modeling differences are taken into account.

One weakness is that both codes are using the same reactivity feedback values, even though there are good reasons to believe that these reactivity feedback numbers are at least approximately correct.

(2) Does the staff agree with the RI and ANL interpretation of the analyses?

ANL has made a major effort to bound the uncertainties in the reactivity feedback values. However, ANL has not considered uncertainties in pump coastdown and other key system values. Thus, the staff notes the uncertainties may be larger than RI and ANL assume, particularly for the loss-of-flow (LOF) events. Because of this, the staff is cautious about giving credit (e.g., in the PRA) for "inherent shutdown," particularly for unscrammed LOF events.

(3) Have key events been overlooked?

The capability of the SAFR design to respond to various unscrammed challenges is a big advantage. However, the existence of a large positive void worth is a clear vulnerability. Events must be considered to see whether there are any improbable events that could trigger power excursions. For the SAFR, the possibility of pump seizures is a concern; these could result in rapid reductions in sodium flow to the reactor. Although RI and ANL have recently worked to address the questions of pump seizure, some questions remain in this area.

A.4 CONCLUSIONS

Independent analysis shows that the RI and ANL calculations presented are reasonable. Because of concerns regarding uncertainties, the staff is cautious regarding the reliability and predictability of the SAFR passive shutdown, particularly for the loss-of-flow events. The staff is convinced of the importance of the safety tests, and is not convinced of the RI arguments that SAFR does not need a safety-grade scram system (see Section 4.6.5 of this SER).

Because of the positive void worth, unscrammed events are important for licensing the SAFR, particularly for containment and emergency planning. As a result, the staff believes that such plausible initiators as pump seizure must be considered under unscrammed circumstances.

A.5 REFERENCES

- A.1 DOE, "SAFR -- Sodium Advanced Fast Reactor," Preliminary Safety Information Document, AI-DOE-13527, Initial Issue - October 1985, with amendments. (DOE Applied Technology.)
- A.2 NRC, G.J. Van Tuyle, G.C. Slovick, et al., "Summary of Advanced LMR Evaluations -- PRISM and SAFR," NUREG/CR-5364, Brookhaven National Laboratory (BNL-NUREG-52197), October 1989.
- A.3 NRC, "Key Licensing Issues Associated With DOE Sponsored Advanced Reactor Designs," SECY 88-203, July 15, 1988.

Appendix A

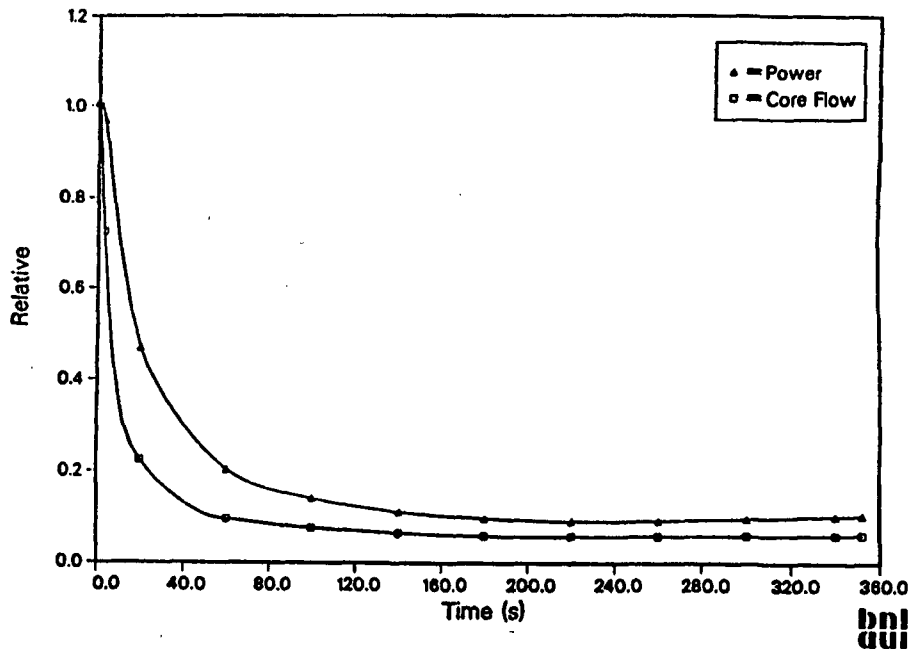


Figure A.1 SUPERSYSTEM code prediction for SAFR ULOF: Power and core flow.

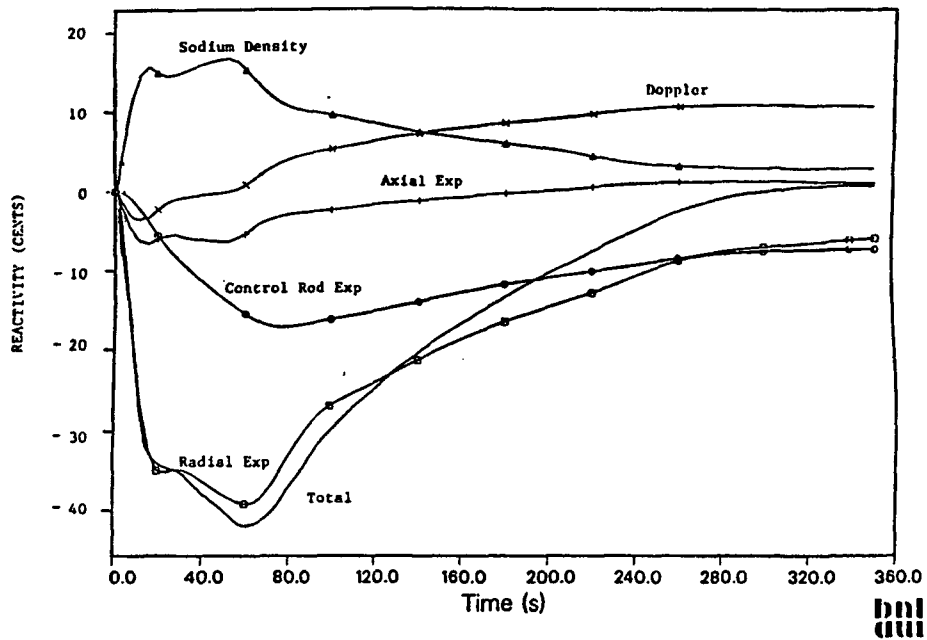


Figure A.2 SUPERSYSTEM code prediction for SAFR ULOF: Reactivity feedback.

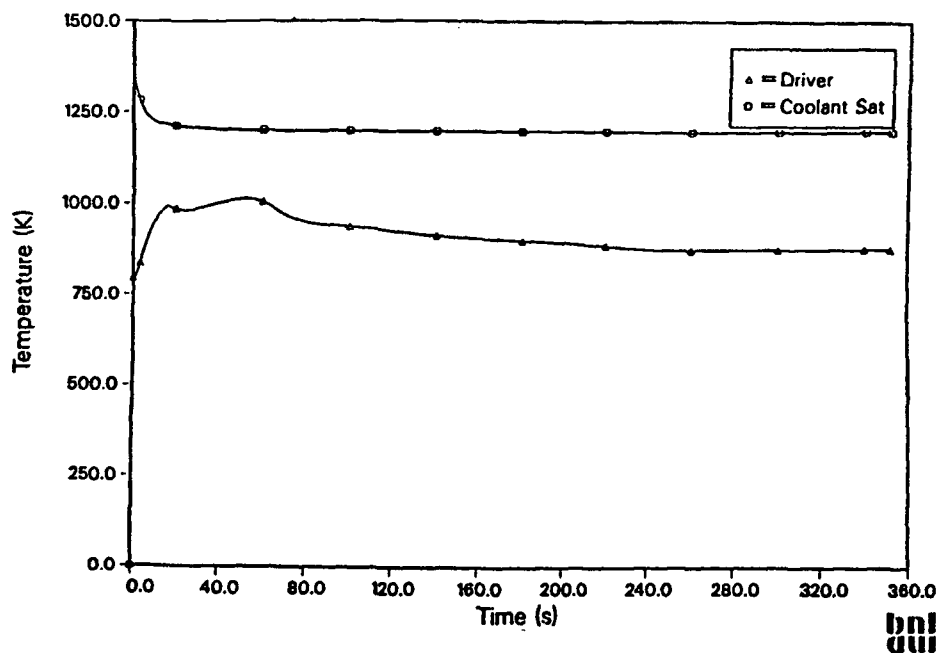


Figure A.3 SUPERSYSTEM code prediction for SAFR ULOF: Saturation and peak coolant temperature.

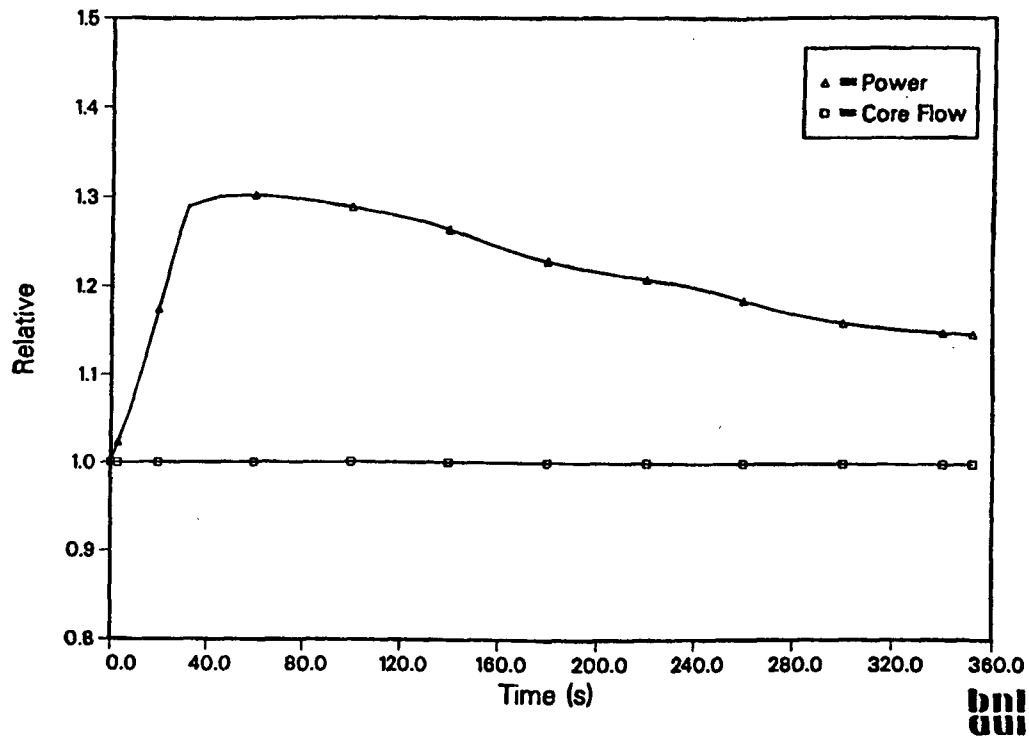


Figure A.4 SUPERSYSTEM code prediction for SAFR 20c UTOP:
Power and core flow.

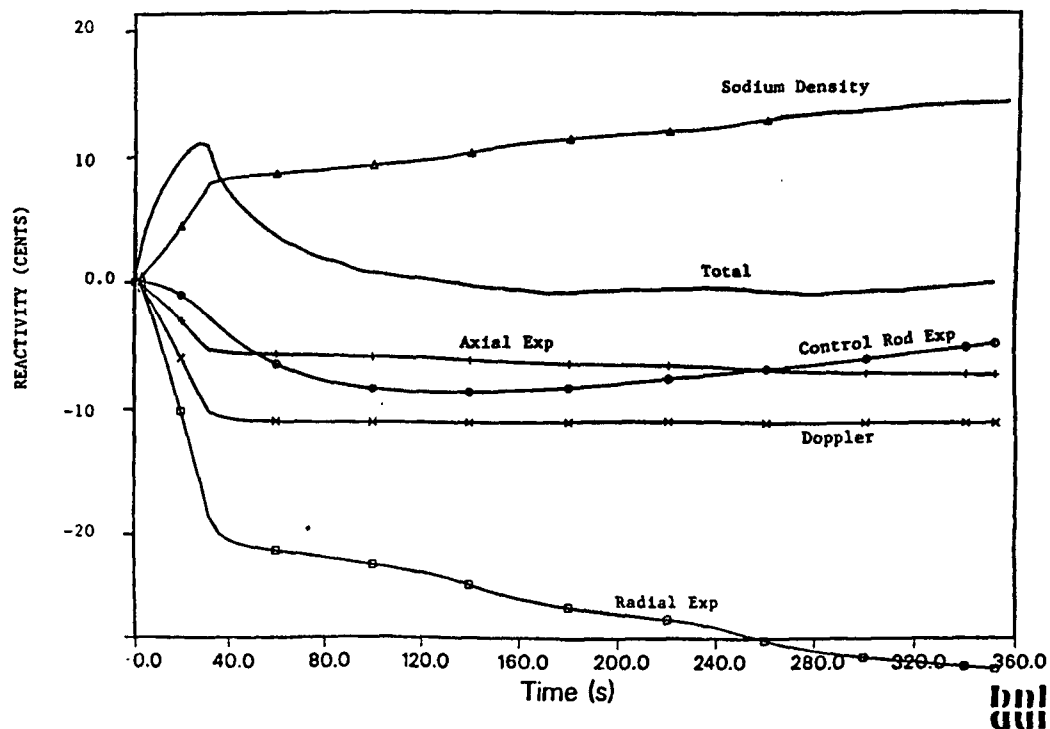


Figure A.5 SUPERSYSTEM code prediction for SAFR 20c UTOP:
Reactivity feedback.

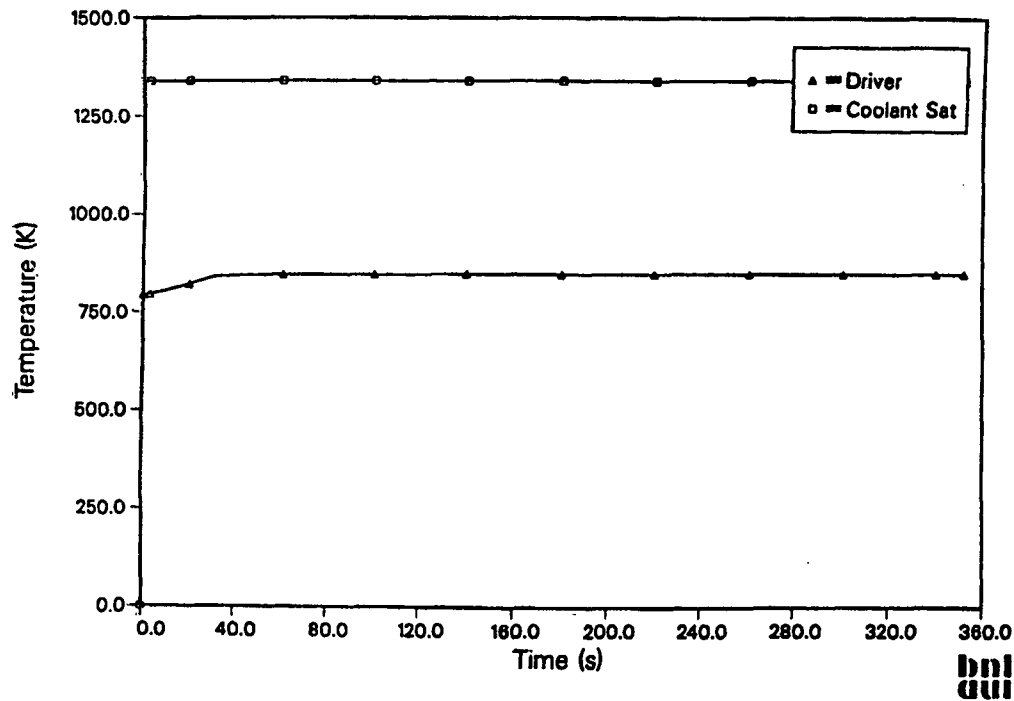


Figure A.6 SUPERSYSTEM code prediction for SAFR 20c UTOP: Saturation and peak coolant temperature.

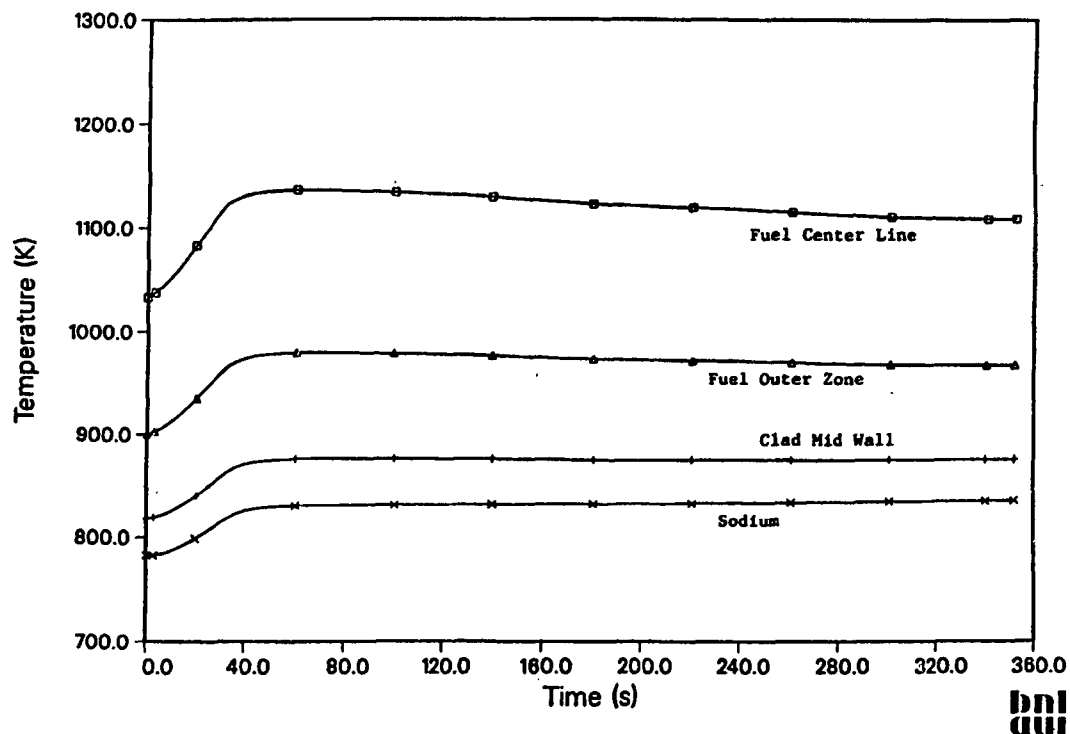


Figure A.7 SUPERSYSTEM code prediction for SAFR 20c UTOP: Peak fuel temperature distribution.

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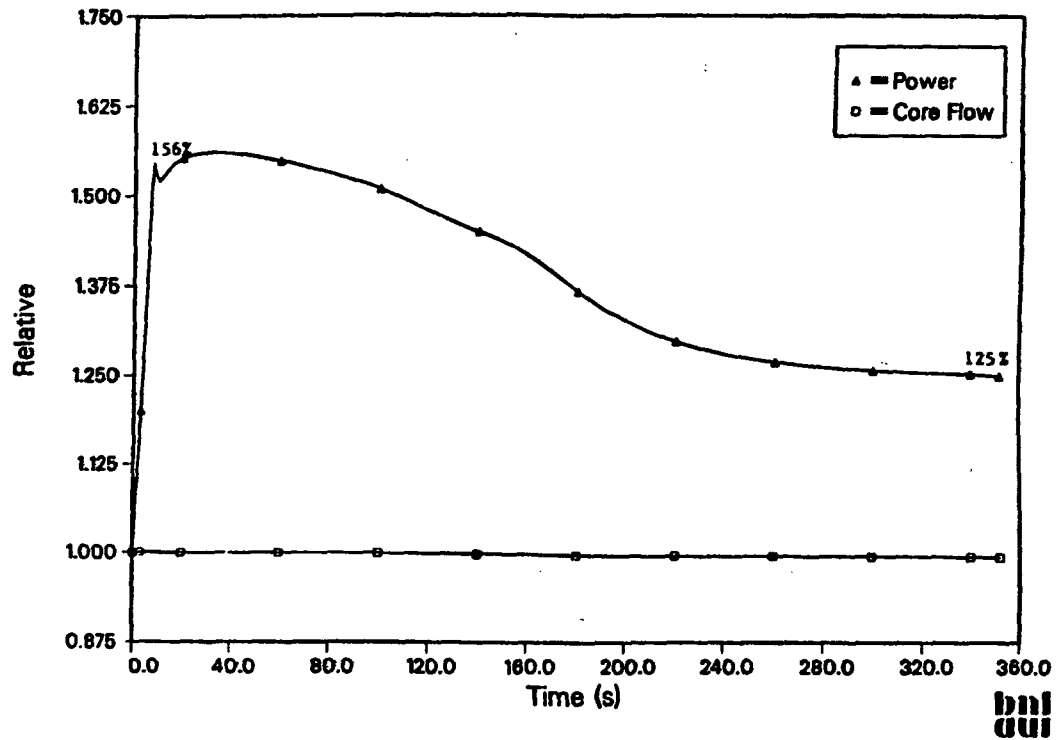


Figure A.8 SUPERSYSTEM code prediction for SAFR 36c UTOP:
Power and core flow.

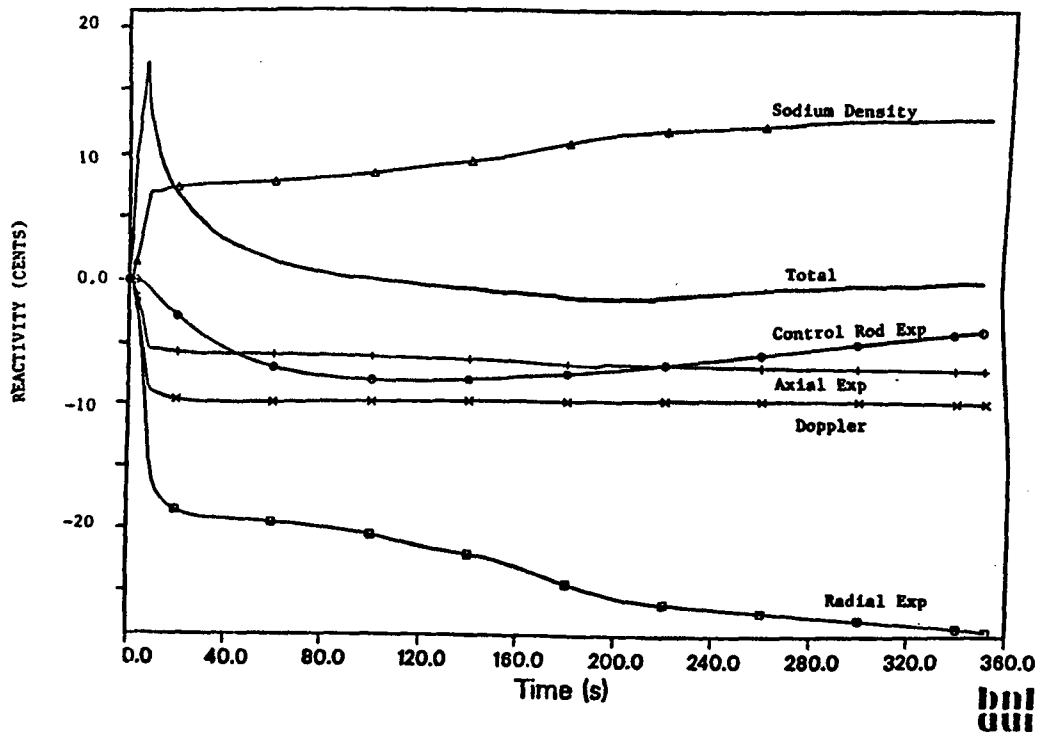


Figure A.9 SUPERSYSTEM code prediction for SAFR 36c UTOP:
Reactivity feedback.

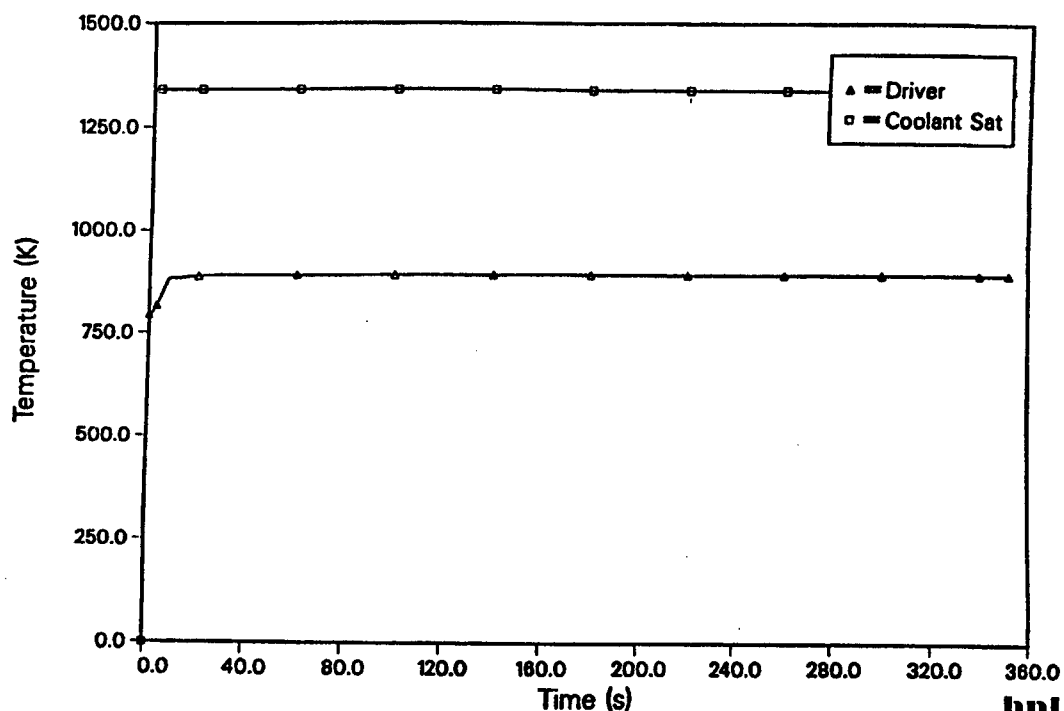


Figure A.10 SUPERSYSTEM code prediction for SAFR 36c UTOP: Saturation and peak coolant temperature.

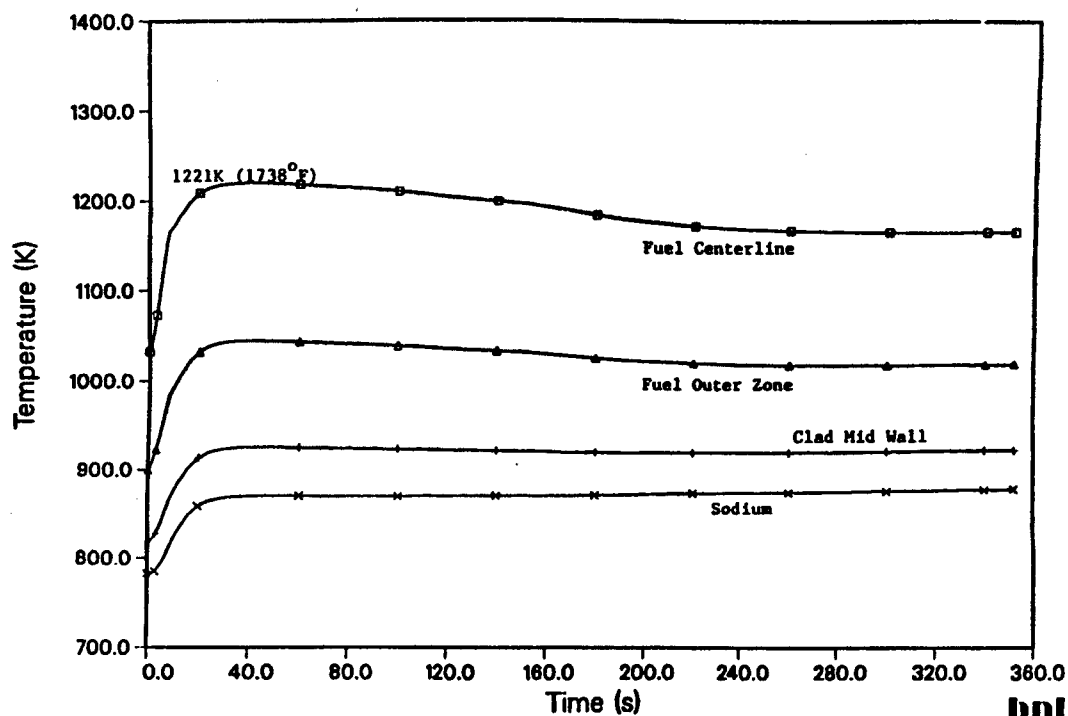


Figure A.11 SUPERSYSTEM code prediction for SAFR 36c UTOP: Peak fuel temperature distribution.

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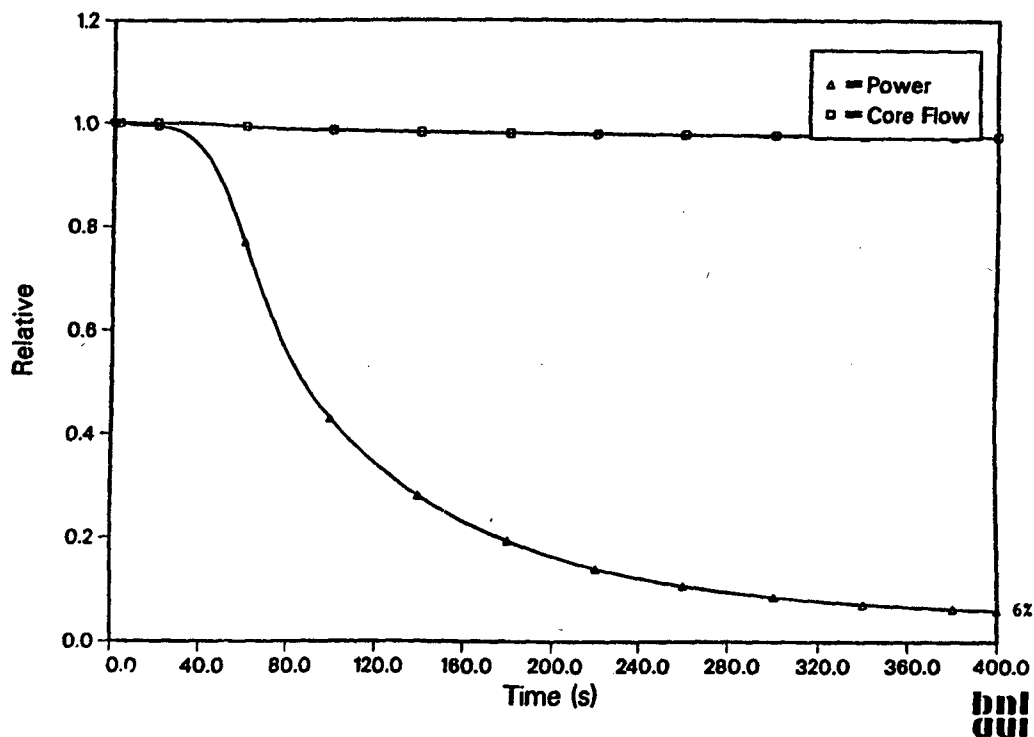


Figure A.12 SUPERSYSTEM code prediction for SAFR ULOHS: Power and core flow.

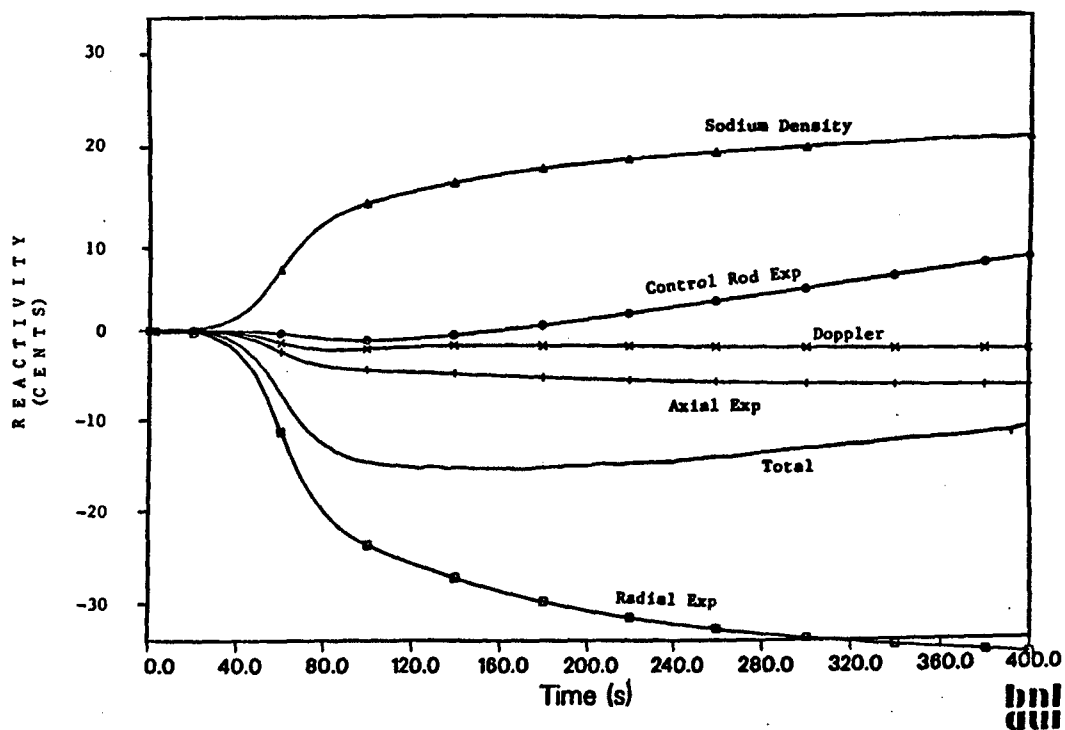


Figure A.13 SUPERSYSTEM code prediction for SAFR ULOHS: Reactivity feedback.

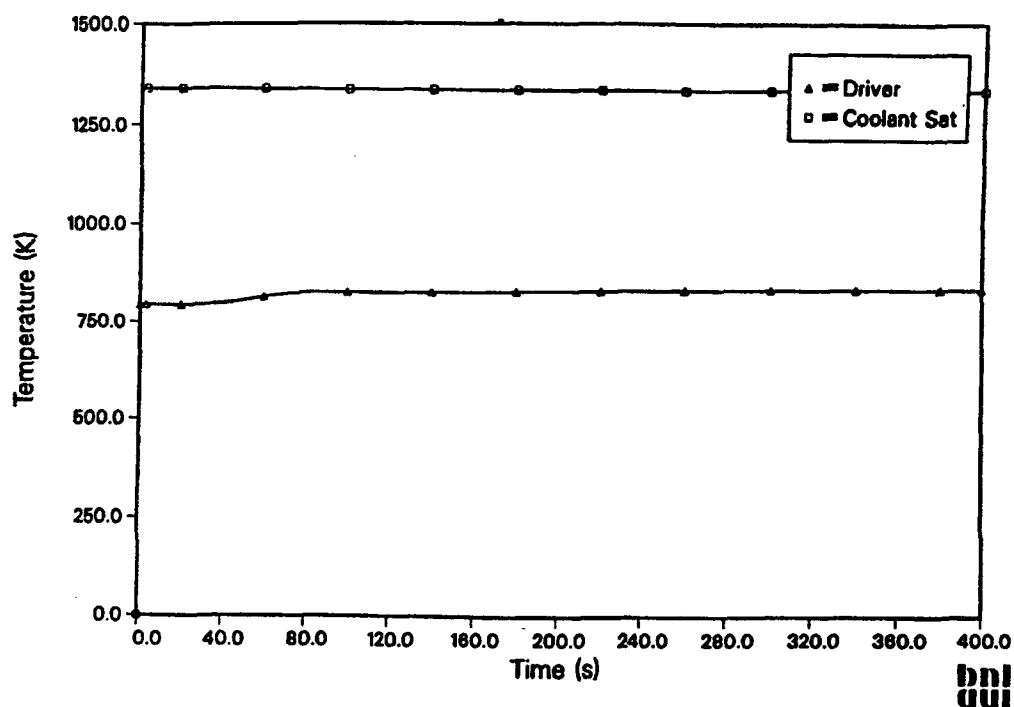


Figure A.14 SUPERSYSTEM code prediction for SAFR ULOHS: Saturation and peak coolant temperature.

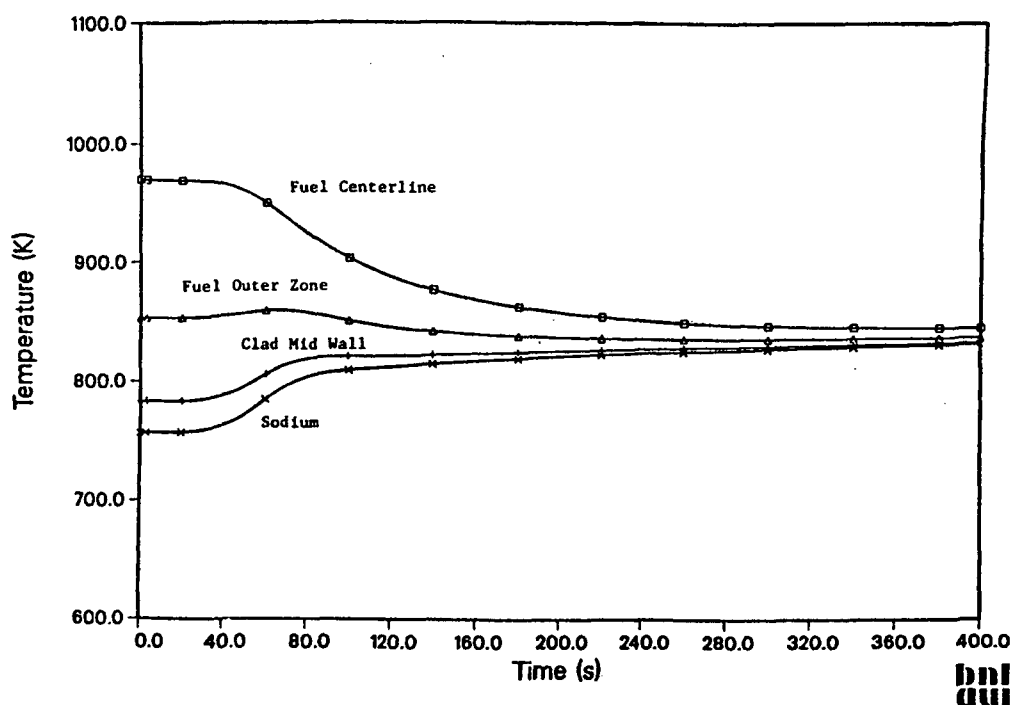


Figure A.15 SUPERSYSTEM code prediction for SAFR ULOHS: Peak fuel temperature distribution.

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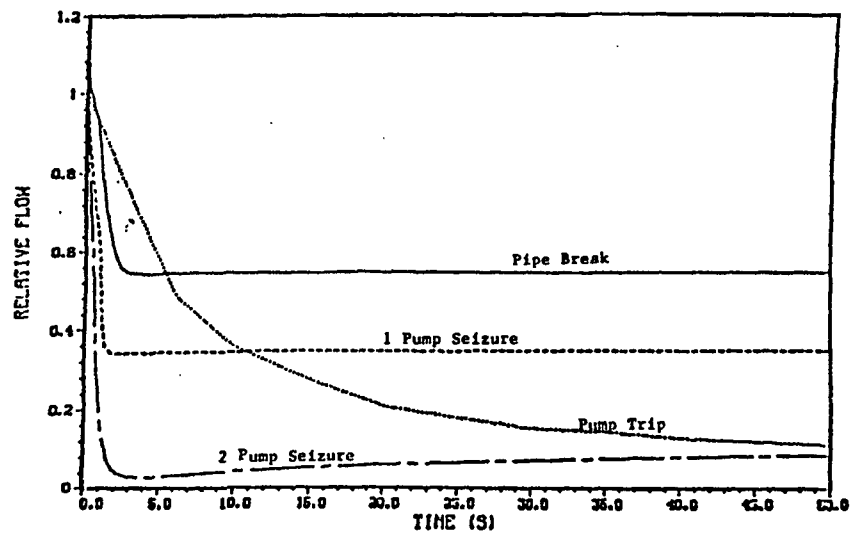


Figure A.16 MINET prediction of SAFR core flow rate for various LOF cases.

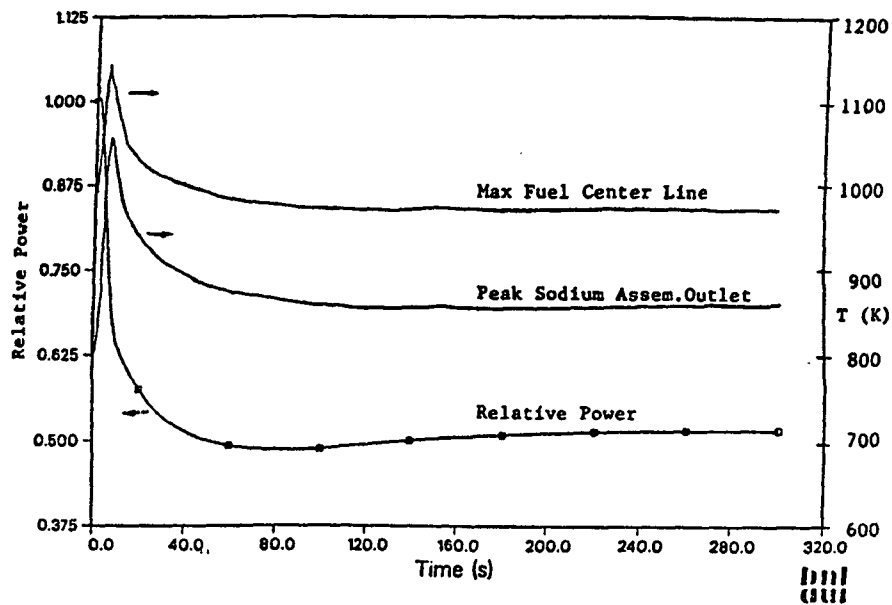


Figure A.17 SUPERSYSTEM code prediction for SAFR pump seizure: Power and key temperatures.

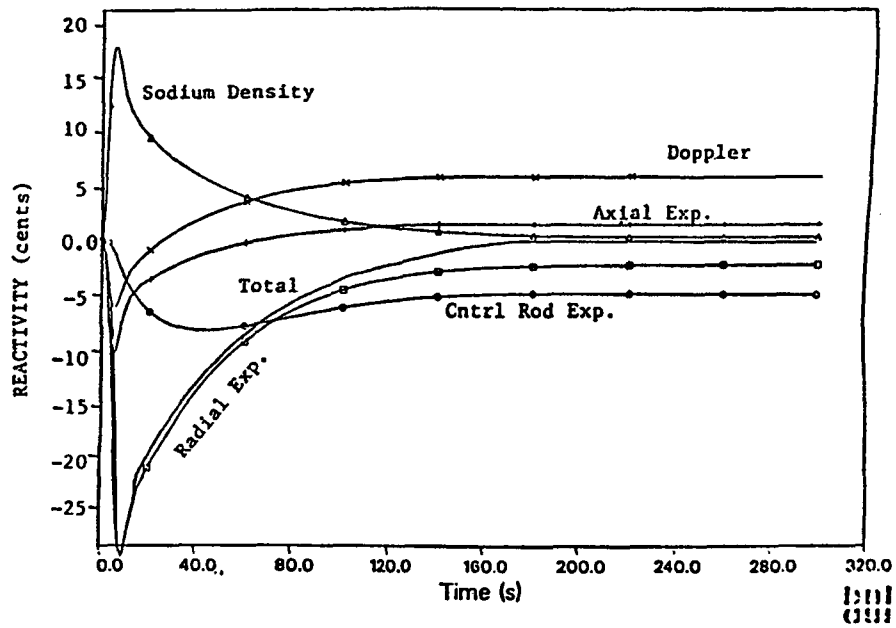


Figure A.18 SUPERSYSTEM code prediction for SAFR pump seizure: Reactivity feedback.

Appendix A

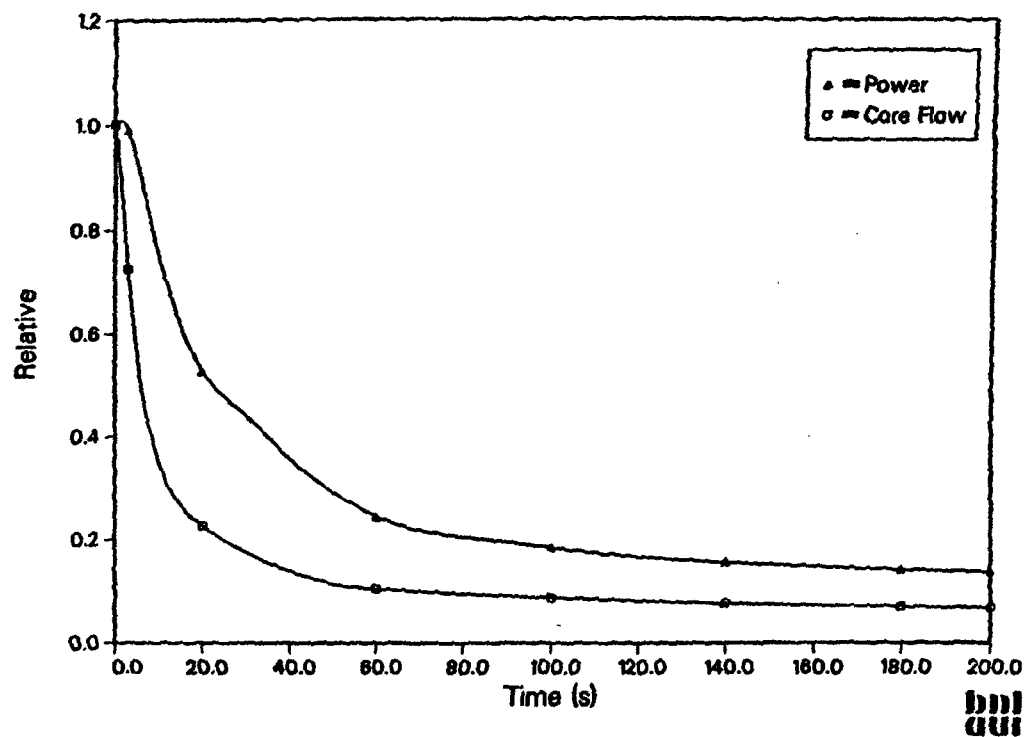


Figure A.19 SUPERSYSTEM code prediction for SAFR UTOP and LOF : Power and core flow.

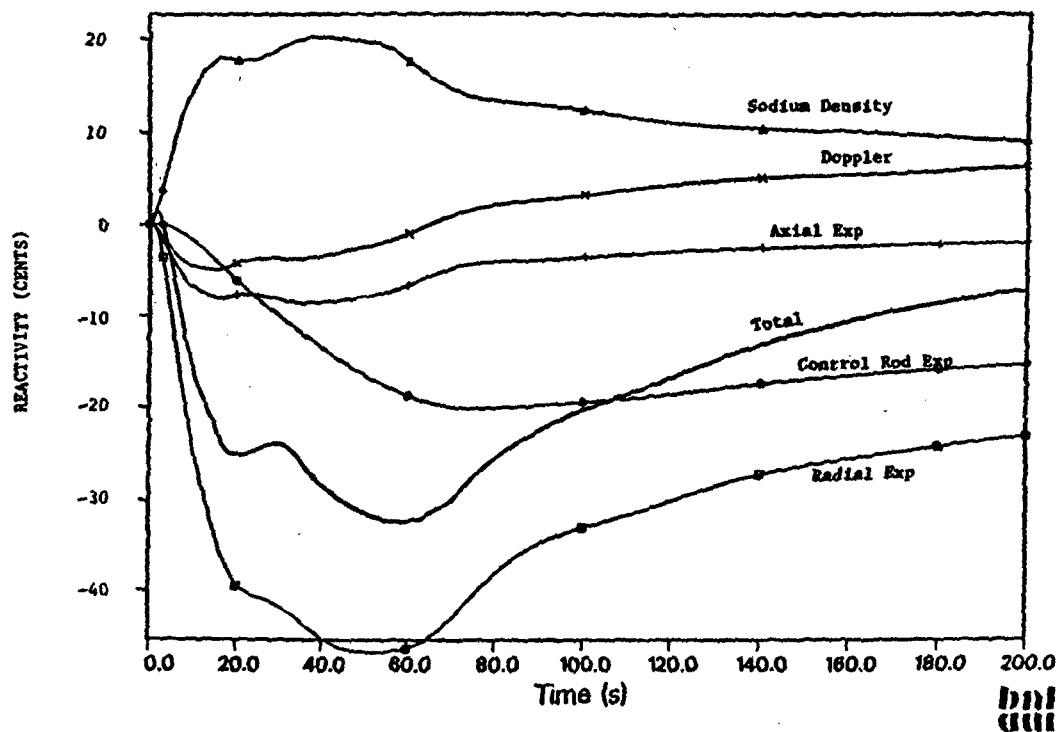


Figure A.20 SUPERSYSTEM code prediction for SAFR UTOP and LOF: Reactivity feedback.

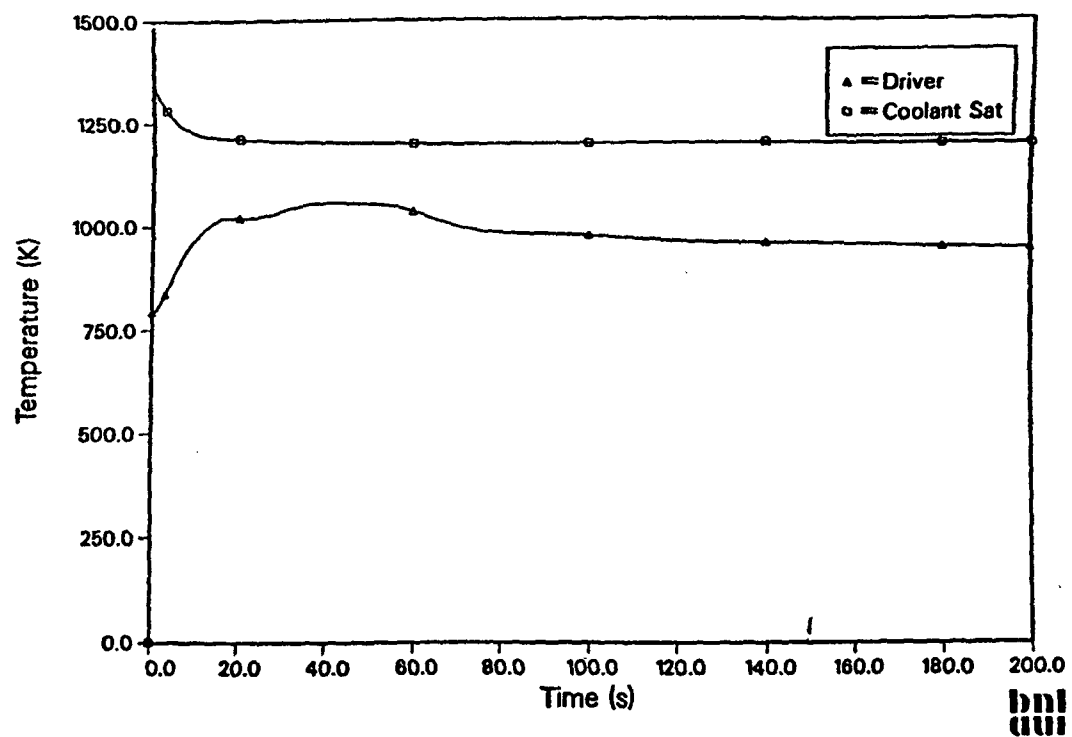


Figure A.21 SUPERSYSTEM code prediction for SAFR UTOP and LOF: Saturation and peak coolant temperature.

APPENDIX B

APPLICATION OF NUREG-0933 TO THE SAFR DESIGN

The designer presented a SAFR-specific response to the generic safety issues described in NUREG-0933, "A Prioritization of Generic Safety Issues" (Ref. B.1). TMI Action Plan items, Series N generic issues, and the human factors issues were also addressed. The SAFR designer reviewed NRC concerns to determine impact on the SAFR design and to see how the SAFR concept either resolved or addressed the concern. A detailed review of this assessment will be required at the next review stage.

B.1 TMI Action Plan Items

The original list of Three Mile Island Unit 2 (TMI-2) action items was published in NUREG-0660 (Ref. B.2) to provide a comprehensive and integrated plan for the actions judged necessary to correct or improve the operation of nuclear facilities based on the experience from the accident at TMI-2. The generic safety issues identified in NUREG-0660 have been prioritized in NUREG-0933 (Ref. B.1). The SAFR assessment is based on Supplement 3 to NUREG-0933, first published in July 1985.

B.2 Task Action Plan Items, Series A, B, C, and D

The task action plan items are documented in NUREG-0371 (Ref. B.3) and NUREG-0471 (Ref. B.4). The unresolved safety issues documented in NUREG-0606 (Ref. B.5) are also addressed for the SAFR design. The staff has identified more than 130 generic safety issues, as documented in NUREG-0410 (Ref. B.6). These issues are grouped into four categories.

Category A: Those proposed generic technical activities judged by the staff to warrant priority attention in terms of manpower or funds or both to attain early resolution. These matters include the resolution of issues that could (1) provide a significant increase in assurance of the health and safety of the public or (2) have a significant impact on the reactor licensing process.

Category B: Those proposed generic technical activities judged by the staff to be important in ensuring the continued health and safety of the public but for which early resolution is not required or for which the staff perceives a lesser safety, safeguard, or environmental significance than Category A matters.

Category C: Those proposed generic technical activities judged by the staff to have little direct or immediate safety, safeguards, or environmental significance but which could lead to improved staff understanding of particular technical issues or refinements in the licensing process.

Category D: Those proposed generic technical activities judged by the staff not to warrant the expenditure of manpower or funds because little or no importance to the safety, safeguards, or environmental aspects of nuclear reactors or to improving the licensing process can be attributed to the activity.

B.3 New Generic Issues, Series N

The NRC staff continually evaluates safety requirements used in its reviews against new information as it becomes available. Licensee event reports (NUREG-0572 (Ref. B.7) for example), Advisory Committee for Reactor Safeguards (ACRS) reports, and other activities identify new generic safety issues as reported in NUREG-0705 (Ref. B.8). These issues are generally referred to as Series N, as compared to the task action plan Series A through D.

Appendix B

B.4 Human Factors Issues

The investigation following the accident at TMI-2 identified the need to incorporate human factors considerations into the regulations and guidance for the design and operation of nuclear power plants (see NUREG-0660 (Ref. B.2)). The issues evaluated for the SAFR design include those documented in NUREG-0985 (Ref. B.9), as well as those in NUREG-0660.

B.5 References

- B.1 NRC, "A Prioritization of Generic Safety Issues," NUREG-0933 including Supplements 1- 6, August 1987.
- B.2 NRC, "NRC Action Plan Developed As a Result of the TMI-2 Accident," NUREG-0660, Volumes 1 and 2, May 1980.
- B.3 NRC, "Task Action Plans for Generic Activities," NUREG-0371, November 1978.
- B.4 NRC, "Generic Task Problem Descriptions: Category B, C, & D Tasks," NUREG-0471, June 1978.
- B.5 NRC, "Unresolved Safety Issues Summary. Data As of August 16, 1985," NUREG-0606, Volume 7 (Aqua Book), August 1985.
- B.6 NRC, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," NUREG-0410, January 1978.
- B.7 NRC, "Review of LERs (1976-1978)," NUREG-0572, September 1979.
- B.8 NRC, "Identification of New Unresolved Safety Issues Relating to Nuclear Power Plants. Special Report to Congress," NUREG-0705, March 1981.
- B.9 NRC, "U.S. Nuclear Regulatory Commission Human Factors Program Plan," NUREG-0985, Revision 1, September 1984.

APPENDIX C

PROBABILISTIC RISK ASSESSMENT

C.1 INTRODUCTION

Objectives and Limitations

The probabilistic risk assessment (PRA) for the Sodium Advanced Fast Reactor (SAFR) has been evaluated within the framework of criteria stated in the Commission's safety goal policy (Ref. C.1) and advanced reactor policy statement (Ref. C.2). The safety goals set the risk-based criteria; the advanced reactor policy anticipates that future designs will increase the margin of safety over current-generation LWRs. In order to determine whether the SAFR conceptual design can meet the above criteria, the following review objectives were established:

- Evaluate the PRA methodology and assumptions for accuracy and completeness.
- Identify the weaknesses and limitations of the PRA.
- Scrutinize the systems and features relied upon the most for protection against severe-accident vulnerabilities.

Because the SAFR design is at the conceptual stage, it lacks design detail and operational experience. A limited amount of experimental data from research performed at the Fast Flux Test Facility (FFTF), Experimental Breeder Reactor II (EBR-II), and the TREAT facility were used to partially validate computer codes that model how the SAFR plant would respond to transients. The codes themselves, however, are continuing to develop, and code limitations coupled with a lack of data have meant that reviewers need to lean heavily on engineering judgment for assessing the probability of risk. Although designers claim to have made conservative judgments, those judgments can only be validated by sufficient operating experience with prototype and commercial reactors.

The Brookhaven National Laboratory (BNL), under contract to the NRC, has reviewed the SAFR PRA and has provided substantial input to this evaluation.

Excluded Areas

The SAFR PRA is a Level 3 conceptual PRA which includes the systems, the containment, and a consequence analysis. The assumed location for population distribution is intended to bound 75 percent of the licensed light-water-reactor (LWR) sites. The PRA does not consider

- sabotage
- startup accidents and accidents at power levels other than full power
- multiple power pak interactions
- accidents related to radioactive sources outside the reactor vessel
- external events other than (limited) seismic events
- normal releases of plant effluents

The areas excluded here need to be considered when more design information becomes available.

C.2 METHODOLOGY AND DATA

C.2.1 Overview

The SAFR PRA employed standard LWR matrix formalities to estimate the risk from internal and seismically initiated events. LWR experience and the Clinch River PRA were used to identify and estimate frequencies of initiating events and probabilities of component failures. Five generic internal initiating events and eleven seismic initiating events were used to represent and bound the event spectrum. These initiators pass through three types of event trees that model the plant, core, and vessel response. The CRAC-2 code was used to perform the final dose calculations. All branch probabilities were termed "mean best estimate." Each had an arbitrarily chosen log normal distribution (error factor of 10) to account for uncertainty. The "discrete probabilistic arithmetic"

method and "method of moments" were used to propagate uncertainty, although both methods contain pitfalls as noted by BNL in its review. Quantification of the passive feedbacks and phenomenological aspects of the PRA relied upon engineering judgment supported by limited FFTF and TREAT experimental data.

Because of the conceptual nature of the design, detailed fault trees were not constructed. Failure-state diagrams were used to estimate reliability of the shutdown system and shutdown heat removal system. Common-cause failures were calculated using the "statistical correlation method," a method that relies strongly on engineering judgment. Failure rates and repair rates were also based on engineering judgment.

C.2.2 Initiating Events

All internal initiating events were consolidated into the following five generic LMR initiators:

- loss of flow (LOF) at 0.2 per year
- reactivity insertion at 0.1 per year
- loss of normal heat removal at 0.18 per year
- all remaining demands on reactor shutdown system at 1.0 per year
- combined transient overpower (TOP) and LOF at 2.5×10^{-3} per year

These generic initiators attempt to bound all potential internal initiating events. For example, the contribution of loss-of-offsite-power events (frequency of 0.1/yr) is included in the LOF initiator. The resulting sequences are therefore burdened with a spectrum of initiating events, some of which are not obvious. Two very important initiators were not explicitly represented, and may not have been adequately bounded by the LOF and TOP/LOF initiators:

- primary pump seizure
- withdrawal of a large number of control rods

Both of these initiators impact the systems and features needed to mitigate the transient, and therefore require explicit analytic consideration. Implicitly, they were considered to be of low probability, but assurance that these probabilities are indeed low can only be confirmed with confidence after in-depth fault trees have been developed.

Steam generator tube rupture is an event sufficiently unique that it should be addressed explicitly in the PRA. The SAFR design's superheated steam cycle places a 2700-psi differential pressure across the steam generator tubes, thus leading to a potential steam blowdown through the intermediate sodium loop and in-vessel heat exchanger. The risk from such an event has not been assessed in the PRA.

Considering LWR experience, initiating frequencies for the initiators "loss of normal heat removal" and "all remaining demands on the reactor shutdown system" are deemed optimistic. On the basis of LWR experience, the staff believes an order of magnitude increase in frequencies would have been more realistic.

Seismic initiators were treated separately. A modified Maine Yankee seismic hazard curve, (intended to include 75 percent of the sites east of the Rocky Mountains) was used to generate the 11 seismic initiators. Each initiator then represented a specific range of peak ground accelerations with a specific probability. The seismic initiating event frequency spectrum as a function of peak ground acceleration is given in Table 6.2 of Appendix C to the PSID (Ref. C.3).

C.2.3 Initiator Event Trees

Associated with each of the five internal initiators are simple event trees. Each tree contains branch probabilities for either reactor shutdown or decay heat removal, as applicable to the scenario considered. The PRA implicitly assumes that the probability of simultaneous loss of both reactor shutdown and decay heat removal systems is negligibly small. The initiator event tree sequences either terminate in safe shutdown (no damage) or initiate one of five types of accidents. The accident frequencies are strictly a function of the reliability of the reactor shutdown system or the decay heat removal system. The five accident types and their frequencies of occurrence are

• unprotected loss of flow (ULOF)	1.3×10^{-9} per year
• unprotected transient overpower (UTOP)	6.3×10^{-11} per year
• protected loss of heat sink (PLOHS)	3.6×10^{-8} per year
• unprotected loss of heat sink (ULOHS)	6.3×10^{-9} per year
• unprotected transient overpower combined with a loss of flow (UTOP/LOF)	1.6×10^{-11} per year

"Unprotected" means scram failure. Very low accident frequencies can be attributed to a low probability of scram failure and a high reliability of decay heat removal. These two systems are addressed below.

Seismic event trees (S1, S2,...,S11) have branch points for only offsite power, reactor shutdown, and decay heat removal. These sequences either terminate in safe shutdown, or initiate a LOF or a PLOHS accident.

C.3 SYSTEM ANALYSIS

C.3.1 Reactor Shutdown System

Each SAFR power pak is protected by two shutdown systems:

- Primary Control Shutdown System (PCSS). This is a non-safety-related reactivity control system that is used for reactor startup, shutdown, and operational control. The design is similar to the one used in the Clinch River Breeder Reactor (CRBR), although only one of the six primary rods for the SAFR design is required for shutdown.
- Self-Actuated Shutdown System (SASS). This is a unique non-safety-related shutdown system consisting of temperature-sensitive magnets latched to safety rods. During reactor overtemperature conditions, the magnet's holding force (Curie point) is reduced, which releases the rods independently into the core. Only one of the three rods is required for shutdown.

Both the PCSS and SASS interface with the automatic plant trip system (APTS). The APTS monitors primary plant parameters and trips primary and secondary rods if the parameters are exceeded. The APTS consists of four independent divisions utilizing a two-out-of-four trip logic. Although stated to be "non-safety-grade," the APTS logic is estimated to have a low failure rate of 7.9×10^{-6} per demand for all internal events. Potential primary control rod and scram breaker failures raise the overall APTS unavailability to 8.1×10^{-6} per demand.

Should the APTS fail to scram the reactor during a transient, the subsequent rise in reactor temperature is postulated to release the secondary (SASS) rods. This feature lowers the overall scram system unavailability for internal events to 6.3×10^{-9} per demand. The unique SASS feature introduces large quantification uncertainties in the PRA. For the LOF initiators, reduced sodium flow through the core would result in reduced sodium flow to the magnetic latches, delaying the SASS response time. Such behavior is not reflected in the PRA. Other potential failure modes include sodium flow blockages and buildup of metallic particles on the SASS magnets. These modes are difficult to

evaluate probabilistically, but are nevertheless needed to support the system reliability estimates. In addition, human errors and support system failures are essentially unmodeled. It is also not clear how the SAFR plant would, or could, achieve such high shutdown system reliability without having to meet safety-related criteria at least equivalent to LWRs. (Passive reactivity feedback is discussed in Section C.4 below.)

C.3.2 Primary Pump Coastdown System

To keep the core from voiding after the primary pumps trip, a controlled primary pump coastdown is required. The SAFR plant's two primary pumps are mechanical and should have sufficient inertia for coastdown, although failure to scram on loss of primary flow [no coastdown] could lead to severe consequences, that is, core disassembly. Assurance must be maintained that the primary pumps will not trip before neutronic shutdown occurs. Failure of the pumps to coast down could in itself increase the internal LOF risk by two orders of magnitude. The PSID did not focus on this constraint.

Coastdown failure of the primary pumps does not appear explicitly in the PRA. Because of its estimated low failure probability (1.9×10^{-9} per coastdown), failures are implicitly assumed to be bounded by passive feedback failures (10^{-3} per demand). The primary pumps are considered to be "non-safety-related," although they are needed to allow the passive feedbacks (which are safety related) to perform properly.

The SAFR PRA does not specifically address pump seizures because Rockwell International (RI) believes this to be a very unlikely event while the pumps are operating at rated speed, although pump seizure frequency has been estimated at 10^{-2} to 10^{-3} per year. Should the scram system not respond to a pump seizure, a second pump trip would result in an energetic excursion, even with successful coastdown of one pump. Given the uncertainty of the SASS response under these conditions, certain defense-in-depth measures may be justified, for example, installation of a third primary pump. Additional data and fault-tree analysis of the primary pump system are needed to explore this issue further.

C.3.3 Shutdown Heat Removal System

The shutdown heat removal system (SHRS) consists of three paths by which decay heat can be removed:

- through the intermediate heat transport system (IHTS) steam generator and then by steam flow through the turbine or turbine bypass and main condenser
- through the direct reactor auxiliary cooling system (DRACS) unavailability (1.6×10^{-3} /demand)
- through the containment vessel and then through natural circulation of air through the safety-related reactor vessel air cooling system (RACS) unavailability (2.2×10^{-6} /demand).

In general, high SHRS availability is attributed to

- diversity and redundancy in decay heat removal
- passive response which is continuously monitored
- the long grace periods which allow time for repair
- high balance-of-plant availability (Decay heat can be removed with one of three feedwater pumps and one of two main condensers.)

Unlike conventional systems, RACS efficiency depends on the environmental conditions present at the site. Atmospheric conditions and the accumulation of foreign objects inside the system will impair RACS performance. Although, with some events, the RACS may prevent core damage, the

high temperatures associated with the cooldown (up to 1200°F) could weaken the vessel, the supporting structure, and other reactor components. The effects of high temperature on these components and systems will have to be considered throughout the design stage.

To prevent excessive temperatures during RACS operation, the DRACS is brought into operation. The DRACS failure probability is considered to be within the range of a well-maintained LWR auxiliary feedwater system, that is, 1.6×10^{-3} per demand. It is not obvious at this review stage how the DRACS will maintain such high reliability, considering that the system is not safety related, involves operator action, is susceptible to sodium freeze failure, and lacks operating experience. The DRACS is expected to operate once every 5 years.

C.4 PHENOMENOLOGICAL ANALYSIS

C.4.1 Plant Model

Associated with each of the five internal initiators is a corresponding core-vessel response tree. Although treated probabilistically, the response tree branches address phenomenological aspects of the accident sequence:

- passive shutdown features
- energetics significance
- energetics level
- internal vessel structural response
- early vessel failure
- core debris coolability in vessel
- subsequent vessel structure response

Sequences formed by the response tree branches exit the tree as 1 of 15 possible plant damage states. These damage states are the input to the containment model described later.

For external (seismic) events, two core-vessel response trees are associated with each g-level initiator. One tree couples to the seismically induced loss-of-flow (LOF) initiator, the other to the seismically induced protected-loss-of-heat-sink (PLOHS) initiator. The seismic trees are somewhat simplified, as some branch probabilities are set equal to unity. Only 8 of the 15 defined internal plant damage states are therefore accessible through the seismic core-vessel response trees.

The SAFR plant's early releases will be dominated by two (energetic) plant damage states defined as

- EVF.T, Early Lower Vessel Failure, Subsequent Thermal Breach of Vessel
- EHF.T, Early Head Failure, Subsequent Thermal Breach of Vessel.

The seismically initiated loss-of-flow (LOF) accident is the dominant contributor to both energetic states. For internal events, early releases are also dominated by LOF sequences with about equal contribution from the EVF.T and EHF.T plant damage states.

The SAFR plant's late releases will be dominated by two plant damage states defined as

- LHF.T, Late Head Failure, Subsequent Thermal Breach of Vessel
- LHF.ET, Late Head Failure, Subsequent Energetics and Melt-Through.

Both damage states are driven by the seismically induced, protected-loss-of-heat-sink (PLOHS) accident. For internal events, late releases are also dominated by the PLOHS accident, with the LHF.T as the dominant plant damage stage. The energy released during the rapidly developing transients (i.e., ULOF with failure of the passive feedback features to be discussed in the next section

Appendix C

of this appendix) contain large uncertainties. Energetics were determined using qualitative reasoning that extrapolates experience from oxide-core FFTF and CRBR analyses to the metallic-core SAFR plant analysis. No fuel-failure data for the LOF conditions exist for metal fuels.

A range of energies should be explored in future PRA activities. The SAFR PRA estimates that energetics will lead to a release of only 1 percent of the core; BNL believes that "tens of percent" cannot be ruled out. Uncertainties in the energetics ultimately impacts vessel behavior, radioactivity released, and consequence estimates. Mechanistic analyses have not been performed that could otherwise support the generic sequences in this portion of the PRA.

For the long term, protected-loss-of-decay-heat removal uncertainties are driven by the phenomenological behavior of the core and sodium as the temperature increases. The level of sodium at the time of core-melt, and the effectiveness of cooling under natural convective boiling, molten fuel relocation, and recriticality, all contribute to the uncertainties. It appears reasonable to the staff that long grace periods might evolve, that is, 30 hours to sodium boiling and 115 to 120 hours to core uncover, although more work is needed to provide confidence.

C.4.2 Passive Reactivity Feedback

As primary temperature increases, sodium density decreases, adding positive reactivity to the core. Passive feedback effects limit the rate and extent of increase in power by adding negative reactivity. Instantaneous or prompt negative feedback results from

- Doppler absorption
- fuel axial expansion

Delayed negative feedback results from

- control rod motion
- fuel subassembly bowing or dilation of the subassembly load pads
- core support grid plate expansion.

Under certain conditions, prompt negative reactivity feedback alone cannot overcome the positive sodium reactivity addition. A LOF without scram and primary pump coastdown on both pumps, for example, would result in voiding the core of sodium. Core voiding would soon be followed by core disassembly and release of radioactivity. To protect against such an event, the designer must ensure low probability of scram failure and a highly reliable coastdown system.

Delayed feedback is conditional and depends on control rod movement and structural integrity. Stuck control rods and structural failures resulting from seismic events can defeat or severely limit feedback effects. Passive feedback is, therefore, dependent on the initiating event and the ability to keep the control rods free in their guide tubes. For internal events, the probability of passive feedback failure has been estimated to be 10^{-3} per demand. For external events, the failure probability of passive feedback ranges from 10^{-3} per demand for earthquakes of less than 0.5g, to 0.29 per demand for earthquakes greater than 1.0g.

In the risk dominant sequences (0.6g to 0.9g earthquake), it appears that reduction in core damage frequencies as a result of the passive feedbacks cannot be justified. The seismically induced failure of the core support would likely result in common-cause failure of the scram system and the passive feedback features. The PRA sequences assume scram failure, but take one to two orders of magnitude credit for the passive feedbacks. These energetic sequences could therefore be underestimated by one or more orders of magnitude.

The ability of the passive feedbacks to shut down the reactor depends strongly on the scenario. For transients that are slow to develop, time will allow the temperature-dependent feedbacks to work.

For fast transients, the temperature response of the core is critical. The PRA does not reflect these differences, nor are data available on which to base such quantification. Although future testing on a prototype will help reduce these uncertainties, the failure probabilities are almost entirely subjective at this stage of development.

C.4.3 Containment Model

The 15 core-damage states (output from the core-vessel response tree) provide a reasonable range of states that were subsequently binned into 8 core-vessel response states. Each state has an associated containment response tree. The branch points on the tree determine the following phenomenological core and vessel behavior during an accident:

- early breach
- debris coolability
- source-term transport

The containment response trees transform the 15 core-damage input states into 20 release categories. Each release category defines the time of release and percentage of core inventory released (source term). Categories range from "no significant release" (R1) to "early energetic release" (R4A and R4B). The R4A and R4B release categories dominate the early fatality risk. Each category has a release frequency of 2.2×10^{-9} per year.

The R4A and R4B release categories are driven by EHF.T and EVF.T core-vessel response states. These states stem from the seismically induced LOF events. Both states dominate the early fatalities, having the largest early release which includes 100 percent of the noble gases and 1 percent of the core inventory at time $t=0$. Considering the catastrophic nature of the event, which includes early head or lower vessel failure, the 1 percent is believed to be optimistic at this review stage. Although additional radioactivity is released to the environment later in the course of the accident, the releases are spread out in time (up to 150 hr) which reduces the fatalities substantially.

The latent fatalities are driven by the two release categories: R6A (frequency 1.3×10^{-6} /yr) and R8A (frequency 2.3×10^{-6} /yr). Both result from the LHF.T and LHF.ET plant damage states. Dominant contributors to these states are the seismically induced, protected, loss-of-heat-sink accidents (PLOHS). Both the R6A and R8A categories have long grace periods. Noble gases, for example, are not released until 115 hours after accident initiation. By invoking ad hoc evacuation procedures, the long delay in release time reduces the latent fatalities significantly.

The dominant release categories for internal events are R5 and R6A. Both are loss-of-heat-sink (LOHS) accidents that do not lead to early fatalities. Because of the very low probability of failure to scram, and the high reliability of the passive feedback features, the energetic sequences for internal events are many orders of magnitude less than those for the external contributors.

C.4.4 Site Model

The CRAC-2 code was used to perform the radiological transport and dose calculations. (An independent assessment of the CRAC-2 calculations was not performed as part of this review.) Standard strategies for unplanned evacuation were assumed in the site model. Code input included the 20 release categories, core fission product inventory, evacuation strategies, population distributions, and site meteorology. The population distribution was chosen so that it would bound 75 percent of the current LWR sites. Acute fatalities were found to occur only for release categories R4A and R4B.

Source terms were not determined mechanistically. They were based on an ANL study of oxide-fueled liquid-metal fast-breeder reactor (LMFBR) releases and then scaled to account for SAFR's metal fuel. Oxide fuels and metal fuels behave differently.

Appendix C

On increasing temperature, metal fuel will tend to:

- release noble gases and volatile material at lower temperatures
- involve an additional energy source because of the fuel's reactivity with oxygen
- release elemental forms of barium and strontium

In-vessel releases for metal fuels would, therefore, be expected to be higher, earlier in the sequence, although little information is available on how metal fuels actually behave during accident progression. Other potential contributors to an increased source term include potentially higher energetics than that assumed in the PRA, and corium-concrete interactions which would be highly exothermic for metal fuels.

Three reasons were cited for the SAFR design's low dose estimates:

- low inventory release fractions (The maximum early release category allows just 1 percent of the core inventory to escape.)
- long time periods between accident initiation and actual release of inventory (The highest frequency sequences begin to release radioactivity 115 hours or more after accident initiation.)
- releases spread out over long time periods (The sodium tends to retain most fission products.)

Except for the early energetic releases, a gradual heatup and boiloff of sodium is expected to occur. There may be late energetics, but these accidents result in long grace periods which, in general, would allow time for people to be evacuated.

C.5 RESULTS

According to the designers' PRA, the risk estimates for a single SAFR power pak were determined to be orders of magnitude less than the NRC safety goal quantitative health objectives. Societal risk, or the probability of latent cancer fatality per person per single year of operation (out to 10 miles) was estimated to be 1.2×10^{-9} per year, compared with the NRC safety goal figure of less than 1.9×10^{-6} per year. The individual risk, or probability of a prompt fatality per person per single year of operation (out to 1 mile), was reported to be less than 10^{-10} per year, compared with the safety goal figure of less than 5×10^{-7} per year. The large release frequency (based on the probability of exceeding one or more early fatalities) was also reported to be 4.4×10^{-9} per year, and total core-melt frequency, which includes both internal and external events, was reported to be 4.4×10^{-6} per year.

According to the PRA, seismic events in the 0.7g to 1.0g range dominate the SAFR design's acute risk, while events in the 0.4g to 0.8-g range dominate the latent risk. For such seismic events, the dominant risk scenarios are as follows:

- Latent Risk: Scram is successful; however, common-cause structural failure of the decay heat removal systems results in a slow core-melt. This event, known as a seismically induced, protected-loss-of-heat-sink (PLOHS) accident, has an estimated frequency of 4.4×10^{-6} per year.
- Acute Risk: Failure to scram, loss of primary flow, and failure of the passive feedback features, caused by structural failures, lead to a high energetic excursion accident. These seismic-related LOF events have an estimated frequency of 4.4×10^{-9} per year. The large release stems predominantly from these scenarios.

The total core-melt frequency for internal events is 3.7×10^{-8} per year and the dominant accident sequences include:

- **Latent Risk:** Loss of all three independent decay heat removal paths (PLOHS), estimated accident frequency is 3.6×10^{-8} per year.
- **Acute Risk:** Loss of primary flow (LOF) coupled with failure to scram and failure of the passive reactivity feedback features, estimated accident frequency is 8.6×10^{-14} per year.

The total core-melt frequency for internal events contributes less than 1 percent to overall risk. According to the SAFR PRA results, the latent risk scenarios are dominated by loss of all independent decay heat removal paths (PLOHS). The staff questions the concentration of risk in one area of vulnerability and believes that more study will be required at a later stage to determine the acceptability of these results.

The advanced reactor policy statement anticipates that future designs will have increased margins of safety over current-generation LWRs. Several innovative design features should contribute to the SAFR design's safety capacity:

- smaller core size, which limits energetics, core inventory, and decay heat
- passive decay heat removal features, which provide high reliability and redundancy to decay heat removal
- passive scram system, which eliminates operator error and provides diversity and redundancy to the scram system
- fewer control rods needed for reactor shutdown, increasing shutdown system reliability
- passive reactivity feedback features that suppress the probability of an energetic excursion
- large sodium pool, which allows time for mitigative and corrective action
- low (atmospheric) primary pressure reducing the probability of a large loss-of-coolant accident (LOCA)
- small changes in control rod worth during fuel burnup

However, the absolute risk benefit of these innovative design features over current-generation reactor systems is somewhat reduced because safety-related systems normally found to be important for LWRs (i.e., emergency feedwater, safety-related diesel generators, conventional containment building), have been eliminated in preference to these more simple, passive, and inherently safe features. Other systems, such as the reactor protection system, pump coastdown system, and control room systems, have been designated as not safety related, although very high reliability is expected of them. Assurance that non-safety-related equipment would continue to be highly reliable over the plant's lifetime has not been adequately demonstrated. In addition, the unique features found on SAFR, such as SASS, DRACS, RACS, and the passive feedbacks, have little data to support their reliability estimates. In particular, the risk contribution from the unscrammed LOF class of events appears to be significantly underestimated. The scram failure probability of 6.9×10^{-9} is optimistic when compared to LWRs which are above 10^{-5} per demand, and the margins for the passive shutdown are too narrow to justify 10^{-3} failure probability. Thus, the probability of a LOF leading to a large release of radioactivity is likely to be greater than 10^{-8} per year rather than 10^{-13} per year as reported in the SAFR PRA.

Appendix C

Three initiating events were not considered explicitly in the SAFR PRA, and need further probabilistic analysis:

- steam generator tube rupture
- primary pump seizure
- withdrawal of a large number of control rods

In addition, on the basis of LWR experience, the following estimates appear optimistic at this stage of development:

- station blackout frequency (Not specified in the PRA but could be high because of lack of safety-related diesels.)
- loss of normal heat removal function at 0.18 per year (LWRs are at least an order of magnitude higher.)
- demands on the reactor protection system at 1.0 per year (Some LWRs have frequencies above 10^{-5} per year.)
- scram failure probability at 6.93×10^{-9} per demand (LWRs are above 10^{-5} per year.)

Uncertainties are not well understood at this conceptual design stage. Major sources of uncertainties stem from

- lack of design detail on which to estimate system reliabilities (Additional information is needed on support systems, common-cause failures, human factors, and repair rates.)
- lack of test data and lack of experience with regard to SAFR's unique features:
 - primary pump coastdown capability
 - passive scram system
 - natural-convection decay heat removal systems
 - passive feedback features
- lack of human reliability information on operator recovery action and maintenance activities
- nonmechanistic source-term estimates which were based on generic accident sequences and oxide fuels which were extrapolated to metal fuel
- lack of data needed to support the engineering judgments used to address accident sequence phenomenology, for example, energetics and subsequent behavior
- passive feedback features which appear optimistic following seismic events and LOF accidents
- unknown surveillance requirements needed to ensure the operability of the shutdown, decay heat removal, and coastdown systems

Issues that have not been addressed, including the impact of fire and multiple power pak interactions, remain to be considered within the PRA.

C.6 CONCLUSIONS

The SAFR PRA has provided a preliminary overview of the plant's vulnerabilities and has gone a long way in attempting to quantify them. Although it appears that seismically induced LOF sequences dominate the early fatalities, and seismically induced PLOHS sequences dominate the latent fatalities, the staff believes that further study is required to support these conclusions. Internal events contribute less than 1 percent to overall risk.

Redundancy, diversity, and passive safety features designed into the SAFR resulted in very low PRA risk estimates and appear to strengthen safety, as anticipated by the advanced reactor policy statement. These estimates meet the quantitative health objectives stated in the Commission's safety goal policy, although it is not clear at this time if the large release criteria (which is currently undefined) will be met. There are caveats, however, when using these risk estimates as a means of judging the SAFR's safety capacity. Any decisions based on these estimates must take into consideration the following facts:

- The SAFR PRA lacks the detail and data required to substantiate the optimistic system reliability estimates. Major weaknesses include essentially unmodeled common-cause failures, human errors, and support-system failures. Surveillance requirements are not known.
- External events other than seismic have not been quantified, and factored into the final risk estimates. Seismic analysis is limited to a hazard curve that is intended to represent 75 percent of the LWR sites. Fragilities relied heavily on engineering judgment supported by very limited test data.
- Interactions among front line systems, support systems, and other power paks were not assessed.
- Three initiating events were not explicitly addressed in the PRA:
 - steam generator tube rupture
 - pump seizure
 - withdrawal of a large number of control rods
- Two of the five generic initiators appear to have been underestimated.
- Passive feedback features are believed to be optimistic, especially for seismically initiated events. The risk significance of the energetic sequences are, therefore, believed to be underestimated.
- Source-term estimates may be low for some scenarios as a result of extrapolating the behavior of oxide fuel to metal fuel.
- The accident sequences were not analyzed mechanistically. Generic assumptions made in the PRA do not appear to properly represent expected response under specific conditions.
- Uncertainties have not been properly represented, nor are they well understood at this preliminary design stage.
- The role of the operator is not apparent from the PRA. Credit in the form of operator recovery had been taken, although it is not known what actions would be performed or what operators would be available to perform such actions.
- In order to substantiate the very low risk estimates reported in the SAFR PRA, a greater effort will be needed to achieve completeness. Completeness is not apparent, as several important sequences have been identified which were not properly represented.

Appendix C

Large uncertainties in the front end of the PRA exacerbate the large uncertainties in the phenomenological treatment of the core response and consequence analysis. Deterministic engineering analyses, complemented with prototype testing on a full-scale reactor module, and operational experience will, therefore, be essential in the determination of the SAFR's safety capacity.

C.7 REFERENCES

- C.1 NRC, "Safety Goals for the Operation of Nuclear Power Plants," 51 Federal Register 28044, August 4, 1986.
- C.2 NRC, "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," 51 Federal Register 32138, August 8, 1985.
- C.3 DOE, "SAFR - Sodium Advanced Fast Reactor," Preliminary Safety Information Document, AI-DOE-13527, Initial Issue - October 1985, with amendments. (DOE Applied Technology.)

APPENDIX D

ADVISORY COMMITTEE FOR REACTOR SAFEGUARDS LETTER



UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

January 19, 1989

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: SAFETY EVALUATION REPORT FOR THE SODIUM ADVANCED FAST
REACTOR (SAFR) DESIGN

During the 345th meeting of the Advisory Committee on Reactor Safeguards, January 12-14, 1989, we completed our review of a draft of the subject safety evaluation report (SER). This subject was also considered during our 344th meeting on December 15-17, 1988. Our Subcommittee on Advanced Reactor Designs met on December 13, 1988 to discuss this matter. During these meetings, we had the benefit of discussions with representatives of the NRC staff and its consultants, with representatives of the Department of Energy (DOE) and its contractors, including representatives of Rockwell International, the lead design contractor. We also had the benefit of the documents referenced.

The SAFR conceptual design is a product of a DOE program to develop designs for possible future power reactor systems that would have enhanced safety characteristics. Other design projects in the program are the Modular High Temperature Gas Cooled Reactor (MHTGR) and the Power Reactor Inherently Safe Module (PRISM). The NRC staff has reviewed these designs in accordance with the Commission Policy on Advanced Nuclear Power Plants. These preapplication reviews are intended to provide NRC guidance on licensing issues at a relatively early stage of design development. The ACRS has previously commented to you in June 1987 on NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," in July 1988 on key licensing issues associated with the entire program, in October 1988 on the SER for the MHTGR, and in November 1988 on the SER for PRISM.

We understand that issuance of the SER will not constitute approval of the SAFR design. Further engineering development and documentation would be required to support a future application for design certification.

The SAFR design incorporates small modular reactors cooled by liquid sodium. The standard SAFR plant would consist of one or more "power paks." Each "power pak" would comprise four reactor modules that would

produce a total of 3600 Mwt (1400 MWe). Each reactor, along with its intermediate heat exchangers and pumps is immersed in a pool of sodium. A steel vessel containing this pool is surrounded by a secondary steel container and each module is installed within a concrete structure above grade. Secondary sodium coolant will flow from each reactor module to a pair of steam generators, located above grade along with the remainder of the balance of plant (BOP) equipment.

The SAFR modular design provides several desirable features for enhancing safety of a nuclear power plant:

- ° a passive system for emergency removal of decay power
- ° inherent mechanisms for negative feedback of reactivity
- ° two independent scram systems, one capable of self-actuation
- ° large thermal inertia in the pool of sodium coolant
- ° metal fuel, offering greater opportunity for on-site fuel reprocessing
- ° small component sizes, providing opportunities for factory fabrication
- ° opportunity for prototype testing of a single module
- ° separation of safety-related functions from BOP systems

SAFR, while similar to PRISM, has some important differences. Each SAFR reactor module is larger and would generate 900 Mwt compared with 425 Mwt for PRISM. SAFR primary sodium would run hotter than in PRISM with a nominal core exit temperature of 950°F compared with 875°F for PRISM. SAFR steam conditions are 850°F and 2700 psig, compared with 545°F and 990 psig for PRISM. SAFR has two reactivity control and scram systems while PRISM has one. SAFR's main coolant pumps are conventional centrifugal while PRISM's are electromagnetic.

The DOE has decided to discontinue its development of the SAFR design and concentrate liquid metal reactor (LMR) efforts in the PRISM design organization, but has requested that the NRC staff complete its review of both SAFR and PRISM. The NRC staff has expressed no opinion that there appears to be a net advantage in the PRISM design over that of SAFR, or vice versa.

On the basis of its review, the NRC staff has concluded that the SAFR design has the potential for a level of safety at least equivalent to

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current light water reactor (LWR) plants. We have no reason to disagree and believe that SAFR, like PRISM, could be licensed if continuing development work is pursued successfully.

A number of safety issues remain to be completely addressed. A continuing program of research and development will be necessary to support further design. Plans for extensive prototype testing should be included. In the following paragraphs, we comment on a number of specific safety issues which we believe should be considered by the staff in its final SER, and by DOE if it continues design and development of this concept.

Positive Sodium Void Coefficient

SAFR, like PRISM, will experience a large increase in reactivity in the event of significant boiling or other voiding of the sodium coolant. The designers' analyses cannot show that such voiding is impossible, but they have concluded that it is very improbable. Whether it is improbable enough and whether the consequences of such voiding can be tolerated is the major safety issue that must be resolved before these reactor designs could be licensed. The simultaneous and sudden loss of both main circulation pumps, without scram, in a reactor module might cause significant sodium boiling and a reactivity increase. If the positive voiding coefficient is to be accepted, such events must be shown to be of extremely low probability. We believe that additional design and safety analysis work is needed in this area.

Other Reactivity Coefficients

The satisfactory performance of the system in certain low probability transients is very dependent on the changes in core reactivity with variations in power, temperature, and flow that can make subtle changes in the core geometry. For these transients there are small margins between the calculated response and unacceptable responses. A considerable design and development effort will be necessary to assure that response of the core will be acceptable over a wide range of potential challenges.

Scram Systems

The SAFR design includes two sets of control rods either of which can independently shut down the reactor in response to a scram signal and maintain it subcritical. One set would be released automatically by the loss of holding power in a special clutch containing a magnet. Abnormally high sodium temperature, greater than 1050°F, would cause the Curie point temperature of the magnet to be exceeded. We note, however, that this feature depends on there being maintained a sufficient flow of

sodium coolant over the magnet. This flow must be assured if the automatic shutdown is to be assured.

Neither of the control rod systems is fully safety grade. Apparently, the systems do have some of the most important features of safety grade systems, e.g., tolerance of single failures. While we agree that experience with LWRs indicates the designation of a system as safety grade is not a guarantee of high reliability, we suggest that designation of a system as fundamentally important as a scram system as non-safety grade is flouting not only convention but good sense.

Use of PRA

The NRC staff seems to have been disappointed in the extent to which PRA has been useful in reviewing the design of SAFR, as well as the earlier review of the PRISM and MHTGR. Apparently the designs at this stage are developed in so little detail that risk analysts have little to work with and the benefits of the analysis are limited. Decision makers should regard with caution quantitative claims of high safety performance for reactor systems still at the conceptual design stage.

Containment

Although a secondary vessel is provided to contain leakage of sodium coolant, the SAFR design does not include a conventional containment capable of resisting high temperatures and pressures. It is contended that the potential for accidents, for which such a containment might provide mitigation, is so low that a conventional containment is not needed. Both deterministic and probabilistic arguments are made in support of this contention. Although these arguments have technical merit, we are not yet convinced. Our position is as stated in our report to you of July 20, 1988 on the key licensing issues associated with DOE-sponsored reactor designs and our report to you of October 13, 1988 on the preapplication safety evaluation report for the Modular High Temperature Gas Cooled Reactor.

However, there is a problem in specifying containment design criteria. One reason for providing a strong physical containment is to protect the public against unforeseen accidents. But, precisely because they are not foreseen, the design requirements for a containment are not obvious. Therefore, engineering and policy judgments must be made about the need for, and nature of, containment that might be used with SAFR. We believe that further study is appropriate before final judgments are made.

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Individual Rod Worth

There are two shutdown systems utilized in SAFR. Neither is currently safety grade. The automatic plant trip system can drive in all six of the primary control rods, which have a net reactivity worth of about ten dollars. It can also interrupt power to the electromagnetic latch and drop three secondary control rods, with a net reactivity worth of about seven dollars. The minimum number of primary control rods needed for reactor shutdown is two out of the six to insert about three dollars. The secondary system needs but one rod (about 2.2 dollars) to enter the core. With this very large reactivity worth for each rod, there is a potential for serious consequences from a rod ejection accident. We believe that this requires further study.

Need for Local Flow and Temperature Monitoring

The SAFR safety analysis indicates that blockage of flow through one fuel assembly may damage that assembly, but will not damage adjacent assemblies. Early work with oxide fuel has demonstrated that propagation is unlikely, but experiments and analysis with metal fuel have not been as extensive. Especially because the design does not provide for monitoring flow and effluent temperature from individual assemblies, we believe that this requires further study.

Role of the Operator

We believe that insufficient attention has been given to the role of the operator. Claims that a SAFR plant would have such inherently stable and safe characteristics that the operator will have essentially no safety function are unproven. Operation of four reactors, possibly in several different operational states at any given time, may be a significant challenge for the small operations crew envisioned. Opportunities for cognitive error, which might defeat favorable safety characteristics of the reactor, might be more abundant than is now recognized. Further study is needed.

Other Operational Considerations

In addition, certain features that have been found to be desirable in LWR plants are not provided in the SAFR design. Although remote shutdown capability is provided, it appears to lack some of the attributes of such systems in current LWR plants. Also, the design does not include Class 1E AC electric power systems, but relies entirely on Class 1E DC power from batteries. We recommend that further consideration be given to the potentially large power needs of essential auxiliary functions such as space cooling.

January 19, 1989

Protection Against Sabotage

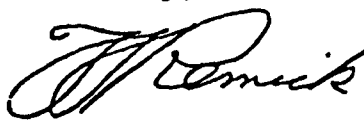
With regard to the need for designing protection against sabotage, the following statement from our report of July 20, 1988 should be given early consideration as the design of this plant progresses:

"It is often stated that significant protection against sabotage can be inexpensively incorporated into a plant if it is done early in the design process. Unfortunately, this has not been done consistently because the NRC has developed no guidance or requirements specific for plant design features, and there seems to have been no systematic attempt by the industry to fill the resulting vacuum. We believe the NRC can and should develop some guidance for designers of advanced reactors. It is probably unwise and counterproductive to specify highly detailed requirements, as those for present physical security systems, but an attempt should be made to develop some general guidance."

Sodium Fires

Further study of the potential for and suppression of sodium fires and consideration of their possible consequences is needed. Such studies should include the possibility of fires resulting from earthquake effects.

Sincerely,



Forrest J. Remick
Chairman

References

1. Office of Nuclear Regulatory Research, "Safety Evaluation Report for the Sodium Advanced Fast Reactor (SAFR)," November 9, 1988 (Predecisional Draft).
2. Rockwell International (DOE contractor), AI-DOE-13527, "SAFR Preliminary Safety Information Document," Volumes I through III, October 1985.

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This safety evaluation report (SER) presents the final results of a preapplication design review for the Sodium Advanced Fast Reactor (SAFR) liquid metal reactor (Project 673). The SAFR conceptual design was submitted by the U.S. Department of Energy (DOE) in accordance with the U.S. Nuclear Regulatory Commission (NRC) "Statement of Policy for the Regulation of Advanced Nuclear Power Plants" (51 FR 24643 which provides for the early Commission review and interaction. The standard SAFR plant design consists of four identical reactor modules, referred to as "paks," each with a thermal output rating of 900 MWt, coupled with four steam turbine-generator sets. The total electrical output was to be 1400 MWe. This SER represents the NRC staff's preliminary technical evaluation of the safety features in the SAFR design. It must be recognized that final conclusions in all matters discussed in this SER require approval by the Commission. During the NRC staff review of the SAFR conceptual design, DOE terminated work on this design in September 1988. This SER documents the work done to that date and no additional work is planned for the SAFR.

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