GEFR-00793
UC-87Ta
November 1986

PRISM™

Preliminary Safety Information Document

Prepared for U.S. Department of Energy
Under Contract No. DE-AC03-89SF17445

Volume VI

Appendix G
Responses to Issues in Draft SER

Applied Technology

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Letter dated 5/26/93

GE Nuclear Energy

Advance Nuclear Technology
San Jose, California

90-090-01

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GEFR - 00793
UC-87TA
MAY 1990

TITLE: APPENDIX G, AMENDMENT 13 TO THE PRISM (ALMR) PRELIMINARY SAFETY INFORMATION DOCUMENT

EDITORS: R. W. Hardy
G. L. Stimmell

Reviewed by: G. L. Gyorey, Manager
Safety & Licensing

F. E. Tippets, Manager
Engineering

Approved by: R. C. Berglund, Manager
ALMR Program,
Advance Nuclear Technology

Prepared for the
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APPLIED TECHNOLOGY

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Letter dated 5/26/93

GE Nuclear Energy

Advance Nuclear Technology
San Jose, California

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Department of Energy
Washington, DC 20585

MAY 2 6 1993

Mr. Dennis M. Crutchfield
Associate Director for Advanced Reactors
and License Renewal
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Crutchfield:

As you indicated in your letter, dated April 29, 1993, you are completing the final Preapplication Safety Evaluation Report (PSER) for the "Power Reactor Innovative Small Module" (PRISM) Advanced Liquid Metal Reactor design. You expressed concern about meeting one of the Commission's objectives of public disclosure since the PSER will be based on documents on which the Department of Energy (DOE), Office of Nuclear Energy, placed a restrictive distribution labeled "Applied Technology." We hereby approve your request for public disclosure and you are authorized to remove the "Applied Technology" (AT) distribution limitation from all of the DOE documents titled Preliminary Safety Information Document. The documents are:

"PRISM - Preliminary Safety Information Document" (PSID) - GEFR-00795

Volume I - December 1987, Chapters 1-4
Volume II - December 1987, Chapters 5-8
Volume III - December 1987, Chapters 9-14
Volume IV - December 1987, Chapters 15-17 and Appendices A-E
Volume V - February 1988, Amendment to PSID
Volume VI - March 1990, Appendix G

With regard to the Modular High Temperature Gas-Cooled Reactor (MHTGR), we would like to request that public disclosure of its AT information be delayed until publication of the MHTGR PSER becomes more imminent. We would appreciate your understanding of this situation and assure you that we will release MHTGR AT for public disclosure when needed to support the PSER issuance. We will be happy to meet with you and your staff to discuss this further at your convenience.

Sincerely,

Jerry D Griffith

Jerry D. Griffith
Director
Office of Advanced Reactor Programs
Office of Nuclear Energy

cc:
Salma El-Safwany, DOE/SF
James Quinn, GE
Richard Hardy, GE
Robert Pierson, NRC
Ray Mills, PDCO

GEFR - 00793
UC-87Ta
March 1990

TITLE: **APPENDIX G, AMENDMENT 12 TO THE PRISM (ALMR) PRELIMINARY SAFETY INFORMATION DOCUMENT**

EDITORS: R. W. Hardy
G. L. Stimmell

Reviewed by: G. L. Gyorey, Manager
Safety & Licensing

F. E. Tippets, Manager
Engineering

Approved by: R. C. Berglund, Manager
ALMR Program,
Advance Nuclear Technology

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GE Nuclear Energy

Advance Nuclear Technology
San Jose, California

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ABBREVIATIONS AND ACRONYMS

ACLP	ABOVE CORE LOAD PAD
ACRS	ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
ACS	AUXILIARY COOLING SYSTEM
ALMR	ADVANCED LIQUID METAL REACTOR
ANL	ARGONNE NATIONAL LABORATORY
ARIES	TRANSIENT COMPUTER CODE
ATWS	ANTICIPATED TRANSIENTS WITHOUT SCRAM
BOEC	BEGINNING OF EQUILIBRIUM CYCLE
BOP	BALANCE OF PLANT
BWR	BOILING WATER REACTOR
CONTAIN	CONTAINMENT RADIOLOGICAL RELEASE CODE
CP/OL	COMBINED CONSTRUCTION PERMIT AND OPERATING LICENSE
CR	CONTROL ROOM
CRBRP	CLINCH RIVER BREEDER REACTOR PLANT
CRD	CONTROL ROD DRIVE
CRDM	CONTROL ROD DRIVE MECHANISM
CREDO	CENTRALIZED RELIABILITY DATA ORGANIZATION
CSDT	COMPOUND SYSTEM DOUBLING TIME
DN	DELAYED NEUTRON
DNB	DEPARTURE FROM NUCLEATE BOILING
DOE	DEPARTMENT OF ENERGY
DPA	DISPLACEMENTS PER ATOM
EBR-II	EXPERIMENTAL BREEDER REACTOR NO. II
EM	ELECTROMAGNETIC
EMF	ELECTROMOTIVE FORCE
EOEC	END OF EQUILIBRIUM CYCLE
ETEC	ENERGY TECHNOLOGY ENGINEERING CENTER

ABBREVIATIONS AND ACRONYMS
(Cont'd)

FCF	FUEL CYCLE FACILITY
FDA	FINAL DESIGN APPROVAL
FFTF	FAST FLUX TEST FACILITY
FMEA	FAILURE MODE AND EFFECTS ANALYSIS
FMF	FUEL MANUFACTURING FACILITY
FSAR	FINAL SAFETY ANALYSIS REPORT
FSF	FUEL SERVICE FACILITY
FTC	FUEL TRANSFER CASK
GDC	GENERAL DESIGN CRITERION
GEM	GAS EXPANSION MODULE
HAA	HEAD ACCESS AREA
HCDA	HYPOTHETICAL CORE DISRUPTIVE ACCIDENT
HDPE	HIGH DENSITY POLYETHYLENE
HFEF/N	HOT FUEL EXAMINATION FACILITY - NORTH (at EBR-II)
HFEF/S	HOT FUEL EXAMINATION FACILITY - SOUTH (at EBR-II)
HVAC	HEATING, VENTILATING, AIR CONDITIONING
IFR	INTEGRAL FAST REACTOR
IHTS	INTERMEDIATE HEAT TRANSFER SYSTEM
IHX	INTERMEDIATE HEAT EXCHANGER
IVTM	IN-VESSEL TRANSFER MACHINE
LED	LIGHT EMITTING DIODE
LMR	LIQUID METAL REACTOR
LODHR	LOSS OF DECAY HEAT REMOVAL
LOF	LOSS OF FLOW
LWR	LIGHT WATER REACTOR
MTTF	MEAN TIME TO FAILURE

NEPA	NATIONAL ENVIRONMENTAL POLICY ACT
NI	NUCLEAR ISLAND
NRC	NUCLEAR REGULATORY COMMISSION
NUCLARR	NUCLEAR COMPUTERIZED LIBRARY FOR ASSESSING REACTOR RELIABILITY
OBE	OPERATING BASIS EARTHQUAKE
ORNL	OAK RIDGE NATIONAL LABORATORY
O&M	OPERATION AND MAINTENANCE
PAG	PROTECTIVE ACTION GUIDELINES
PAM	POST ACCIDENT MONITORING
PCS	PLANT CONTROL SYSTEM
PCU	POWER CONDITIONING UNIT
PRA	PROBABILISTIC RISK ASSESSMENT
PRISM	POWER REACTOR INNOVATIVE SMALL MODULE
PSAR	PRELIMINARY SAFETY ANALYSIS REPORT
PSID	PRELIMINARY SAFETY INFORMATION DOCUMENT
PSPS	PRIMARY SODIUM PURIFICATION SYSTEM
PSST	PRIMARY SODIUM STORAGE TANK
PWR	PRESSURIZED WATER REACTOR
Q/A	QUALITY ASSURANCE
RAM	RELIABILITY, AVAILABILITY, AND MAINTAINABILITY
RCSS	REACTIVITY CONTROL AND SHUTDOWN SYSTEM
RC&IS	ROD CONTROL & INFORMATION SYSTEM
RE	REFUELING ENCLOSURE
RPS	REACTOR PROTECTION SYSTEM
RPST	REACTION PRODUCTS SEPARATION TANK
RSF	REMOTE SHUTDOWN FACILITY
RSS	ROD STOP SYSTEM

ABBREVIATIONS AND ACRONYMS
(Cont'd)

RTE	RESIDUAL TOTAL ELONGATION
RV	REACTOR VESSEL
RVACS	REACTOR VESSEL AUXILIARY COOLING SYSTEM
SASSYS	NATIONAL STANDARD SAFETY CODE
SDC	STANDARD DESIGN CERTIFICATION
SER	SAFETY EVALUATION REPORT
SG	STEAM GENERATOR
SGB	STEAM GENERATOR BUILDING
SGS	STEAM GENERATOR SYSTEM
SGTS	STANDBY GAS TREATMENT SYSTEM
SMART	RADIOLOGICAL DOSE CODE
SS	STAINLESS STEEL
SSE	SAFE SHUTDOWN EARTHQUAKE
SWR	SODIUM WATER REACTION
SWRPRS	SODIUM WATER REACTION PRESSURE RELIEF SUBSYSTEM
TLP	TOP OF CORE LOAD PAD
TREAT	TRANSIENT REACTOR TEST FACILITY
TSS	THERMAL SHUTOFF SYSTEM
UBC	UNIFORM BUILDING CODE
UCLA	UNIVERSITY OF CALIFORNIA - LOS ANGELES
UIS	UPPER INTERNALS STRUCTURE
ULOF	UNPROTECTED LOSS OF FLOW
ULOHS	UNPROTECTED LOSS OF HEAT SINK
USS	ULTIMATE SHUTDOWN SYSTEM
UTOP	UNPROTECTED TRANSIENT OVERPOWER
WHC	WESTINGHOUSE HANFORD COMPANY

G.1 INTRODUCTION

G.1.1 Purpose of Appendix G

Preliminary Safety Information Document (PSID), GEFR-00793, was submitted to the Nuclear Regulatory Commission (NRC) in November 1986 to provide safety information for the PRISM (Power Reactor Innovative Small Module) liquid metal reactor design. The safety information was presented in four volumes of descriptive material organized into 17 chapters and five appendices, A through E. This material, along with subsequent amendments, defined the PRISM design as it existed in 1986-1987. A fifth volume containing Appendix F was subsequently added to provide responses and clarifications to NRC Staff comments raised during the Staff's review of the PSID during 1986-1988. Amendments 1 through 11 were issued during 1987-1988 to revise the PSID based on these responses and clarifications.

Since then, two significant events have occurred. First, the Department of Energy (DOE) awarded the Advanced Liquid Metal Reactor (ALMR) design contract to an industrial team led by General Electric, with direction to continue advanced conceptual design based on the PRISM concept starting in January 1989. Second, the NRC Staff issued NUREG 1368 in September 1989 presenting the Draft Preapplication Safety Evaluation Report (SER) for PRISM.

The draft SER identifies 18 safety issues concerning the original PRISM design. Review of these safety issues shows that many of them have already been addressed or are being addressed in recent design work being performed on the ALMR. Since the final SER is not scheduled to be issued until after the NRC Commissioners resolve a number of policy issues related to advanced reactors, the opportunity exists to update the information in the PSID to the current reference ALMR design and have the Staff incorporate an evaluation of this current information into the SER prior to its release in final form.

The purpose of Appendix G is to update the PSID by describing the current reference ALMR design. This appendix is organized into three major sections comprising (1) a summary of the current ALMR reference design and the major design changes made since 1987, (2) a summary of the safety related R&D results and plans for the ALMR Program, and (3) a discussion of the ALMR design details and rationale for addressing each of the safety issues identified in the draft SER. The 18 safety issues are listed below:

1. Containment
2. Shutdown Systems
3. 60 Year Plant Life
4. Seismic Isolators
5. Sodium Void
6. Flow Blockage
7. Electromagnetic Pumps
8. Sodium/Water Reaction Pressure Relief System
9. Reactor Vessel Auxiliary Cooling System
10. Control Room
11. Emergency Preparedness
12. Role of Operator
13. Multi-Module Control
14. Security
15. Prototype Test
16. Safety Analyses
17. Station Blackout
18. Risk Assessment

In addition, because the reference ALMR design now incorporates significant containment and mitigation capabilities for severe core accidents, an additional subsection has been added to Appendix G entitled:

19. Mitigation of Severe Core Accidents

Each of the 19 safety issues is discussed in Section 4 of Appendix G. For each safety issue, the GE understanding of the issue is stated, generally paraphrasing statements in the draft SER. Significant ALMR design features and approaches are then described as they relate to each safety issue. Finally, analyses and rationale are described justifying the reference ALMR design features and approaches in response to each safety issue.

The material in Appendix G has been scheduled for submittal to the NRC Staff in two separate amendments, Amendment 12 dated March 1990, and Amendment 13 dated May 1990, in order to expedite both the submittal of the material and the NRC Staff review of this material. A sixth volume has been provided to contain Appendix G.

The intent has been to make Appendix G a stand alone document, such that reference to the original PSID is not required for an understanding of how the reference ALMR design addresses the safety issues.

When reviewing Appendix G, it is important to note that changes in both design and approach have been made since the original PSID was issued. The emphasis in the original PSID was on prevention of accidents to reach the safety goals. Containment and mitigation of severe accidents were not addressed in detail. Consistent with this approach, the design had an unconventional containment in that portions of the primary coolant boundary doubled as containment. In the current reference ALMR design, the design and approach have been changed. While it is still claimed that the safety goals can be reached by prevention of accidents alone, Appendix G discusses how the reference ALMR design is expected to contain and mitigate the effects of severe accidents without breach of the primary coolant boundary. In addition, Appendix G describes the low leakage pressure-retaining containment dome which has been added to provide additional assurance that a severe accident can be mitigated if it did breach the primary coolant boundary.

G.1.2 Cross Reference to Draft SER

The material in Appendix G is intended to respond directly to the safety issues identified in the draft SER. In order to facilitate correlation of the responses to the issues, this section provides a cross reference between the material in Appendix G and sections in the draft SER where the issue was originally identified. Also noted is whether the material in Appendix G is part of Amendment 12 or Amendment 13.

Appendix G Section No. & Topic		Draft SER Section Addressed	Amendment
G.1	Introduction	N/A	12
G.2	Design Description	N/A	12
G.3	Safety R&D Results & Plans	N/A	13
G.4	Discussion of Safety Issues		
G.4.1	Containment	3.1.2.3 3.2 6.2.6 15.10.6	13
G.4.2	Shutdown Systems	3.2 4.3.5 4.5.5 4.5.6 4.6.5 4.6.6 7.2.5 7.2.6	13
G.4.3	60 Year Plant Life	3.3.4	12
G.4.4	Seismic Isolators	3.3.4	12
G.4.5	Sodium Void	4.3.5 4.6.5	12
G.4.6	Flow Blockage	4.4.5 4.4.6 15.10.2	13
G.4.7	Electromagnetic Pumps	4.4.5 4.6.5 5.4.1.3 5.4.5.1 5.4.6 7.2.5.3 8.3.1 8.3.2 A.4.2	12
G.4.8	Sodium/Water Reaction Pressure Relief System	5.5.5 5.6.5.2	13
G.4.9	Reactor Vessel Auxiliary Cooling System	5.7.5	13
G.4.10	Control Room	7.3.3	12

Appendix G Section No. & Topic	Draft SER Section Addressed	Amendment
G.4.11 Emergency Preparedness	3.1.2.4 13.1.4	13
G.4.12 Role of Operator	13.2.3	12
G.4.13 Multi-Module Control	13.2.4	12
G.4.14 Security	13.3.3	12
G.4.15 Prototype Test	14.3.2 14.4.4	13
G.4.16 Safety Analyses	15.3.5 15.10.5 15.10.6	13
G.4.17 Station Blackout	A.3.2	12
G.4.18 Risk Assessment	A.7	13
G.4.19 Mitigation of Severe Core Accidents	N/A	13

G.1.3 Summary of Safety Issues and Responses

Table G.1-1 provides a capsule summary of the major Staff concerns for each of the 19 safety issues discussed in the draft SER, and the ALMR response provided in Appendix G.

Table G.1-1

SUMMARY OF SAFETY ISSUES AND RESPONSES

NAME OF ISSUE	STAFF CONCERN	ALMR RESPONSE
1. CONTAINMENT	A. CONTAINMENT DESIGN UNACCEPTABLE SINCE ACCEPTANCE CRITERIA NOT MET FOR ALL BOUNDING EVENTS B. MECHANISTIC SOURCE TERMS CAN BE ACCEPTABLE IF THEY ARE SHOWN TO BE BOUNDING, AND IF PERFORMANCE OF REACTOR AND FUEL IS WELL UNDERSTOOD	A. A LOW LEAKAGE, PRESSURE RETAINING CONTAINMENT DOME HAS BEEN ADDED TO CONTAINMENT VESSEL TO COMPLETE CONTAINMENT BOUNDARY B. DESIGN GOAL IS TO RETAIN HCDA AND CORE MELT ACCIDENTS WITHIN PRIMARY SYSTEM BOUNDARY C. SOURCE TERM TO CONTAINMENT DOME CONSERVATIVELY ASSUMES REACTOR CLOSURE BREACH AND SODIUM FIRE D. CONTAINMENT DOME AND REFUELING ENCLOSURE MITIGATE REFUELING AND MAINTENANCE ACCIDENTS E. DESIGN MEETS PAG, 10CFR100 LIMITS

Table G.1-1 (continued)

SUMMARY OF SAFETY ISSUES AND RESPONSES

NAME OF ISSUE	STAFF CONCERN	ALMR RESPONSE
2. SHUTDOWN SYSTEMS	A. PASSIVE REACTIVITY FEEDBACK IS ACCEPTABLE AS A DIVERSE MEANS OF SHUTDOWN, CONFIRMATION OF FEEDBACKS AND ACTIONS TO ACHIEVE EVENTUAL SUBCRITICALITY ARE REQUIRED B. POSITIVE SODIUM VOID COEFFICIENT SHOULD BE REDUCED C. FEEDBACKS WILL BE VERIFIED OVER LIFE OF PLANT D. MUST SHOW CONTROL ASSEMBLIES CANNOT FLOAT IN EVENT OF PRIMARY PUMP STARTUP DURING REFUELING E. LOSS OF FLOW EVENTS WITHOUT PUMP COASTDOWN ARE OF CONCERN	A. ULTIMATE SHUTDOWN SYSTEM HAS BEEN ADDED TO PROVIDE COLD SHUTDOWN, REACTIVITY FEEDBACKS WILL BE CONFIRMED IN PROTOTYPE B. POSITIVE VOID COEFFICIENT IS ACCEPTABLE BASED ON LOW PROBABILITY OF OCCURRENCE AND TOLERABLE CONSEQUENCES C. FEEDBACKS NEED TO BE VERIFIED OVER LIFE OF PLANT D. ABSORBER BUNDLE DESIGN PRECLUDES LIFTING BY HYDRAULIC FORCES DUE TO CAPABILITY OF GRAVITY INSERTION AGAINST FULL PUMP FLOW E. ROD STOPS AND GAS EXPANSION MODULES HAVE BEEN ADDED TO LIMIT THE CONSEQUENCES OF ROD WITHDRAWAL AND LOSS OF FLOW EVENTS
3. 60 YR PLANT LIFE	A. CURRENT LEGISLATION LIMITS LICENSE TO 40 YEARS B. EXTENSION TO 60 YEARS WOULD REQUIRE DEGRADATION AND AGING STUDIES, EXTRA ISI & MAINTENANCE	A. LEGISLATION CONCERN WILL BE SETTLED BY LWR LIFE EXTENSION PROGRAMS B. AGREE THAT DEGRADATION AND AGING STUDIES, EXTRA ISI & MAINTENANCE ARE REQUIRED C. ALSO IMPORTANT ARE PROPER DESIGN & COMPONENT REPLACEMENT
4. SEISMIC ISOLATORS	A. FURTHER EVALUATION BASED UPON ADDITIONAL DESIGN WORK AND R&D IS REQUIRED	A. CONSIDERABLE DESIGN WORK AND R&D HAVE ALREADY BEEN COMPLETED, SHOWING THAT THE SEISMIC ISOLATOR PERFORMS AS EXPECTED, AND HAS CONSIDERABLE MARGIN B. ADDITIONAL DESIGN WORK AND R&D ARE PLANNED

Table G.1-1 (continued)

SUMMARY OF SAFETY ISSUES AND RESPONSES

NAME OF ISSUE	STAFF CONCERN	ALMR RESPONSE
5. SODIUM VOID	A. MAGNITUDE OF POSITIVE SODIUM VOID COEFFICIENT SHOULD BE REDUCED B. PROBABILITY OF VOIDING MUST BE LOW C. CONSEQUENCES OF VOIDING MUST BE TOLERABLE	A. MEANINGFUL REDUCTIONS IN VOID WORTH ADVERSELY IMPACT OTHER SAFETY PARAMETERS AND REQUIRE UNECONOMIC REACTOR DESIGNS B. PROBABILITY OF VOIDING IS SHOWN TO BE EXTREMELY LOW C. DESIGN GOAL IS TO RETAIN HCDA AND CORE MELT ACCIDENTS WITHIN PRIMARY SYSTEM BOUNDARY D. CONTAINMENT DOME AND VESSEL PROVIDE ADDITIONAL MITIGATION
6. FLOW BLOCKAGE	A. FLOW BLOCKAGE HAS POTENTIAL FOR SODIUM BOILING AND ENERGETICS B. PREVENTION AND DETECTION OF FLOW BLOCKAGE DUE TO FABRICATION ERRORS ARE REQUIRED	A. FLOW BLOCKAGE DUE TO FABRICATION ERRORS IS EXTREMELY LOW PROBABILITY B. PRELIMINARY ANALYSES SHOW THAT CONSEQUENCES OF FLOW BLOCKAGE ARE TOLERABLE C. ADDITIONAL R&D IS PLANNED TO SUPPORT ANALYSES D. FLOW TESTING OF ASSEMBLY PRIOR TO LOADING INTO REACTOR IS PLANNED, IN-REACTOR FLOW TESTING BEING EVALUATED
7. EM PUMPS	A. PUMP STARTUP DURING REFUELING MUST NOT BE ABLE TO FLOAT CONTROL ROD ABSORBER BUNDLES B. SYNCHRONOUS MACHINES SHOULD BE SEISMICALLY ISOLATED C. ADEQUACY OF COASTDOWN MUST BE VERIFIED BY R&D PROGRAM D. PUMPS MUST NOT TRIP BEFORE CONTROL RODS ARE INSERTED E. MUST BE ABLE TO MONITOR SYNCHRONOUS MACHINE PERFORMANCE F. FAILURE MODES, RISK ESTIMATES, ENVIRONMENTAL EFFECTS, SYSTEM INTERACTIONS, AGING, MAINTENANCE EFFECTS, PERFORMANCE MONITORING, OPERATION UNDER EXTREME CONDITIONS ARE NOT WELL UNDERSTOOD	A. ABSORBER BUNDLE DESIGN PRECLUDES LIFTING BY HYDRAULIC FORCES DUE TO CAPABILITY OF GRAVITY INSERTION AGAINST FULL PUMP FLOW B. SYNCHRONOUS MACHINES ARE SEISMICALLY ISOLATED C. A RIGOROUS R&D PROGRAM WILL BE COMPLETED TO QUALIFY PUMPS AND SYNCHRONOUS MACHINES D. LOGIC IN RPS DELAYS PUMP TRIP UNTIL INDICATION OF CONTROL ROD INSERTION IS RECEIVED (NORMAL SCRAM) E. SAFETY GRADE THERMAL SHUTOFF SYSTEM ADDED TO ENSURE EM PUMP SHUTOFF AND ELIMINATION OF HEAT SOURCE FOR ULOHS EVENT F. EXTENSIVE SET OF DIAGNOSTIC SENSORS MONITORS SYNCHRONOUS MACHINE PERFORMANCE

Table G.1-1 (continued)

SUMMARY OF SAFETY ISSUES AND RESPONSES

NAME OF ISSUE	STAFF CONCERN	ALMR RESPONSE
		G. ADDITIONAL STUDIES HAVE BEEN, AND WILL BE, COMPLETED ON FAILURE MODES AND RISK ESTIMATES, ENVIRONMENTAL EFFECTS, SYSTEM INTERACTIONS, AGING, MAINTENANCE, PERFORMANCE MONITORING, OPERATION UNDER EXTREME CONDITIONS
8. SWRPRS	A. BOTH SWRPRS AND THE WATER/STEAM DUMP SYSTEM SHOULD BE SAFETY GRADE	A. SWRPRS RUPTURE DISKS ARE SAFETY GRADE, BUILDING IS SEISMIC II
9. RVACS	A. CORE INTEGRITY IS THREATENED ON LOSS OF ALL DECAY HEAT TRANSIENT (BOUNDING EVENT NO. 3) B. FREQUENCY, HIGH TEMPERATURES, RECOVERY FROM RVACS TRANSIENT ARE OF CONCERN	A. CORE CAN TOLERATE REDEFINED BOUNDING EVENT NO.3 WHICH PRESUMES LOSS OF 75% RVACS FLOW B. ONLY FOUR RVACS TRANSIENTS ARE EXPECTED DURING PLANT LIFE, WITH ONLY ONE TRANSIENT TO MAXIMUM TEMPERATURE - REDUCTION IN PLANT LIFE DUE TO RVACS TRANSIENTS IS SMALL C. TEMPERATURES WILL BE MAPPED DURING RVACS TRANSIENT IN PROTOTYPE TEST
10. CONTROL ROOM	A. THE CONTROL ROOM AND REMOTE SHUTDOWN AREA SHOULD BE SAFETY GRADE, WITH SAFETY GRADE CONTROLS AND INSTRUMENTATION	A. CONTROL ROOM RELOCATED TO INSIDE HIGH SECURITY BOUNDARY, UPGRADED TO SEISMIC CATEGORY II WITH TORNADO HARDENING, UPGRADED HVACS B. REMOTE SHUTDOWN FACILITY UPGRADED TO SEISMIC CATEGORY I WITH TORNADO HARDENING, CLASS 1E C&I, SAFETY GRADE HVACS C. SEISMIC CATEGORY II, TORNADO HARDENED TUNNEL BETWEEN CONTROL ROOM AND REMOTE SHUTDOWN FACILITY
11. EMERGENCY PREPAREDNESS	A. USE OF A LIMITED OFF-SITE EMERGENCY PLAN REQUIRES MEETING THE LOWER LEVEL PROTECTIVE ACTION GUIDELINES B. THE STAFF CONCLUDES PRISM CANNOT MEET LOWER LEVEL PAGs BECAUSE OF REACTOR RESPONSE TO FOUR BOUNDING EVENTS	A. USE OF A LIMITED OFF-SITE EMERGENCY PLAN IS JUSTIFIED SINCE ALMR MEETS LOWER LEVEL PAGs BY COMBINATION OF ACCIDENT PREVENTION CAPABILITY, LONG RESPONSE TIMES, DESIGN FOR HCDA AND CORE MELT ACCOMMODATION, ADDITION OF CONTAINMENT DOME TO COMPLETE CONTAINMENT BOUNDARY B. ALMR MEETS ACCEPTANCE CRITERIA FOR ALL BOUNDING EVENTS

Table G.1-1 (continued)

SUMMARY OF SAFETY ISSUES AND RESPONSES

NAME OF ISSUE	STAFF CONCERN	ALMR RESPONSE
12. ROLE OF OPERATOR	A. OPERATORS MUST BE PROTECTED, PROVIDED WITH APPROPRIATE COMMUNICATIONS, CONSIDERED BACKUP TO SAFETY SYSTEMS	A. CONTROL ROOM RELOCATED TO INSIDE HIGH SECURITY BOUNDARY, UPGRADED TO SEISMIC CATEGORY II WITH TORNADO HARDENING, UPGRADED HVACS B. REMOTE SHUTDOWN FACILITY UPGRADED TO SEISMIC CATEGORY I WITH TORNADO HARDENING, CLASS 1E C&I, SAFETY GRADE HVACS C. SEISMIC CATEGORY II, TORNADO HARDENED TUNNEL BETWEEN CONTROL ROOM AND REMOTE SHUTDOWN FACILITY D. OPERATOR HAS RESPONSIBILITIES FOR SUPERVISION AND MANAGEMENT OF PLANT, SECURING PLANT TO COLD SHUTDOWN, MONITORING POST ACCIDENT CONDITIONS, COMMUNICATING PLANT CONDITIONS TO OUTSIDE PERSONNEL, INITIATING RECOVERY ACTIONS
13. MULTI-MODULE CONTROL	A. OPERATION WITH MULTI-MODULE CONTROL NEEDS DEMONSTRATION	A. ALMR CONTROL SYSTEM WILL BE MODELED, SIMULATED IN REAL TIME, TESTED ON EBR-II, TESTED FOR MAN-MACHINE INTERFACE, TESTED ON PROTOTYPE BY USE OF ONE ACTUAL REACTOR MODULE CONTROL SYSTEM AND TWO SIMULATED REACTOR MODULE CONTROL SYSTEMS
14. SECURITY	A. CONCERNED ABOUT SABOTAGE INDUCED LOSS OF FLOW EVENTS B. ISOLATION ZONE CAPABILITY NEEDS UPGRADING C. PERIMETER FENCE AND RESPONSE TEAMS NEED RELOCATION D. RVACS SECURITY NEEDS UPGRADING E. DOOR ALARMS REQUIRED IN VITAL AREAS F. ON-SITE POWER SUPPLIES FOR SECURITY EQUIPMENT NEEDS PROTECTION G. LOCAL LAW ENFORCEMENT RESPONSE TIMES NEED TO BE CONSIDERED	A. PASSIVE GAS EXPANSION MODULES, ELECTRONICALLY POSITIONED MECHANICAL ROD STOPS AND ULTIMATE SHUTDOWN SYSTEM MITIGATE SABOTAGE INDUCED EVENTS B. ISOLATION ZONE HAS BEEN UPGRADED C. PERIMETER FENCE LOCATION MEETS REQUIREMENTS, VULNERABILITY ANALYSIS SHOWS RESPONSE TIMES ARE ADEQUATE D. INTRUSION DETECTION SENSORS AND ALARMS ARE PROVIDED ON RVACS STACKS E. DOOR ALARMS ARE PROVIDED IN VITAL AREAS

Table G.1-1 (continued)

SUMMARY OF SAFETY ISSUES AND RESPONSES

NAME OF ISSUE	STAFF CONCERN	ALMR RESPONSE
	H. INSIDER SABOTAGE NEEDS TO BE CONSIDERED I. OPERATIONS CENTER SHOULD BE LOCATED WITHIN THE HIGH SECURITY BOUNDARY J. ADDITIONAL JUSTIFICATION REQUIRED TO LOCATE SWRPRS OUTSIDE HIGH SECURITY BOUNDARY	F. UNINTERRUPTABLE POWER WILL BE PROVIDED FOR SECURITY G. VULNERABILITY ANALYSIS SHOWS ONLY ON-SITE RESPONSE FORCE REQUIRED, WITH LOCAL LAW ENFORCEMENT TO BE INFORMED H. VULNERABILITY ANALYSIS SHOWS SUCCESSFUL INSIDER THREAT IS NOT CREDIBLE I. CONTROL BUILDING RELOCATED INSIDE HIGH SECURITY FENCE J. SWRPRS UPGRADED TO HIGH SECURITY AREA
15. PROTOTYPE TEST	A. SCOPE OF PLANT TO BE CERTIFIED IS AN OPEN ITEM B. PROTOTYPE TEST MUST USE TRUE PROTOTYPIC REACTOR AND KEY SUPPORT SYSTEMS C. ADDITIONAL INSTRUMENTATION MAY BE REQUIRED DURING PROTOTYPE TEST D. PORTION OF PLANT NOT TO BE CERTIFIED MUST BE SHOWN TO NOT AFFECT SAFE OPERATION OF PLANT	A. COMPLETE POWER BLOCK AND KEY SUPPORT SYSTEMS ARE TO BE CERTIFIED; REACTOR MODULE AND IHTS WILL BE INCLUDED IN PROTOTYPE TEST USING AN AIR-COOLED HEAT EXCHANGER SYSTEM INSTEAD OF THE STEAM GENERATOR B. REACTOR AND KEY SUPPORT SYSTEMS WILL BE FULL SCALE PROTOTYPIC, AND INCLUDED IN THE PROTOTYPE TEST OR DEMONSTRATED SEPARATELY C. ADDITIONAL INSTRUMENTATION WILL BE INCLUDED IN THE PROTOTYPE TEST TO PROVIDE REQUIRED DIAGNOSTIC INFORMATION D. IT WILL BE SHOWN THAT THE PORTION OF THE PLANT DESIGNATED AS NON-NUCLEAR SAFETY GRADE CANNOT ADVERSELY AFFECT SAFE OPERATION OF THE PLANT
16. SAFETY ANALYSES	A. FOUR BOUNDING EVENTS SHOW POTENTIAL FOR CORE MELTING, ADDITION OF SIGNIFICANT POSITIVE REACTIVITY, POTENTIAL FOR LARGE RADIOLOGICAL RELEASE	A. DESIGN CHANGES MADE TO IMPROVE SAFETY MARGINS INCLUDE AN INCREASE IN NUMBER OF FUEL PINS, AND THE ADDITION OF GEMS, ROD STOPS, AND ULTIMATE SHUTDOWN SYSTEM; FEATURES ALSO ADDED TO CONTAIN HCDA AND CORE MELT ACCIDENTS WITHIN PRIMARY SYSTEM BOUNDARY, AND TO PROVIDE CONTAINMENT DOME TO COMPLETE CONTAINMENT BOUNDARY

Table G.1-1 (continued)

SUMMARY OF SAFETY ISSUES AND RESPONSES

NAME OF ISSUE	STAFF CONCERN	ALMR RESPONSE
		B. RESPONSE TO BOUNDING EVENTS IS ACCEPTABLE, WITH RESPONSE TO ASSEMBLY FLOW BLOCKAGE TO BE DEMONSTRATED
17. STATION BLACKOUT	A. LACK OF CLASS 1E DIESELS MAY MAKE BLACKOUT FREQUENCY TOO HIGH	A. FREQUENCY OF STATION BLACKOUT WITHOUT NON-1E AUXILIARY POWER OR TURBINE RUNBACK TO PICK UP HOUSE LOAD IS SMALL B. CLASS 1E BATTERIES AND PASSIVE SAFETY FEATURES ASSURE SAFE SHUTDOWN AND DECAY HEAT REMOVAL
18. RISK ASSESSMENT	A. PRA LACKS DETAIL AND DATA B. EXTERNAL EVENTS HAVE NOT BEEN QUANTIFIED C. SYSTEM INTERACTION STUDIES HAVE NOT BEEN PERFORMED D. EXTRAPOLATION OF SOURCE TERM DATA FROM OXIDE TO METAL FUEL MAY NOT BE CONSERVATIVE E. RETENTION OF FISSION PRODUCTS IN HEAD ACCESS AREA NEEDS ADDITIONAL ANALYSIS F. MECHANISTIC ANALYSES OF ACCIDENT SEQUENCES HAVE NOT BEEN PERFORMED G. UNCERTAINTIES HAVE NOT BEEN QUANTIFIED H. ROLE OF OPERATOR IS NOT APPARENT I. NEED MORE WORK ON LOWER END OF PROBABILITY SPECTRUM	A. ADDITIONAL INFORMATION WILL BE DEVELOPED AS PRA EVOLVES AND MATURES TO ADDRESS PRELIMINARY AND FINAL DESIGN B. A NUMBER OF DESIGN CHANGES INCREASE MARGIN FOR ACCIDENT PREVENTION AND MITIGATION, WHICH REDUCES SENSITIVITY TO LACK OF DETAIL AND UNCERTAINTIES C. ADDITIONAL EFFORT IS UNDERWAY TO DEVELOP MORE DETAIL AND TO REDUCE UNCERTAINITES, STARTING WITH INITIATING EVENTS AND CORE DAMAGE FREQUENCIES D. STUDIES WILL BE PERFORMED ON METAL FUEL TO SUPPORT SOURCE TERM ANALYSES
19. MITIGATION OF SEVERE CORE ACCIDENTS	A. DRAFT SER DID NOT IDENTIFY MITIGATION OF SEVERE CORE ACCIDENTS AS A CONCERN	A. ANALYSES SHOW IT APPEARS FEASIBLE TO CONTAIN HCDA AND CORE MELT ACCIDENTS WITHIN THE PRIMARY SYSTEM BOUNDARY - THIS IS NOW A DESIGN GOAL

G.2 DESIGN DESCRIPTION

G.2.1 Summary of ALMR Plant Reference Design

This section summarizes the current Advanced Liquid Metal Reactor (ALMR) reference design. Changes made in the reference ALMR design since the PRISM PSID was issued in November 1986 and amended in December 1987, are summarized in Section G.2.2. Additional details for the ALMR reference design are described in Section G.4, as necessary, to support the responses to each of the safety issues presented in that section.

The ALMR design is based on the General Electric PRISM (Power Reactor Innovative Small Module) design described in the initial issue of Preliminary Safety Information Document, GEFR-00793. An objective of the ALMR program, conducted under Department of Energy Contract DE-AC03-89SF17445, is to develop a conceptual design of an ALMR power plant which improves safety margins, licensability, constructibility, operations, maintenance, and cost such that it is a viable option for commercialization shortly after the year 2000.

The GE ALMR design emphasizes passive safety, modular construction, and factory fabrication. Reactor modules for an ALMR power plant are sized to be fabricated in a factory and shipped to the sites by the most economic combination of barge, rail, and road transport. The reactor facilities, reactor auxiliary systems, fuel service facility, remote shutdown facility, and the optional co-located fuel cycle facility will be nuclear safety grade. The remaining nuclear island (NI) and balance of plant (BOP) facilities will be of high quality, industrial grade construction.

The ALMR features simple and reliable safety systems, seismic isolation, passive decay heat removal, passive reactivity control, and substantial margins to structural and fuel damage limits during potential accident situations. These features result in significant gains in the public safety and protection of the owner's investment. Standardized modular construction and extensive factory fabrication result in a plant design that is economically competitive against projected coal plants and other nuclear design approaches.

Trade studies, conducted during 1988 and 1989, identified and evaluated design alternatives for the ALMR. In addition, the NRC issued NUREG 1368 in September 1989 summarizing their safety evaluation based on review of the PRISM PSID. As a result of these trade studies and the safety issues identified in NUREG 1368, design improvements have been incorporated into the reference ALMR design. It is the reference ALMR plant design with these improvements incorporated that is summarized in this section.

G.2.1.1 Overall Plant Description

The reference commercial ALMR plant, shown in Figure G.2.1-1, utilizes nine reactor modules arranged in three identical 465 MWe power blocks for an overall plant net electrical rating of 1395 MWe. Each power block features three identical reactor modules, each with its own steam generator that jointly supply power to a single turbine-generator. Smaller plant sizes of 465 MWe and 930 MWe can be provided by using one or two of the standard power blocks. With incremental power block construction, early revenue can be produced by operating initial power blocks while awaiting completion of subsequent power blocks.

The main power system flow diagram for a standard power block is shown in Figure G.2.1-2. Major plant performance characteristics are summarized in Table G.2.1-1. Each of the three 471 MWt reactor modules has its own steam generator which is heated by secondary sodium piped from the intermediate heat exchangers in the reactor module. The three steam generators supply 965 psia dry saturated steam to a single power block 465 MWe (net output) turbine.

All nuclear safety grade systems and buildings are enclosed within a fenced and barricaded high security area surrounding the nuclear island (NI) as shown in Figure G.2.1-1. These Seismic Category I, safety-grade facilities include the reactor module, electrical equipment vaults, remote shutdown facility, NI guard house, and fuel service facility. Non-safety grade, but functionally related, facilities such as the control building, reactor maintenance facility, NI personnel facility, and assembly facility are also located within the NI high security fence. A Seismic Category II personnel tunnel connects the remote shutdown facility to the Seismic Category II control building.

Table G.2.1-1

PLANT PERFORMANCE CHARACTERISTICS

Overall Plant	
- Net Electrical Output	1395 MWe
- Net Station Efficiency	32.9%
- Number of Power Blocks	Three
- Number of Reactor Modules:	
per power block	Three
per plant	Nine
Power Block	
- Number of Reactor Modules	Three
- Net Electrical Output	465 MWe
- Steam Generator Number	Three
- Steam Generator Type	Helical Coil
- Steam Cycle	Saturated
- Turbine Type	1800 rpm, Tandem Compound Four Flow - 38-inch Last Stage Bucket
- Turbine Throttle Conditions	965 psia/540°F
- Feedwater Temperature	420°F
Reactor Module	
- Thermal Power (Core)	471 MWt
- Primary Sodium Inlet/Outlet Temperature	640°F/905°F
- Primary Sodium Flow Rate	46,000 gpm
- Intermediate Sodium Inlet/Outlet Temperature	540°F/830°F
- Intermediate Sodium Flow Rate	41,250 gpm
Reactor Core	
- Fuel	U-Pu-Zr Metal (Oxide Backup)
- Refueling Interval	18 Months (12 Mo. for oxide backup)
- Compound System Doubling Time for Breeding	~100 Years

For the reference design, a central fuel cycle facility is located at a remote site away from the ALMR plant. Capability for an optional co-located fuel cycle facility within the fenced plant area has been retained.

The intermediate heat transport system (IHTS), steam generator system (SGS), and related structures, which are not designated nuclear safety-related, are located outside the high security fence in the BOP area. These facilities and systems are designed and built to high quality industrial standards.

Other BOP buildings include the following facilities and equipment: the power conversion and support systems equipment, facilities for plant administration, training, and security, personnel access control, and laboratory, maintenance, auxiliary, and storage facilities for operation of the plant. Each of the BOP buildings, except the steam generator buildings and the IHTS pipe tunnels, are classified as Seismic Category III structures. The IHTS pipe tunnels and steam generator buildings are classified as Seismic Category II structures.

Several NI and BOP facilities will be constructed with factory fabricated modules in order to minimize cost. These facility modules include: structural steel; equipment; piping; heating, ventilating, and air conditioning (HVAC) ducting; cable trays; conduit; wiring and cable; and roof deck and siding which function as exterior walls. For the reference ALMR design, the maximum facility module size is 15 feet wide by 15 feet high and approximately 80 feet long. The maximum facility module weight is about 300 tons.

G.2.1.2 Reactor Module

The reactor module consists of the reactor vessel, reactor closure, containment vessel, internal structures, internal components, reactor module supports, and reactor core. Figure G.2.1-3 shows the reactor module installed in its facility and Figure G.2.1-4 shows the reactor module and its internals. The reactor module is located below grade in a concrete reactor silo. The outermost structure of the reactor module is the containment vessel which is made of one-inch thick 2-1/4Cr-1Mo steel. The reactor vessel is made of two-inch thick 316 SS. A five-inch gap between the reactor vessel and the containment vessel is filled with argon at about 12 psi above the reactor cover gas pressure. The reactor cover gas is helium at a pressure of about one atmosphere at normal power conditions; during power operation the reactor is hermetically sealed. The reactor closure at the top is a 12-inch thick 304 SS plate with a single rotatable plug and penetrations for the reactor equipment and primary sodium and cover gas service lines. Primary sodium purification is accomplished during reactor shutdown by a single cold trap system in each power block. There are no penetrations in the reactor vessel or the containment vessel. The reactor vessel is butt-welded to a skirt that is integral with the underside of the closure. The containment vessel is bolted to the closure and sealed by welding. The reactor module is supported entirely at the top by bolted brackets which transfer the load to the seismically isolated head access area (HAA) floor structure.

The reactor core is supported by a redundant beam structure attached at the bottom and the sides of the reactor vessel. A core support cylinder, extending from the core inlet plenum to an elevation above the core, has storage racks attached to its inner surface for storage of up to 30 spent fuel and blanket assemblies. A redundant structure designed to retain molten fuel is located immediately below the core inlet plenum. Two intermediate heat exchangers (IHX) and four 11,500 gpm electromagnetic (EM) primary pumps are suspended from the reactor closure. In addition, six control rod drives (CRD), an ultimate shutdown system (USS) assembly, in-vessel instrumentation, and an in-vessel transfer machine (IVTM) for refueling are also suspended from the rotatable plug in the closure. The closure mounted components are removable from the top for inspection, repairs, and replacement.

The reactor module is about 62 feet high and slightly under 20 feet in diameter. The module, less removable components, weighs about 640 tons and is capable of rail shipment using an existing special car (a 36-axle, 880-ton capacity Schnabel car). The reactor modules will be fabricated in a factory and transported to a particular site by the most economical combination of barge, rail, or road.

G.2.1.3 Core and Fuel

The reference ALMR core is a heterogeneous, metal alloy fuel design with 199 assemblies: 42 fuel assemblies, 24 internal blanket assemblies, 33 radial blanket assemblies, 42 reflector assemblies, 48 shield assemblies, 3 gas expansion modules (GEMs), 6 control assemblies, and 1 ultimate shutdown assembly. This configuration is shown in Figure G.2.1-5. Table G.2.1-2 summarizes the overall core design parameters. The core is designed to produce 471 MWt with an average temperature rise of 265°F. The inlet temperature is 640°F and the bulk outlet temperature is 905°F. The core height is 53 inches and there are no upper or lower axial blankets.

A heterogeneous arrangement of blankets and driver fuel is used, with six control rod locations, as shown in Figure G.2.1-5. The reference fuel for the equilibrium ALMR core is a metallic alloy of U-27%Pu-10%Zr. A single enrichment is used for the fuel assemblies. The blanket alloy is depleted U-10%Zr. The ferritic alloy HT9 is used for cladding and channels to minimize swelling associated with long burnups. At equilibrium, the design basis refueling interval follows 18 months of operation, with one-third of the driver fuel and one-fifth of the blankets being changed at each refueling outage. The fuel has a 4.5-year life (135 MWd/kg peak burnup); the blankets have a life of 7.5 years. Metal fuel provides excellent negative reactivity feedback for loss of cooling and transient overpower events. Metal fuel also provides competitive fuel costs.

Forty-two reflector assemblies are located at the core perimeter. Each reflector assembly consists of 61 Inconel 600 rods within the HT-9 assembly duct. The reference core has been designed with reflectors and without axial blankets so that excess Pu is not produced; breeding in the reference core is close to breakeven. The core is designed for the addition of fertile materials to increase breeding should the design goals be changed.

Table G.2.1-2

REFERENCE CORE DESCRIPTION AND OPERATING CONDITIONS

Core Thermal Power (MWt)	471
Reactor Mixed Mean Outlet Temperature (°F)	905
Reactor Temperature Rise (°F)	265
Core Height (in.)	53
Number of Core Enrichment Zones	1
Core Configuration	Radial Heterogeneous
Number of Assemblies in Core	
Core Fuel	42
Internal Blanket	24
Radial Blanket	33
Reflector	42
Shield	48
Control	6
Gas Expansion Modules	3
Ultimate Shutdown	1
Total	199
Plant Capacity Factor	0.85
Refueling Interval (months)	18
Number of Batches:	
Core Fuel	3
Internal Blanket	2*
Radial Blanket	3*
Assembly Structural Material	HT-9
Duct Pitch (in.)	6.282
Duct Gap (in.)	0.175
Fission-Gas Plenum Location	Upper
Fission-Gas Plenum Length (in.)	74.0
Above-Core Load Pad Length (in.)	6.0
Above-Core Load Pad Thickness (in.)	0.2225
Top Load Pad Length (in.)	4.0
Load Pad Gap (in.)	0.010
Total Core Mass Flow Rate (lbm/hr)	1.99×10^7
Flow Split (%)	
Core Fuel	62.45
Internal Blanket	15.14
Radial Blanket	17.70
Reflector and Shield	1.08
Control	0.75
Bypass	2.87

* Blankets are shuffled every cycle to achieve a lifetime of 7.5 years

Forty-eight shield assemblies, each consisting of seven boron carbide pins within the HT-9 assembly duct, are provided to prevent excessive irradiation damage to reactor structures and components surrounding the core. The shield assemblies also limit activation of intermediate sodium and materials carried in the reactor vessel auxiliary cooling system (RVACS) air circuit.

Three gas expansion modules (GEM) are located at the periphery of the active core. A GEM is the same external size and configuration as the ducts on the other core assemblies. The GEMs are filled with inert gas and sealed at the top. Each GEM communicates with the inlet plenum through an opening in the nose piece. With the primary pumps on, the high pressure in the inlet plenum compresses the gas captured in the GEMs and raises the sodium level in the GEMs to a region above the active core. When the pumps are turned off, the gas expands, displacing the sodium in the GEMs to a level below the active core. This change in sodium level introduces significant negative reactivity and limits the peak temperatures attained during loss of flow events. The GEMs enhance the ALMR capability of safely withstanding severe undercooling accidents without scram, including loss of all cooling by the IHTS from a full power condition.

Mixed uranium-plutonium oxide fuel is an alternative for the ALMR. This alternative can replace the reference metal fuel in the same basic core volume without requiring changes in the reactor structure or refueling system equipment.

Plutonium for the prototype and the commercial plant startup is planned to come from reprocessing light water reactor (LWR) spent fuel. The minor actinides, primarily Np, Am, and Cm, in the LWR spent fuel will be included with the Pu to produce the initial core and first two reload cores for the ALMR. Subsequent reload cores will be produced at the fuel cycle facility from reprocessed LMR spent fuel and blanket assemblies. In the ALMR hard neutron spectrum, the actinides largely fission as part of the fuel, creating thermal energy while being reduced to shorter-lived fission products. Ultimately, the fission products are removed from the fuel cycle as waste products whose radioactive lives for biological toxicity will be less than their source natural uranium in a few hundred years.

G.2.1.4 Reactivity Control and Shutdown

Reactivity control for normal operations of startup, load following, and shutdown is accomplished by a system of six identical control rods arranged as shown in Figure G.2.1-5. The six control rods provide scram diversity and shutdown redundancy. A stepping motor, controlled by the plant control system (PCS), actuates a lead screw to insert and withdraw the absorber for normal operation. The PCS actuates only one control rod at a time as shown schematically in Figure G.2.1-6.

Each control unit consists of a drive mechanism, a driveline, and a control assembly (absorber bundle and outer duct). Each control rod unit provides two diverse means of scramming the absorber bundle. For rapid emergency shutdown (scram), the Class 1E reactor protection system (RPS) causes the electromagnets on all six control rod assemblies to de-energize which opens the mechanical latch and allows the absorbers to drop into the core by gravity. Unlatch time is less than 0.2 seconds with full stroke insertion accomplished in about two seconds. The second means is by an irreversible high speed drive-in motor controlled by the RPS from an uninterruptible power supply. High speed drive-in is initiated at the same time as latch release and can exert up to 2000 pounds to force the absorber into the core. Fast drive-in provides a full stroke insertion time of 18 seconds. Each of the six rods has sufficient worth for reactor shutdown, providing a six-to-one redundancy.

A Class 1E electronically positioned mechanical rod stop system (RSS) prevents the unprotected rod withdrawal event from exceeding 0.40$ reactivity insertion. The ALMR core and fuel can safely accommodate this reactivity addition without fuel melting or sodium voiding; the consequences of this unprotected transient overpower (UTOP) event are benign. In practice, the rod stops will be set at a lower limit to accommodate uncertainties and improve margins.

The conceptual design of the electronic system for positioning the mechanical rod stop is illustrated in Figure G.2.1-6. Components in the rod stop system include a redundant, Class 1E controller, a rod stop drive selector, and a limited capacity power supply which controls power to each of the six rod stop adjustment drive motors, one for each control rod. The rod stop block, attached to an Acme screw and positioned by the drive motor, is sized to prevent the control rod from exceeding the preset stop position. Redundant, absolute position sensors are used to determine control rod and stop positions. The rod stop controller is separate from the reactor protection system (RPS) controller. The RSS obtains reactor power and absolute control rod position data from the redundant class 1E sensors through the RPS controller. The RSS is activated by operations only as required to adjust the rod stop position. The actual rod stop position is determined automatically by the controller with a permissive required by the operator which enables the stop to be repositioned.

The unprotected loss of flow (ULOF), loss of heat sink (ULOHS), and transient overpower (UTOP) events without scram result in the reactor inherently and passively achieving a stable condition at temperatures below design limits. This condition can be safely sustained until operator action is taken to bring the reactor to a cold, subcritical shutdown condition. To bring the reactor to a cold shutdown condition in the unlikely event the PCS and the RPS have both failed, a manually activated ultimate shutdown system (USS) is provided which releases B_4C absorber material into a center core assembly. The USS provides a diverse and reliable method of shutting down the reactor in the unlikely event the normal control rods fail to insert when required. The USS contains B_4C balls stored in a dry canister within the reactor vessel above the sodium. Upon actuation, the balls fall freely down a guide tube into a center core assembly. The worth of the inserted B_4C balls is sufficient to bring the reactor from 135% of full power (which includes the all-rod UTOP event) to a cold shutdown.

G.2.1.5 Containment

The ALMR containment, shown in Figures G.2.1-3 and G.2.1-7, provides a low leakage, pressure retaining boundary which completely encloses the reactor coolant boundary. It consists of a lower containment vessel surrounding the reactor vessel and an upper containment dome over the reactor closure.

The upper portion of the containment is a 48-foot diameter cylindrical steel (SA516 Grade 70) dome which extends between the reactor closure and the tornado hardened roof structure of the reactor facility located at grade. The upper containment dome region is designed to limit leakage to less than 1% per day at 25 psig and 700°F. This region is designed to mitigate accidents which release radionuclides through the reactor closure. Manned access to the reactor containment dome is accomplished through a personnel air lock. All piping and instrument penetrations are through the containment dome and are above the coolant boundary and the operating sodium level. IHTS piping penetrations are provided with bellows to accommodate thermal expansion. An isolation valve is provided in each of the 20-inch IHTS pipes at the exterior of the containment dome.

The lower portion of the containment consists of a one-inch thick, 19-foot 10-inch diameter, 2-1/4 Cr-1Mo steel vessel. The containment vessel has no penetrations and is designed to remain essentially leak tight at 60 psig and 800°F. A five-inch annulus between the reactor and containment vessels is sized to retain the primary sodium such that the reactor core, the stored spent fuel, and the inlets to the intermediate heat exchangers will remain covered in the event of a reactor vessel leak. This ensures that the internal sodium flow path will not be interrupted and shutdown heat removal via RVACS will maintain safe temperatures within the core and reactor system following a postulated reactor vessel leak. The argon filled annulus is maintained at a higher pressure than the reactor cover gas, which is at atmospheric pressure, and is continuously monitored with pressure sensors, sodium ionization detectors, and sodium liquid detectors for early warning of any leak in either vessel.

A schematic of the primary system and containment boundaries is shown in Figure G.2.1-8. The primary system boundary includes the reactor vessel, reactor closure, closure penetrations, below-head ducting of the two IHX units, and the primary sodium and cover gas clean-up system piping up to and including the first isolation valve (immediately outboard of the reactor closure). During power operation, all sodium and cover gas service lines are closed with double isolation valves and all other penetrations in the reactor closure are seal-welded. Thus, the primary system operates in a totally sealed manner during power operation.

Extremely severe accidents that could challenge the containment have been evaluated on a probabilistic basis. These assessments indicate that the risk of a primary system boundary breach by these events is less than 10^{-9} per reactor year. There are two major factors in achieving this low probability: 1) an excellent ability to prevent severe accidents due to the high reliability of the reactor protection system, the inherent negative reactivity feedback characteristics, the invulnerability of the passive safety grade shutdown heat removal system (RVACS), and the large temperature and structural design margins, and 2) the intended ability of the design to withstand the effects of extreme accidents involving gross core melting or abrupt internal energy releases from prompt critical reactivity excursions (hypothetical core disruptive accidents, HCDA) without breaching the primary coolant and cover gas boundary. Preliminary calculations show that the primary system can be expected to contain, without breach, energetic events producing in the order of 500 MJ of work energy. This level is more than an order of magnitude greater than the anticipated energetics from any conceivable HCDA. Nevertheless, to provide in-depth defense, the steel containment described above has been provided and is being designed to mitigate a reactor closure breach caused by a HCDA.

The design basis event for the containment dome assumes that: a relatively large breach in the reactor closure has been created by a HCDA, that 100 % of the noble gas (Xe, Kr), 0.1 % of both the volatile solids (Cs, Rb) and halogens (Br, I), and 0.01 % of the fuel is instantly released to the containment volume. In addition, it is assumed that the breach in the reactor closure is large enough to allow the He cover gas to escape into the containment and air to enter the reactor cover gas region which initiates a sodium pool fire that continues until all the oxygen in the containment has been consumed. Preliminary analysis has shown that the peak pressures and structural temperatures produced as a consequence of this event are far below the design conditions (13 psig vs 25 psig, and 250°F vs 700°F) and that the radiological consequences are below the protective action guidelines (PAG) limits (<1 REM at the site boundary for 36 hours) and far below 10CFR100 limits.

A secondary containment is provided during refueling or maintenance operations by a portable refueling enclosure, which is moved over the reactor, and a standby gas treatment system is then activated to maintain a negative pressure within both the refueling enclosure and the upper portion of the containment dome. The standby gas treatment system contained within the refueling enclosure provides an aerosol decontamination factor of 100. Figure G.2.1-9 depicts both the primary coolant and containment boundaries which are employed during refueling and maintenance activities. Prior to performing refueling and maintenance operations, the reactor is shut down, the primary sodium is cooled to 400°F, and the cover gas is replaced. Fuel and equipment removal and replacement operations are accomplished using dual isolation valves, one on a transfer adapter and one on the fuel or equipment transfer cask. This ensures that the integrity of the primary system boundary is always maintained. Access for refueling and equipment removal and replacement is provided by four removable ports in the head of the steel containment dome. Fuel transfer operations and control rod drive line replacement is accomplished through 24-inch diameter ports and IVTM and sodium purification pump replacement is accomplished through 36-inch diameter ports. Replacement of the primary pumps and IHX units will require cutting and re-welding the raised 24-foot diameter center portion of the domed structure. However, this will be an infrequent operation since the IHX is expected to last the life of the plant and the EM pumps are expected to be replaced only once (after 30 years).

G.2.1.6 Seismic Isolation

The reactor module, reactor vessel auxiliary cooling system (RVACS), head access area (HAA) and containment structures, and the reactor protection system and EM pump electrical equipment vaults are supported by an arrangement of 31 seismic isolators located as shown in Figure G.2.1-10. The isolator system reduces the horizontal seismic accelerations that are transmitted to the reactor module by a factor of more than three and facilitates adaptation of the standard ALMR design to the seismic conditions of a large range of sites by adjustment of the isolator characteristics.

Each isolator is an assembly of steel plates laminated with layers of a natural rubber compound and encased in rubber as illustrated in Figure G.2.1-10. A doweled configuration is used for the seismic isolators in the reference design. The reactor is very stiff in the vertical direction, so vertical isolation is not needed.

The safe shutdown earthquake (SSE) requirement for the ALMR is 0.3g and the operating basis earthquake (OBE) requirement is 0.15g. To provide additional margins, the safety related equipment, systems, and structures and the IHTS and SGS are being designed for a 0.5g peak ground acceleration earthquake. This includes the reactor module and facility, and NI facilities which are designated Seismic Category I. The remaining non-safety related systems and structures (designated Seismic Category II and III) are designed to the Uniform Building Code (UBC) requirements with a 0.17g peak ground acceleration earthquake. Seismic Category II structures are evaluated for a 0.5g earthquake and strengthened if necessary to ensure failure will not adversely affect safety related systems and functions.

G.2.1.7 Intermediate Heat Transport System

The intermediate heat transport system (IHTS) for each reactor module, shown schematically in Figure G.2.1-11, consists of piping and components required to transport the reactor heat from the primary system, through the intermediate heat exchanger, to the steam generator system (SGS).

The IHTS is a closed loop system with an expansion volume that is integral to the steam generator and argon cover gas to accommodate thermally induced system volume changes. Intermediate sodium is circulated by a constant speed mechanical pump, located in the cold leg of the loop, through the tube side of the IHX and the shell side of the steam generator. A permanent magnet flowmeter located in the cold leg monitors sodium flow in the loop. The major components within the head access area (HAA) consist of the main loop hot and cold leg piping and vents. Guard pipes surrounding the IHTS pipes prevent intermediate sodium leakage into the HAA by containing the sodium. The guard pipes are sealed at the reactor closure and the containment wall. Safety grade isolation valves, provided in each of the 20-inch IHTS pipes immediately outboard of the containment dome, can be closed to isolate the IHXs from the SGS in the unlikely event of a sodium-water reaction in the SG, and to complete the containment boundary.

The arrangement and relative elevation of the IHTS piping and components, shown in Figures G.2.1-12 and G.2.2-13, are designed to promote natural circulation for decay heat removal. The initial natural circulation rate following shutdown from normal full power operating conditions is 9% of normal flow.

The main IHTS piping includes gimballed bellow joints to accommodate thermal expansion and differential motion in both the horizontal and vertical directions arising from relative motions between the seismically isolated reactor module and the non-isolated steam generator building. Rigid supports restrain in the vertical and horizontal directions while spring hangers support the dead loads.

Hot leg sodium exits the two IHXs from separate 20-inch 304 stainless steel pipes and is merged at a tee within the pipe tunnel into a 30-inch pipe leading to the steam generator. The cold leg piping arrangement is similar to the hot leg but is located above the hot leg for ease of maintenance.

Sodium enters the steam generator at 830°F and exits at 540°F. Sodium flow in the IHTS is provided by a 41,250 gpm centrifugal pump located in the steam generator facility. The pump, located in the cold leg, is a vertically oriented single stage, double suction, free surface centrifugal pump driven by a 2700 hp constant speed induction motor. An auxiliary pony motor provides 10% flow for decay heat removal during low power or standby conditions. The pump design is similar to that developed and tested for the Clinch River Breeder Reactor Plant (CRBRP). The relative elevations of the reactor module and the steam generator are such that during shutdown conditions the IHTS sodium will naturally circulate at a flow rate sufficient to remove decay heat from the reactor. The IHTS includes a sodium leak detection system to provide early warning of any sodium leaks.

In the event of a steam generator tube leak, the sodium water reaction pressure relief subsystem (SWRPRS) provides overpressure protection of the IHTS and IHXs. The SWRPRS consists of two safety-grade 28-inch rupture discs in series, a reaction products separation tank (RPST), two sodium dump tanks, a vent stack, and a hydrogen ignitor. The SWRPRS dumps the IHTS sodium, except the IHX inventory, and simultaneously initiates a rapid water-side blowdown of the steam generator system. The SWRPRS has the capability to expel the sodium and sodium-water reaction products from the shell side of the SG within 30 seconds of a three-tube guillotine size failure, while reducing the shell side pressure from the 300 psig rupture disk setpoint to less than 100 psig. In addition, safety-grade isolation valves in each of the 20-inch IHTS lines are closed to ensure that the steam-sodium interface within the IHTS cannot be driven backward into the IHX under any condition. Analysis has shown that if all the tubes in the SG eventually fail due to a postulated failure of the steam-side isolation system, the backpressure in the SG shell will not exceed 700 psi. The IHX secondary and the IHTS piping are capable of withstanding the full 1000 psi steam pressure.

The RPST is a vertically oriented 14-foot diameter, 23-foot long low alloy steel tank. Liquid and solid reaction products, along with displaced sodium, flow through a 30-inch SWRPRS line, and rupture discs, connecting the steam generator lower head to the RPST. There the gaseous reaction products are separated and the liquid and solid products are drained into the dump tanks through two 24-inch drain lines.

The two 14-foot diameter, 33-foot long carbon steel dump tanks are interconnected by two 24-inch sodium equalization lines and one 30-inch gas equalization line as shown in Figure G.2.1-14. The gaseous products are released and burned through the stack and flare tip ignitor into the atmosphere and the system is flooded with nitrogen.

G.2.1.8 Steam Generator System

The steam generator system (SGS) is comprised of the steam generator, steam drum, recirculation pump, leak detection subsystem, and water dump subsystem. There is one steam generator system for each reactor module. Three steam generators are headered together to feed a single turbine-generator system in each power block.

The ALMR steam generator is a vertically oriented, helical coil, sodium-to-water counterflow shell-and-tube exchanger shown in Figure G.2.1-15. It is designed and fabricated to the requirements of the ASME B&PV Code Section VIII, Division 2. The unit is designed for 479 MWt and generates steam at 1000 psig and 545°F with steam/water in upflow on the tube side and 830°F inlet sodium in downflow on the shell side. The mass ratio of recirculated water to generated steam is 1.1:1 giving a steam outlet quality of 91%. Departure from nucleate boiling (DNB) effects are mitigated by the helical coil tube geometry and the high average water mass flux of 1.2×10^6 lbm/hr-ft^2 at full load operating conditions.

The steam generator consists of the helical coil tube bundle and support, feedwater inlet tube assembly, steam outlet tube assembly, sodium inlet distribution assembly and cover gas space with provision for sodium expansion. The steam generator material is 2-1/4 Cr-1 Mo steel. There are 323 single wall tubes, 1.25 inches outside diameter, 0.105 inch wall thickness, and 178 feet long, in the 20-foot high tube bundle. The overall size of the steam generator is 67 feet-2 inches high and 12 feet in diameter.

The steam generator includes an internal cover gas space to accommodate sodium expansion. Cover gas in the steam generator head mitigates the pressure transients during large sodium-water reaction events. A cover gas hydrogen meter in the upper head detects small sodium-water reactions within the steam generator under hot standby as well as normal operating and upset conditions.

The inner shroud serves as a bypass channel to equalize pressure differentials between the inlet and outlet sodium nozzles. The bypass flow channel and the low pressure drop in the tube bundle (0.6 psi at full load) ensure that steam cannot be forced down the hot leg pipe into the IHX by the differential pressure between the inlet and outlet sodium nozzles in the event the steam isolation valves fail to close. Thus, passive protection of IHX tubes from a worst case steam generator tube leak is provided.

The single wall helical coil concept is similar to the steam generators used in Japan (Monju plant) and Europe (SNR-300 and Super Phenix plants). A 76 MWt prototype helical coil steam generator, with 40 full length tubes and thermal/hydraulic parameters equivalent to those in the plant unit, has successfully completed 17,900 hours of testing at ETEC.

A 12-foot 9-inch diameter, 34-foot long horizontal steam drum is located 15 feet above at grade outside the steam generator building. Two stage separators and chevron dryers separate the water-steam mixture. Dry steam exits the steam drum through two 24-inch nozzles at a rate of 2.05 x 10^6 lb/hr.

In the event of a steam generator tube break, rapid depressurization is accomplished through a steam and water-side blowdown system which is initiated in conjunction with the sodium dump of the IHTS by the SWRPRS. Steam generator pressure is reduced from 1000 psig to 300 psig in less than 60 seconds. Two water dump valves direct the water and flashed steam mixture to a 14-foot diameter, 25-foot long, vertically oriented water dump tank.

Leakage of water/steam into the sodium stream is monitored by hydrogen diffusion leak detectors located in the main sodium outlet and vent lines. Redundant detectors are provided on each sodium line.

G.2.1.9 Turbine-Generator

The turbine generator for each power block is a 1,800 rpm, tandem compound four-flow unit with rated inlet steam conditions of 965 psia, 540°F and exhausting to two twin shell surface condensers at 2.0 inches Hga while extracting steam for six stages of feedwater heating. The turbine has a single flow high pressure casing and two double flow low pressure casings. The turbine is provided with moisture separator-reheaters, each with one stage of reheat.

G.2.1.10 Plant Control

The ALMR plant is controlled by a highly automated state-of-the-art digital control system especially designed to optimize control of the multi-module (three reactors and one turbine) power block configuration and provide well integrated NSSS and BOP operation under normal and off-normal conditions. The overall plant control function is performed by the plant control system (PCS) and the reactor protection system (RPS). The PCS and RPS are separate and independent systems totally isolated from each other as shown in Figure G.2.1-16. The RPS is a highly reliable Class 1E system, designed on a per reactor basis, that scrams the affected reactor automatically whenever the reactor safety limits are reached. RPS electronics for each reactor are located in a safety-grade, seismically protected vault adjacent to the reactor. The PCS is a plant-wide control system which provides reliable and efficient plant operation for high plant availability and investment protection. The PCS has no safety role. PCS electronics are throughout the plant with the main computers residing in the control building.

An overview of the ALMR plant control system is shown in Figure G.2.1-17. The architecture is hierarchical with highly distributed processing and features modern model-based controller technology. Plant data is transmitted using fiber optics and multiplexing systems. Intelligent processors are distributed throughout the plant to make control decisions and generate diagnostics based on inputs they receive from the plant and commands from higher level controllers. All controllers and data communication systems are redundant and fault tolerant. Interface with the control room operator and the administrative and technical support staff is through interactive CRT-based consoles connected to the plant data highway. Each power block has one operator console in the control room which receives all the processed information regarding the operating status of the block (and its modules), and from which all block operations are directed.

During normal load-following operation, all modules in a block are operated as a unit and all module power changes are equal. However, during refueling or during transients which limit power from a single module, power levels from unaffected modules are varied independently through supervisory control strategies to improve plant availability.

A Seismic Category I, tornado hardened, remote shutdown facility (RSF), connected by a tunnel to the control building, houses the remote shutdown console. Safety grade reactor scram and post-accident monitoring capabilities are provided in the RSF. The RSF also contains the safety grade operator interface for the rod stop system and the ultimate shutdown system. Battery backup ensures habitability conditions are maintained in the RSF for a minimum of 36 hours.

G.2.1.11 Shutdown Heat Removal

Reactor shutdown heat is normally removed by the turbine condenser using the turbine bypass (Figure G.2.1-2). An auxiliary cooling system (ACS) is provided for cases when, due to maintenance or repair needs, an alternative shutdown heat removal method is required. The ACS induces natural circulation of atmospheric air past the shell side of the steam generator. The ACS consists of an insulated shroud around the steam generator shell with an air intake through the annulus at the bottom and an isolation damper located above the steam generator building roof. Normal, natural circulation ACS operation is initiated by opening the exhaust damper. ACS operation in a natural circulation mode has the capability to maintain reactor temperatures well below design limits. To increase the heat removal rate and reduce maintenance outages, an auxiliary fan located in a separate exhaust stack, equipped with an isolation damper, may be activated. In the natural circulation mode, the natural exhaust stack damper is open while the forced exhaust stack damper is closed.

In the highly unlikely event that the intermediate heat transport system (IHTS) becomes unusable during power operation, for example because of a main sodium pipe break or sodium dump, the reactor will scram and the reactor vessel auxiliary cooling system (RVACS) will automatically come into full operation. Temperatures of the reactor sodium and reactor vessel will rise, increasing the radiant heat transfer across the argon gap to the containment vessel (95% by radiation) and the heat transfer from the containment vessel to the upwardly flowing atmospheric air around the vessel as depicted in Figure G.2.1-18. The temperatures and heat transfer by RVACS will continue to increase until equilibrium between reactor heat generation and RVACS cooling is established.

Figure G.2.1-19 shows calculated temperatures for RVACS cooling alone after loss of all cooling by the IHTS with reactor scram from full power. The temperatures are well below the "faulted" design limit of 1300°F and the margin to sodium boiling temperature in the core (1750°F) is large.

The RVACS stacks incorporate four separate inlets and exhausts and are tornado-hardened (Figure G.2.1-3). The redundant air inlet and exhaust ducting combined with substantial margins in the design make the RVACS extremely tolerant of accidental flow blockages and surface fouling. Because of its simplicity, passive operation, internal redundancy and resistance to operational failure, RVACS is the only shutdown heat removal system designated as safety-related in ALMR.

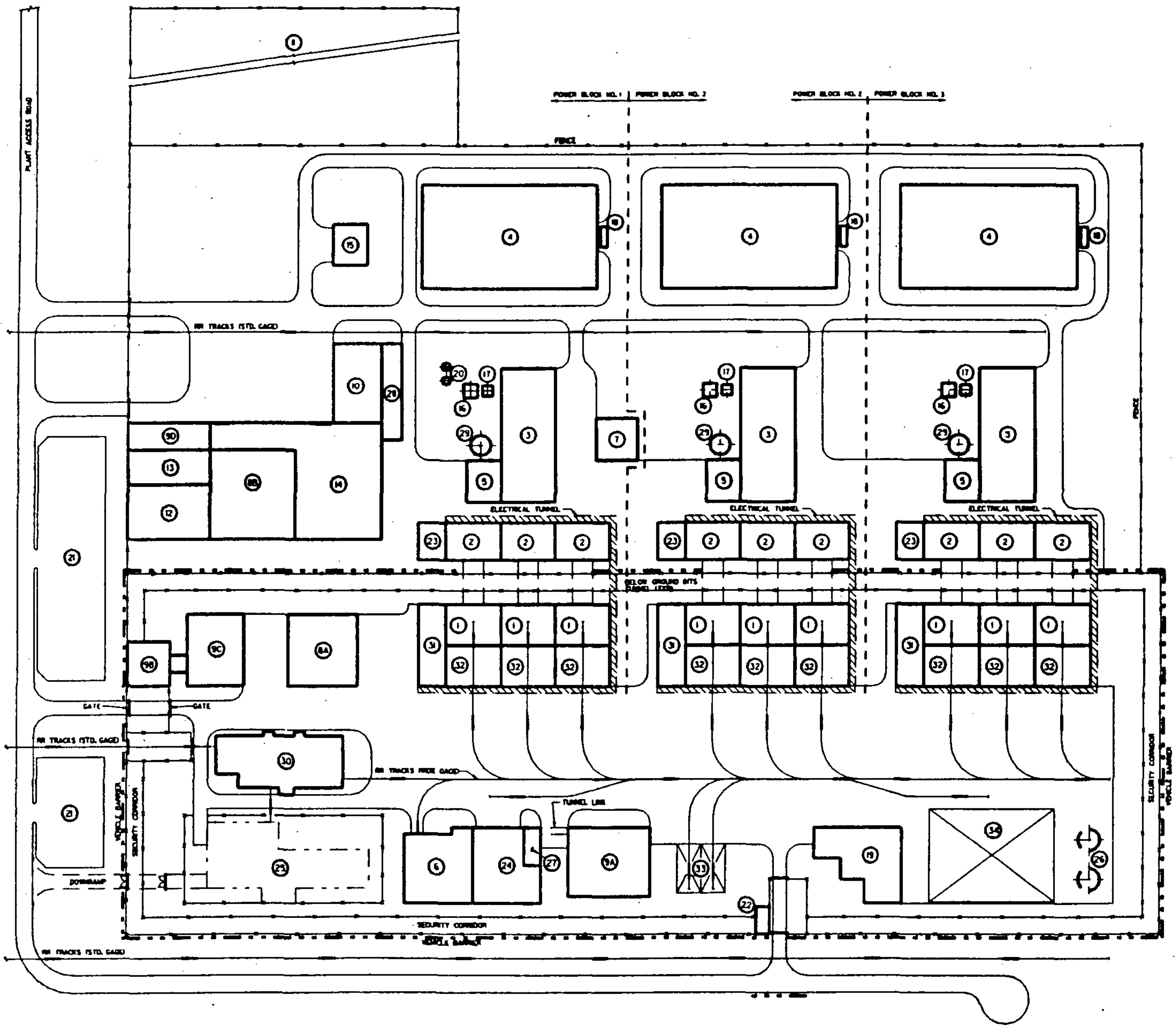
G.2.1.12 Refueling System

Reactor refueling occurs every 18 months. For refueling, the reactor is shut down and the sodium cooled to 400°F. A portable refueling enclosure with integral overhead crane, shown in Figure G.2.1-20, is rolled into place and secured over a port of the HAA above the reactor. The tornado hardened refueling enclosure is designed to provide a low leakage secondary containment when positioned over the reactor module. Redundant gas treatment systems, integral with the refueling enclosure, maintain a negative pressure at about 0.25 inches WG within the refueling enclosure during refueling and maintenance operations. Before refueling, the reactor helium cover gas is replaced and taken in shielded tanks on a transporter to the radioactive waste facility for analysis and cleanup. An adapter with floor valve is installed through HAA roof and the containment boundary penetrations and is attached to the transfer port in the reactor closure. The transfer cask on its transporter is positioned in the refueling enclosure and a leaktight connection is made to the adapter. The transfer cask is used to exchange spent fuel and other core assemblies in the reactor (six assemblies at a time) with new assemblies from the fuel service facility. Within the reactor, core assemblies are moved between the core, storage racks, and a transfer station below the transfer port by the in-vessel transfer machine. Fuel and blanket assemblies are allowed to decay in the reactor storage positions for one cycle before removal and transfer to the fuel service facility. A special hoist (straight pull type machine) inside the transfer cask raises and lowers core assemblies between the in-vessel transfer station and the transfer cask.

G.2.1.13 Fuel Cycle Facility

A central, off-site metal fuel cycle (reprocessing) facility is the reference approach for the commercial plant. Capabilities for an optional, co-located fuel cycle facility on-site are retained (Figure G.2.1-1). The fuel cycle facility reprocessing is based on the pyrometallurgical process being developed by Argonne National Laboratory. The optional on-site reprocessing facility could be operated either by the power plant owner-operator or by a separate organization.

LWR recycle Pu, including minor actinides, is used for the reference ALMR startup core and the first two reloads. Later ALMR cores are produced from the recycled LMR fuel processed by the central fuel cycle facility.



LEGEND

1. Reactor Facility
2. Steam Generator Facility
3. Turbine Generator Facility
4. Cooling Tower
5. Circulating Water Pump House
6. Reactor Maintenance Facility
7. Gas Turbine Facility

8A Warehouse (NI)
8B. Warehouse (BOP)
9A. Control Building
9B. Nuclear Island Guardhouse
9C. Nuclear Island Health Physics /Personnel Service Building
9D. Balance of Plant Guardhouse /Personnel Service Building

10. Balance of Plant Services Facility
11. Switchyard
12. Administration Building
13. Training Center
14. Maintenance Building
15. Sanitary Waste Treatment
16. Gen. Step Up Transformer
17. Auxiliary Transformer
18. Transformer
19. Assembly & Storage Facility
20. Station Service Transformer
21. Parking Lot
22. Construction & Outage Guardhouse
23. Inert Gas Storage
24. Radwaste Facility
25. Fuel Cycle Facility (Optional)
26. Spent Components Temporary Storage
27. Remote Shutdown Facility
28. Storage Tanks
29. Condensate Storage Tanks
30. Fuel Service Facility
31. Sodium Purification Vault
32. Electrical Equipment Vault
33. Refueling Enclosure & Cask Transporter Parking
34. Transilift Operating Area

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Figure G.2.1–1 ALMR POWER PLANT (3 Power Blocks)

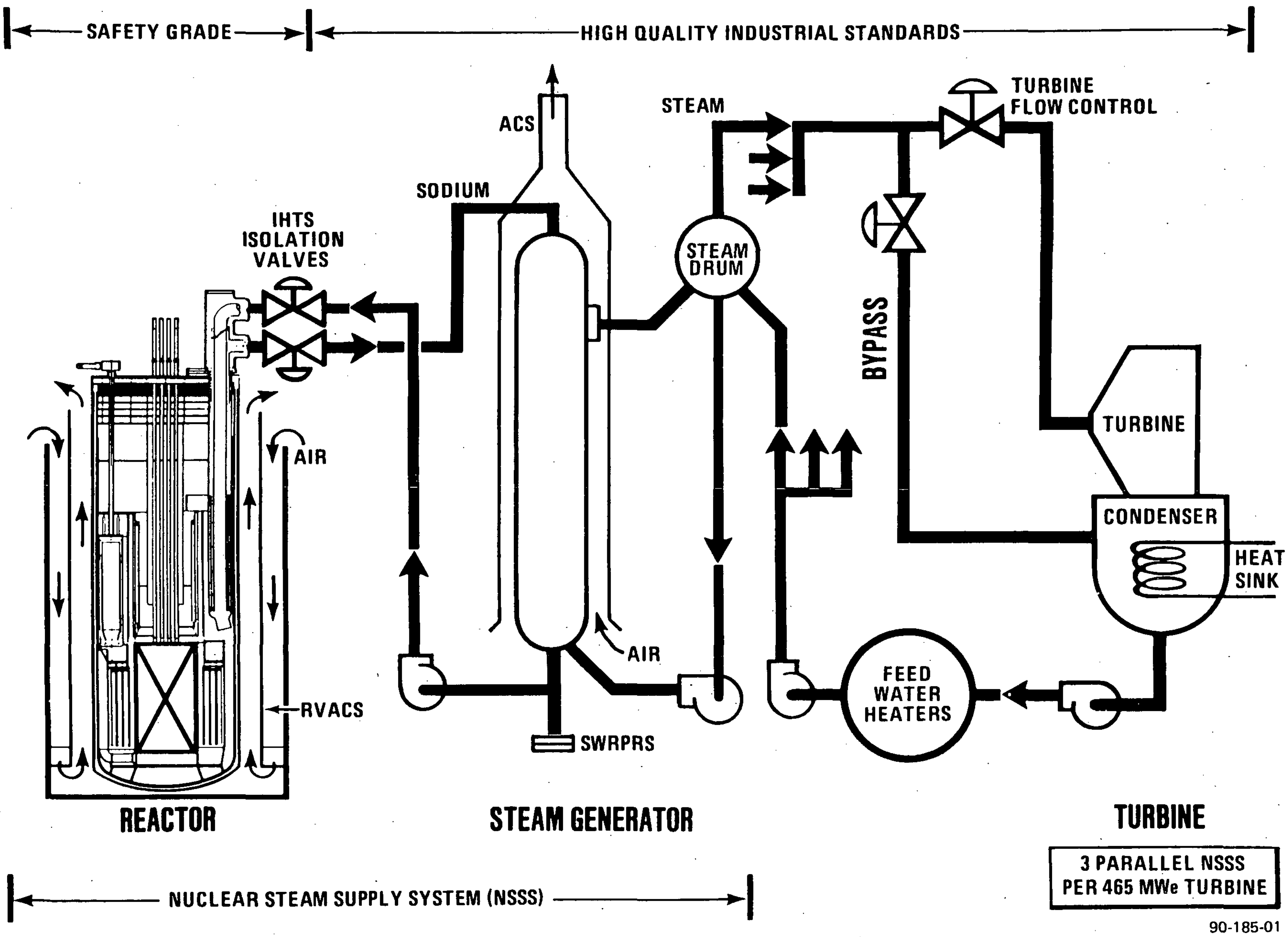

Figure G.2.1–2 ALMR MAIN POWER SYSTEM

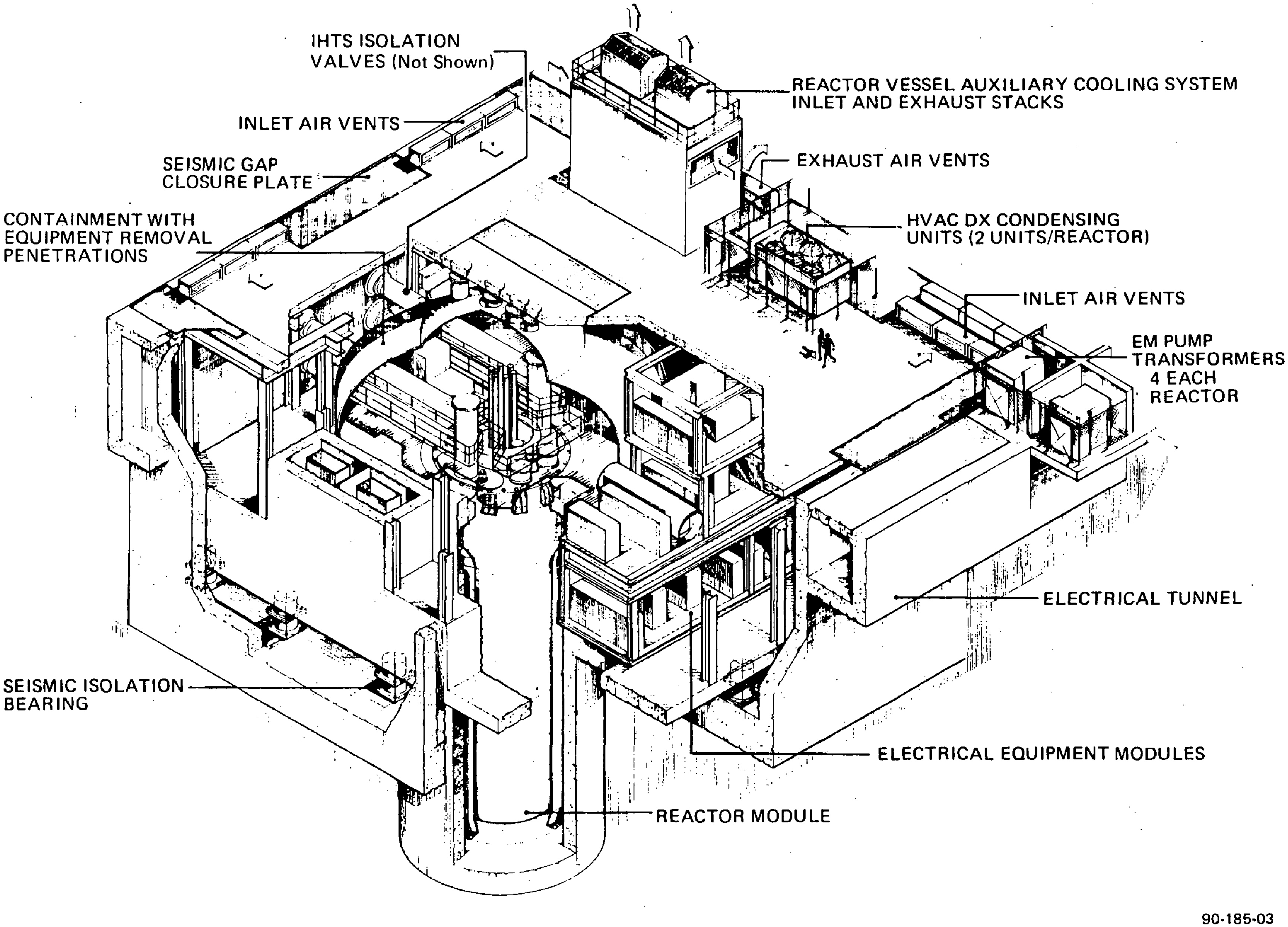

Figure G.2.1—3 REACTOR FACILITY GENERAL ARRANGEMENT

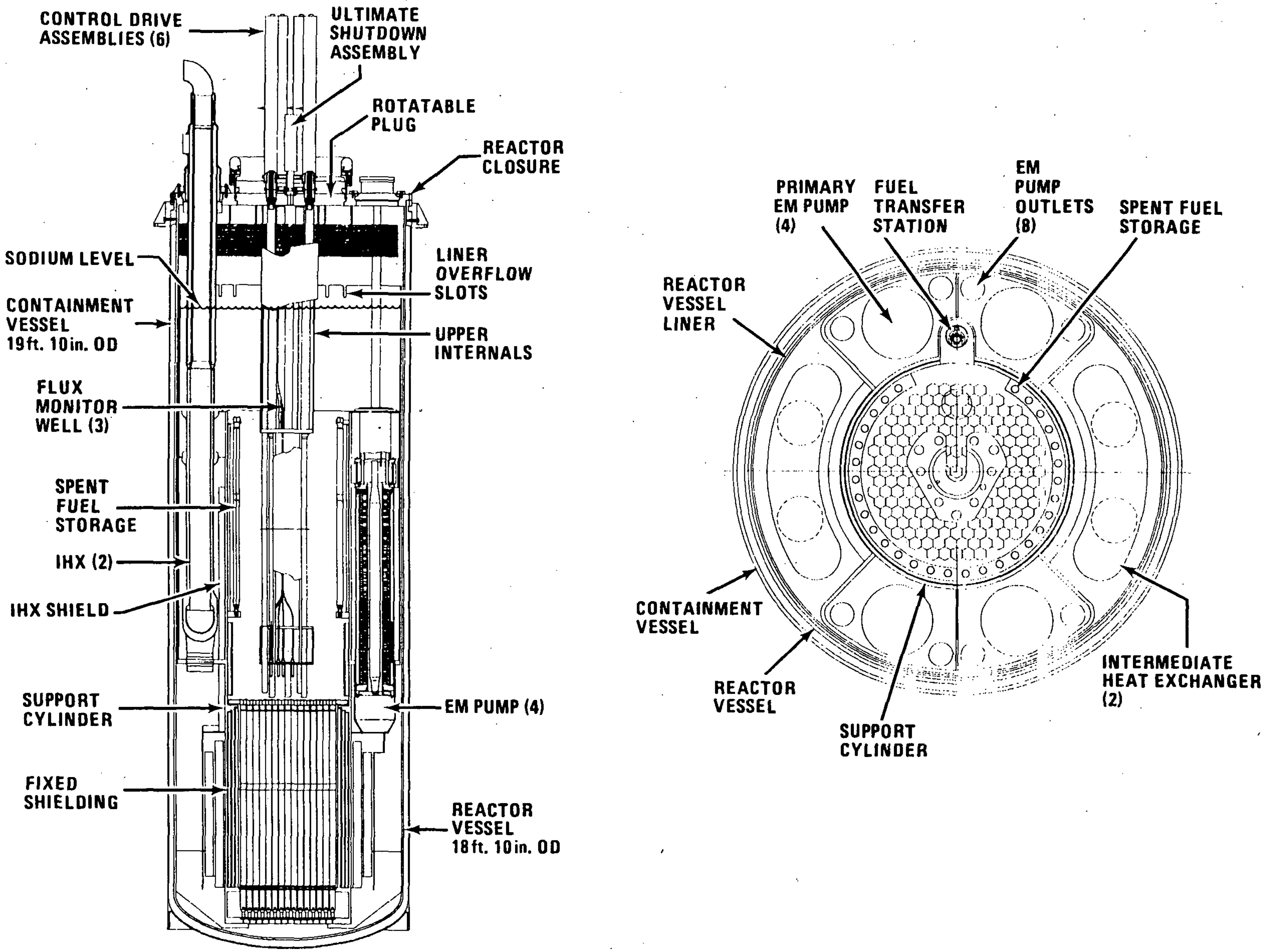

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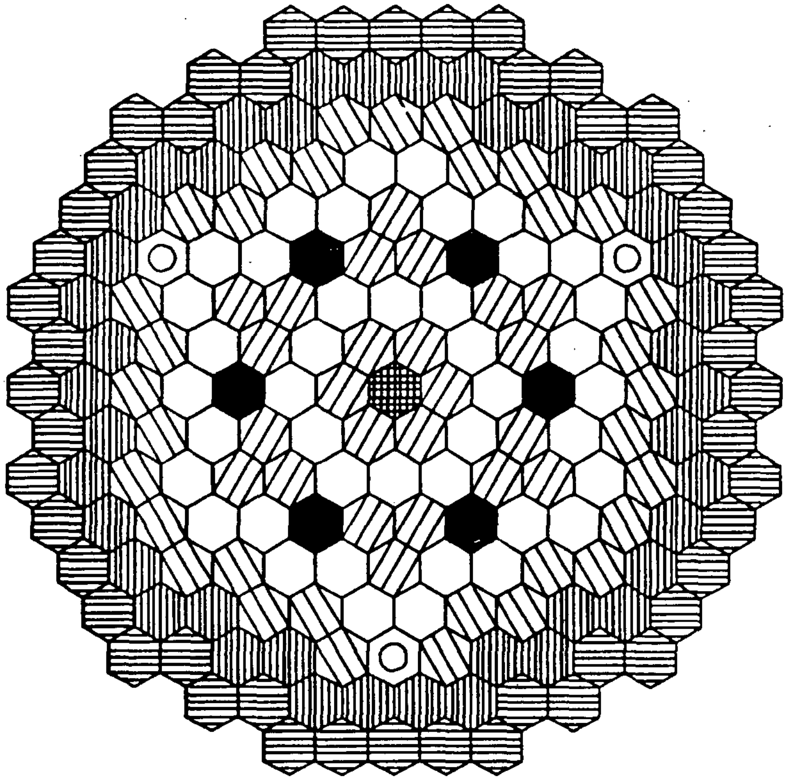
Figure G.2.1–4 ALMR REACTOR MODULE



DRIVER FUEL	42
INTERNAL BLANKET	24
RADIAL BLANKET	33
CONTROL	6
ULTIMATE SHUTDOWN	1
GAS EXPANSION MODULE	3
REFLECTOR	42
SHIELD	48
TOTAL	199

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Figure G.2.1–5 **REFERENCE METAL CORE**

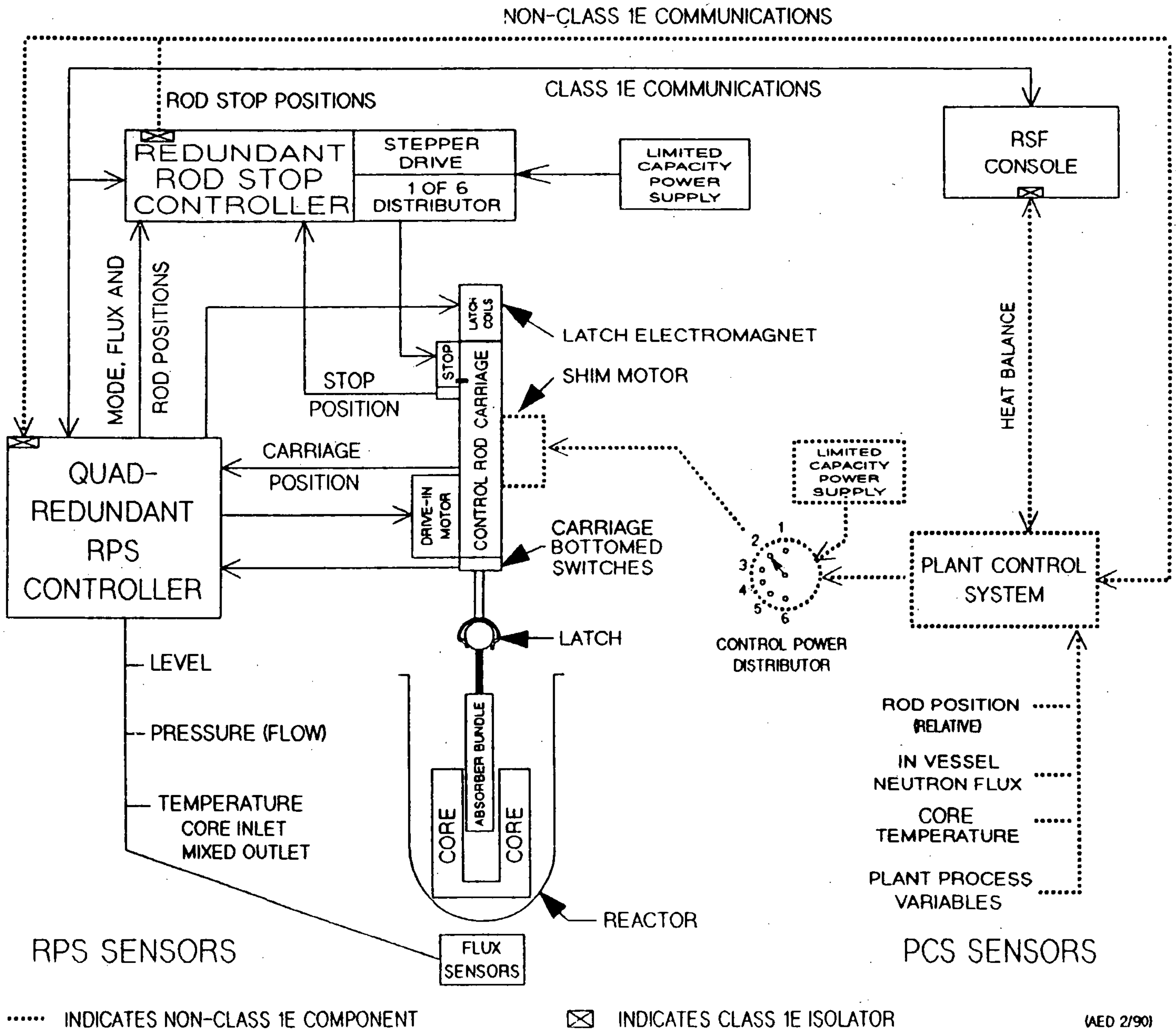

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Figure G.2.1—6 REACTIVITY CONTROL AND SHUTDOWN SYSTEM

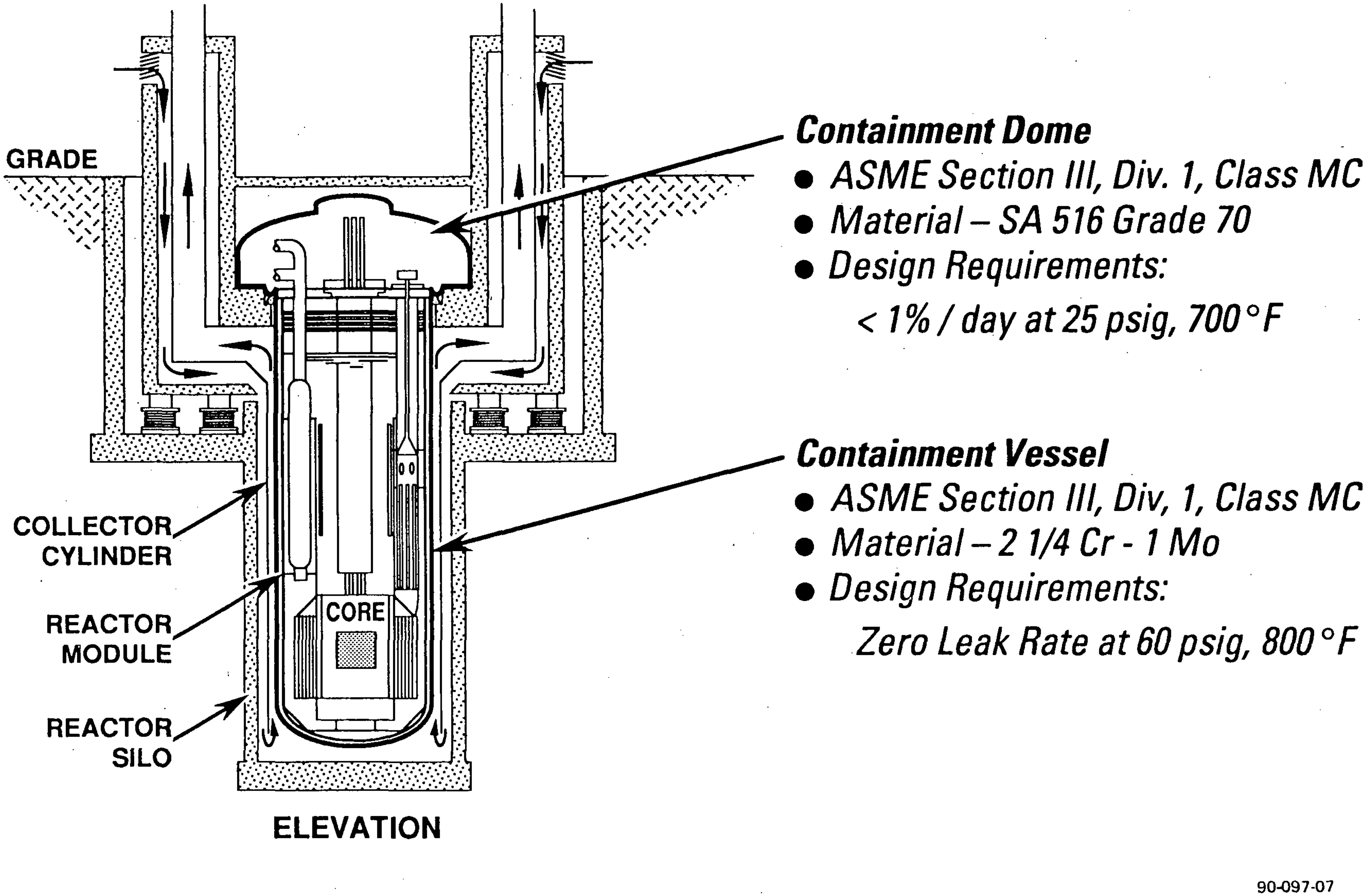

Figure G.2.1–7 REACTOR CONTAINMENT

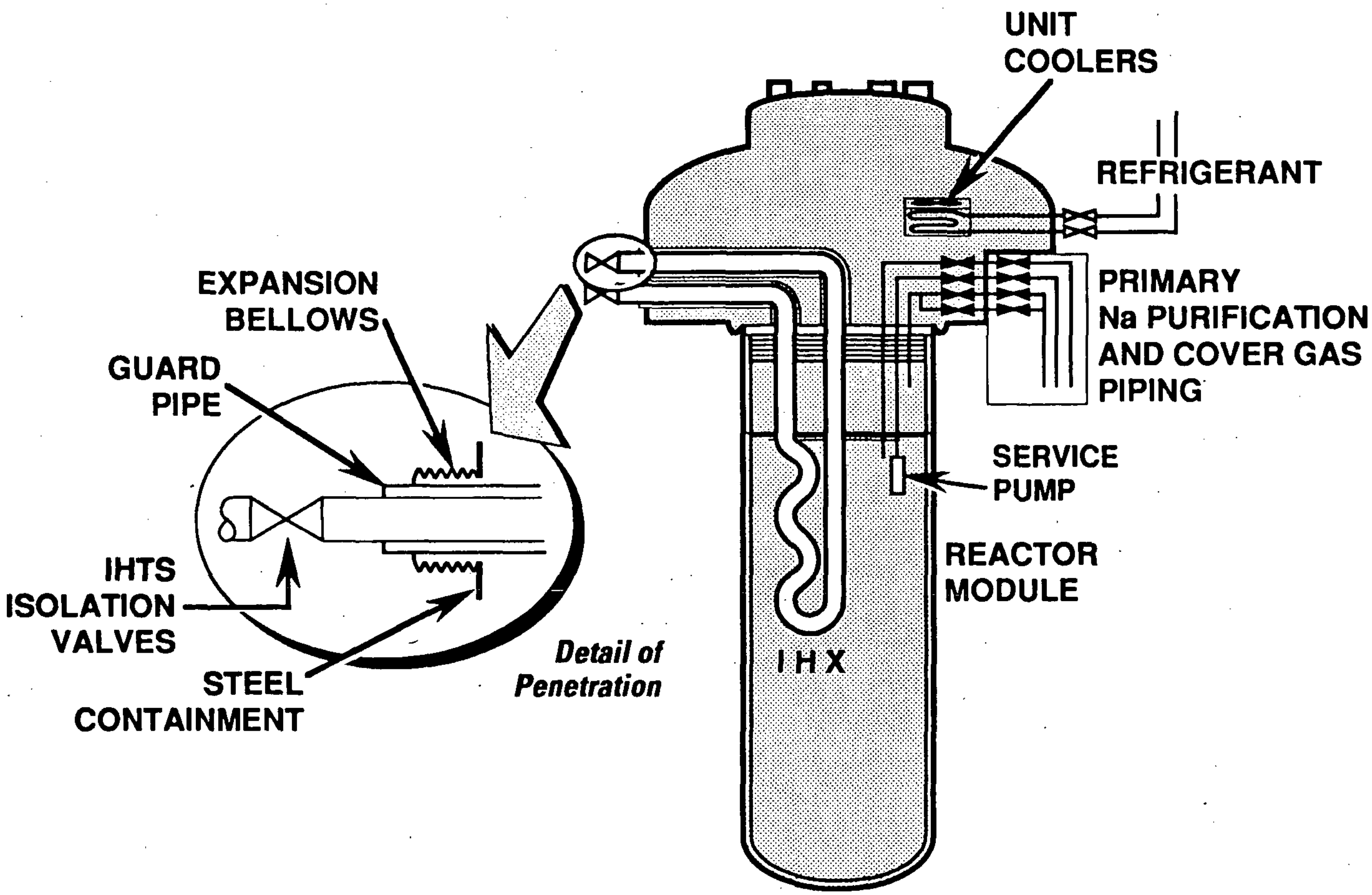

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Figure G.2.1–8 CONTAINMENT BOUNDARY DURING OPERATION

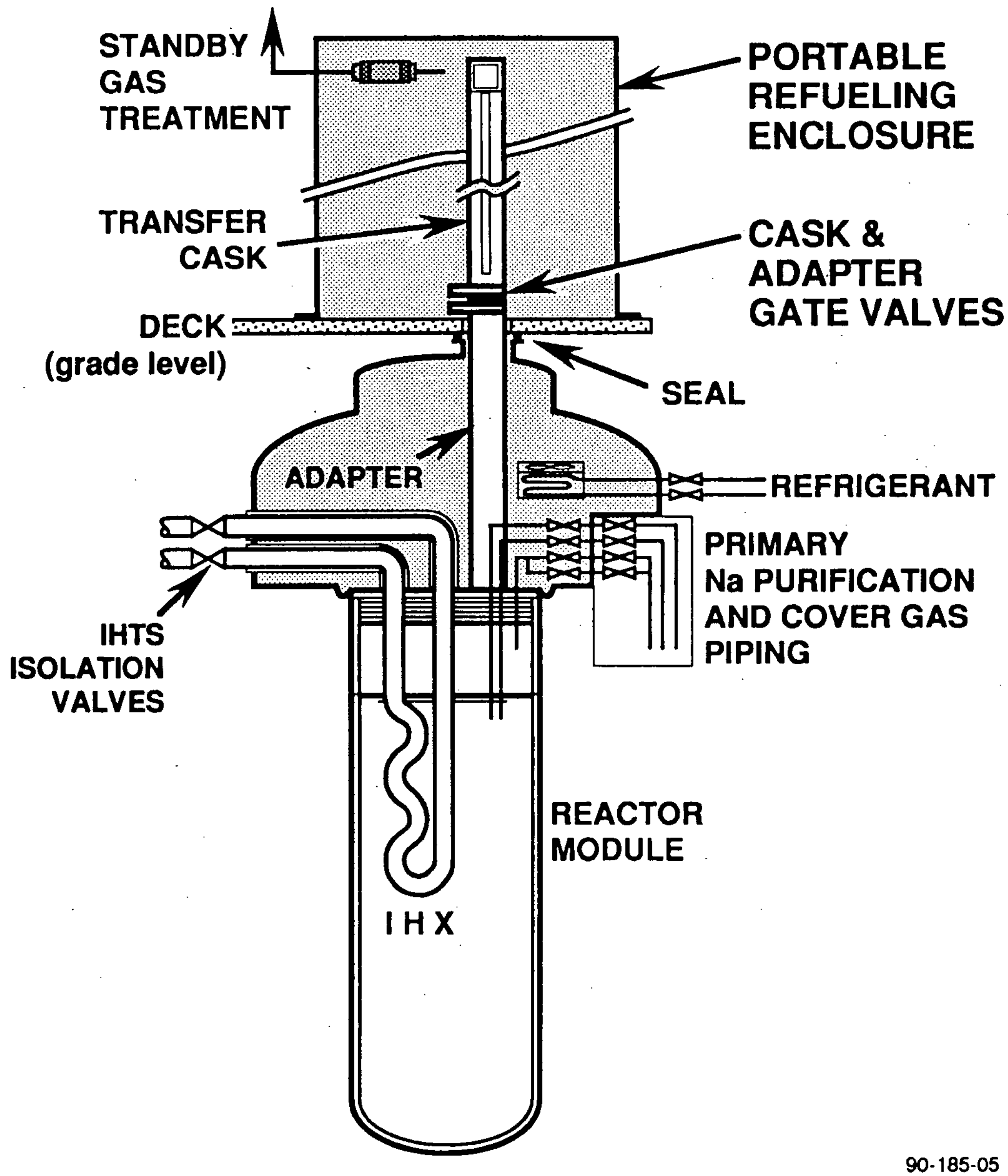
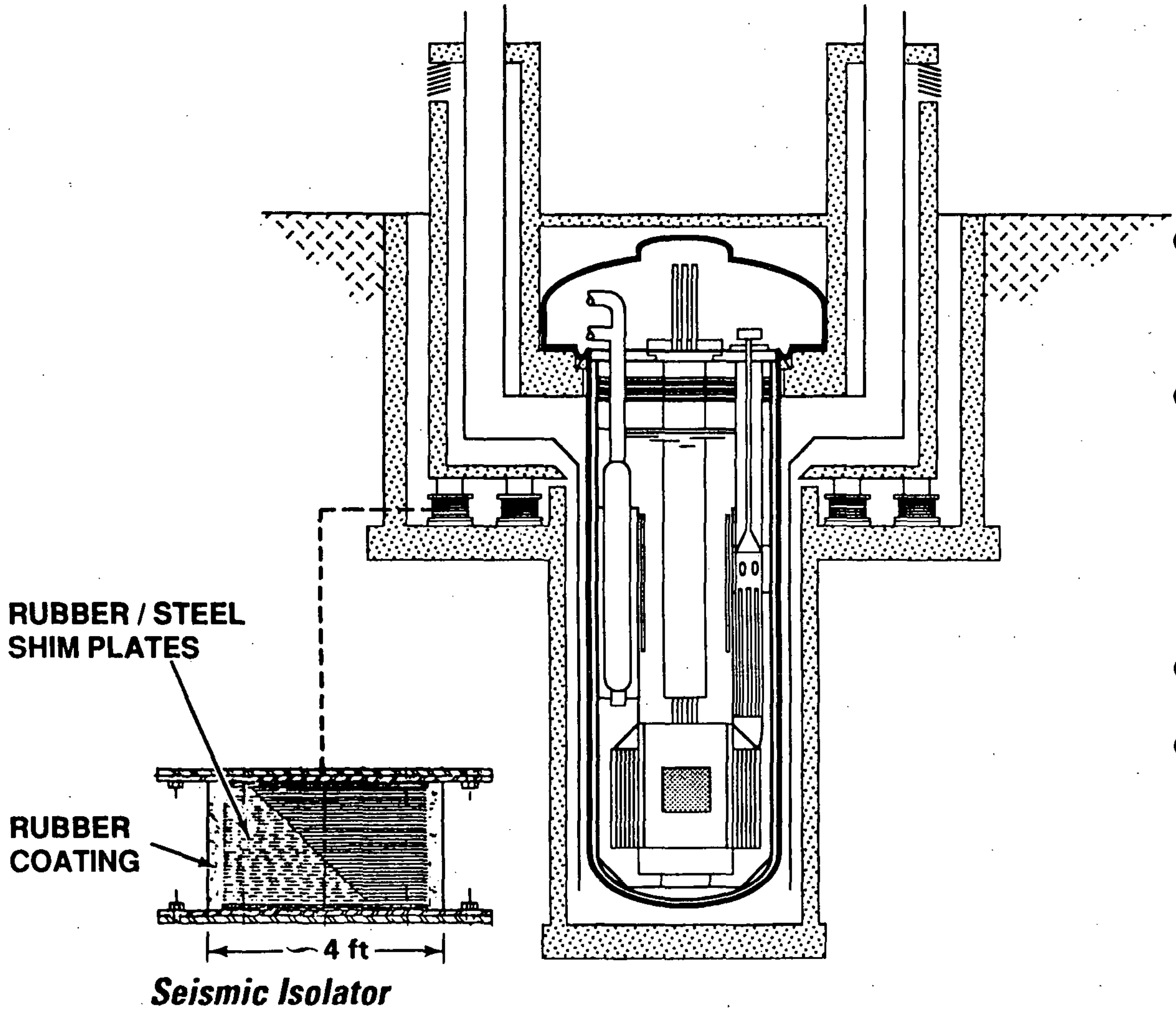


Figure G.2.1-9 CONTAINMENT DURING MAINTENANCE & REFUELING



Seismic Isolation System Characteristics

- Safe Shutdown Earthquake (SSE)
  - — Design Requirement 0.3g (ZPA)
  - — Design Capability 0.5g
- Lateral Displacement
  - — at 0.3g 7.5 in.
  - — Space Allowance
    - - Reactor Cavity 20 in.
    - - Remaining Structures 28 in.
- Natural Frequency 0.75 Hz
- Lateral Load Reduction >3

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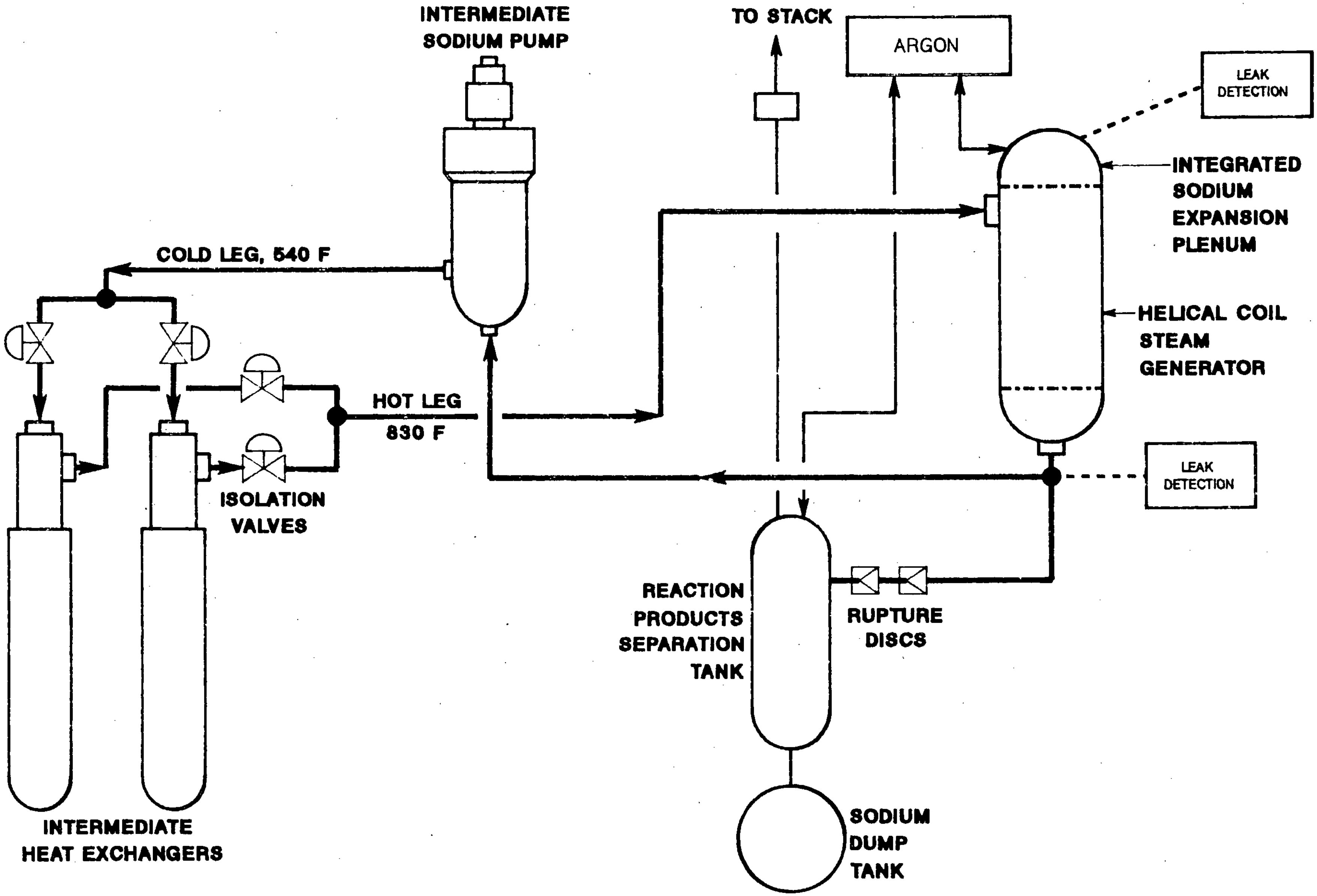
Figure G.2.1–10 SEISMIC DESIGN



90-185-02

Figure G.2.1–11 ALMR INTERMEDIATE HEAT TRANSPORT SYSTEM FLOW DIAGRAM

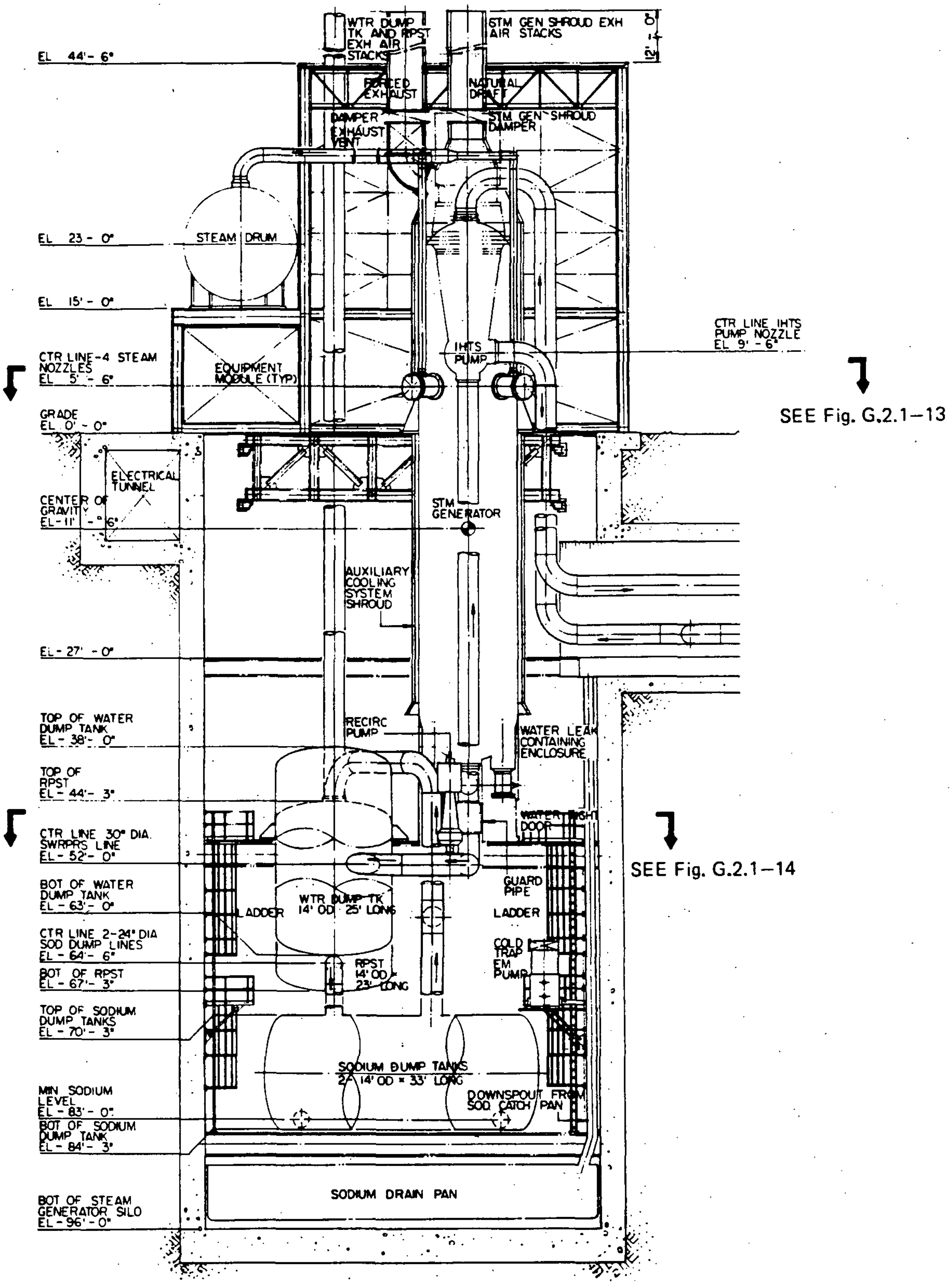


Figure G.2.1–12 GENERAL ARRANGEMENT STEAM GENERATOR FACILITY

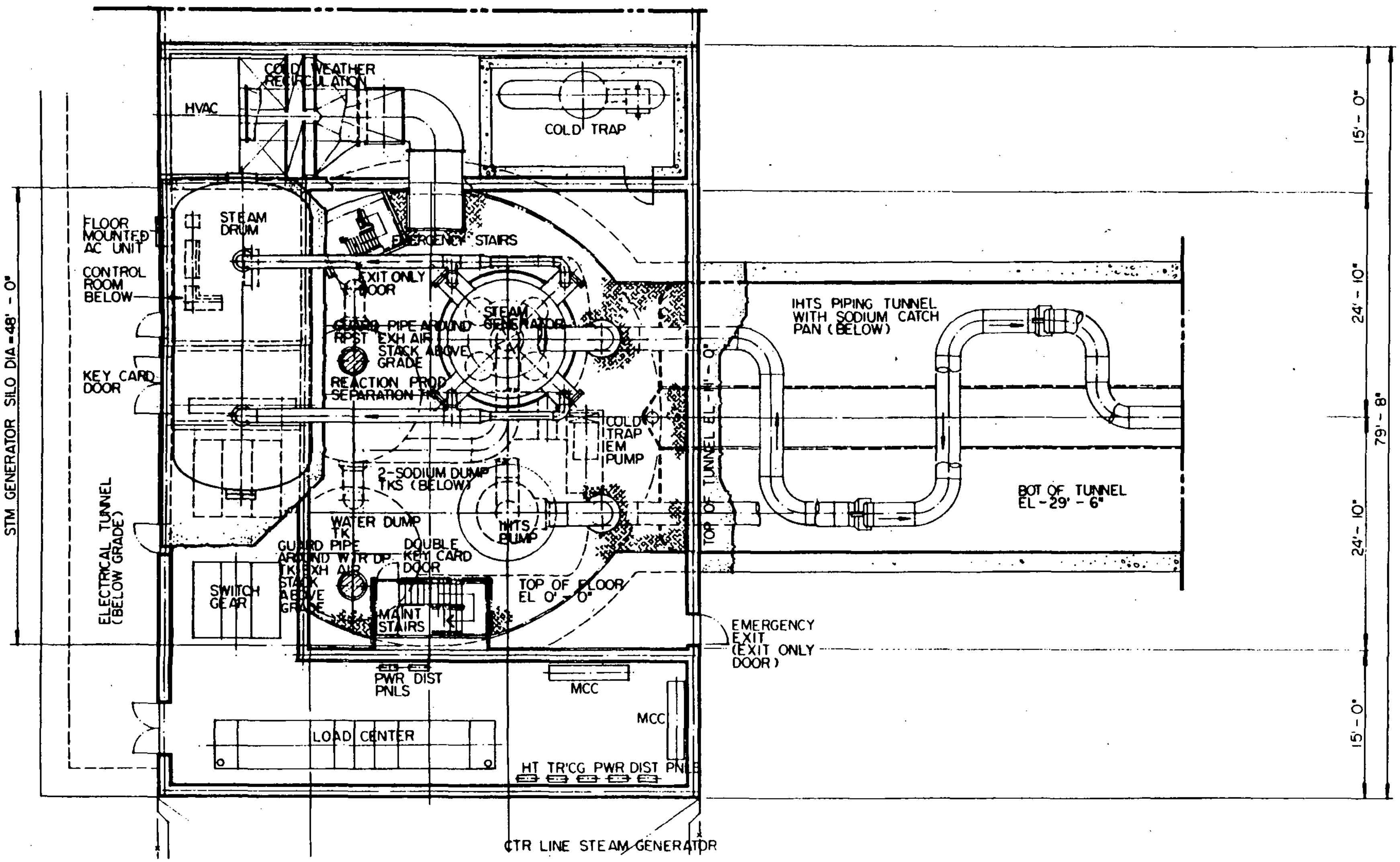


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Figure G.2.1—13 STEAM GENERATOR FACILITY PLAN VIEW

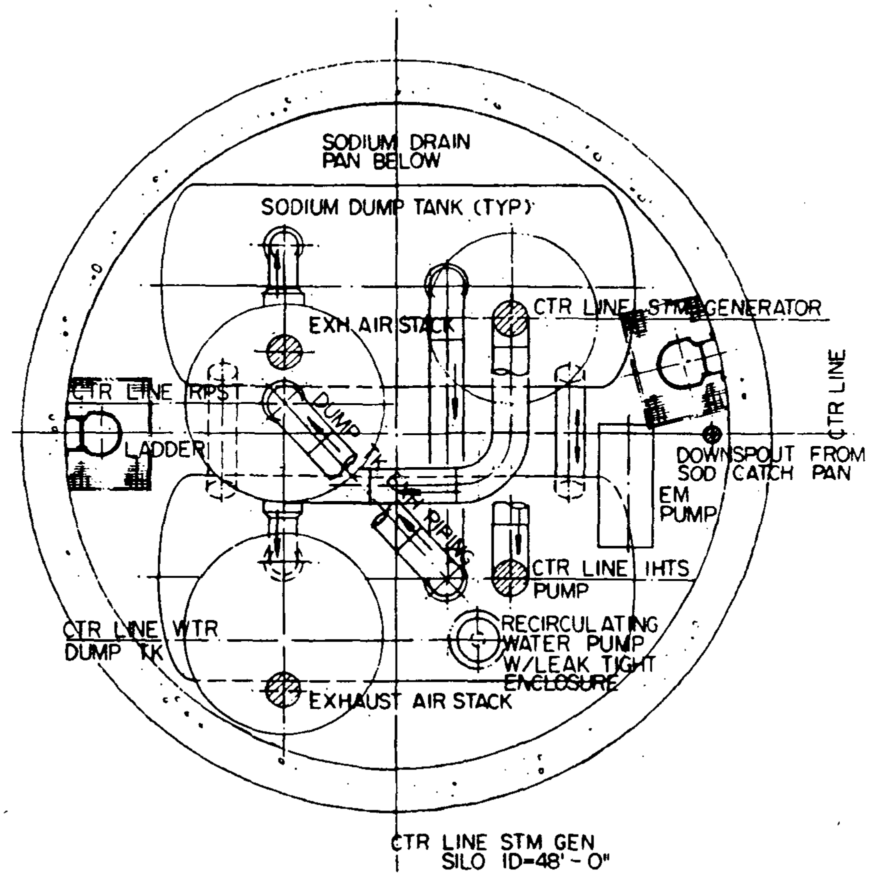

Figure G.2.1–14 STEAM GENERATOR FACILITY – PLAN AT ELEV.-45 FT.

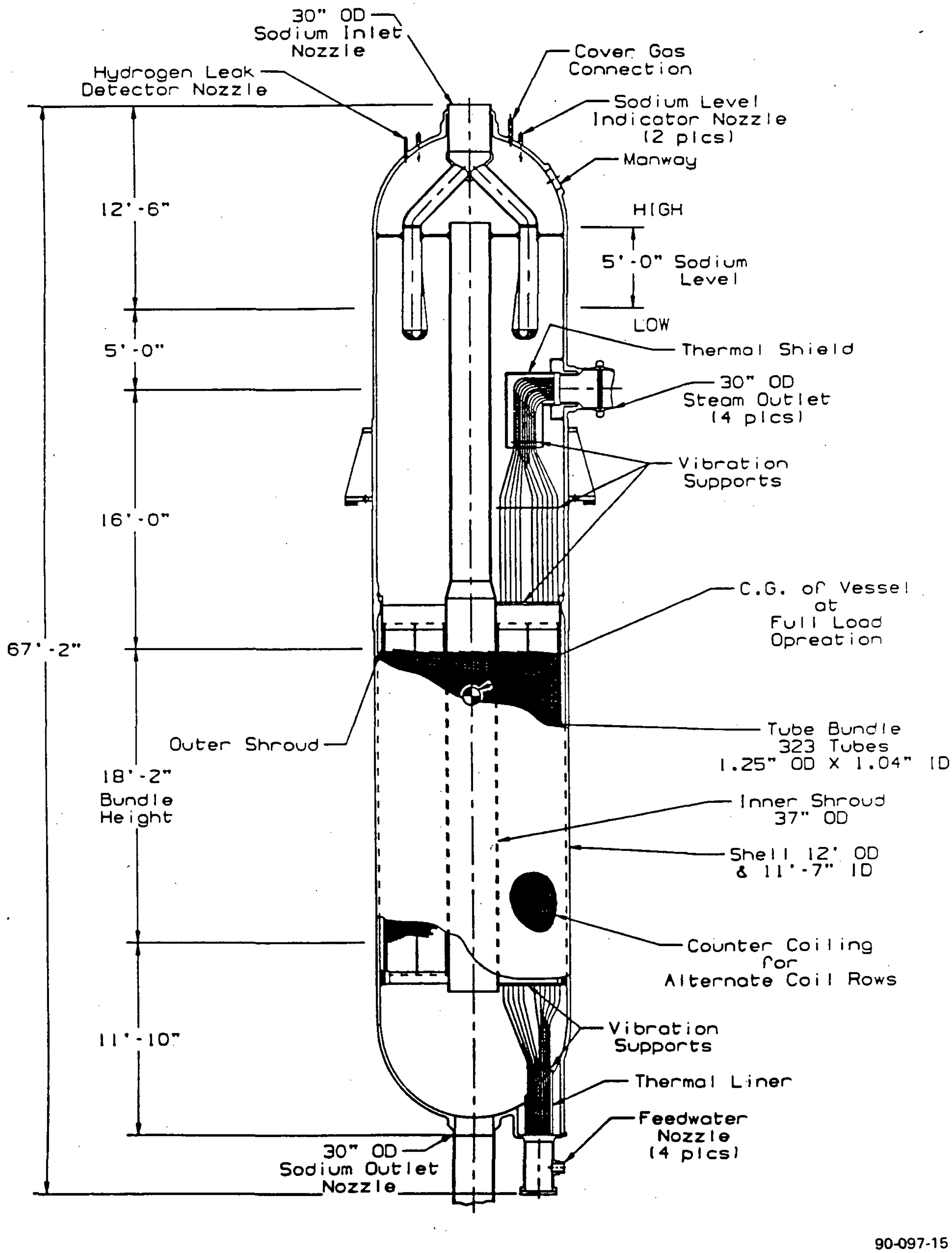

Figure G.2.1—15 REFERENCE HELICAL COIL STEAM GENERATOR ARRANGEMENT

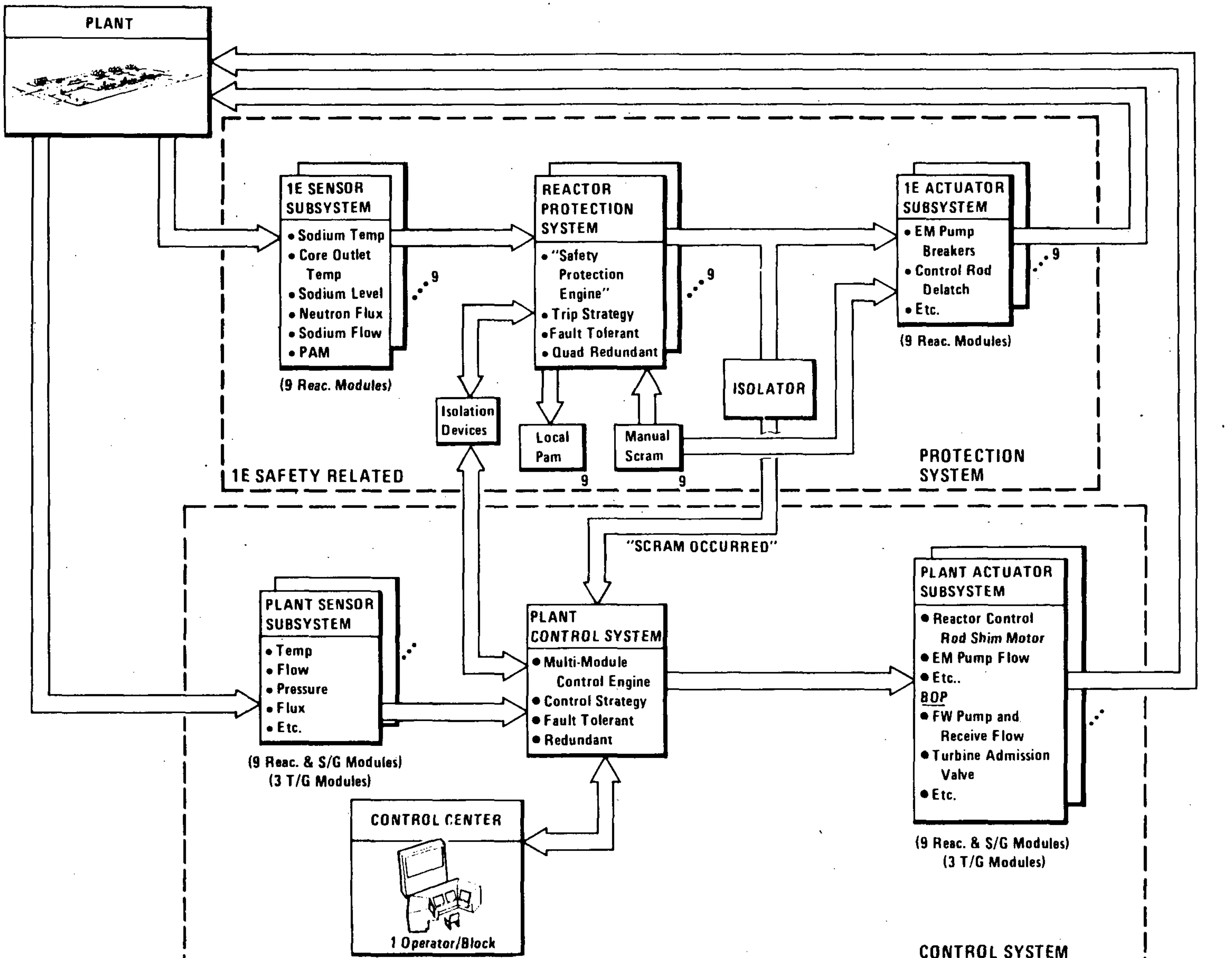

Figure G.2.1–16 PLANT CONTROL AND REACTOR PROTECTION SYSTEM INTERFACES

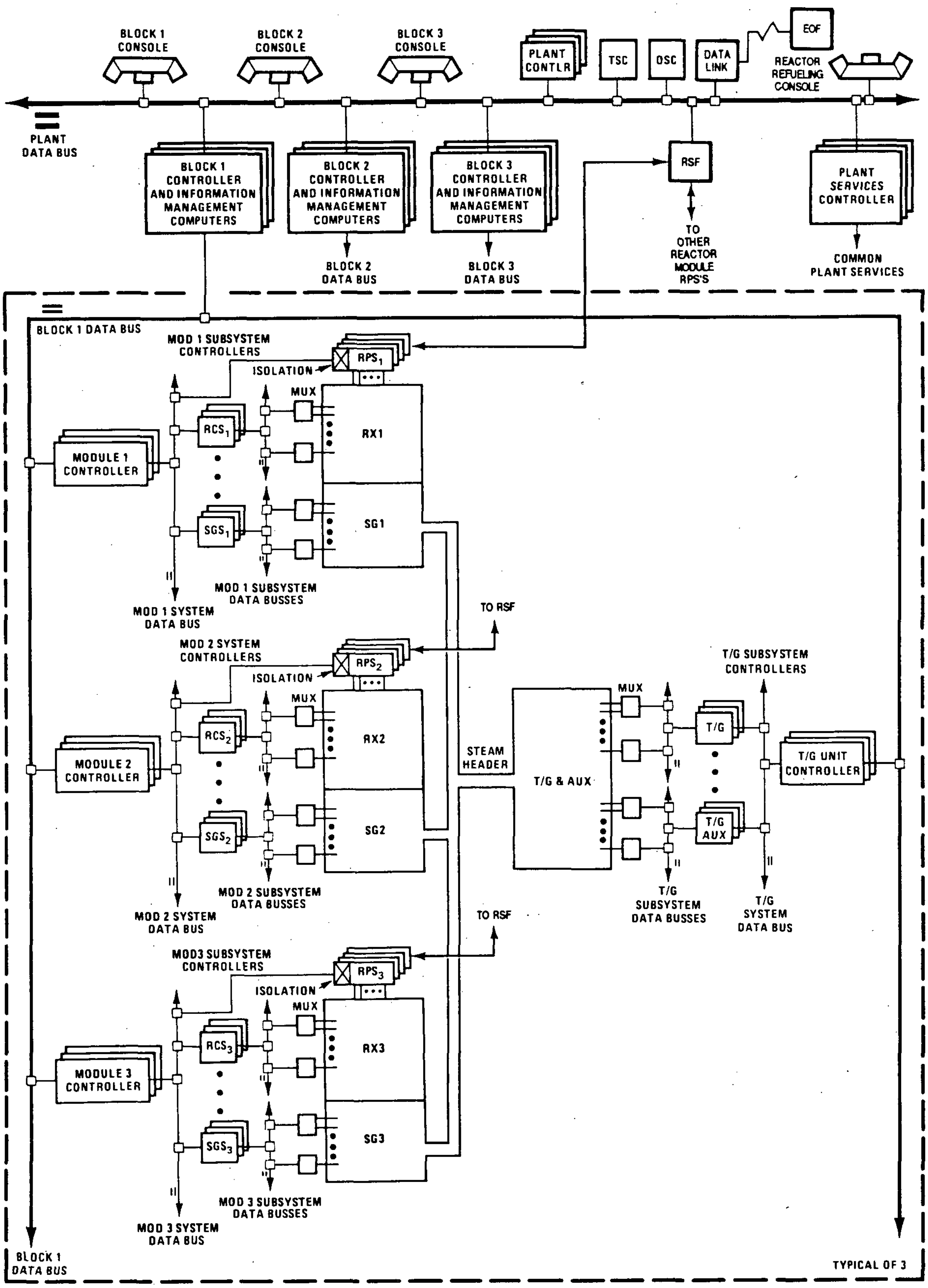

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Figure G.2.1—17 PLANT CONTROL SYSTEM ARCHITECTURE

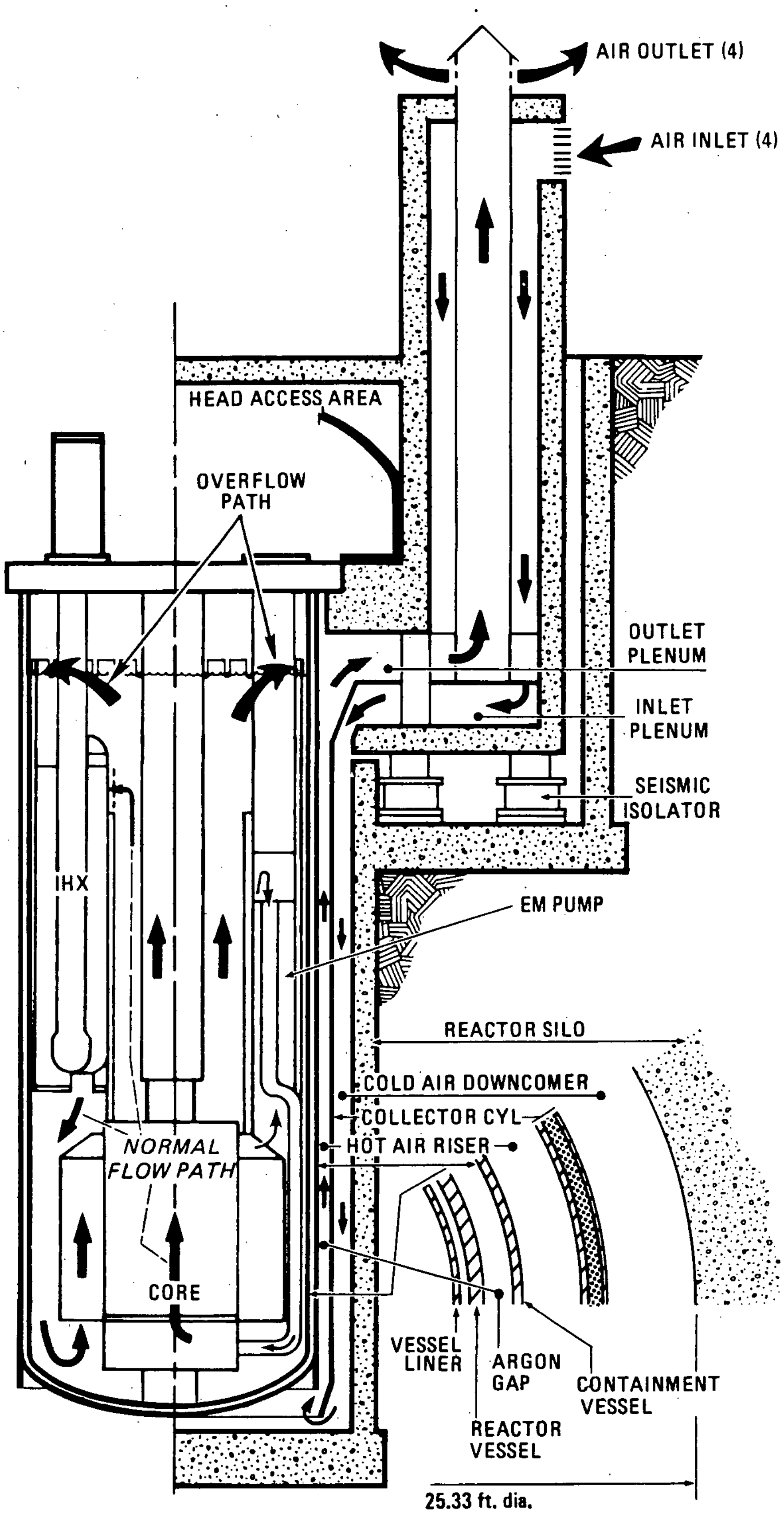

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Figure G.2.1–18 PRIMARY SODIUM AND AIR FLOW CIRCUITS DURING RVACS HEAT REMOVAL OPERATION

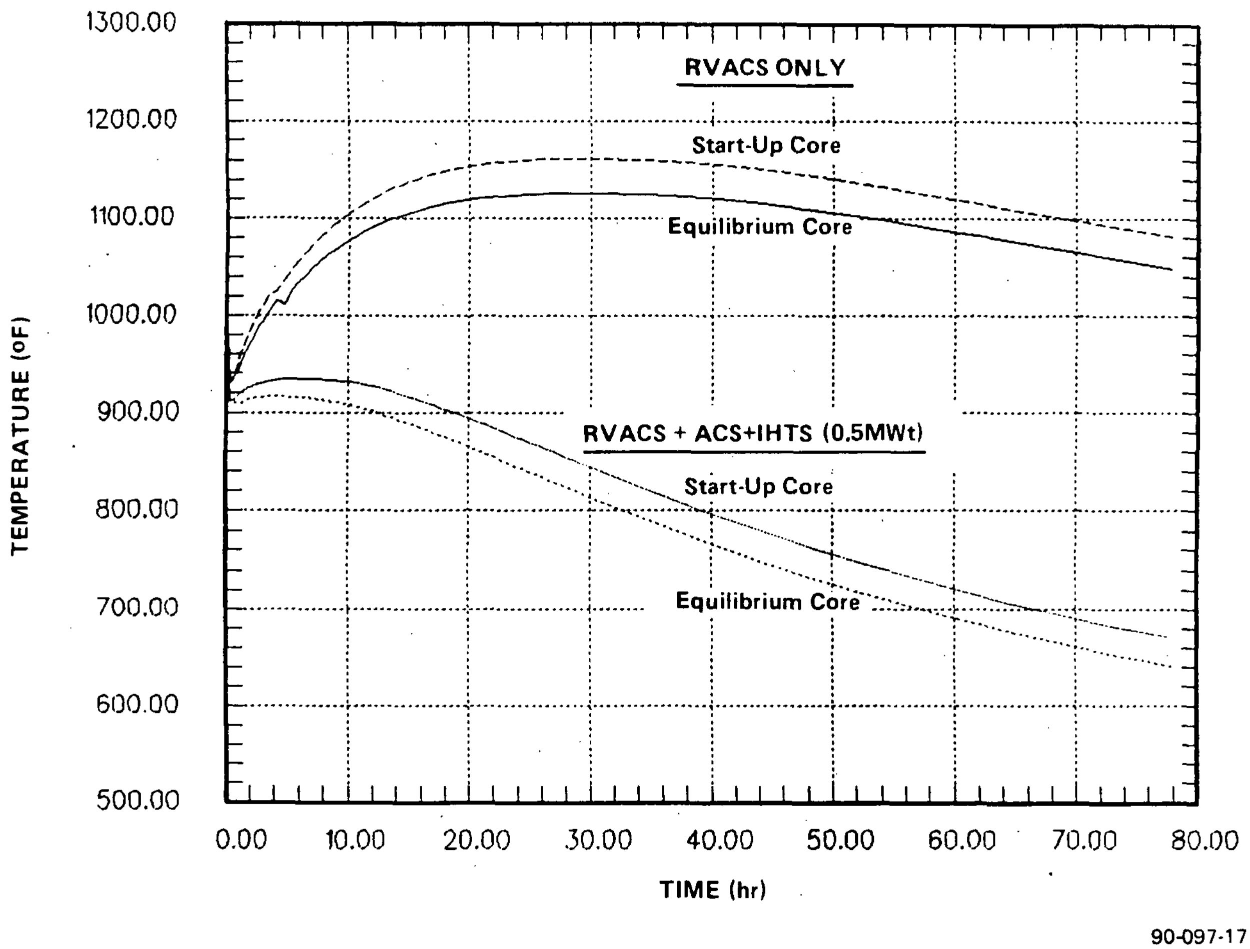


Figure G.2.1–19 AVERAGE CORE OUTLET TEMPERATURES AS FUNCTIONS OF TIME FOR RVACS ONLY AND RVACS PLUS ACS CASES

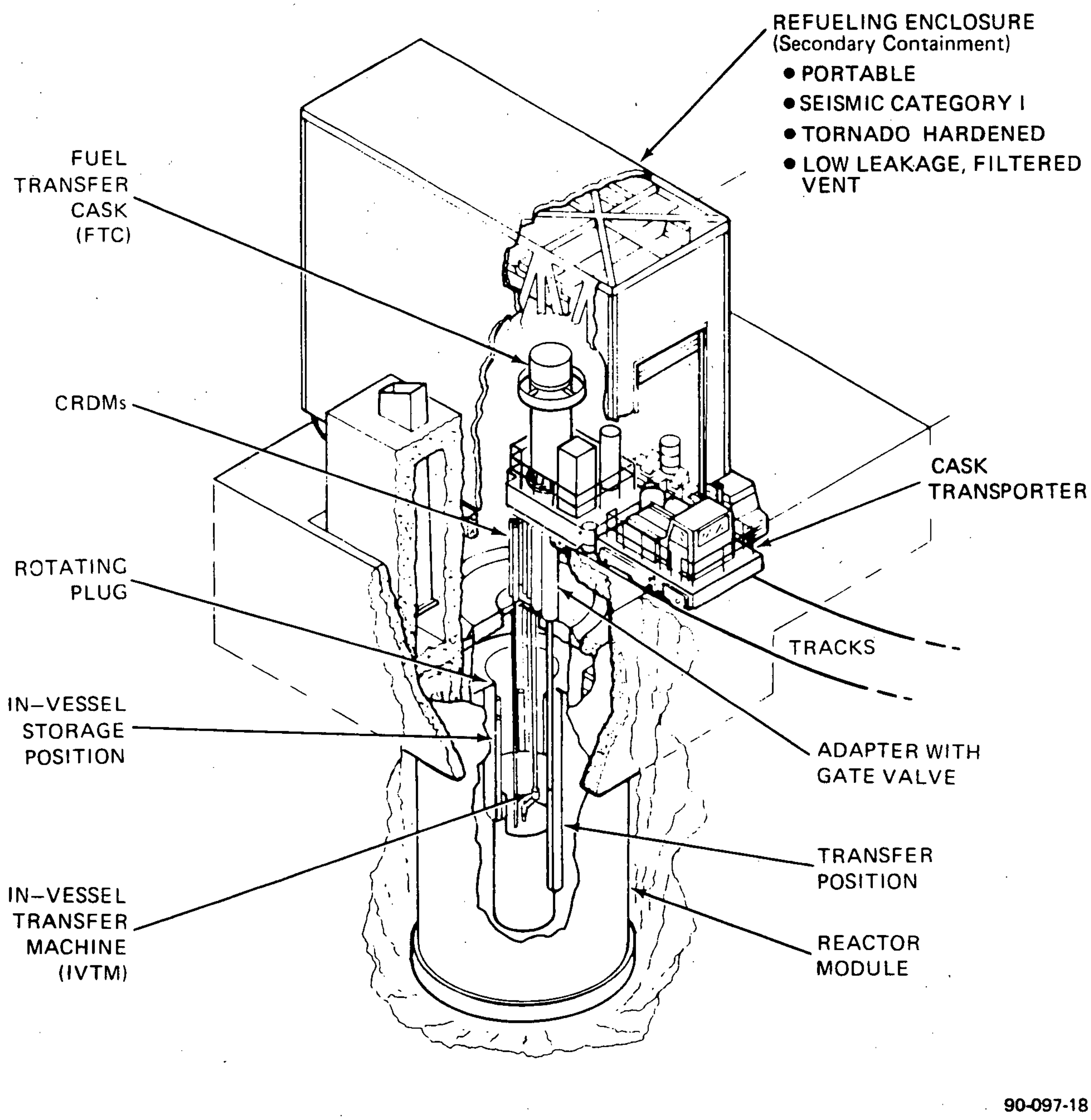


Figure G.2.1-20 REACTOR REFUELING SYSTEM ARRANGEMENT

G.2.2 Summary of Major Design Changes Since 1986-1987 PSID

The reference design, described in Chapter 1 of the PSID, has been revised to incorporate the results of trade studies subsequently completed and to address safety issues identified by the NRC staff in NUREG-1368. The discussion in Section G.2.1 provides a summary of the reference ALMR design; changes made to the 1986-1987 reference design are summarized in Table G.2.2-1.

Significant plant parameters of the 1986-1987 design, listed in Table 1-1, Chapter 1, of the PSID, are compared to the ALMR reference design in Table G.2.2-2.

Table G.2.2-1

SUMMARY OF MAJOR DESIGN CHANGES SINCE 1986-1987 PSID

ALMR Plant Feature	1986-1987 PSID	ALMR Ref. Design
Reactor power	425 MWt	471 MWt
Plant Electrical Rating	1245 MWe	1395 MWe
Cold Shutdown After Stabilization by Inherency for ATWS events	Unspecified	Poison (B_4C) Ball Insertion
Accommodate ULOF/LOHS Accidents	Negative Reactivity Feedbacks	Addition of three GEMs at reference metal core boundary (six GEMs for oxide core)
Accommodate UTOP Accidents	Negative Reactivity Feedbacks	Addition of electronically positioned mechanical control rod withdrawal limiters (mechanical stops)
Accommodation of Core Melt	Prevention	Prevention plus capability to contain melted fuel (whole core melt) within core support structure

Table G.2.2-1
(Cont'd)

SUMMARY OF MAJOR DESIGN CHANGES SINCE 1986-1987 PSID

ALMR Plant Feature	1986-1987 PSID	ALMR Ref. Design
Accommodation of HCDA	Prevention	Prevention plus capability to withstand resulting forces without breach of primary pressure boundary
Refueling Interval	20 months	18 months
Refueling Outage Duration	22.5 days	11.3 days
Ex-vessel Storage for Core Unloading	Partial Core (58 Assemblies)	All fuel and blankets (135 assemblies)
Seismic Design Basis: o Reactor Module o NI Seismic Category I Facilities	0.3g SSE, 0.15g OBE	0.3g SSE, 0.15g OBE requirement with capability for 0.5g peak ground acceleration earthquake
Seismic Isolation of EM Pump Synchronous Machine	Not isolated	Isolated
Seismic Isolation of RPS Electronics	Not isolated	Isolated
Containment	Containment vessel below reactor head seal welded to reactor head	Containment vessel below reactor head seal welded to reactor head plus non-venting, leaktight (<1% leakage per day at 25 psig), pressure-containing cylindrical domed steel containment vessel above reactor head, IHTS isolation valves
HAA Containment Capability	Unspecified	Below grade non-venting leaktight (<1% leakage per day at 25 psig), pressure-containing cylindrical domed steel containment vessel
Portable Refueling Enclosure Containment Capability	Unspecified	Low leakage, filtered vent

Table G.2.2-1
(Cont'd)

SUMMARY OF MAJOR DESIGN CHANGES SINCE 1986-1987 PSID

ALMR Plant Feature	1986-1987 PSID	ALMR Ref. Design
Portable Refueling Enclosure Seismic and Tornado Qualification	Unspecified	Seismic Category I, tornado hardened
Steam Generator Building Seismic and Tornado Qualification	Uniform Building Code	Seismic Category II, tornado hardened
Steam Generator Type (Sodium-Water Boundary Failure Rate, 9 units)	Straight tube, double wall SG (<0.01/year)	Helical coil single wall SG (<0.013/year); internal accommodation of sodium expansion; internal bypass for relief of sodium water reaction products
Steam Generator SWRPRS Rupture Disk Qualification	Unspecified	ASME Section III
IHTS Auxiliary Cooling System	Natural circulation air	Natural circulation air (forced circulation capability)
Control Building Location	Outside high security boundary	Inside high security boundary
Control Building Seismic and Tornado Qualification	Uniform Building Code	Seismic Category II, tornado hardened
Remote Shutdown and Post-Accident Monitoring Facility	Non-safety grade Auxiliary Shutdown Facility located in Reactor Service Building; individual safety grade shutdown facilities at each reactor module	Safety grade, Seismic Category I, tornado hardened Remote Shutdown Facility located in Radwaste Facility; individual safety grade shutdown facilities at each reactor module in RPS vaults
Fuel Cycle Facility Reference Location	Co-located FCF on-site (optional off-site)	Central off-site FCF (optional co-located, on-site)

Table G.2.2-2

COMPARISON OF PLANT PARAMETERS

	1986-1987 Design	ALMR Ref. Design
Overall Plant		
Number of Reactor Modules	9	9
Plant Thermal Power, MWt	3825	4239
Net Electrical Output, MWe	1245	1395
Number of Control Rooms	1	1
Capacity Factor, %	80	85
Reactor Module		
Core Power, MWt	425	471
Primary Sodium Inlet Temperature, °F	610	640
Primary Sodium Outlet Temperature, °F	875	905
Primary Sodium Flowrate, gpm	40,800	46,000
Intermediate Sodium		
Cold Leg Temperature, °F	540	540
Intermediate Sodium		
Hot Leg Temperature, °F	800	830
Intermediate Sodium Flowrate, gpm	41,000	41,250
Steam Cycle	Saturated (1000 psig)	Saturated (965 psia, 540°F)
Core, Fuel Description		
Assembly length, inches	176	196
Core height, inches	46	53
Fuel pins/assembly	271	331
Fuel pin OD, inches	0.290	0.263
Cladding thickness, inches	0.022	0 .020
Lifetimes, Cycles:		
Fuel	3	3
Blanket	5	5
Refueling Interval, months	20	18

Table G.2.2-2
(Cont'd)

COMPARISON OF PLANT PARAMETERS

	1986-1987 Design	ALMR Ref. Design
Number of Core Assemblies:		
Fuel	42	42
Internal Blanket	25	24
Radial Blanket	36	33
Control	6	6
Reflector	60	42
Shield	--	48
Ultimate Shutdown	--	1
Gas Expansion Module	--	3
Total:	169	199
Nuclear Performance:		
Fuel Axial Expansion:		
Expansion, %	0	5
Batch Reactivity Worth, $	0	-1.06
Fissile Enrichment, Fissile Pu/Pu+U	22.3	22.5
Burnup Reactivity Swing, $	+0.4	-0.21
Compound System Doubling Time (CSDT), years	45	~100
BOEC fissile Mass, kg	991	1327
Peak Fuel burnup, MWd/kg	161	135
Nominal Peak Linear Power, kw/ft		
Fuel	11.7	9.3
Internal Blanket	13.0	10.6
Peak Fast Fluence, n/cm^2	3.3×10^{23}	3.7×10^{23}
Sodium Void Worth, $:		
Fuel	2.5	3.1
Blanket	2.8	2.2

G.3 SAFETY R&D RESULTS AND PLANS

G.3.1 R&D Results Since 1986-1987 PSID

The technology development tasks have continued since 1987, as outlined in the ALMR Research and Development Requirements Plan. Work performed by the national laboratories has proceeded in key areas and was supplemented by international collaboration programs. Table G.3.1-1 summarizes the major development program results.

The ALMR R&D Program recognizes that a significant safety and licensing data base is available from earlier U.S. liquid metal reactors, including FFTF and CRBRP. For these reactors much of the safety evaluation effort was focused on the assessment of accidents with severe consequences, such as the HCDAs. Typically, the initiating events included loss of flow, loss of heat sink, and transient overpower events without scram.

A goal for the ALMR is to demonstrate that these sequences will lead to benign consequences. Testing in EBR-II and FFTF has verified that physical phenomena, mainly thermal expansion, will transition the ALMR reactor to a new equilibrium state at an elevated system temperature which is structurally acceptable, and at a core power generation level reduced to near zero fission power.

Activities under the safety and licensing support task of the ALMR R&D Program address containment evaluations, including the characterization of radionuclide transport, retention of radionuclides in the sodium pool, evaluation of the consequences of sodium fires, steam generator sodium/water reactions, and support of evaluations of residual risk.

In many areas analytical models have been developed and experimental data supporting the modeling of key phenomena have been generated. The applicability of the data base and models to specific metal fuel related processes are being carefully evaluated.

Table G.3.1-1

SUMMARY OF ALMR TECHNOLOGY DEVELOPMENT

TECHNOLOGY AREA	KEY PERFORMING ORGANIZATIONS	STATUS OF R&D (RESULTS SINCE 1986-87 PSID)
Electromagnetic Pump	ANL UCLA	Accelerated aging tests of electrical insulation materials is in progress with testing of 30 bar samples insulated with 15 layers of mica/glass tape including 15 samples of unbonded (dry) amber and white mica/glass tape and 15 samples of SECON-5 bonded amber mica. The tests proceed at 1500V and temperatures in the range of 680 to 750°C. Twenty-two samples have exceeded 10^4 hrs without failure, older samples with MAP bonding have exceeded $3x10^4$ hrs. Testing of 6 full diameter coils (23 in. dia) proceeds at 1500V at temperatures of 500 and 550°C. Testing times have exceeded $2x10^4$ hrs without failure. Slightly decreasing leakage currents have been observed. Tests for evaluation of the mechanical performance of a 1/4 length segment of the pump have been successfully completed after ~3000 hrs at 370V, and a maximum winding temperature of 870°F. More than 30 startup/shutdown cycles were accommodated. Post-test evaluations are in progress. Evaluations were initiated to confirm the applicability of the Arrhenius principle for life-time predictions of EM pump electrical insulation materials.
In-Vessel Fuel Transfer Machine	International	Review of international data base for operations of pantograph refueling machine in sodium (PEC* components qualification program). Test included the evaluation of bearings and seals in sodium environment, positional accuracy, and functional performance verification.

* PEC - Prova Elementi Combustibili (Italian Sodium-Cooled Fuel Element Test Reactor; construction discontinued in 1988)

Table G.3.1-1 (continued)

SUMMARY OF ALMR TECHNOLOGY DEVELOPMENT

TECHNOLOGY AREA	KEY PERFORMING ORGANIZATIONS	STATUS OF R&D (RESULTS SINCE 1986-87 PSID)
Control Drive	To be determined	No activities planned before 1991.
Steam Generator	ETEC	Testing of the 70 MW helical coil steam generator was discontinued by DOE in 1989 after $1.6x10^4$ hrs of operations at various power levels. Tests were completed for a broad range of test conditions covering both normal and off-normal plant unit-operating conditions, including various startup, shutdown, load maneuvering and controllability sequences, transients, etc. Confirmation 1 tests with 40 tubes were completed after 243 test runs. Post-test evaluations will be specified.
Instrumentation	International ORNL	Completed the test specification for a passive fission gas monitor test and initial test design. Completed development and fabrication of high temperature source range flux monitor for in-reactor life tests.
Plant Controls	ORNL ANL	Completed automated controller development for turbine bypass and tested module at EBR-II. Completed development of supervisory technique for module power allocation.
Seismic Isolation	UC Berkeley ETEC ANL International	Completed static and dynamic testing of twelve 1/2 size and twelve 1/4 size ALMR seismic isolator bearings to determine structural characteristics and performance margins including vertical and horizontal stiffness, damping, and horizontal shear strains at failure. Bearings with dowelled connections achieved, as predicted, 150% horizontal shear strain or three times the expected SSE displacement; bolted bearing achieved higher horizontal displacements, up to 350%, but not consistently. Building tests at Tohoku University, Sendai, Japan with ~1/3 scale bearings were successfully completed.

Table G.3.1-1 (continued)

SUMMARY OF ALMR TECHNOLOGY DEVELOPMENT

TECHNOLOGY AREA	KEY PERFORMING ORGANIZATIONS	STATUS OF R&D (RESULTS SINCE 1986-87 PSID)
Seismic Isolation (continued)		The San Bernardino Law and Justice Center with ~1/2 size seismic isolator bearings (with a similar shape factor) experienced a 0.15g acceleration at the basemat during the 2/28/90 Upland earthquake with responses as predicted.
		The development of finite element codes for predicting the structural performance of the bearings and for design optimization is in progress.
		The development of acceptance test procedures for seismic isolator bearings is in progress.
		An initial set of seismic isolation design guidelines has been developed and distributed for peer review.
Shielding	ORNL	Tests on the TOWER shielding facility are in progress in support of Japanese reactor designs with generic application to the ALMR program.
Materials	WHC	30 B_4C assemblies (with 1642 pins, 20% B^{10} enrichment) have been irradiated in FFTF for a peak burnup of 3.3×10^{22} captures/cm^3 B_4C or 1100 EFPD. Excellent performance was observed. Post-test examinations are planned for additional assemblies.
		Completed RVACS surface emissivity characterization under environmental conditions, thermal cycling, and expected long-term changes.
Thermal-Hydraulics	ANL	Completed first series of water simulation tests of reactor system with a 1/5 scale model using laser techniques for flow vizualization. Evaluated flow fields under normal and off-normal conditions, for comparison with 3D code predictions.

Table G.3.1-1 (continued)

SUMMARY OF ALMR TECHNOLOGY DEVELOPMENT

TECHNOLOGY AREA	KEY PERFORMING ORGANIZATIONS	STATUS OF R&D (RESULTS SINCE 1986-87 PSID)
Passive Reactivity Reduction	ANL WHC	Integral transient tests to demonstrate the inherent shutdown characteristics have been completed in EBR-II for a small metallic core and in FFTF for a mixed oxide core. The transient tests involved loss of flow and loss of heat sink conditions without reactor scram. These conditions were previously considered as potential initiators for core disruptive accidents. However, for the metal fueled core (EBR-II) these events were accommodated with benign consequences, either a short-term temperature peak of 1300°F for less than 100 seconds or an increase of the core support structure temperature by 80°F. For FFTF, nine gas expansion modules were included to perform loss-of-flow tests from 100% flow and 50% power. A sodium outlet temperature increase of 150°F in 90 seconds reduced the fission power to zero.
Passive Shutdown Heat Removal	ANL	The performance of the air-side shutdown heat removal was tested with a full length annular segment of the RVACS for various temperature and heat flux boundary conditions. Heat transfer correlations were developed. Tests were performed for blocked and partially blocked inlets and test evaluations are in progress. Adequate heat removal capability was demonstrated for a completely blocked inlet.
Safety & Licensing Support	ANL	Supplemental data base and models for characterization of radionuclide transport and residual risk as necessary.
Fuel Safety	ANL	IFR activities including M-series tests in TREAT, EBR-II irradiation tests, and out-of-reactor materials tests are in progress. TREAT tests were performed to establish the margins to failure for metal fuel, and to validate the analysis of metallic fuel transient behavior. Safety experiments and analyses addressed the key phenomenology in fuel behavior under accident conditions. Supporting ex-reactor experiments were conducted with unirradiated and irradiated fuel. Analyses were conducted of operational transients and local faults to establish margins of safety for metallic fuel.
Fuel Cycle Safety	ANL	IFR program in progress.

G.3.2 R&D Plans

G.3.2.1 ALMR R&D Requirements

An overall ALMR R&D Requirements Plan has been specified for ALMR technology development. The program is organized into three categories: (1) tasks important to safety, (2) tasks related to component development and design verification, and (3) tasks related to investment protection. The development work supports the safety evaluation and licensing process. A key element in the general licensing strategy is the operation of a prototype module for performance demonstration and safety tests.

In general, technology development has been included only in areas where significant safety improvements or design simplifications could be expected. Otherwise, use has been made of the existing extensive data base for liquid metal reactor technology. The technology development described in the R&D Requirements Plan therefore addresses only the qualification of key innovative components and features, such as the self-cooled electromagnetic pump, a pantograph type in-vessel refueling machine, an improved control drive system, an advanced instrumentation and control system, and advanced promising technologies, such as seismic isolation, passive reactivity reduction, and passive shutdown heat removal.

The schedule for this development is aligned with the ALMR project schedule supporting the completion of an advanced conceptual design in 1991, the completion of a preliminary design in 1993, and prototype reactor module criticality in 1999.

As indicated in Figure G.3.2-1, key features tests will be performed for developmental components consistent with the reliability development and growth plan during the advanced conceptual design phase. Prototype tests for these components will be initiated during the preliminary design phase and completed during the subsequent detailed design phase. The characterization and qualification of the passive safety features and safety enhancing mechanisms, including passive reactivity reduction, the passive shutdown heat removal or the seismic isolation system, will be completed by performing systems tests with the first prototype reactor module. Agreement will be reached with the NRC on the scope of these safety tests.

Most of the development tasks support the implementation of the first prototype reactor module. Some development activities, such as the development of an advanced control system for a multi-module plant, the steam generator, improved structural materials, or robots for maintenance and repair activities, will extend in time consistent with requirements for certification of a complete ALMR power plant in the Year 2003.

The work breakdown structure for the ALMR technology development is shown in Figure G.3.2-2. Some of the work breakdown structure elements are supported by the IFR Program conducted by ANL. Some overlap in the specification of the ALMR technology and the IFR program plan exists in the area of safety and licensing. However, this area is of very high importance to the advanced liquid metal reactor program, and potential overlaps in specifications will be adjusted later, if necessary.

Significant international contributions are planned for the advanced components and systems tasks which include the development of the electromagnetic pump, the in-vessel transfer machine, the control drive and the steam generator. The university program will make more fundamental contributions to the components reliability and safety evaluation program.

The planned contributions of the national laboratories, the universities, the GE Team and international organizations to the key technology development tasks are shown in matrix form in Table G.3.2-1.

The technology development program is conducted consistent with requirements established by the Reliability, Availability and Maintainability (RAM) Program Plan (GE Nuclear Energy, GEFR-00843, April 1989). Specific reliability tasks are included to ensure that, for a given technology area, adequate reliability development and growth and a reliable end product is obtained, and the time phasing is consistent with the ALMR design development and implementation. The RAM scheme represents an evolution process from predominantly qualitative to quantitative assessments consistent with the evolution of the design from the advanced conceptual phase to the detailed design phase.

Table G.3.2-1

ALMR R&D PROGRAM CONTRIBUTIONS (Preliminary Allocation)

PROGRAMS		ANL	ORNL	WHC	ETEC	UNIVER.	OTHER	GE TEAM	INT'L
AT100	ADVANCED COMPONENTS AND SYSTEMS								
AT110	Self-Cooled EM Pump	X				X		X	X
AT120	In-Vessel Transfer Machine				X			X	X
AT130	Control Drive							X	X
AT140	Steam Generator				X			X	X
AT200	ADVANCED INSTRUMENTATION AND CONTROL								
AT210	Advanced Instrumentation	X	X	X	X			X	X
AT220	Advanced Controls	X	X					X	X
AT230	Robotics		X			X			X
AT300	ADVANCED TECHNOLOGY								
AT310	Seismic Isolation	X			X	X	X	X	X
AT320	Shielding		X					X	
AT330	Mat'ls & Structures		X	X					X
AT340	Thermal-Hydraulics	X			X	X			
AT410	REACTOR SAFETY								
AT410	Passive Reactivity Reduction	X		X					
AT420	Passive Shutdown Heat Removal	X				X			X
AT430	Safety & Licensing	X	X	X				X	X

A schematic of the integrated RAM approach is shown in Figure G.3.2-3. The RAM development and growth testing applies to all phases of the ALMR program, and the testing of prototypes is done at the components and the systems level.

Reliability and growth testing will be completed prior to the detailed design phase; however, tests to support the reliability qualification of components will proceed throughout the design and fabrication phase into the pre-operations phase and safety testing of the prototype module. Reliability test plans will be prepared in all key technology development areas.

G.3.2.2 IFR Program Plan

The Integral Fast Reactor (IFR) concept, advanced by ANL, is a complete advanced fuel cycle concept which capitalizes on the unique characteristics of metallic fuel and liquid metal cooling to offer significant improvements in safety, fuel cycle economics, environmental protection, and safeguards.

The metal fuel provides high fissile atom density, high thermal conductivity, and superior compatibility with the liquid metal coolant. The use of metallic fuel in turn makes possible the utilization of innovative fuel cycle processes (termed "pyroprocessing") which will permit fuel cycle closure with ultra-compact, low-cost reprocessing facilities, co-locatable with the reactor plant if required. The pyroprocessing method, in addition to its inherent economic advantages, generates minimal waste volumes, and can be tailored to recycle actinides which presently complicate conventional nuclear waste disposal options. The extraction and recycling of actinides, however, does not eliminate the need for a waste repository as planned for implementation by DOE.

ANL-West facilities play a crucial role in the metal fuel and fuel cycle development and demonstration. These facilities include EBR-II for irradiation tests and plant testing, FMF for EBR-II driver fuel manufacturing, TREAT for accident-simulating transient fuel tests, ZPPR for criticality tests, HFEF/N for destructive and nondestructive fuel examinations, and HFEF/S for fuel fabrication and fuel cycle demonstration.

A major new mission under ANL-West facilities is the refurbishment of the original EBR-II fuel cycle facility (now called HFEF/S), which would allow a prototype demonstration of the entire IFR fuel cycle in conjunction with EBR-II. The HFEF/S refurbishment involves facility modifications to meet the present-day safety and environmental standards and the design, fabrication and installation of the IFR process equipment systems.

Although based in large part on the technology which has resulted in the successful operation of the Experimental Breeder Reactor II (EBR-II) for over 25 years, the current metal fuel cycle technology program is new in many respects. When the IFR concept was conceived in the latter part of FY84, the IFR technology development and demonstration were planned in the following three phases, as shown in the overall program schedule, Figure G.3.2-1.

Phase I	Technology Feasibility	FY84-86
Phase II	Technology Development	FY87-90
Phase III	Technology Demonstration	FY91-95

The goal during Phase I was to establish the technical feasibility of the concept. Phase I consisted largely of scoping tests, analyses and critical reviews, intended primarily to establish the feasibility of the concept.

During this period a landmark series of demonstration tests were carried out in EBR-II, clearly showing the passive, inherent safety advances achievable with the IFR. Steady-state and transient testing of metallic fuels in EBR-II and TREAT further demonstrated the potential for improved reactor performance, both in normal and off-normal operation modes. Laboratory-scale experiments with pyroprocessing operations proved the feasibility of the electro-refining and injection casting processes. Reactor design studies, including support for the PRISM and SAFR design concepts, served to further enforce the conclusion that the metal fuel concept is a preferred option for future advanced reactor development. This was given additional credence with the findings of an independent, top-level review committee which asserted the technical feasibility of the metal fuel cycle and recommended continued development as a high-priority effort.

Major accomplishments during Phase I included:

- o Feasibility demonstration of electro-refining on a laboratory scale.
- o Inherent safety demonstration tests in EBR-II.
- o Adoption of the metal fuel cycle to the PRISM and SAFR designs.

The successful conclusion of the Phase I feasibility demonstration was followed by initiation of Phase II of the Program, which is the period during which the detailed technology will be developed to enable a subsequent full-scale demonstration (Phase III). The Phase II technology development activities deal with all aspects of the metal fuel cycle technology, from reactor design development support to waste disposal. Particular emphasis is placed on the characterization and performance evaluation for the binary and ternary (U-Zr and U-Pu-Zr) fuel compositions.

Also receiving major programmatic emphasis is work related to the design and testing of demonstration-scale pyroprocessing unit operations systems, including electro-refining, fuel fabrication, pyroprocess flowsheet optimization, and waste management processes. Phase II activities include major facility modifications, principally in the HFEF/S hot cells, to prepare for the demonstration phase. Because EBR-II reactor operations comprise an important part of the integrated metal fuel cycle technology demonstration, work in the reactor operations area continues to be directed toward evolution of operational practices, incorporating advanced instrumentation and control systems technologies.

Phase II points toward further enhancement of core design and analysis capabilities, and to the development of designs or operating strategies which utilize most effectively the unique features of the metal fuel cycle. In preparation for increased activity in the area of safety analyses and interactions with licensing authorities, efforts are being extended during Phase II to update the safety data base and analytical models, with added emphasis placed on analysis of severe accident initiating events and consequences.

Major accomplishments expected during Phase II include:

- o Demonstration of high burnup potential and fuel performance characterization.
- o Engineering-scale demonstration of electro-refining.
- o Safety data base to support ALMR interactions with the NRC.
- o EBR-II core conversion with the new U-Zr and U-Pu-Zr fuels.
- o Refurbishment of the original EBR-II fuel cycle facility (HFEF/S).

A final review of the accomplishments of Phase II and the status of the required technology development will be held at the end of FY90, whereafter the demonstration phase will begin. Further technology development requirements and activities will be dictated by experience accruing from the technology demonstration during Phase III.

Phase III, the metal fuel cycle technology demonstration, is the stage when individual aspects of the IFR technology will be brought together and integrated to prove the overall systems performance. This phase will be centered upon the extended operation of EBR-II with U-Pu-Zr metal fuel. Core conversion will be completed, and the reactor will be operated with fuel having a variety of fuel and cladding compositions. Reactor operation will provide a substantial fuel performance data base for future utilization. The spent fuel will be subjected to the full spectrum of pyroprocessing operations, using engineering-scale unit operations equipment installed in the modified fuel cycle facility (FCF) at the ANL-West site. The performance of recycled fuel will be evaluated by irradiation of a number of EBR-II fuel subassemblies fabricated in the FCF with fuel compositions typical of steady-state recycle operation, and the influence of this fuel on various reactor passive inherent safety characteristics will be assessed by direct measurements. Waste handling and treatment practices representative of future IFR plant operations will also be developed and tested during this demonstration phase.

A continuing activity during Phase III will be the support of technology development efforts, safety analyses, and licensing interactions for the ALMR. Design of a commercial fuel cycle facility will proceed apace with the industrial reactor design activities, with a conceptual design for such a facility to be available by the end of FY91 for use in commercialization strategy planning. Upon completion of Phase III, the metal fuel cycle technology will be fully developed for commercial application.

The end products of the IFR Program, scheduled to be completed by the end of FY95, as shown in Figure G.3.2-4, are as follows:

- o Fuel performance demonstration of recycled metal fuel alloys up to 150,000 MWd/T burnup level.
- o Demonstration of inherent safety potential of the IFR concept through actual EBR-II plant tests with recycled fuels.
- o Demonstration of the entire fuel cycle on a prototype scale.
- o Waste form certification.
- o Demonstration of actinide recycle capability.
- o Licensing data base in support of the ALMR project interactions with the NRC.

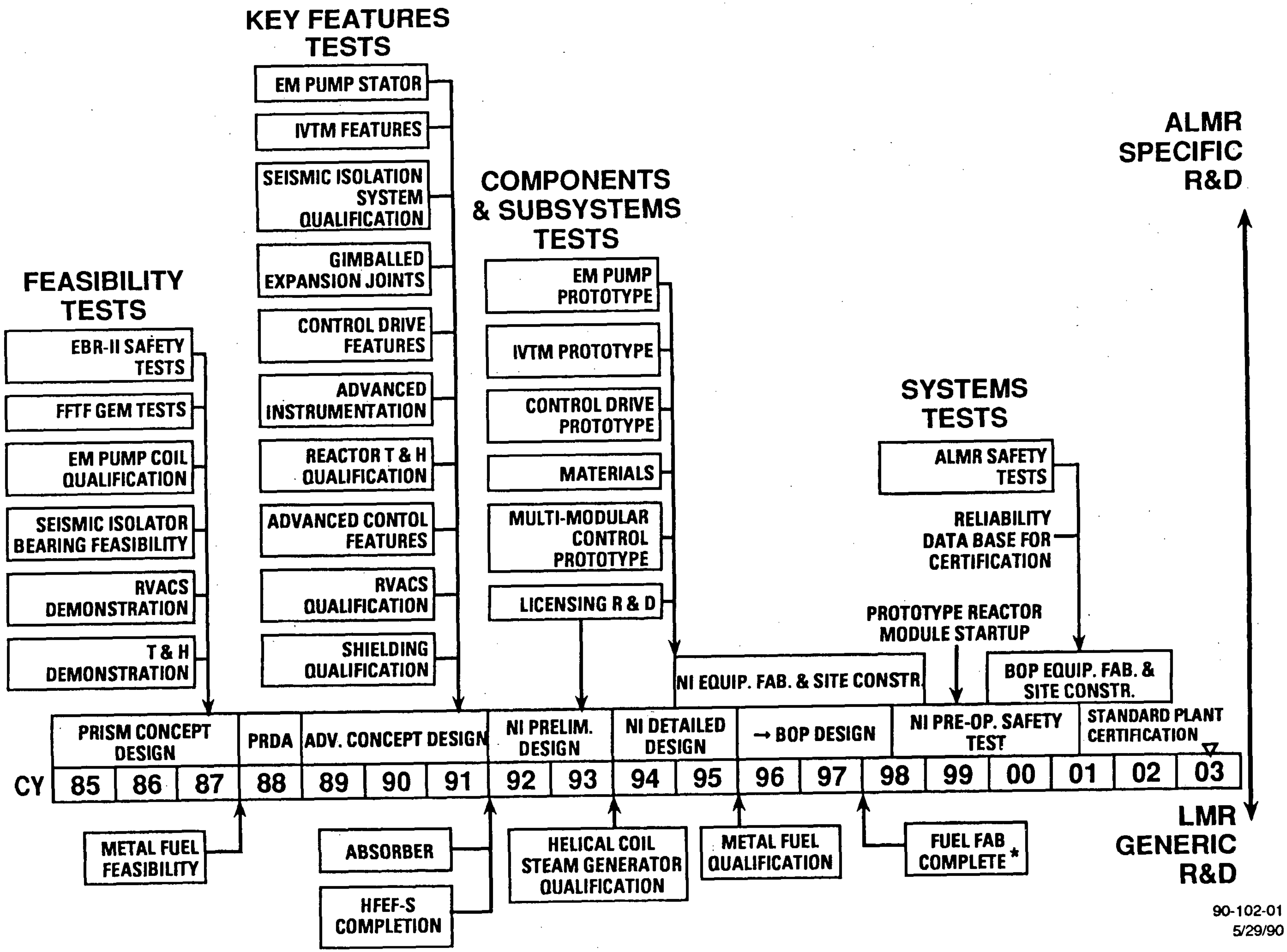
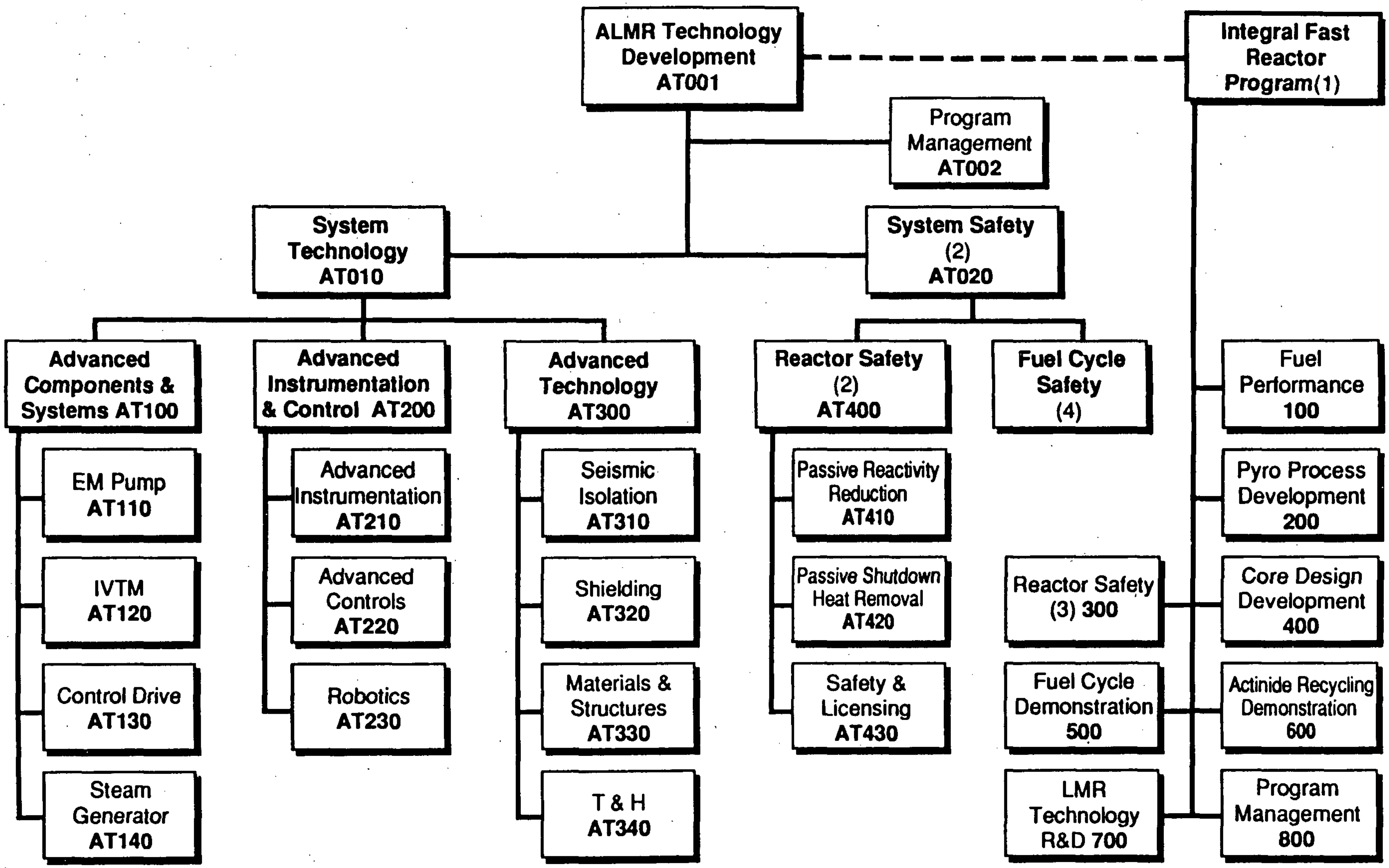


Figure G.3.2-1 ALMR R&D SUMMARY



(1) IFR PROGRAM PLAN AS SPECIFIED BY ANL (9/89)
(2) SOME WORK ELEMENTS ARE COMMON WITH IFR PROGRAM PLAN
(3) SOME WORK ELECMENTS ARE COMMON WITH ALMR DESIGN ACTIVITIES
(4) IFR PROGRAM

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Figure G.3.2-2 ALMR R&D WORK BREAKDOWN STRUCTURE

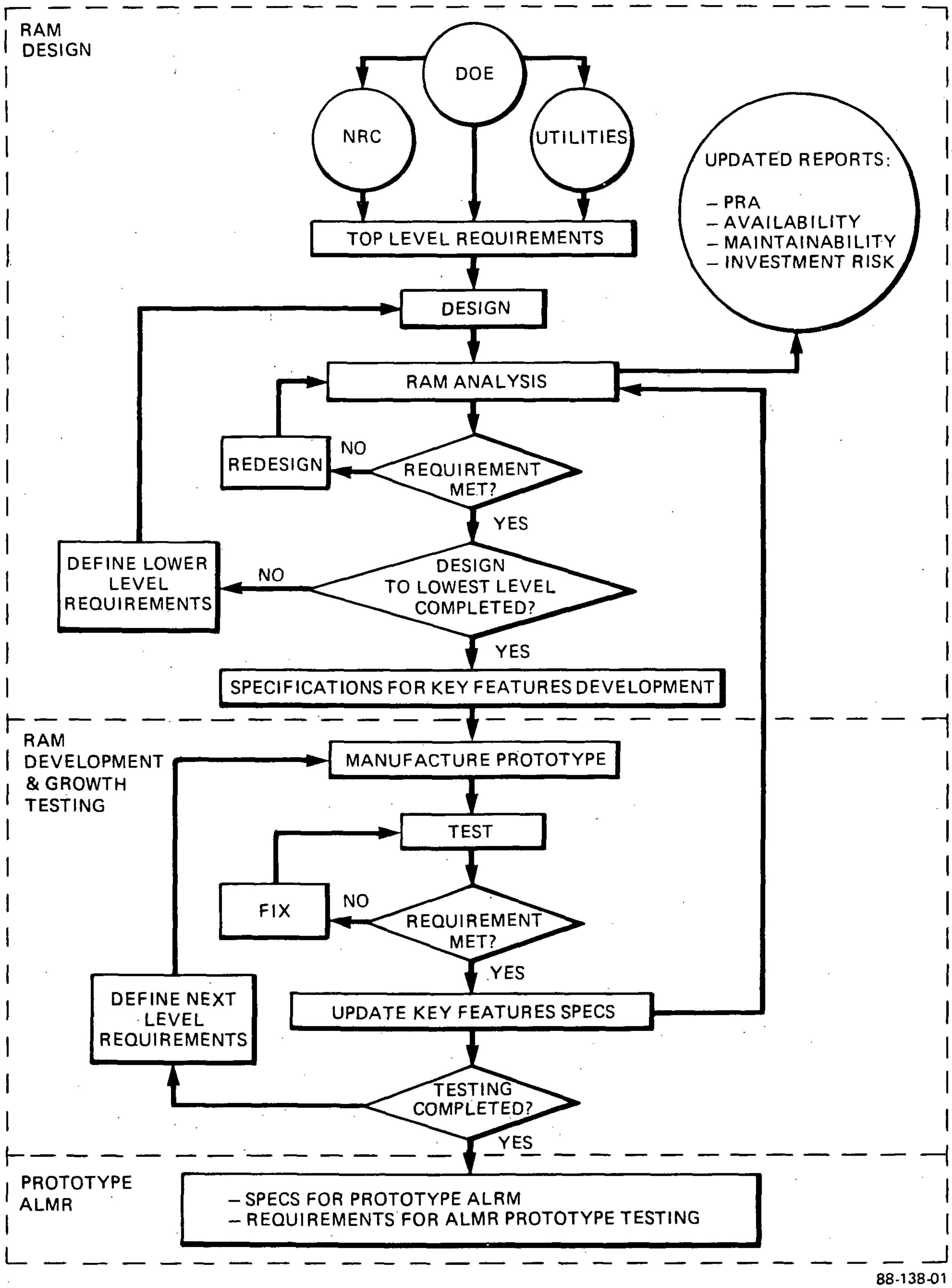

Figure G.3.2–3 INTEGRATED RELIABILITY, AVAILABILITY, MAINTAINABILITY APPROACH

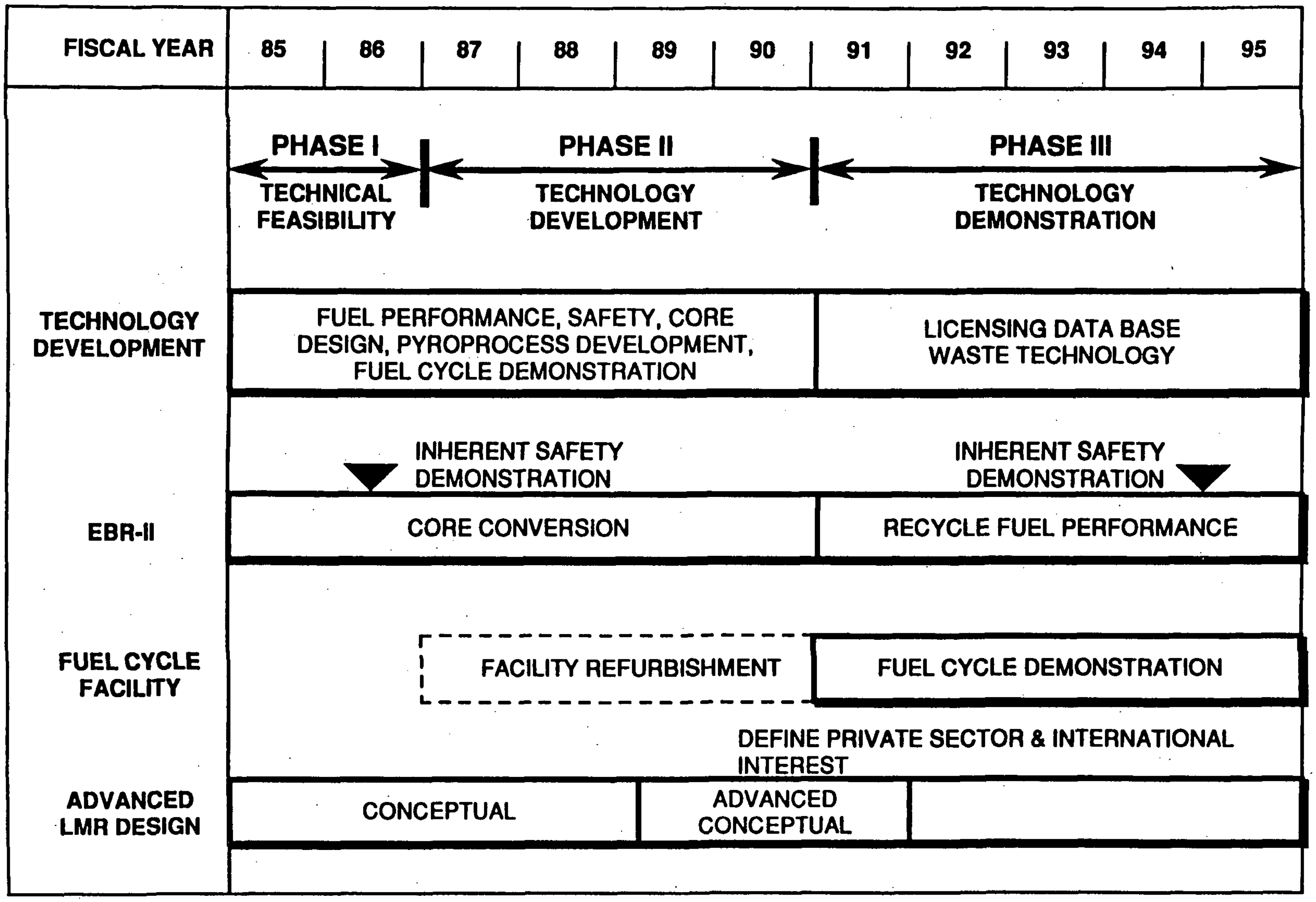

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Figure G.3.2-4 - PROGRAM SCHEDULE FOR THE METAL FUEL CYCLE PROGRAM

G.4.1 Containment

G.4.1.1 SER Position on Containment

Sections 3.1.2, 3.2, 6.2.6 and 15.10.6 of the draft SER (NUREG 1368) address the issue of containment. Section 3.1.2 presents specific licensing criteria concerning the issue of containment and the related key issues of accident selection, siting source term calculation and use, and off-site emergency planning. Section 3.2 discusses general design criteria for containment design (GDC-16), containment design basis (GDC-50), and containment related issues (GDC-38 through -43 and GDC-51 through -56). Under GDC-16 the staff states:

> "At the present time the response of PRISM to certain events does not convince the Staff the present containment design is acceptable..."

Under GDC-50 the Staff states:

> "The containment should be designed to withstand, with sufficient margin, the temperature and pressure conditions resulting from all EC-I through EC-III events, including primary sodium leakage from the reactor vessel, without exceeding the design leakage rate. Margins should be included to account for uncertainties in the accident phenomena and calculations.
>
> "...the adequacy of the current containment design and the acceptability of a design without a conventional containment [are] not resolved."

The Staff reiterates these positions in Section 6.2.6 which states:

> "The staff cannot accept the present PRISM containment design. Specifically the response of the PRISM design to certain of the Bounding Events...does not meet our proposed criteria...for accepting a design without a conventional containment building. The bounding events of concern are:

"1. BE-1 (inadvertent withdrawal of all control rods without scram)...

"2. BE-3 (loss of all decay heat removal for 36 hours)...

"3. BE-4 (instantaneous loss of flow from one primary pump with failure to scram)...

"4. BE-7 (flow blockage of a single fuel assembly)...

"Since the above 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 containment) they represent a fundamental concern with the PRISM design... Resolution of these concerns remains an open item."

In Section 15.10.6, the Staff states:

"Since certain of the Bounding Events identified by the staff for inclusion [in] EC-III have the potential to lead to core melt and/or energetic reactivity accidents, the acceptability of the PRISM design (particularly the containment and off-site emergency planning proposals) is of concern. Until resolved, the staff cannot conclude that PRISM has the potential to achieve a level of safety at least equivalent to current generation LWRs."

The criteria referred to above are described in Section 3.1.2.3 of the draft SER, and can be summarized as follows:

- Meet 10CFR50, 40CFR190, 10CFR100 limits for EC-I, EC-II, and EC-III events
- Demonstrate via a full-size prototype test
- Employ enhanced quality assurance, surveillance, in-service inspection, and in-service testing
- Protect against sabotage and external events

- Take measures to ensure that no core melt accidents, accidents with significant positive reactivity feedback, or other accidents with the potential of a large radiation release are in the EC-I, EC-II, or EC-III spectrum

- Assess the potential improvement in safety if a containment building were added

For the related issue of determining a source term, to be used in evaluating the effectiveness of containment, the Staff states in Section 3.1.2.2:

> "The staff believes source terms can be developed for advanced reactors based on mechanistic analysis provided (1) those source terms are used in conjunction with dose guidelines consistent with those applied to LWRs, (2) the events considered in the mechanistic analysis are selected to bound credible severe accidents and design-dependent uncertainties, and (3) the performance of the reactor and fuel under normal and off-normal conditions is sufficiently well understood to permit mechanistic analysis..."

The dose guidelines referenced above are specified in Sections 3.1.2.2 and 3.1.2.3 of the draft SER as follows:

- For EC-I, meet limits of 10CFR50, Appendix I, and 40CFR190

- For EC-II, meet 10% of the dose limits of 10CFR100, calculated using conservative accident scenarios and conservative meteorology

- For EC-III, meet the dose limits of 10CFR100, calculated using best estimate accident scenarios but conservative meteorology

Section 3.1.2.4 of the draft SER also specifies dose limits for off-site emergency planning. The section states:

"An off-site emergency plan should be prepared, however, such a plan would not have to include early notification, detailed evacuation planning, and provisions for exercising the plan if:

- "The lower level PAGs are not predicted to be exceeded at the site boundary within the first 36 hours following any event in Categories EC-I, II, and III, and

- "A PRA for the plant that includes at least all events in Categories EC-I through EC-IV ...indicates that the cumulative mean value frequency of exceeding the lower level PAGs at the site boundary within the first 36 hours does not exceed approximately 10^{-6}/yr."

G.4.1.2 Reference Containment Design

In the 1986-1987 PRISM PSID, the emphasis was on prevention of accidents to reach the safety goals. Containment and mitigation of severe accidents were not addressed in detail. Consistent with this approach, the design had an unconventional containment in that portions of the primary system boundary doubled as containment. However, in response to concerns stated by the Staff in the draft SER, and quoted above in Section G.4.1.1, the ALMR design has been upgraded with the following three levels of defense:

a. Addition of design provisions to ensure that none of the EC-III bounding events of concern leads to core damage or sodium boiling. This is discussed in Section G.4.16.

b. Addition of design provisions to ensure the integrity of the reactor vessel and vessel closure under hypothetical core disruptive accidents or core meltdown accidents. This is discussed in Section G.4.19.

c. Addition of a low leakage pressure-retaining containment dome and isolation valves in the IHTS piping. The containment dome, the original containment vessel, and the IHTS isolation valves (when closed) now provide a complete containment boundary surrounding the primary system boundary. Containment is discussed in this Section G.4.1.

It is still claimed, however, that the safety goals can be reached by prevention of accidents alone.

G.4.1.2.1 Goals and Analysis Results

As stated above, the reference ALMR design contains modifications to provide additional defense in depth for a full spectrum of severe accidents, including a hypothetical core disruptive accident (HCDA) and a core melt. These modifications include changes to the reactor closure and lower internal structure (see Section G.4.19), and the addition of a low leakage pressure-retaining containment dome and isolation valves on the IHTS piping. The goals of these modifications are to:

- o Limit the probability of severe core damage to less than 10^{-6} per plant year,
- o Assure that the integrity of the primary system and containment boundaries are maintained under postulated core melt and core energetic events, and
- o Assure that the probability of a 1 rem radiation dose at the site boundary over a 36-hour period following a severe accident is less than 10^{-6} per plant year.

It is also a goal of the ALMR program to perform mechanistic analyses to develop source terms and evaluate the site doses for EC-I, -II, and -III events in accordance with the draft SER criteria. Argonne National Laboratory's Integral Fast Reactor (IFR) program will provide the data and analytical tools to meet this goal. For now, however, a conservative source term, selected by engineering judgment, has been used to evaluate the containment. The source term selected is based on the site suitability source term discussed in Section 6.2.3.3 of the 1986-1987 PSID, with the following conservatisms:

a. Release to the containment dome is assumed to occur at time zero (previously the PSID source term was assumed to be located in the cover gas region which leaks at 0.1%/day).

b. Consistent with the above assumption, a leak path is assumed to occur in the reactor closure, as a result of an unidentified cause, which allows (1) He cover gas to escape from the cover gas region, and (2) air from the containment dome to replace the He, initiating a sodium fire.

c. Before the sodium fire, the complete core and in-vessel stored irradiated fuel are assumed to melt, with all fission products uniformly distributed in the primary sodium.

d. The sodium fire is assumed to continue until all the oxygen in the containment dome is consumed.

The above assumptions led to the containment design basis summarized in Table G.4.1-1.

In order to determine if the dose goals are met, the source term has been used as input to the CONTAIN computer code for calculation of the radiological release. The code predicts the fraction of radioactive materials which leak into the environment. Site boundary doses from the leaking radionuclides were then calculated for different weather conditions using the SMART code.

Results of the analyses lead to the following main conclusions:

a. Consumption of oxygen by the sodium fire causes a negative containment pressure seven hours into the accident.

b. Maximum containment pressure during the accident is less than 10 psig (compared to a design pressure of 25 psig) and maximum temperature is less than 370°F (compared to a design temperature of 700°F).

c. Stopping the containment leakage (due to the development of negative pressures) after seven hours leads to a containment attenuation factor of 1000 for noble gases. For aerosols, the attenuation factor is further increased by a factor of 20 because of fallout and plateout.

d. Minimum PAG limits are met with margin, even with conservative meteorology and exposure to ground deposition radiation for one week.

These results indicate that the containment concept selected is viable. The containment performance analysis is presented in Section G.4.1.3.

Table G.4.1-1

CONTAINMENT DESIGN BASIS

	Item	Magnitude: Early Phase (0-10 Sec)	Magnitude: Sodium Fire Phase (10 Sec - 6 Hrs
A.	Materials Released to Containment Through Reactor Closure		
	Noble Gases (Xe, Kr)	100%	0%
	Halogens (Br, I)	0.1%	0.8%
	Alkali Metals (Cs, Rb)	0.1%	1.6%
	Te, Ru	0.1%	0.004%
	Sr, Ba	0.01%	0.0016%
	Fuel & Other Fission Products	0.01%	0.0008%
	Na-22, Na-24	None	0.4%
B.	Energy Sources		
	Sodium Fire (Within Reactor)	None	~1700 lbs
	Decay Heat	Yes	Yes
C.	Leak Rate (Containment Dome)	<1%/day @ 25 psig/700°F	

G.4.1.2.2 Containment Description

Containment During Plant Operation

The ALMR containment provides a second, low leakage, pressure-retaining boundary which completely surrounds the primary system boundary. As Figure G.4.1-1 shows, it includes a lower containment vessel designed to contain reactor vessel leaks and an upper containment dome which will mitigate severe events, such as a HCDA, which are postulated to cause an expulsion of radionuclides through the reactor closure into the region above the reactor. The upper and lower regions of the containment boundary have different requirements since the upper containment dome region is not required to contain primary sodium leaks as is the case for the lower portion which extends below the sodium level in the reactor vessel.

The upper containment is a cylindrical steel (SA516 Grade 70) tori-spherical dome located between the reactor closure and the tornado hardened roof structure of the reactor facility, which is located at grade. The one-inch thick steel lower cylindrical portion of the containment dome is 12 feet high with an inner diameter of 48 feet. The upper cylindrical portion is 24 feet in diameter. The maximum height of the containment dome from the operating floor to the top of the 1.5-inch thick steel dome is 24 feet at the center line. The containment dome is designed to limit leakage to less than 1% of its volume per day at 25 psig and 700°F. Manned access into the operating deck region (above the reactor closure) is accomplished through a personnel air lock. All piping and instrument penetrations through this containment boundary are located above the reactor primary system boundary and well above the operating sodium level. The containment penetrations are similar to those used in LWR containments, including the main loop IHTS piping penetrations which are provided with bellows and single isolation valves immediately outside the containment dome. Open loop containment penetrations, such as the sodium and cover gas cleanup lines, employ double isolation valves.

The upper and lower portions of the containment are connected to each other by a horizontal plate located at the same elevation as the reactor closure and the top of the containment vessel. The lower containment consists of a one-inch thick 19 foot - 10 inch diameter 2-1/4 Cr-1Mo steel vessel. The containment vessel has no penetrations and is designed to remain leak tight at 60 psig and 800°F. A five-inch argon-filled gap between the reactor vessel and lower containment is sized to ensure that the reactor core, the stored spent fuel, and the inlets to the intermediate heat exchangers will remain covered with primary sodium in the event of a reactor vessel leak. This ensures that the internal sodium flow path will not be interrupted and shutdown heat removal via RVACS will operate to maintain safe temperatures within the core and reactor system following a postulated reactor vessel leak. The argon gas is maintained at a higher pressure (~12 psig) than the reactor cover gas which is at atmospheric pressure. The annulus is continuously monitored with pressure sensors, sodium aerosol detectors, and sodium liquid detectors for detection of a leak in either vessel.

Figures G.4.1-2 and G.4.1-3 provide elevation and plan views of the reactor facility, including the containment. Figure G.4.1-4 provides a schematic of the reactor containment boundary and shows the main elements of the primary system boundary.

The primary system boundary includes the reactor vessel, reactor closure, control rod drive housings, instrument dry wells, below-head ducting and tubing of the two IHX units, and the primary sodium and cover gas cleanup system piping up to and including the first isolation valve. During power operation, all sodium and cover gas service lines are closed with double isolation valves, and all other penetrations in the reactor closure are seal-welded. Thus, the primary system is totally sealed during power operation.

Containment During Maintenance and Refueling

During shutdown, when refueling or maintenance operations are being performed, a tornado hardened refueling enclosure (RE) is moved over the reactor. Figure G.4.1-5 depicts the containment boundary which is employed during refueling and maintenance activities. Prior to performing refueling and maintenance operations, the reactor is shut down, the primary sodium is cooled to 400°F, and the cover gas is replaced.

Fuel transfer and equipment removal and replacement operations are accomplished with the use of dual isolation valves, one on the transfer adapter, which provides a sealed leak tight transfer path between the reactor closure and the transfer cask which is located above grade, and one on the fuel or equipment transfer cask. This ensures that a closed primary system boundary is always maintained. Access for refueling, and small equipment removal and replacement, is provided by four ports with removable sealed closures in the head of the containment dome. Thirty-six inch diameter ports are used for the IVTM and sodium purification pump, and 24-inch diameter ports are used for control rod drive line replacement and fuel transfer operations. Replacement of the primary EM pumps and IHX units will require cutting and subsequent re-welding of the raised 24-foot diameter center portion of the upper containment. However, this will be an infrequent operation since the IHX is expected to last the life of the plant, and the EM pumps are expected to be replaced once after 30 years.

Replacement of core assemblies during each refueling outage requires assemblies to be removed and installed through the upper containment and reactor closure boundaries. A procedure has been developed to maintain the integrity of these boundaries during the entire sequence of refueling operations.

During normal reactor operation, the refueling port in the reactor closure is sealed with the self-locking shield plug and a seal-welded cover. The refueling access port in the upper containment is also sealed during operation with a mechanically-secured seal plug. Tornado protection for the containment dome is provided by the reinforced concrete roof structure of the reactor facility.

In preparation for refueling, prior to reactor shutdown, the refueling enclosure (RE) is positioned over the reactor, secured to the seismically isolated roof of the reactor facility, and sealed to the roof upper surface. A roof plug is removed to provide access to an upper containment port. The seal plug in the containment port is removed which establishes temporary communication between the RE and the containment atmosphere. During this operation a self-contained standby gas treatment system maintains the combined regions at a slight negative pressure (about 0.25-inches wg) with the RE doors closed. Preparation for refueling continues with installation of the transfer adaptor between the refueling port in the reactor closure and the roof. An inflatable seal between the roof upper surface and the transfer adapter gate valve body is activated to separate the upper containment and RE atmospheres. A buffered seal arrangement is used permitting the seal to be leak checked. This permits the upper containment integrity to be maintained when the RE door is opened for movement of the fuel transfer cask (FTC) into or out of the RE. The transfer adaptor with its upper end gate valve becomes an extension of the primary system boundary during the refueling operation. The reactor closure shield plug is removed into an inerted cask to provide access to the refueling station in the reactor. The FTC, with its integral gate valve, is attached to the transfer adaptor for replacement of core assemblies. Before the gate valves on both the shield plug and fuel transfer casks and the transfer adaptor are opened, the space between them is evacuated and backfilled with helium and leak checked. With gate valves open, core assemblies are exchanged between the reactor transfer station and the inerted FTC.

A similar sequence is followed for replacement of major reactor components. Transfer casks and transfer adaptors are sized to accommodate the larger components, and the technique for maintaining containment integrity during equipment replacement operations developed for the refueling system is applied to these other operations as well.

G.4.1.3 Containment Performance Analysis

G.4.1.3.1 Description of the Operating Design Basis Event

Extremely severe accidents that could challenge the containment have been evaluated on a probabilistic basis. These assessments indicate that the risk that the primary system boundary would be breached is extremely small, less than 10^{-9} per reactor year. There are two major factors in achieving this low probability. First, an excellent ability to prevent severe accidents due to the high reliability of the reactor protection system, the inherent negative reactivity feedback characteristics, the very high reliability of the passive safety grade reactor vessel auxiliary cooling system (RVACS) to remove decay heat, and the large temperature and structural margins in the design. Second, the ability of the design to withstand the effects of extreme accidents involving gross fuel melting or energetic hypothetical core disruptive accidents (HCDA) without breaching the primary system boundary. Preliminary calculations show that the primary system boundary can contain, without breach, energetic events producing more than 500 MJ of work energy, a level which is substantially greater than the anticipated energetics from any credible HCDA. Nevertheless, to provide in-depth defense, the steel containment described above has been provided and is being designed to mitigate a reactor closure breach.

The design basis event assumes that (1) a relatively large breach in the reactor closure has been created by some unknown mechanism and (2) that 100% of the noble gases (Xe, Kr), 0.1% of the halogens (Br, I), 0.1% of the alkali metals (Cs, Rb), 0.1% of Te and Ru, and 0.01% of other fission products (Sr, Ba) and fuel are instantly released to the containment volume. In addition, it is assumed that the breach in the reactor closure is large enough to allow the He cover gas to escape into the containment dome, and air to enter the reactor cover gas region, initiating a sodium pool fire which continues until all the oxygen in the containment dome is consumed.

Burning of primary sodium within the reactor vessel results in release of radioactive isotopes that are carried with the sodium combustion products, such as sodium aerosols and hot air, into the containment dome atmosphere. It has been conservatively assumed that the complete core melts, and all the fission products are uniformly distributed in the primary sodium before burning initiates. This assumption leads to the additional estimated release of 0.8% of the halogens, 1.6% of the alkali metals, 0.004% of Te and Ru, 0.0016% of Sr and Ba, and 0.0008% of the fuel. In addition, 0.4% of the radioactive sodium isotopes, Na-22 and Na-24, contained in the primary sodium inventory, are assumed to be released into the containment dome atmosphere. The basis for this estimate is discussed in Section G.4.1.3.4.

G.4.1.3.2 Description of the Analysis Model

CONTAIN Code Description

The CONTAIN computer code, used in the safety analysis of the ALMR, simultaneously treats thermal-hydraulic, aerosol, and fission product behavior in the reactor containments under severe accident conditions (Reference G.4.1-1). The liquid metal reactor (LMR) version used in the ALMR containment analysis was updated in 1990. Analysis of the ALMR containment has been performed by the Westinghouse Hanford Corporation (WHC), which has previous experience applying the CONTAIN code to the Fast Flux Test Facility. The code calculates the quantities of radiological isotopes released, but does not include the capability to calculate the radiological consequences of any release from the containment. Radiological consequences have been calculated by GE using the SMART code (Reference G.4.1-2) using release quantities calculated by CONTAIN as input. The results of this analysis are described in Section G.4.1.3.3.

ALMR Input Model Description

The ALMR containment dome has been modeled as a right-circular cylinder, divided into cells to allow establishment of convective air currents within the structure. A hot sodium pool is assumed to be in direct contact with the air in the containment atmosphere. A leak path is provided between the containment and the environment to allow release of material present in the containment atmosphere. Specific parameters used in the model are given in Table G.4.1-2.

A diagram of the containment model is given in Figure G.4.1-6. The size and position of the four cells within the containment dome were chosen to allow convective flow within the dome atmosphere.

Table G.4.1-2

PARAMETERS USED IN CONTAIN CODE

Containment Dome Volume	39,250 ft^3
Containment Dome Internal Diameter	48.0 ft
Containment Dome Atmosphere Initial Temperature	100°F
Containment Dome Atmosphere Initial Pressure	14.7 psia
Sodium Pool Diameter	18.5 ft
Sodium Pool Volume	3450 ft^3
Sodium Pool Temperature	905°F
Containment Dome Leak Area	0.0005 in^2

The containment structure was assumed to be a one-inch thick steel shell, and the floor outside of the sodium pool was assumed to be concrete about three feet thick. Equipment within the containment dome was modeled as a one-inch thick slab 12 feet by 48 feet. Heat transfer between the containment atmosphere and these structures was calculated by the code. The environment outside of the containment dome was assumed to be at a nominal temperature of 77°F. Heat was assumed to be passively removed from the containment dome by natural convection of air, which passes into and out of the region next to the containment dome through the 18-inch wide seismic gap.

The radioactive material released within the containment area was assumed to disperse in the following fashion. The noble gases were assumed to associate with the air in the containment atmosphere, and therefore follow the flow of the containment atmosphere. All the other radioactive materials were assumed to be in the form of aerosols, so that they would either remain in the containment atmosphere, attach to structures within the containment, or settle out onto the floor or sodium pool depending on the aerosol algorithms built into the CONTAIN code. Two aerosol groups were chosen to enable separate tracking of the initial and the sodium burning release phases.

G.4.1.3.3 Containment Analysis Results

Figure G.4.1-7 shows the pressure within the ALMR containment calculated by CONTAIN following the initiation of the sodium pool fire and introduction of the radioactive materials from the primary coolant. The pressure peaks at just under 10 psig. The perturbation in the pressure curve between 50 and 60 minutes into the transient is caused by termination of the reaction between water vapor (100 percent humidity assumed) to be present in the containment atmosphere and sodium oxide produced by the pool fire. This termination, due to the availability of water vapor as shown in Figure G.4.1-8, eliminates one of the energy generation sources to the containment atmosphere, causing the effects seen in the containment pressure and temperature calculations. The reaction of the water vapor with the sodium fire products contributes to the containment pressure decrease noted in Figure G.4.1-7 following the pressure peaking.

Figure G.4.1-9 presents the calculated cell temperatures. Cell 1, immediately adjacent to the sodium pool and the location where the fission products are introduced, is at the highest temperature as would be expected. Cell 2, just above Cell 1 is only slightly cooler due the rapid energy exchange caused by the convective air flows. Cell 4, adjacent to Cell 2 at the top of the containment is slightly cooler due to its larger size and energy losses through the containment shell to the environment. Cell 3 is slightly cooler than 4 for the same reasons. The water vapor/ sodium oxide reaction termination shows up quite plainly in this plot.

Figure G.4.1-10 shows the containment oxygen mole fractions, which continually decrease due to the sodium pool fire. As shown on the figure, containment oxygen is consumed at about 400 minutes into the accident. Depletion of the oxygen within the containment also contributes to the decreasing trend in the containment pressure noted in Figure G.4.1-7. Differences in oxygen mole fractions between cells is small due to the mixing effects of the convective flows which are shown in Figure G.4.1-11. Flow rates between the cells are virtually identical. Flows from Cell 3 to Cell 4, and from Cell 1 to Cell 3, are plotted as negative values since the flows are in the opposite direction to that specified in the input flow path. The convection cell or flow pattern is from Cell 1 to 2 to 4 to 3 and back to 1.

Figure G.4.1-12 presents the oxygen consumption rate, Figure G.4.1-13 the sodium combustion rate, and Figure G.4.1-14 the energy generation rate due to the sodium pool fire. Decrease in combustion energy generation rate is due to the decreasing oxygen content in the atmosphere.

The energy from radioactive decay also decreases with time, reducing the rate of energy input to the containment.

Figure G.4.1-15 shows the temperatures of the various structures in the model during the transient. The figure shows Cell 3 equipment to have the highest temperature, with a peak temperature of about 350°F at 300 minutes into the accident. The containment structure walls reach their peak of 230°F to 270°F at 180 minutes into the accident.

The peak pressure of 10 psig (Figure G.4.1-7) and peak temperature of 270°F (Figure G.4.1-15) are well within the design values of 25 psig and 700°F for the containment dome.

Figure G.4.1.3-16 shows the leakage flow from the containment to the environment. The figure shows the leak rate increases to a maximum of about 0.13 lbs/min shortly after the accident, and then drops to ~0.0 lbs/min in less than 400 minutes. As expected, the leak rate follows a trend similar to that of the containment pressure (Figure G.4.1-7).

Figures G.4.1-17 and G.4.1-18 show the mass of fission products aerosol deposited on the containment walls and floor, and the mass of suspended aerosol in the containment atmosphere for the initial release into the containment before the sodium fire. As seen in the figures, almost all the aerosol from this release either deposits or leaks within the first 20 minutes into the accident.

Figure G.4.1-19 and G.4.1-20 show the mass of fission products aerosol deposited on the containment walls and floor, and the mass of suspended aerosol in the containment atmosphere for the sodium burning release phase. As seen in Figure G.4.1-20, the aerosol mass in the containment atmosphere increases initially up to a peak at around 120 minutes into the accident. This is a result of the continuing sodium burning and aerosol release to the containment. After 120 minutes, as the burning rate decreases (Figure G.4.1-13), aerosol is deposited faster than it is produced. This leads to rapid decline in the aerosol mass suspended in the containment atmosphere, which reaches zero in less than 200 minutes into the accident.

Termination of the leak from the containment before 400 minutes (Figure G.4.1-7) is a direct result of the consumption of the containment dome oxygen in the sodium fire. This termination limits the release of noble gases to the environment to about 0.1%. Aerosol deposition of other fission products and the fuel limits the release fraction of these material to the environment to about 0.005%.

G.4.1.3.4 Radiological Consequence Evaluation

The SMART code (Reference G.4.1-2) was used to estimate the site boundary dose for the reference ALMR release case. The code was developed by the Brookhaven National Laboratory for the US NRC as a fast running tool to estimate the early dose and health effects resulting from severe accidents. The code uses CRAC2 and MACCS dose models and has been validated against these models. The code uses any one of seven weather types (A through G). The code calculates the dose to different organs at various distances from the point of release as a result of inhalation and direct radiation shine from a passing radioactive cloud, and the dose from ground deposition for one day or one week of exposure. The dose estimates reported here are based on one week calculations. Consequently, the estimates are conservative for the two-hour dose, 36-hour dose, or any exposure less than one week in duration. The radioactivity inventory, release parameters and weather conditions used in the analysis are described below. This is followed by a presentation and discussion of the results.

Radioactivity Inventory

The end of equilibrium cycle (EOEC) radioactivity inventory was estimated using the ORIGEN2 code for the current reference ALMR core (53 inch core height, no axial blanket, 3 GEMs, 5% axial expansion, 75% smear density, and LMR recycle equilibrium core). The total inventory, including all core assemblies (42 driver fuel, 24 internal blankets, and 33 radial blankets) and the 14 fuel and 12 blanket assemblies in the in-vessel storage, was assumed to have become molten and been dispersed uniformly in the primary sodium coolant before the hypothetical fire of 1700 pounds of primary sodium started. The equilibrium activation product inventory, including Na22 and Na24, was also assumed to be uniformly distributed in the primary sodium.

Release Parameters

The accident defined for evaluating the ALMR containment results in release of radioactivity from the reactor vessel through the reactor closure to the containment dome in two phases. In the first phase, the release is assumed to occur in 10 seconds and to involve 100% of the noble gases, 0.1% of the halogens, alkali metals, Te, Ru, and 0.01% of the remaining core inventory. In the second phase (burning of ~1700 pounds of primary sodium) the release is assumed to follow the initial release for about six hours. The radioactivity released with the burning sodium was estimated using test results reported in Reference G.4.1-3 for the retention factors in burning sodium for various radioactive groups. The retention factor is defined as the ratio of the concentration of a radioactive element in the sodium pool to that in the burning sodium aerosol. Table G.4.1-3 shows the retention factors used in estimating the sodium burning release to the containment.

Using the above information, rates of release into the containment were estimated for 56 radioisotopes (54 default radioisotopes used in the SMART code library, plus Na22 and Na24). The rates obtained were used in the CONTAIN analysis which calculated the leak rate and accumulated release to the environment of the various radioisotopes.

Table G.4.1-3

RETENTION FACTORS FOR BURNING SODIUM

Halogens (I, Br)	0.5
Cs, Rb	0.25
Ru, Te, Rh	100
Sr, Ba, Rare earths	250
Fuel and actinides	500

As discussed in Section G.4.1.3.3, the CONTAIN results indicate that the release of radioactivity to the environment stops in less than 400 minutes due to the drop in containment pressure. Since the SMART code accepts release data in the form of puffs only, the SMART code analysis conservatively assumed that the release to the environment starts at time zero, and that the release of all the radioactive materials predicted by CONTAIN is completed in 1/2 hour. The release height was assumed to be ground level.

Weather Conditions

Three weather conditions were selected for the SMART code analysis which cover nominal conditions and two conservative weather assumptions.

The nominal weather condition corresponds to the 50th percentile of the weather conditions of the ALMR site. This has been estimated to be weather Type D (neutral) with a wind speed of 4 m/sec.

The first conservative weather condition corresponds to the 95th percentile of the above site. This has been estimated to be weather type F (moderately stable) with wind speed of 2 m/sec.

The second conservative weather condition corresponds to the one specified in Regulatory Guide 1.3 or 1.4 for the first eight hours of release. This corresponds to weather type F with a wind speed of 1 m/sec. This weather type is the most conservative.

Dose Results:

The dose to various organs at the site boundary (1/2 mile) was estimated using the SMART code for each of the weather conditions specified above. For the inhalation dose, the breathing rate of Regulatory Guide 1.3 or 1.4 was used. Table G.4.1-4 shows the protective action guidelines (PAG) dose limits for various organs and the estimated doses for the three weather conditions assumed in the SMART analysis. The dose estimates are the sum of (1) the inhalation dose and direct radiation shine dose from the passing cloud over the duration of the release, and (2) the dose from ground deposition for one week of exposure. Therefore, the estimates are conservative for the 36-hour dose requirement. The estimated doses are presented in rems, and as a percentage of the PAG limits. As seen in the table, the ALMR containment meets the PAG limits with substantial margin, even for the most conservative weather conditions.

G.4.1.3.5 Containment Performance Following Maintenance and Refueling Accidents

EC-III events are defined as severe accidents which have a probability of occurrence which is greater than 10^{-7} and less than 10^{-4} per plant year. A major safety goal of the ALMR design is to ensure that accident events with a mean frequency higher than 10^{-7} per reactor year are mitigated such that no radioactive releases exceeding 10CFR100 limits can occur. In addition, to avoid the need for a formal evacuation plan, releases resulting from accidents within EC-I, II, and III shall not cause a dose at the site boundary which exceeds 1 rem during the first 36 hours.

Three severe maintenance and refueling accidents have been defined and analyzed under EC-III criteria (nominal analysis and conservative meteorology), based on the assumption that their probability is sufficient to warrant their inclusion within EC-III. The three accidents analyzed are:

- A major maintenance accident resulting in a large opening in the reactor closure
- A large primary sodium spill
- A major refueling accident

Table G.4.1-4

SITE BOUNDARY DOSE FOR CONTAINMENT DESIGN BASIS EVENT

		Weather Type					
		ALMR 50th Percentile		ALMR 95th Percentile		Reg. Guide 1.3 & 1.4	
Organ	PAG Dose Limit (Rem)	Dose (Rem)	% PAG	Dose (Rem)	% PAG	Dose (Rem)	% PAG
Whole Body	1.0	2.2E-2	2.2	9.8E-2	9.8	1.9E-1	18.6
Bone Marrow	1.25	2.6E-2	2.1	1.2E-1	9.3	2.2E-1	17.6
Lung	1.25	5.0E-2	4.0	3.0E-1	23.7	5.1E-1	40.7
Thyroid	5.0	8.3E-2	1.7	5.2E-1	10.0	8.7E-1	17.3

Each of these accidents is described below and evaluated for the protective mitigation provided by the secondary containment system in use during these off-line maintenance and refueling operations.

Maintenance Accident

The maintenance accident consists of an inadvertent breach in the reactor closure due to an accident during reactor maintenance. The EM pump replacement was chosen for this event because it is the only large component scheduled to be replaced. The scenario for this accident is that: (1) the reactor is in the cold shutdown mode (400°F); (2) the cover gas is at atmospheric pressure and has been replaced with clean helium; and (3) the primary EM pump is being replaced.

Replacement of the pump involves: (1) installing the refueling enclosure over the reactor facility, (2) removing the access hatches in the HAA roof and cutting the 24-foot diameter center portion of the upper containment, (3) installing the transfer adapter with gate valve to provide a leak tight chamber around the pump closure plug between the reactor deck and the transfer cask, (4) installing a pump transfer cask with gate valve in the maintenance enclosure, (5) inerting the adapter chamber and transfer cask, (6) operating the gate valves lifting the EM pump with integral plug into the transfer cask, (7) closing the adapter and transfer cask gate valves and, (8) removing the transfer cask with pump to the maintenance facility. The EM pump transfer cask is shown in Figure G.4.1-21 and the EM pump adapter in Figure G.4.1-22.

The assumed accident occurs when the transfer adapter is removed without installing a replacement EM pump, leaving a 41-inch diameter opening in the reactor closure. Air enters the reactor vessel and quickly displaces the hot helium cover gas. The oxygen in the air supports a pool fire 270 ft^2 in area at the sodium surface in the reactor vessel. The aerosols formed fill the upper containment and refueling enclosure regions with radioactive sodium oxides. The sodium pool burns for one hour at a rate of about 2 lb/hr-ft^2 (~1/4 the combustion rate of sodium in an open air environment) consuming 540 lb of sodium. About 20% or 110 lb of the burned sodium is released into the containment volume (upper containment and refueling enclosure) as Na_2O aerosol (150 lb Na_2O) during this period. During the initial pool burning phases, an oxide crust forms on the surface but gradually settles to the bottom of the pool.

For large pool fires in air, the average burning rate is about 6-8 lbs/hr-ft^2. The burning rate is roughly proportional to the oxygen concentration. Unless finely divided with a high surface-to-weight ratio, sodium does not normally ignite in an air atmosphere at temperatures below its melting point. The minimum oxygen dry air fraction at which a pool of sodium will ignite spontaneously is about 5% at 650°F; 7.5% at 485°F; 10% at 440°F; 15% at 430°F; and 21% (air) at 400°F. Thus, ignition would occur in the reactor vessel only after essentially all the hot helium in the cover gas region was displaced with air.

The pool fire in the reactor vessel is terminated after one hour by inerting the reactor cover gas region with nitrogen gas available from the IHTS/SGS nitrogen supply system. The nitrogen supply system for each power block has a minimum nitrogen inventory of 80,000 scf to be used for inerting and purging the steam generator and SWRPRS in the event of a sodium water reaction. A 1-1/2 inch diameter nitrogen line is run from this system, which has a service station located in the primary sodium processing system equipment vault, to the reactor cover gas helium supply line, which is located outdoors near the vault. The line is equipped with locked closed double block and bleed valves to prevent inadvertent operation. Nitrogen can then be supplied directly to the reactor cover gas region through this line by opening the appropriate remotely operated isolation valves. A nitrogen flow rate of about 40 scfm is established to blanket the sodium surface. The gas is heated to about 1200°F in the reactor vessel and establishes a positive nitrogen gas flow of about 128 scfm to produce a 0.2 ft/sec velocity through the opening. The heated nitrogen gas will purge the 1600 ft^3 cover gas volume in about 15 minutes. After about three volume exchanges, all the oxygen and sodium aerosol present in the vessel will have been vented from below the reactor closure, leaving only makeup required for the loss through the opening. The nitrogen inventory available on site from the three power blocks can support this operation for about 100 hours (~4 days), and this support could be extended infinitely with appropriate deliveries of nitrogen gas from off-site.

Sodium aerosol monitors and radiation detectors in the containment dome region signal the initiation of this event, and activate the standby gas treatment system in the refueling enclosure if this system has not previously been activated per the operating procedures. The standby gas treatment system operates to maintain a negative pressure of 0.25-inch wg so that leakage is into the enclosure, and filtered prior to release. The air temperature within the containment rises slightly (to <150°F) because of the hot gases from the breach in the reactor closure. More than 99 percent of the sodium aerosols are removed from the vented gas by the particulate filters in the standby gas treatment system. After about four days, the radiation level from Na-24 activation in the containment region will decrease so that safe access can be gained to plug the opening in the reactor closure.

This maintenance accident is very unlikely because the high radiation levels in the upper containment, when the pump with its shielding plug is removed, dictate special procedures which would be difficult to circumvent. Personnel access will be limited at this time because of the increased radiation. The radiation level is closely monitored during the maintenance operation, and only after a replacement pump is in place can personnel gain access to complete the bolting and sealing operation at the pump flange. In addition, the transfer adapter is evacuated and purged with clean inert gas prior to removal to mitigate potential contamination from any leaking fuel pins. The absence of a pump closure plug would be detected during this evacuation process by a drop in reactor cover gas pressure.

A radioactive release of aerosol containing 110 pounds of primary sodium into the containment over a two hour period would not result in an excessive dose at the site boundary (1/2 mile). The sodium radioactivity is conservatively assumed to be reduced by about one half due to sodium aerosol deposition, and 99 percent of the remainder is removed by the gas treatment system prior to discharge to the environment. A radiological analysis has been performed conservatively assuming that the sodium discharge of 0.6 pounds occurs over a two-hour period. The total body dose, lung dose and thyroid dose for the radioactivity released with conservative meteorology are a small fraction of a rem at 1/2 mile and are within the acceptance limits of 10CFR100.

Primary Sodium Spill In the PSPS Piping Tunnel

Primary sodium is circulated during shutdown from the reactor vessel through the primary sodium purification system (PSPS) at a rate of 60 gpm. A PSPS cold trap and plugging temperature indicator are located in a vault, and alternately service each of the three reactor modules in a power block. Only one reactor is connected to the PSPS at any given time. The PSPS sodium pipes in the upper containment are protected by guard pipes which prevent sodium leakage into the containment in the event of a pipe leak. Leaks external to the containment will be controlled by a guard pipe and drain system which directs the leakage to catch pans and fire suppression decks in the pipe tunnel. The below grade PSPS pipe tunnel is about seven feet high and five feet wide inside, and extends the entire length of the reactor building on the north side (~200 feet). The sodium inventory in the PSPS equipment and maximum length of non-isolated interconnecting piping is about 1100 gallons. A hydraulic profile drawing of the PSPS is shown in Figure G.4.1-23. Forced circulation of sodium through the PSPS is provided by a submerged EM pump located in each reactor vessel. A sketch of the PSPS EM pump is shown in Figure G.4.1-24.

In the event of major pipe leak, the PSPS EM pump could pump sodium into the guard pipe at a rate of 60 gpm (or at slightly higher rate due to pump runout) until manually shut off or isolated. At refueling conditions, the EM pump suction is submerged six feet below the reactor sodium surface. The reactor vessel contains about 2000 gallons of sodium per foot of height at the pump elevation. The PSPS and the interconnecting piping are located above the reactor normal level and refueling level to avoid the potential for siphoning in the event of a PSPS pipe or component leak. Three sodium spills were considered and the worst one chosen for containment evaluation.

The case selected assumes that the pump electrical power supply is interrupted one hour after the leak and that 3600 gallons (28,000 lbs) of sodium at 400°F are spilled. The sodium discharged results in a pool fire in the guard piping around the PSPS piping and in the PSPS tunnel where most of the sodium would drain. A cross sectional view of the pipe tunnel at one reactor module is shown in Figure G.4.1-25. The tunnel is equipped with a sodium catch pan and a fire suppression deck, contains an air atmosphere, and has a leak rate of 100% per day. The PSPS guard pipe is equipped with downcomers to direct the sodium spillage below the fire suppression deck. The guard pipe is sealed between modules to confine sodium spillage to one module. The reactor itself would not be endangered. However, the sodium spill would result in a release of radioactive sodium aerosols into the PSPS pipe tunnel since the primary Na cleanup system is started 8-12 hours after a reactor shutdown. About one percent of the sodium spill, 280 pounds, is conservatively assumed to burn in the confined pipe tunnel and 10 pounds of sodium as aerosol (~3 percent of the sodium oxide formed) is conservatively assumed to be released during a two hour period. The accident is terminated by the fire suppression deck which extinguishes the fire by cutting off the oxygen supply.

A radioactive release of aerosol containing 10 pounds of primary sodium over a two hour period would not result in an excessive dose at the site boundary (1/2 mile). The direct release to the atmosphere of this amount of aerosol would result in total body, thyroid and lung doses less than 1 rem at 1/2 mile using conservative meteorology, and the doses for the sodium spill accident are within the acceptance limits of 10CFR100.

Refueling Accident

During reactor refueling, fuel transfer is accomplished with a portable passively cooled cask which is permanently attached to the cask transport car. The fuel transfer cask (FTC), shown in Figure G.4.1.-26 is 22 feet-10 inches high and 65 inches in outside diameter. The cask has a 16.5-inch thick shielding cylinder around the 32-inch diameter, six element cavity. Within the cask is a carousel with six storage positions. The carousel is suspended from the top of the cask and can be rotated to align the core assemblies with the closure transfer port. The carousel is motor driven for positioning. The cask is designed for a gas leakage rate of less than one percent per day at 20 psig pressure.

The fuel assemblies are stored in the reactor for 18 months after which the decay power has decreased to a maximum decay heat load varying between 2.5 kW for the equilibrium LMR recycle core to 3.8 kW for a startup core assembly produced from fissile actinides from reprocessing spent LWR fuel. Blanket assemblies are also stored for 18 months before removing them from the core. After 18 months their decay heat level will be less than a tenth of the fuel assemblies. At this low decay level, the fuel assemblies do not require active cooling and can be transferred directly to the fuel service facility (FSF) for storage before being transferred to the fuel cycle facility for reprocessing. Fourteen fuel assemblies, 12 blanket assemblies, and two control assemblies are replaced every 18 months from each reactor module. In addition, about 100 blanket, shield, and reflector assemblies are shuffled or rotated at each refueling.

The postulated refueling accident occurs when one "hot" fuel assembly is inadvertently loaded into the fuel transfer cask instead of a low decay level assembly. The "hot" assembly is assumed to be removed 40 hours after shutdown and has a peak decay power of 38.7 kW (equilibrium recycle), or 44.7 kW (startup core). The maximum total heat load in the cask at this time is 63.5 kW of decay heat rather that the 22.6 kW expected. The fuel pins of all six fuel assemblies in the cask overheat and fail at a temperature of about 2000°F in less than 30 minutes, releasing gaseous fission products and molten fuel into the cask. The cask is designed so that the melted fuel cannot form a critical mass. The molten fuel is contained within the cask. However, all of the gaseous and volatile fission products (xenon, krypton, cesium, rubidium, bromine and iodine) are assumed to leak out of the cask through the damaged seals in the sliding gate valve into the refueling enclosure. One half of the cesium, rubidium, bromine, and iodine is assumed to plate out in the refueling enclosure, and 99 percent of the remainder is assumed to be removed by the vent filters in the standby gas treatment system. The total release from the cask to the environment is conservatively assumed to take place uniformly over a 24 hour period.

This accident is highly unlikely to occur because it requires an error in selecting the fuel assembly for removal, failure of the radiation monitors to detect the "hot" assembly, and failure of the cask instrumentation to detect the excessive increase in the temperature of the He within the cask. Since it takes about 1.5 hours to load each fuel assembly, and additional time to prepare the cask for transport to the FSF, the accident would be detected first by the cask radiation monitors, and then confirmed by the He gas temperature monitors prior to isolation and movement of the cask. Upon detection, the "hot" fuel assembly would then be transferred back to the reactor vessel and submerged in sodium.

If a fuel melt occurred while the cask is positioned in the refueling enclosure and connected to the reactor vessel, the hot debris would drop into the 400°F reactor sodium pool and solidify. Although this accident would result in a reactor cleanup problem, the radiological release from the ruptured fuel assemblies would be essentially confined within the primary coolant boundary and very little release to the refueling enclosure would occur since the cask would not be pressurized. If the accident occurred within the refueling enclosure after the cask was isolated, the release of the fission gas into the refueling enclosure would be mitigated by the standby gas treatment system (SGTS). The case of a fuel melt occurring after the cask is removed from the refueling enclosure has not been evaluated since such a fuel melt would occur long before the cask has been prepared for transport to the FSF.

The environmental radioactive doses resulting from a release to the refueling enclosure of all of the gaseous and volatile fission products uniformly over a two-hour period were determined. The release was assumed to be vented to the atmosphere uniformly over a 24-hour period through the SGTS. All of the xenon and krypton is released and 0.5 percent of the cesium, rubidium, bromine, and iodine is released to the atmosphere. The release results in total body dose of 0.20 rem, a thyroid dose of 4.39 rem and a lung dose of 1.13 rem at 1/2 mile using conservative meteorology. The doses for the refueling accident are within the acceptance limits of 10CFR100 and the PAG.

G.4.1.3.6 Analysis of IHX Failure

The intermediate heat exchanger (IHX) is part of the primary system boundary. The boundary provided by the IHX is backed up by the corresponding closed loop IHTS which functions to contain the radioactive primary sodium within the reactor vessel by operating at a higher pressure during normal and upset conditions to make leakage of the radioactive sodium into the IHTS loop unlikely. Safety-grade isolation valves on each IHTS pipe immediately outside the containment dome provide additional protection. The reactor module is also designed to allow cover gas venting to the primary sodium storage tank (PSST), if necessary, to reduce the reactor cover gas overpressure and to eliminate the potential leakage of radioactive sodium. The primary cold trap system may also be used to transfer sodium from the reactor module to the primary sodium storage tank to reduce the reactor cover gas pressure. Table G.4.1-5 summarizes the results of analyses for three IHX failure categories of events discussed in the following sections. These analyses are relevant to a period during which the IHTS isolation valves remain open. The events analyzed could be terminated earlier by closure of the valves. The required times for valve closure during these events have not yet been determined.

<u>IHX Failure During Normal Operation</u>

The ALMR IHX is designed to the criteria of the ASME B&PV Code Section 111, Division 1, "Nuclear Power Plant Components." The following design margins assure the structural integrity of the component during normal and accident operation conditions: (1) 42% minimum margin of safety for the fluid elastic whirling vibration and vortex shedding, (2) 89% minimum margin of safety for a 0.5g earthquake, (3) a lower plenum head adequate for 6000 cycles of the "Loss of Power to the Intermediate Pump" transient event, (4) an intermediate side of the IHX designed for 1000 psig pressure with margin under the ASME Faulted (Level D) Conditions, and (5) a primary side of the IHX which can withstand 760 psig under the ASME Faulted (Level D) conditions.

Table G.4.1-5

SUMMARY OF IHX FAILURE ANALYSIS

	Normal Operation	Concurrent SG Leak	IHTS Pipe Break
Probability/Plant Year	$<10^{-2}$ (1)	$<10^{-7}$ (2)	$<10^{-8}$ (3)
Event Category	II	IV	IV
Breach of IHTS Boundary	No	Yes	Yes
Potential Primary Na Spill, Gals	None	None	9000
Mitigation Method	N/A	N/A	Close Isolation Valves or Cover Gas Vent or Na Transfer to PSST
Potential Radiological Release, Whole Body	None	None	<1 Rem

(1) Small IHX Leak
(2) Small SG Leak
(3) Medium Size Pipe Break

LMR experience has shown IHXs to be extremely reliable, leak free components having a failure rate of less than 10^{-3} failures per reactor-year. It is not expected to leak during the life of the plant. However, in the event of an IHX leak during normal plant operation, the intermediate sodium pressure will be a minimum of 35 psi higher than the primary sodium pressure so that the sodium leak will be from the non-radioactive intermediate system to the primary system. The reactor cover gas pressure during full power conditions is slightly less than atmospheric (~14.4 psia). The time required to detect the leakage will depend on the leak size. Leak detection on high reactor vessel sodium level will prompt the RPS (Reactor Protection System) to scram the reactor. Detection from high reactor cover gas pressure or low intermediate system sodium level will cause the reactor to be shutdown manually.

With the normal plant heat sink, the reactor system and the IHTS loop will be cooled to hot standby conditions (550°F) within 30 minutes. As the primary sodium temperature drops, the reactor cover gas pressure will decrease. The sodium ingress will continue but at a slower rate as the IHTS pump coasts down to pony motor speed. The leak is terminated by closing the isolation valves, and if necessary, draining the IHTS sodium and flooding the system with nitrogen. No primary sodium leaks out of the reactor system boundary.

IHX Failure Following A Sodium Water Reaction Accident

Sodium water reaction (SWR) is a unique accident event for LMRs, and special features have been incorporated into the IHTS/SGS design to protect the IHX and mitigate the effect of the sodium water reaction. These key features include: (1) selection and development of a reliable SG helical coil concept; (2) a triple redundant hydrogen detection system for early SWR detection and plant shutdown; (3) a sodium-water reaction pressure relief system (SWRPRS), with large (28-inch) safety grade rupture disks set at 325 psig, and a 30-inch relief line for rapid sodium expulsion; (4) redundant water and steam side isolation valves, and a rapid blowdown system to terminate the accident; (5) cover gas space in the SG to attenuate the SWR pressure pulse; (6) steam bypass flow path in the SG to reduce the sodium inlet and outlet nozzle pressure differential and thereby prevent the sodium/steam interface from moving toward the IHX; (7) 1000 psig structural design for the tube side of the IHX with margins in the ASME Faulted (Level D) Conditions; (8) a nitrogen purge system to prevent sodium fire in the SG and IHTS loops; (9) isolation valves on each IHTS pipe at its containment dome penetration; and (10) a cover gas vent system (dual isolation valves are opened) allowing the reactor cover gas to vent to the PSST to prevent the cover gas pressure from exceeding 4 psig during the subsequent 30-hour RVACS heat up transient. The protection of the IHX from SG leak events, in which it is assumed the steam-side isolation and blowdown system fails, is discussed in detail in Section G.4.8, Sodium/Water Reaction Pressure Relief System. Although the IHTS/SG system has been designed to prevent a radiological release in the event that the non-safety grade steam and feedwater systems fail to terminate the event, safety grade main loop isolation valves have been added to assure that the integrity of the containment will be maintained. The reactor will be scrammed and the IHTS isolation valves closed upon detection of a major SG leak event. Two diverse signals will be employed to initiate a reactor scram and closure of the IHTS isolation valves. The first signal will emanate from the sodium detectors downstream of the 28-inch rupture disks in the SWRPRS. The second signal will come from redundant safety grade IHTS pressure sensors located within the NI.

Small, intermediate, and large SG leak events can be postulated. Leak testing experiments performed at ETEC and around the world have demonstrated that small (<50 gm/sec) leaks will progress very slowly by erosion of the SG tubes. Therefore, considerable time (hours) will be available to detect the leak with the hydrogen detectors in the sodium loops or in the cover gas space of the SG so that the plant control system (PCS) can initiate a runback and proceed with a rapid shutdown of that power block. The small amount of sodium oxides and hydrides introduced into the IHTS loop will not cause corrosion damage and affect the structural integrity of the IHX tubes. The reactor system will be cooled to hot standby conditions before the SG is isolated, the steam/water will be blown to the water dump tank, and the ACS/RVACS cooling system will be used to remove the shutdown heat load. The IHTS loop will be drained for repair after the reactor heat generation rate has decayed enough that the auxiliary cooling system (ACS) is not required (~4 days) to prevent the system from exceeding ~600°F. Thus, the overall safety impact of a small SG leak, even if the IHX has a concurrent leak, is minor and similar to that of an IHX failure during normal plant operation with no appreciable safety consequences.

The probability that a small leak will not be detected and progress into a medium to large sodium water reaction is small. However, if a medium or large leak occurs before corrective action can be taken, the shell side pressure of the SG will rise rapidly, but the peak pressure will be limited to about 325 psig by the attenuation of the SG cover gas and the action of the 28-inch diameter rupture disks. The SG will then depressurize as the sodium/water reaction pressure relief system vents the shell side of the SG to the reaction product separation tank (RPST). The relief system is designed to remove the sodium and reaction products from the SG within 30 seconds while venting the steam and gas to the atmosphere through a relief stack. The SGS is also designed to provide fast steam/water isolation and blowdown within 30 seconds, venting steam to the atmosphere and the water to a water dump tank. With the steam/water line isolated, the SG will be flooded with nitrogen to terminate the event. The SG shell side back pressure now will be close to atmospheric as the SWRPRS is opened to the atmosphere through the RPST relief stack.

The IHTS piping configuration is designed to assure that a positive sodium static head relative to the primary sodium in the reactor module is maintained even after a severe sodium water reaction accident. As shown in Figure G.4.1-27, the initial static head in the intermediate sodium loop is about 55 feet above the sodium level in the reactor during normal operation and more than 47 feet above the primary level following a SG leak event (see Figure G.4.1-28).

The IHX leak rate, assuming a concurrent IHX failure has occurred, will depend on the leak size and pressure differential following the SG leak. In the unlikely event the IHTS isolation valves fail to close, the reactor cover gas pressure will rise slowly as the reactor module is experiencing a RVACS heatup and sodium ingression transient. As the secondary sodium is drained into the reactor and its static head drops, the sodium level in the reactor module will rise and pressurize the cover gas. Consequently, the pressure differential across the leak will diminish, resulting in a slower sodium leak rate and a prolonged quasi-steady state condition for the accident.

As the reactor module heats up during the 30 hour RVACS transient, some primary sodium will be forced into the tube side of the IHX. The peak cover gas pressure will be less than 21 feet of sodium (~8 psig). This is less than the static head (47 feet) required to force sodium over the IHTS loop high point and into the RPST. Consequently, radioactive sodium will not pass through the failed IHX into the secondary system due to the higher elevation of the secondary system. To provide additional assurance that this event will not result in a radiological release, IHTS isolation valves have been added to the system.

After RVACS cooling of the reactor module to less than 800°F, the remaining sodium in the IHTS will be drained and purged with inert gas. If the accident requires it, the IHTS piping will be cut and capped outboard of the isolation valves. For example, if an IHTS isolation valve has failed to completely close, the IHTS loop will be purged through the IHX argon gas vent lines, and the pipe jacket and insulation removed to prepare for the IHTS loop isolation. An inflatable bladder will be inserted into the IHTS pipe through a plug just outside the containment dome, as illustrated in Figure G.4.1-29. Following installation, the bladder will be inflated with gas to isolate the loop and to ensure that atmospheric air cannot reach the IHX. The pipe will then be cut and a welded cap installed.

IHX Failure Following An IHTS Pipe Break

A large IHTS pipe break concurrent with an IHX failure accident is a very low probability event (< 10^{-8} per plant-year). The probability of an IHTS pipe break and the impact of the resulting sodium spill are minimized by: (1) the use of ductile material (304SS) to ensure a leak before break piping characteristic so that a large IHTS pipe break is extremely unlikely, (2) the use of guard pipe within the steel containment to contain IHTS sodium, (3) the use of a leak jacket in the IHTS piping to promptly detect a pipe leak and minimize the amount of the IHTS sodium spill, (4) the use of a catch pan and sodium fire suppression deck to collect the sodium and extinguish the fire to minimize the amount of sodium aerosol released into the atmosphere, (5) the use of a nitrogen purge system to prevent sodium fire in the IHTS loops, (6) isolation valves on each IHTS pipe at its containment dome penetration, and (7) a cover gas vent system to prevent the pressurization of the reactor system.

Any IHTS pipe failure is likely to result in only a small sodium leak because of the use of ductile material for the piping design and the fact that it is a moderate energy fluid system (<100 psig). With prompt detection of a sodium leak, rapid shutdown of the reactor, and cooling of the heat transport systems, the safety impact of a small pipe leak followed by an IHX failure accident is similar to that of an IHX failure during normal plant operation. Extra steps will be required to clean up the IHTS loop and the pipe tunnel, but the radiological effect on public safety is inconsequential.

The probability of a large IHTS pipe break will be several orders of magnitude less than that for small pipe leaks. In this event, the intermediate sodium will spill into the catch pan and fire suppression decks located in the IHTS pipe tunnel or the SG building where the non-radioactive sodium will react with air. A rapid IHTS sodium drain may be initiated to minimize the sodium spill. However, only a small fraction of the sodium will burn (<2%) in the pipe tunnel due to the use of catch pans and fire suppression decks.

The pipe break location which would result in the largest spill of radioactive sodium is along the horizontal plane of the IHTS hot leg pipe where only about 4 psig pressure is required to force primary sodium out of the IHX and spill into the pipe tunnel. Other leak locations in the IHTS loop will tend to increase the head required to force primary sodium out of the IHTS loop and the time to reach that pressure head.

Closure of the IHTS isolation valves will prevent primary sodium from being forced into the IHTS piping. However, cover gas pressure control as described below will also prevent a radioactive release. A IHX leak will not impose an immediate reactor module overpressurization danger as the amount of sodium ingression from the intermediate system into the primary system is small. Even if the IHTS isolation valves are not closed, venting of the cover gas during the course of the RVACS heatup transient will prevent the cover gas pressure from exceeding 4 psig. A large IHX failure will allow more intermediate sodium to be drained into the reactor module, raising the cover gas pressure to a level consistent with the secondary sodium column in the intermediate sodium loop. As the pipe break causes the intermediate sodium level to drop and decrease the corresponding static head in the IHTS loop, the resulting higher cover gas pressure in the reactor will force radioactive sodium into the IHTS stream where it will spill into the pipe tunnel together with the non-radioactive sodium. Controlled venting of the reactor cover gas will limit the cover gas pressure to less than 4 psig and prevent the sodium from being forced up to the elevation of the IHTS piping outlet line. An alternative approach to relieving the cover gas pressure is to transfer reactor sodium to the primary sodium storage tank (PSST) through the primary cold trap system, to increase the reactor cover gas volume, and thereby prevent the pressurization of the reactor module. With the cover gas pressure kept below 4 psig, radioactive sodium cannot be forced out of the IHX and there will be no radioactive release to atmosphere.

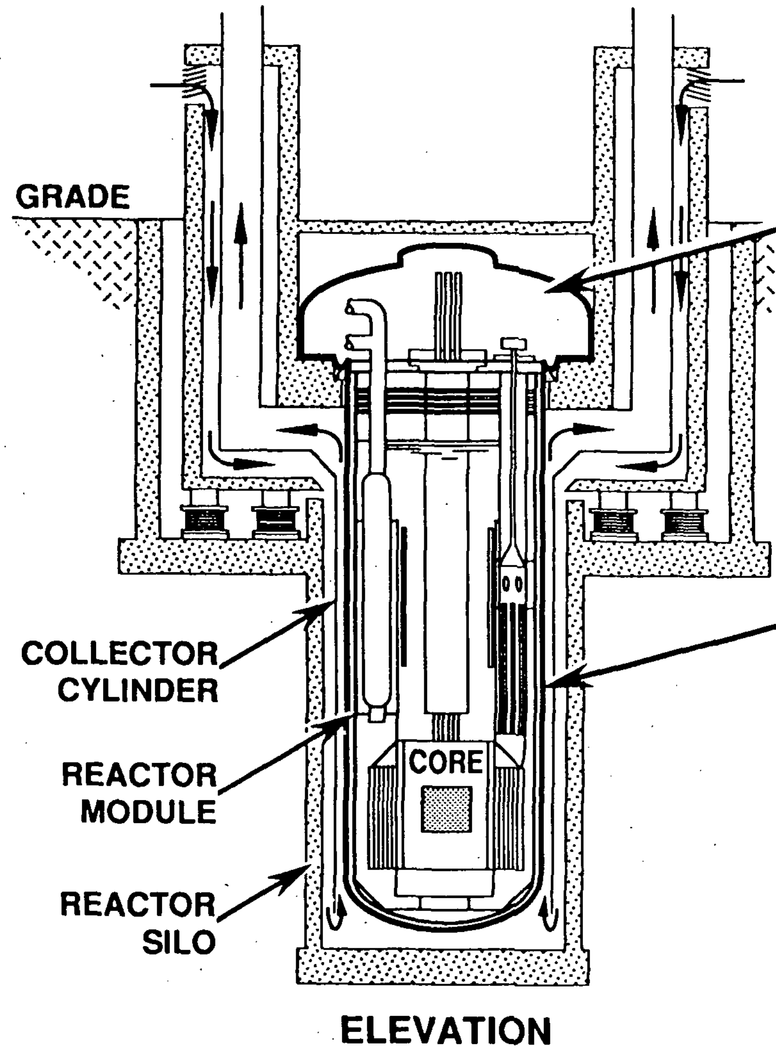
Although highly unlikely, a major spill of primary sodium (~9000 gals) into the pipe tunnel over a period of 25 hours (RVACS heatup transient) could occur if both the IHTS isolation valve fails and the operator fails to control the cover gas pressure within the first few hours after the accident, and a concurrent (passive) failure of the IHX is assumed. However, this event will not cause a site boundary dose exceeding 1 rem whole body. As the reactor slowly heats up, forcing radioactive sodium out of the reactor module, the average leakage rate will be less than 2600 lb per hour. The catch pans and the fire suppression deck will limit the sodium burning to less than 2%, aerosol release fraction to less than 3% of the Na burn, and with 50% of aerosol plateout or fallout, less than 1 lb per hour of aerosol will be released to the atmosphere. The total body dose, lung dose, and thyroid dose with nominal meteorology will be less than 1 rem at 1/2 mile (site boundary), which is within the acceptance limits for a EC-IV event.

G.4.1.4 References

G.4.1-1 Murata, K.K., et al., "User's Manual for CONTAIN 1.1 - A Computer Code for Severe Nuclear Reactor Accident Containment Analysis," SAND 87-2309, Sandia National Laboratories, Albuquerque, NM

G.4.1-2 Madni, I. K., et al., "A Simplified Model for Calculating Early Off-site Consequences from Nuclear Reactor Accidents," BNL-NUREG-52153, July 1988

G.4.1-3 Merlin, M., et al., "Evaluations of the Sodium Retention Factors for Fission Products and Fuel," Proceedings of the LMFBR Safety Topical Meeting, Lyon, France, July 19-23, 1982



Containment Dome
- ASME Section III, Div. 1, Class MC
- Material – SA 516 Grade 70
- Design Requirements:

  < 1% / day at 25 psig, 700 °F

Containment Vessel
- ASME Section III, Div, 1, Class MC
- Material – 2 1/4 Cr - 1 Mo
- Design Requirements:

  Zero Leak Rate at 60 psig, 800 °F

90-097-07

Figure G.4.1-1 REACTOR CONTAINMENT

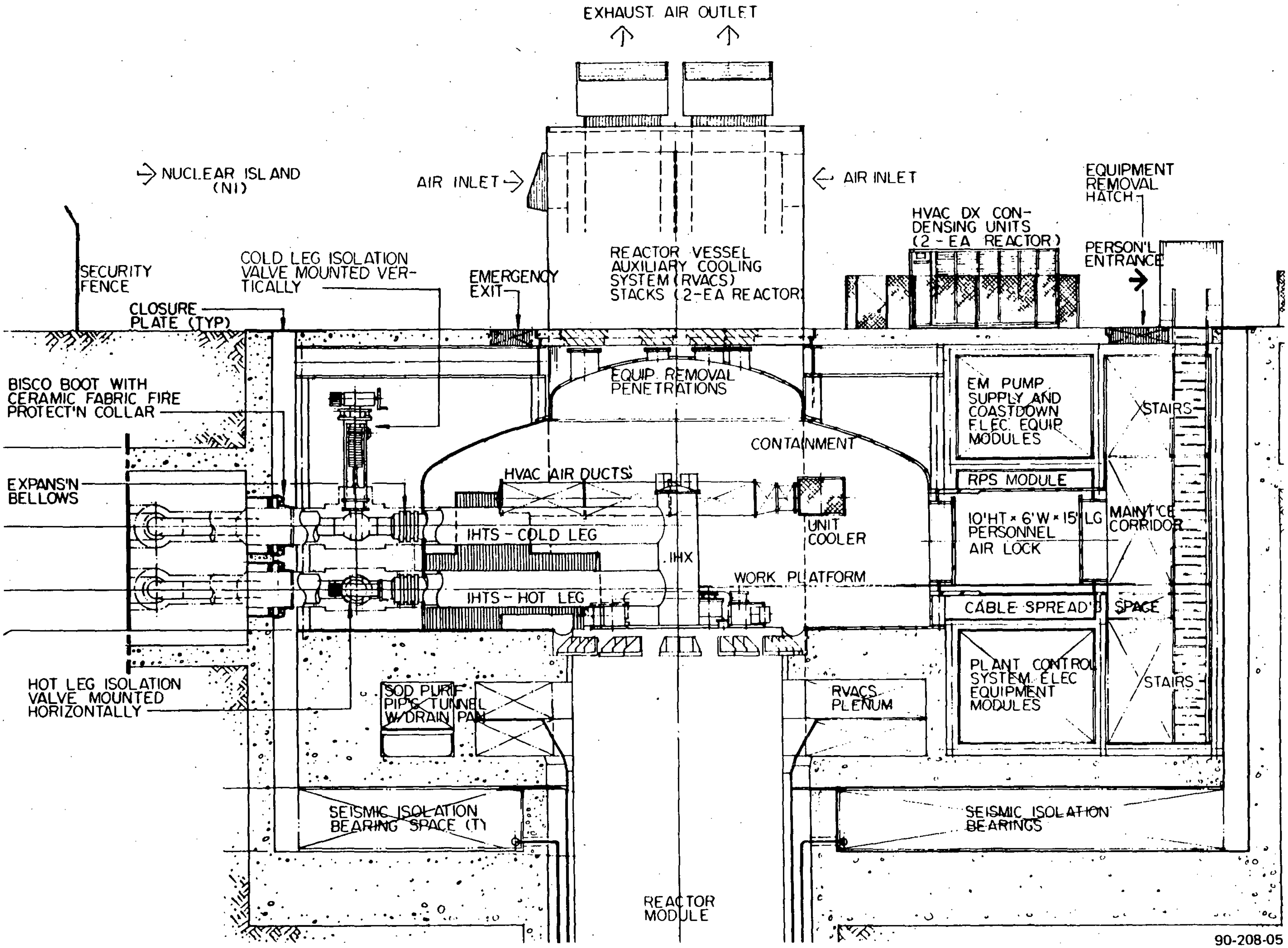


Figure G.4.1–2 REACTOR CONTAINMENT ELEVATION

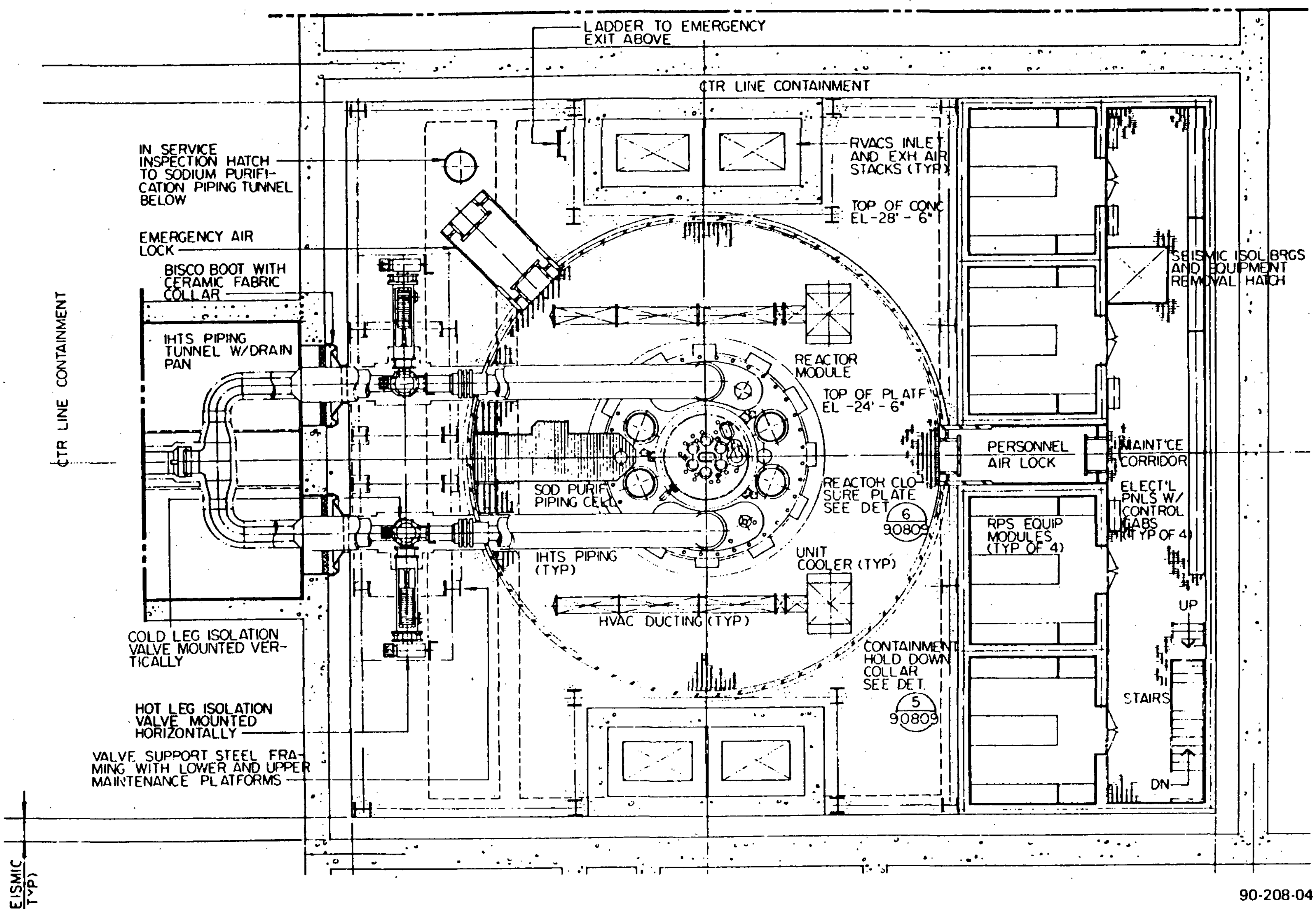


Figure G.4.1—3 REACTOR CONTAINMENT PLAN VIEW

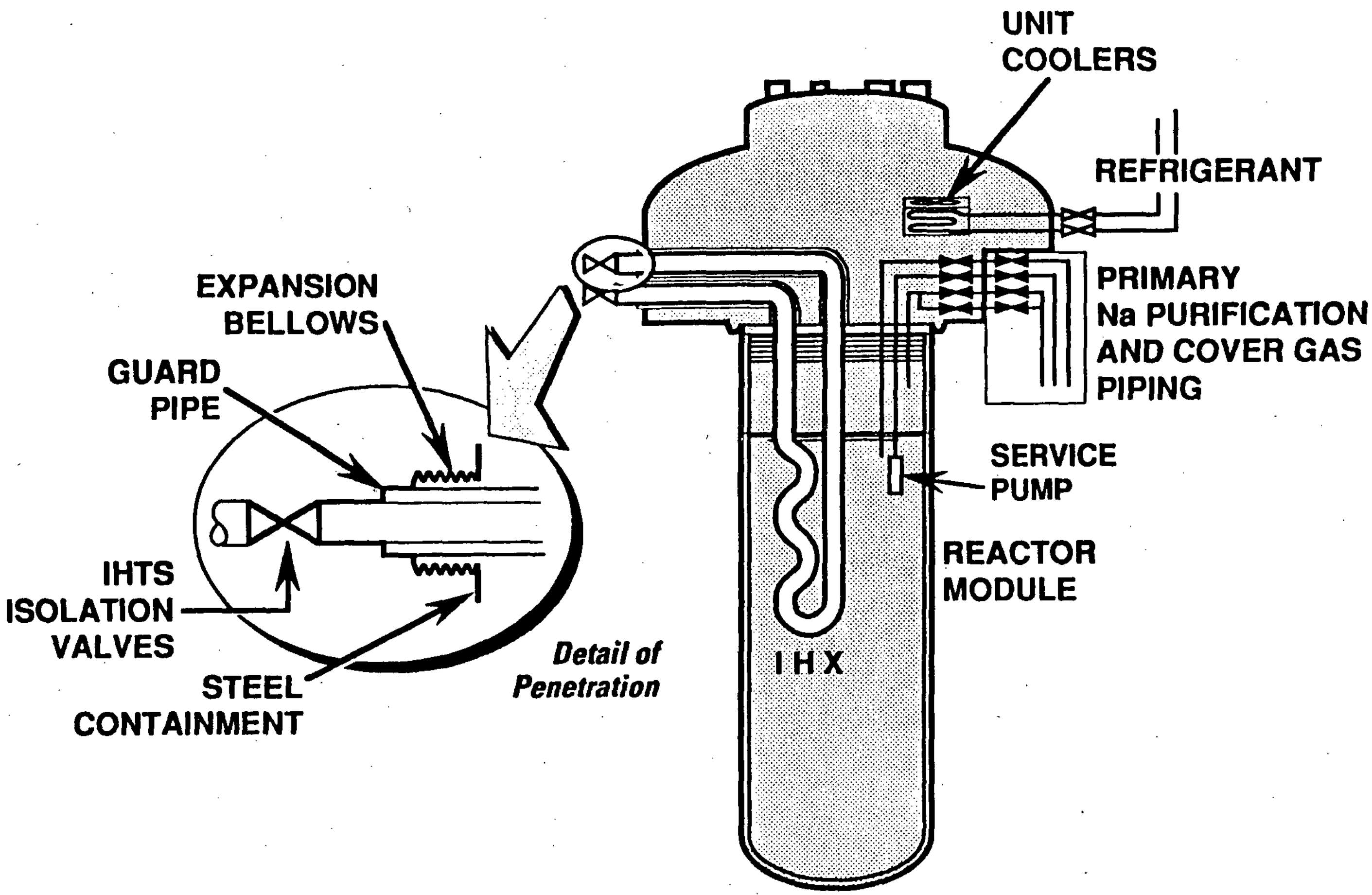


Figure G.4.1-4 - CONTAINMENT DURING OPERATION

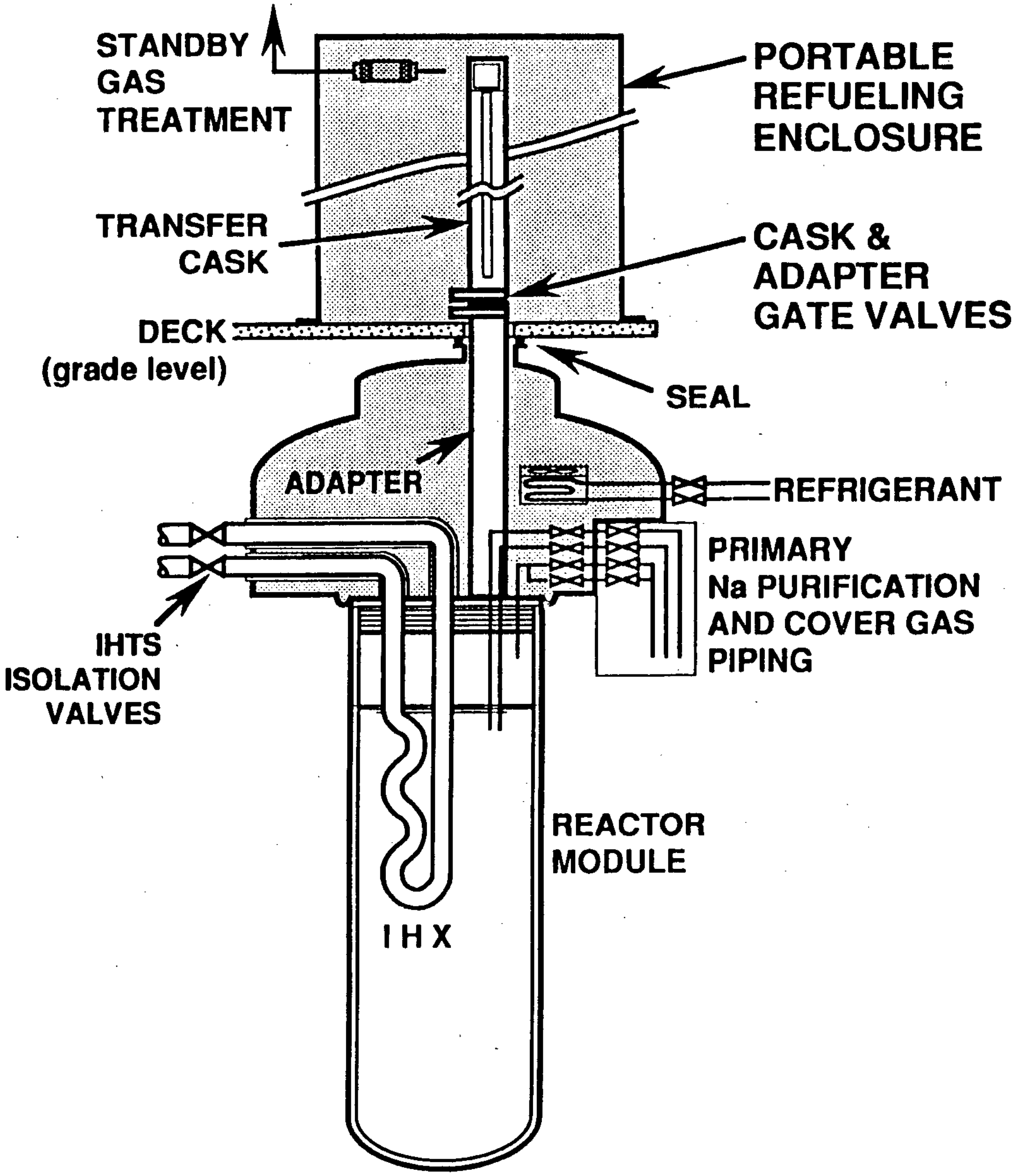


Figure G.4.1-5 - CONTAINMENT DURING MAINTENANCE AND REFUELING

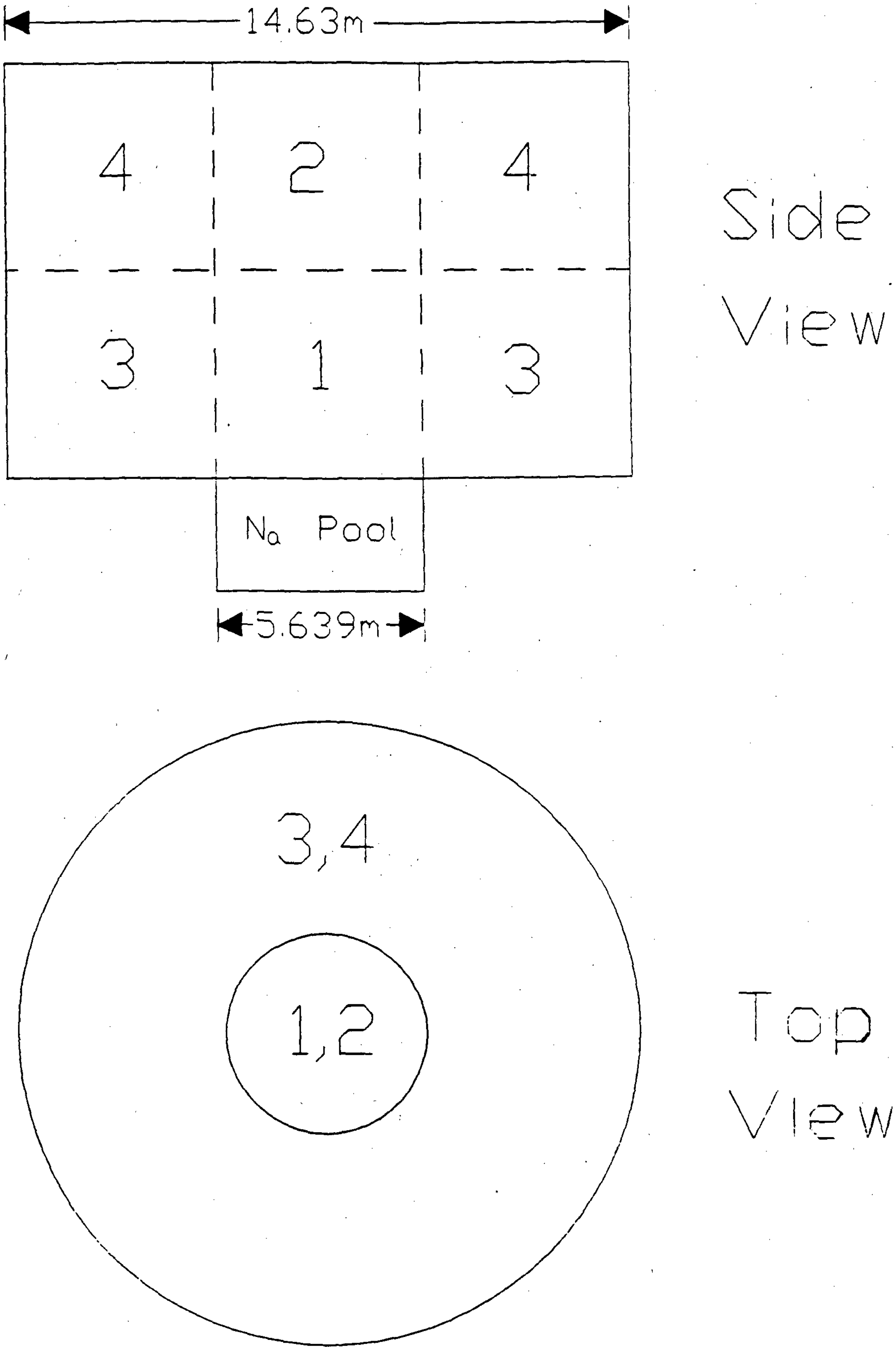


Figure G.4.1-6 - SCHEMATIC OF THE ALMR CONTAIN MODEL

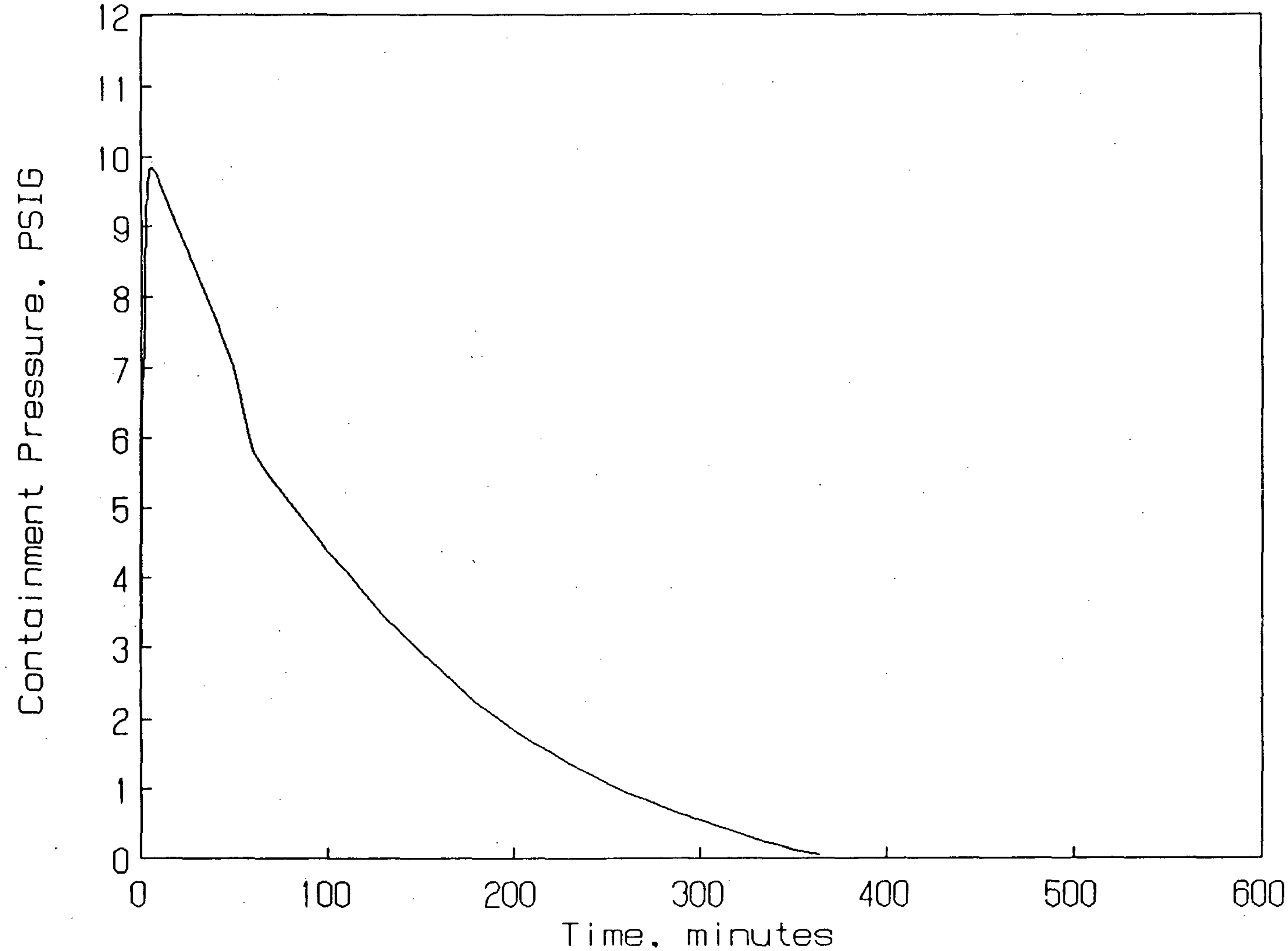


Figure G.4.1-7 - CONTAINMENT DOME PRESSURE vs TIME

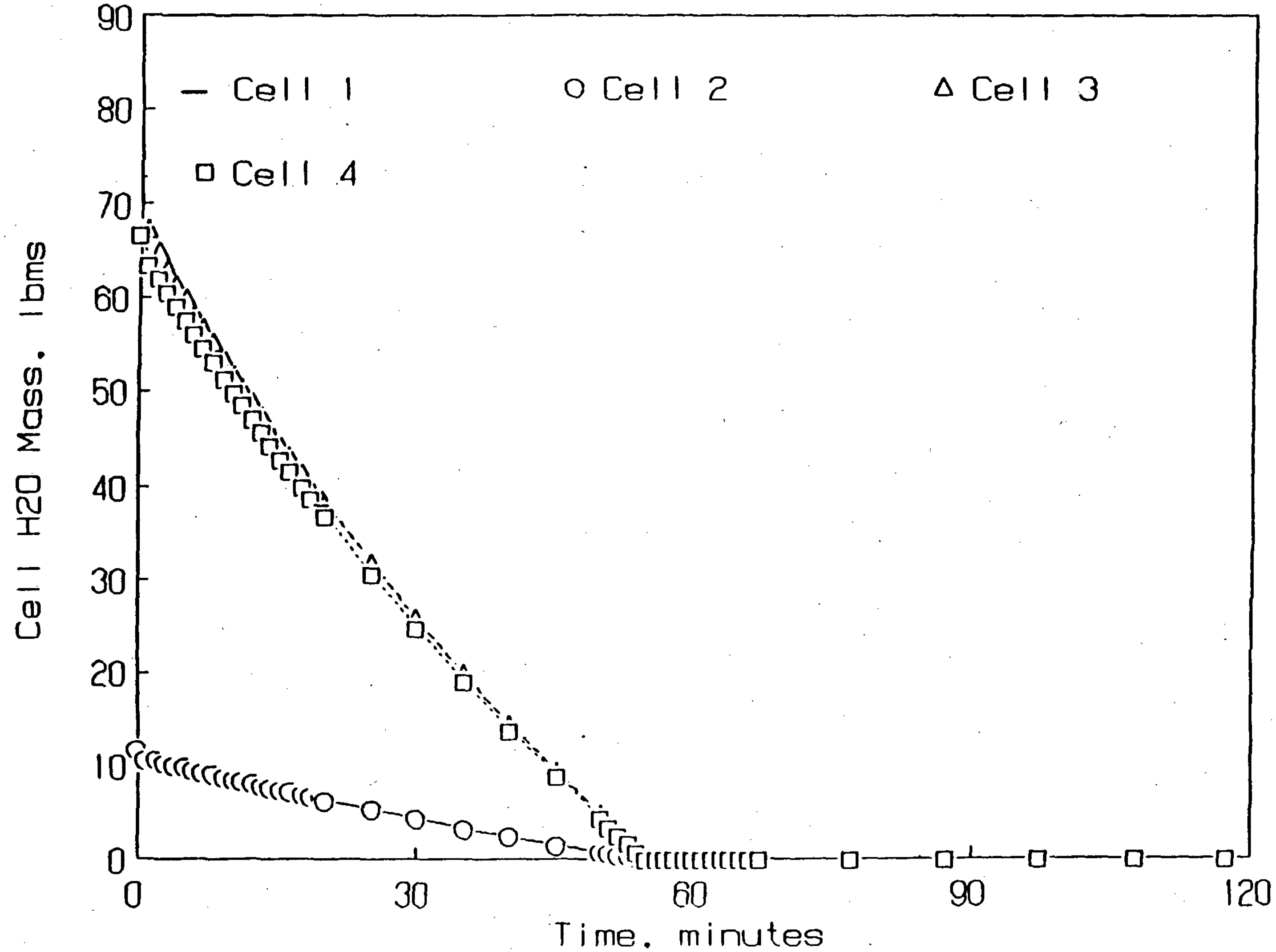


Figure G.4.1-8 - CELL WATER VAPOR MASS vs TIME

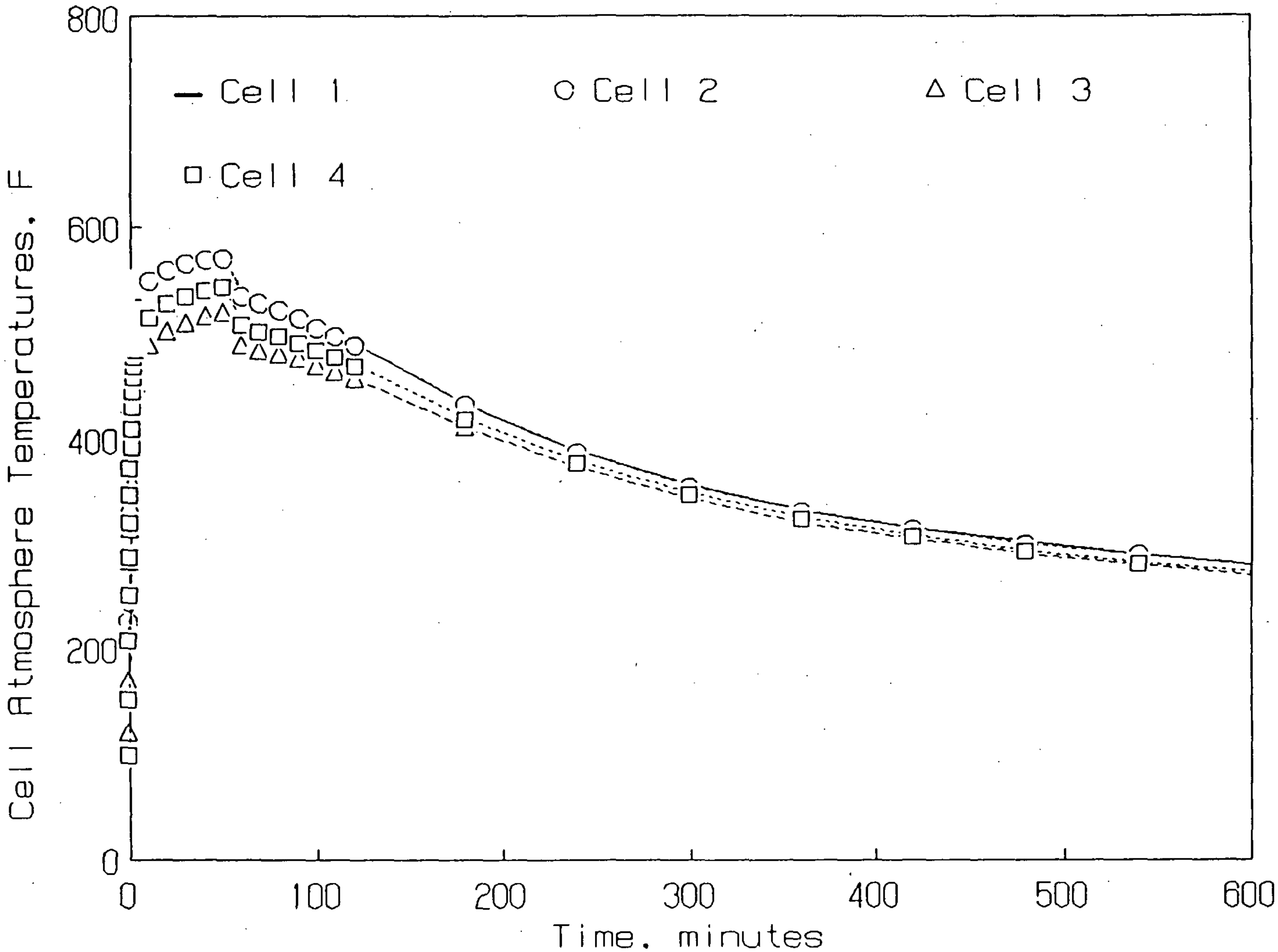


Figure G.4.1-9 - CELL TEMPERATURE vs TIME

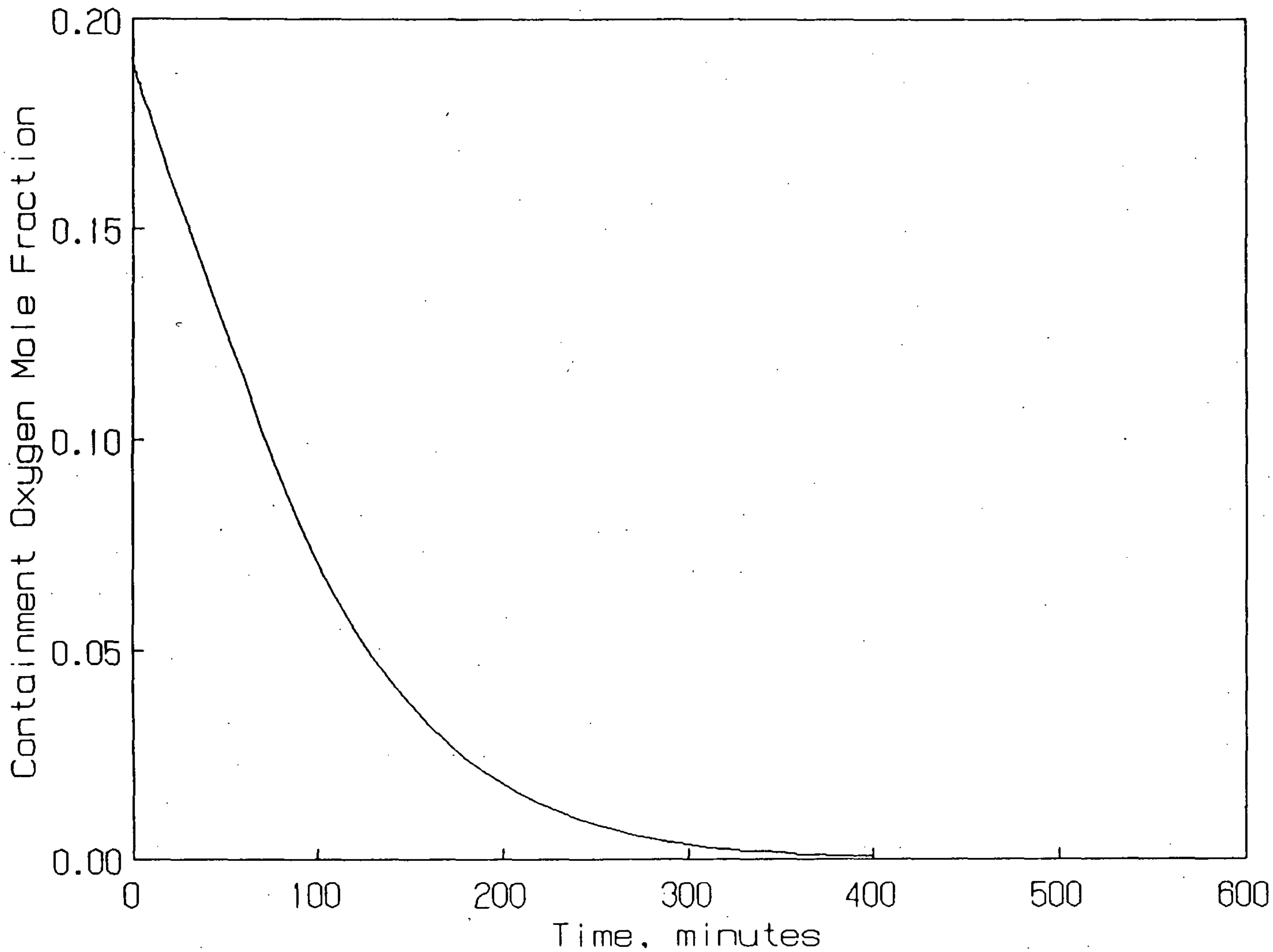


Figure G.4.1-10 - CONTAINMENT OXYGEN MOLE FRACTION vs TIME

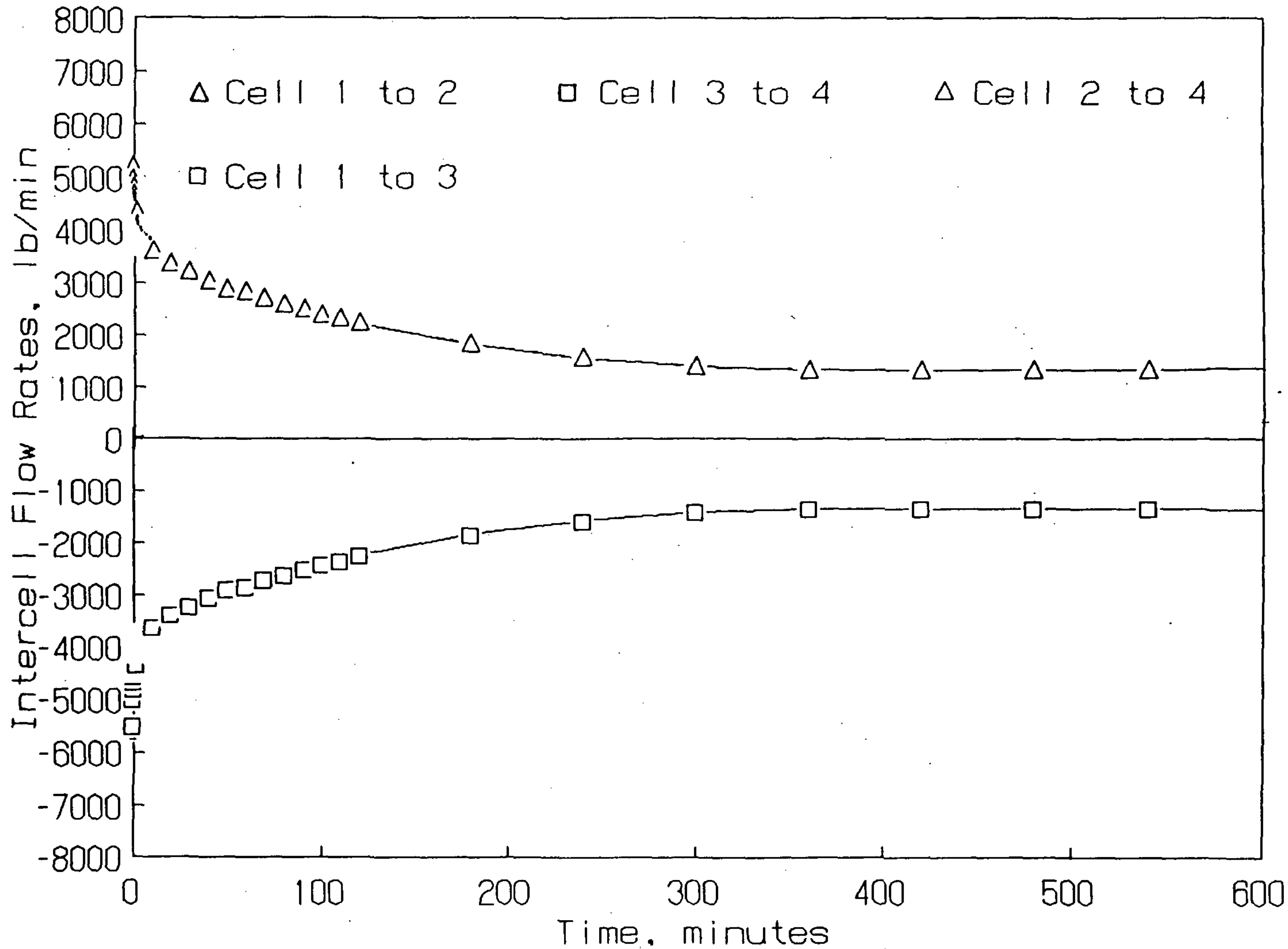


Figure G.4.1-11 - INTERCELL FLOWS vs TIME

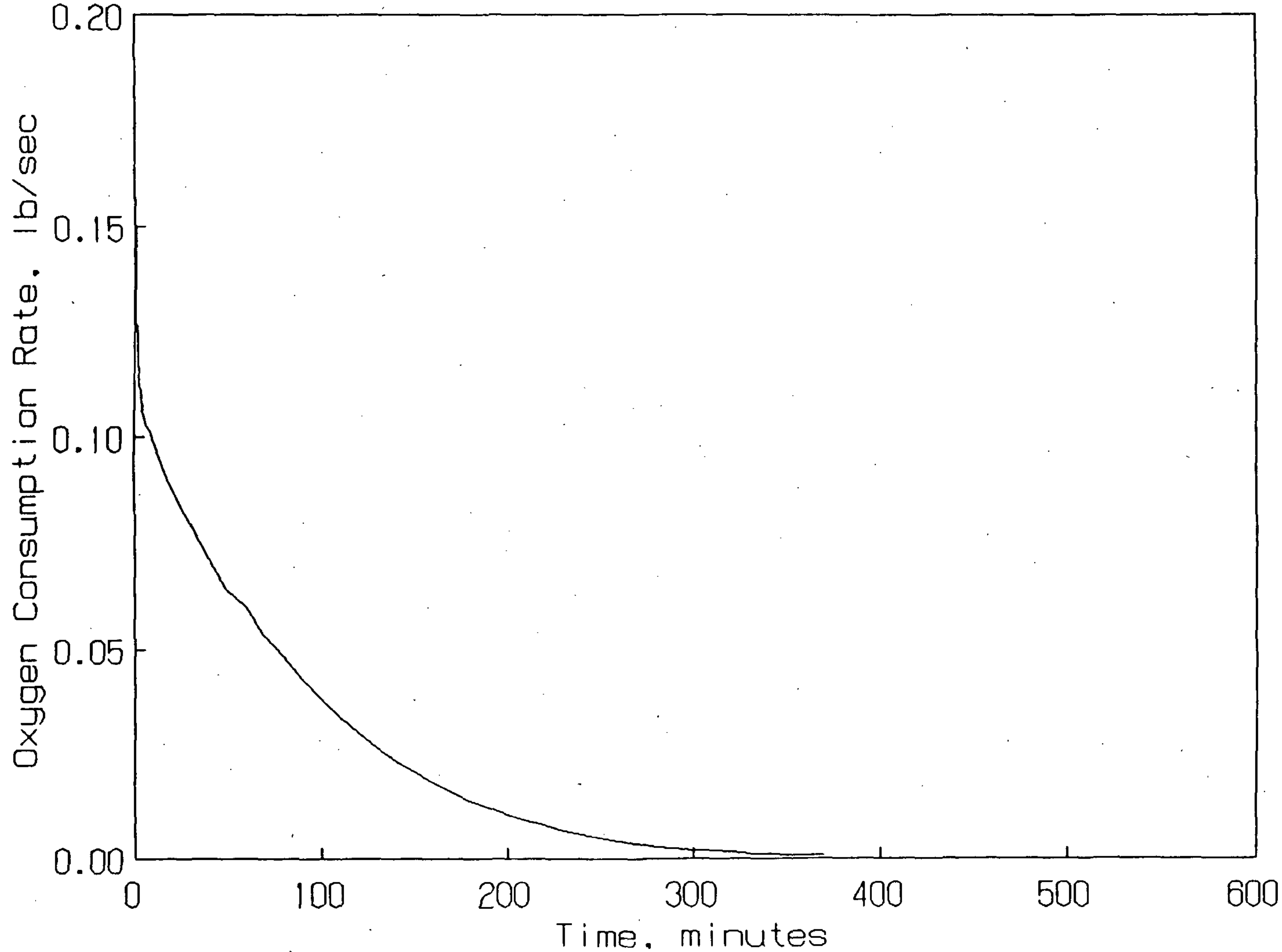


Figure G.4.1-12 - OXYGEN CONSUMPTION RATE vs TIME

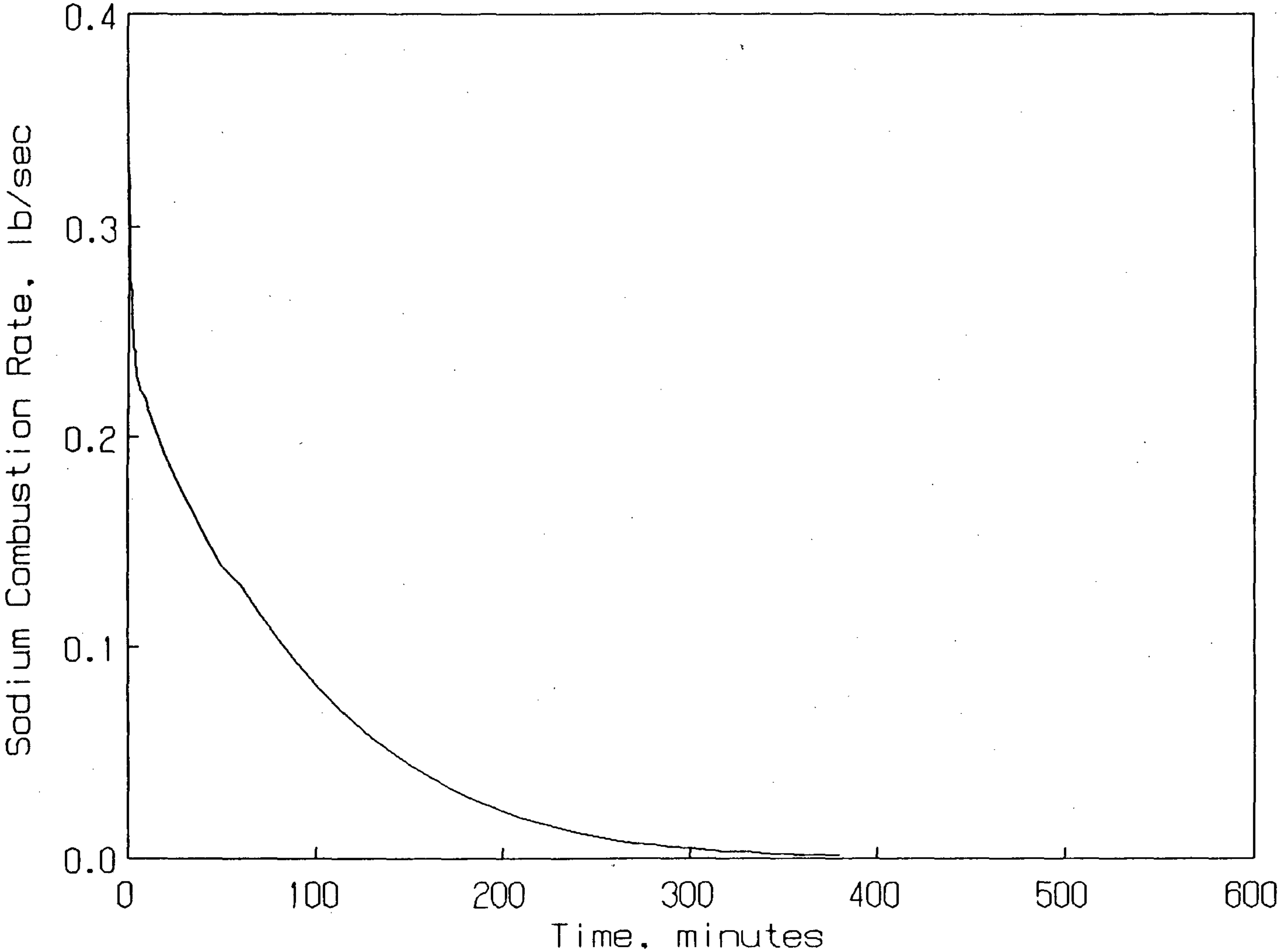


Figure G.4.1-13 - SODIUM COMBUSTION RATE vs TIME

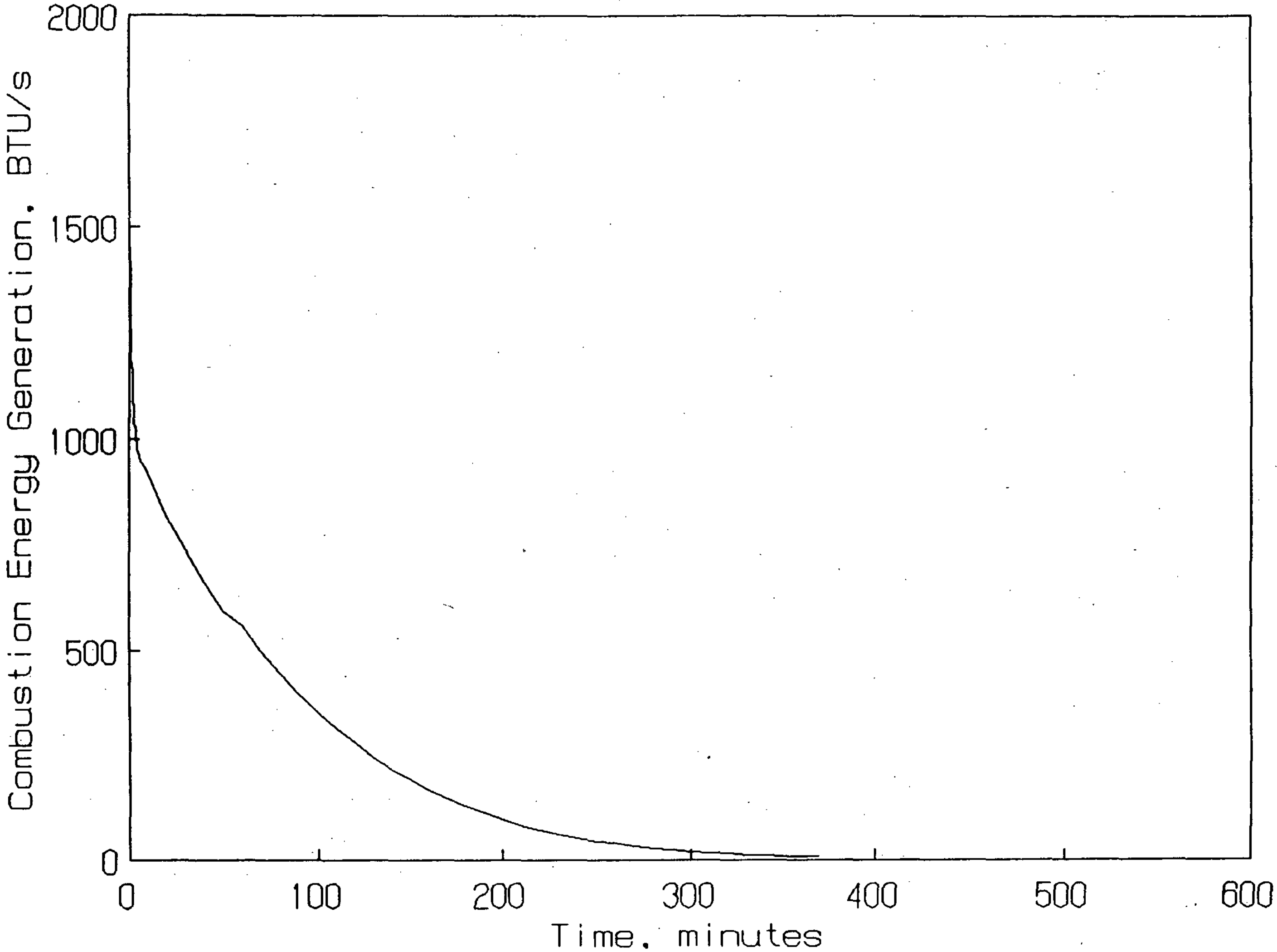


Figure G.4.1-14 - COMBUSTION ENERGY GENERATION RATE vs TIME

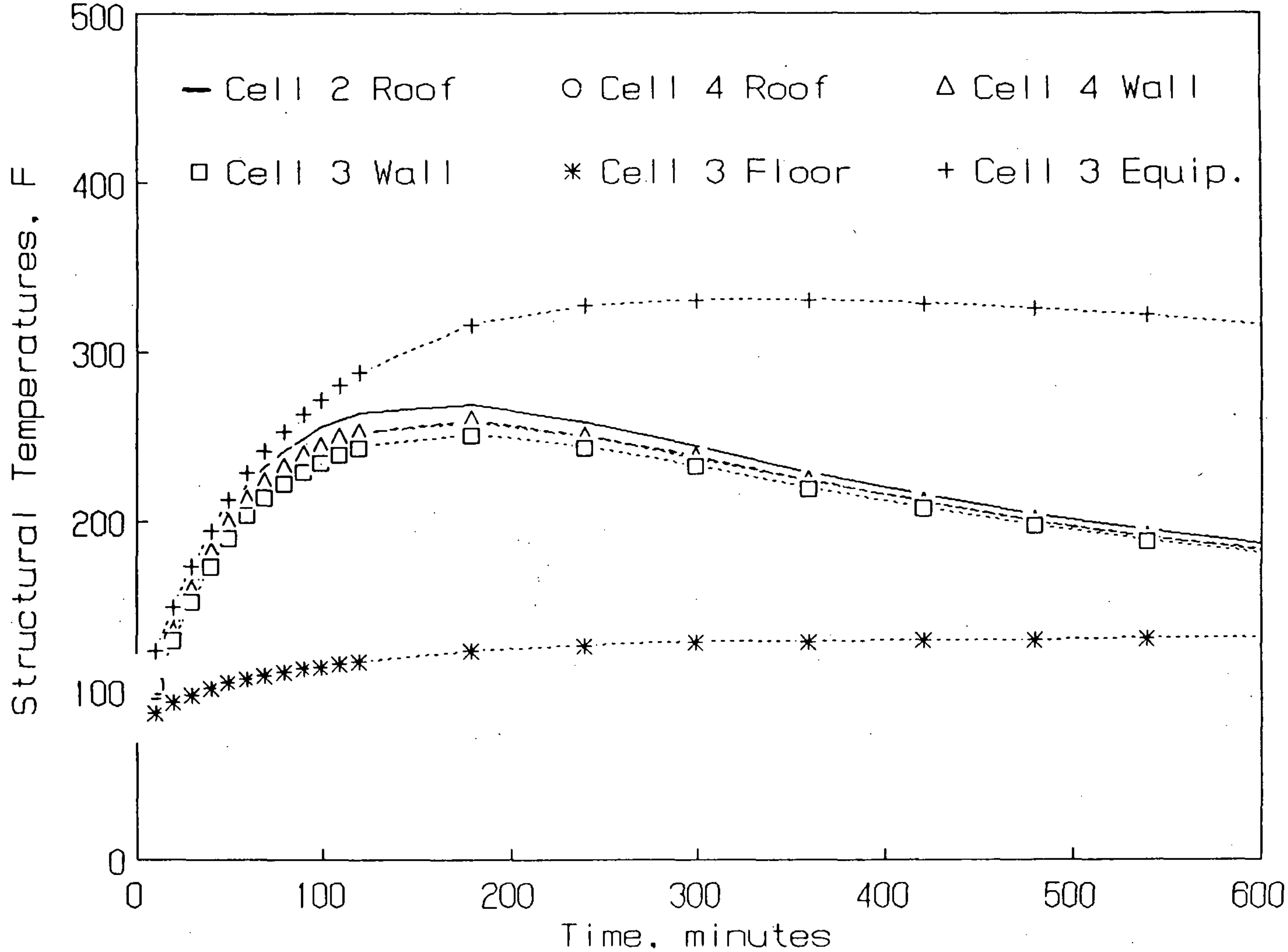


Figure G.4.1-15 - STRUCTURAL TEMPERATURES vs TIME

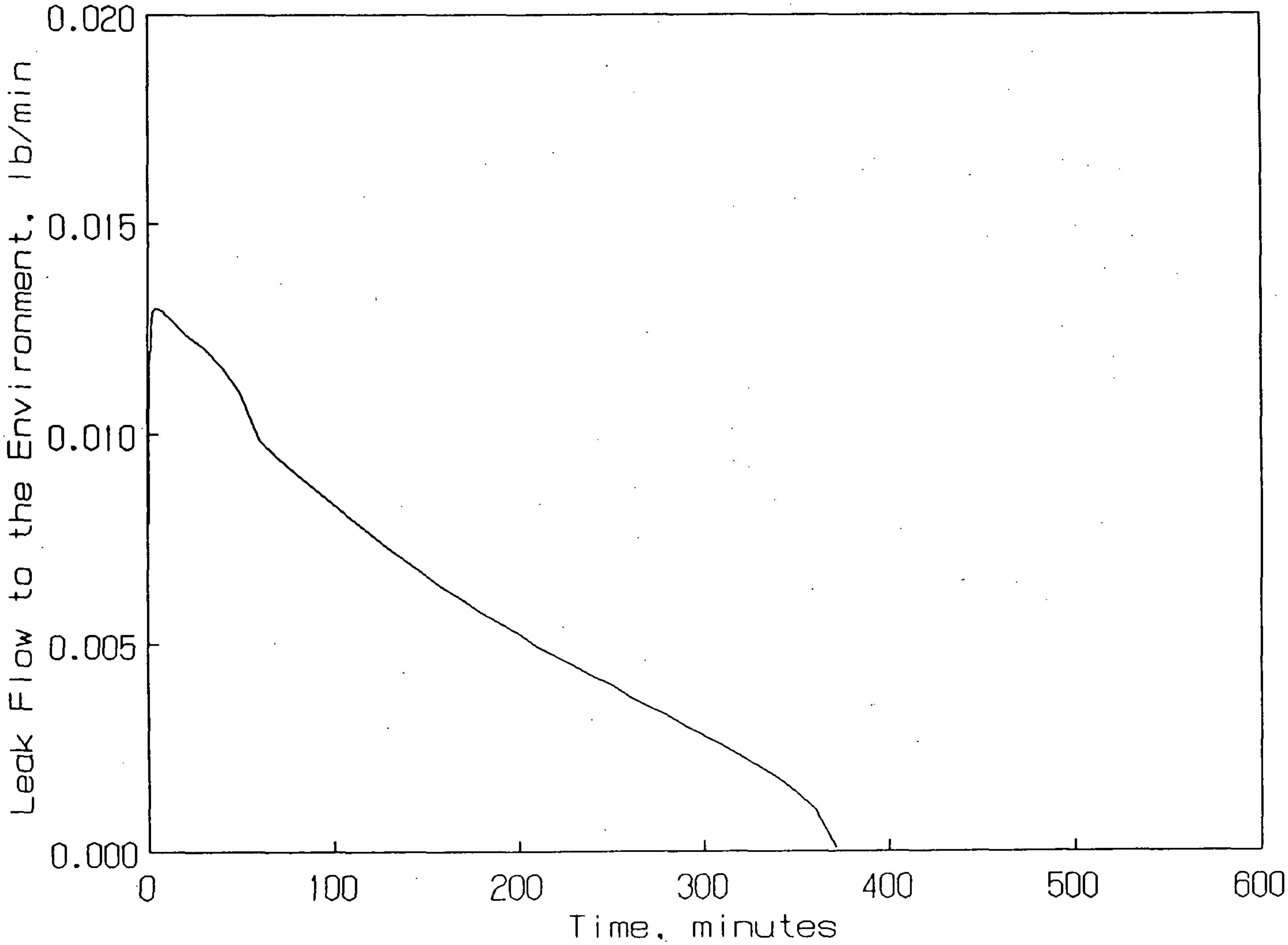


Figure G.4.1-16 - LEAK FLOW TO THE ENVIRONMENT vs TIME

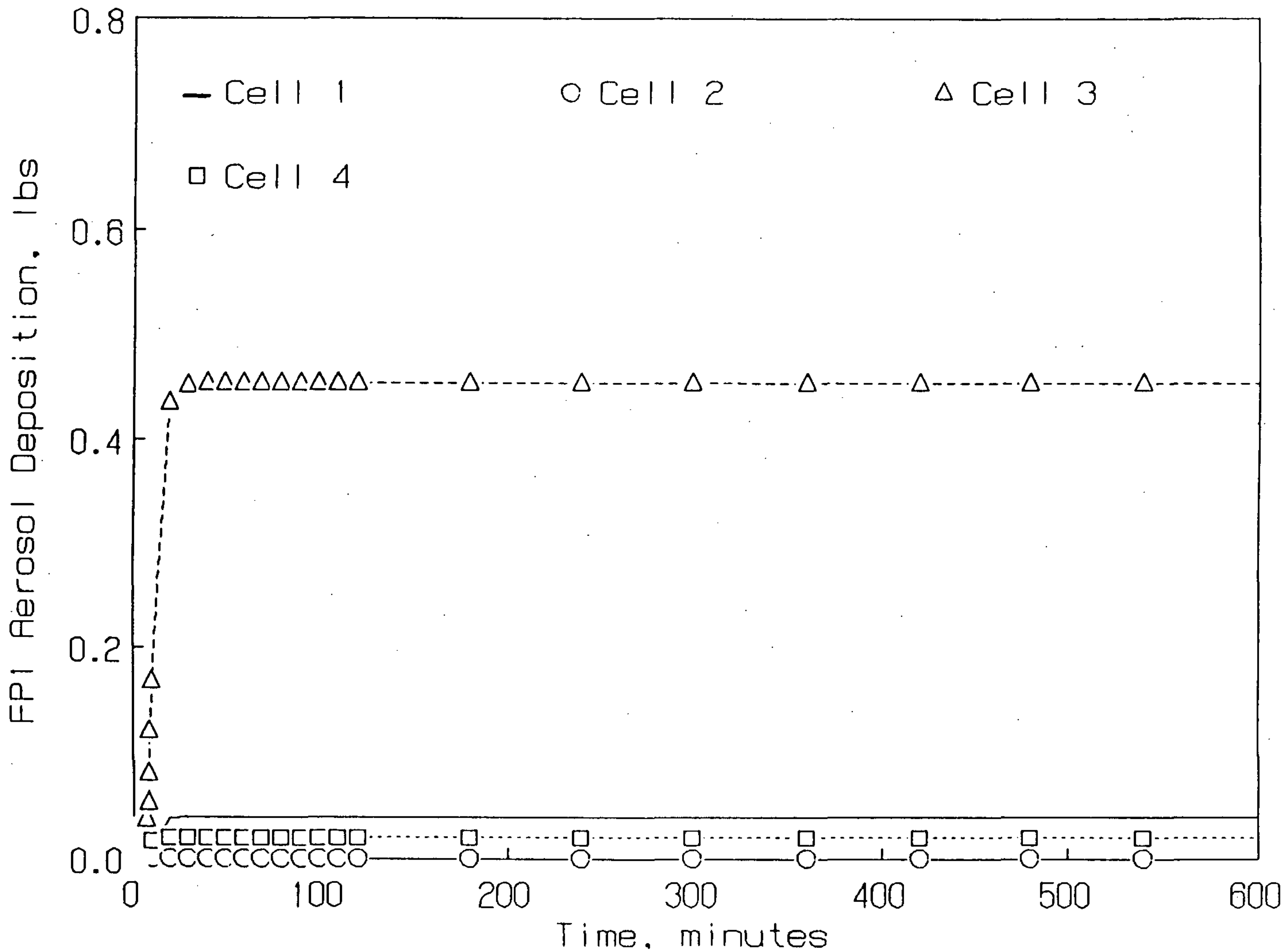


Figure G.4.1-17 - AEROSOL DEPOSITION vs TIME (without fire)

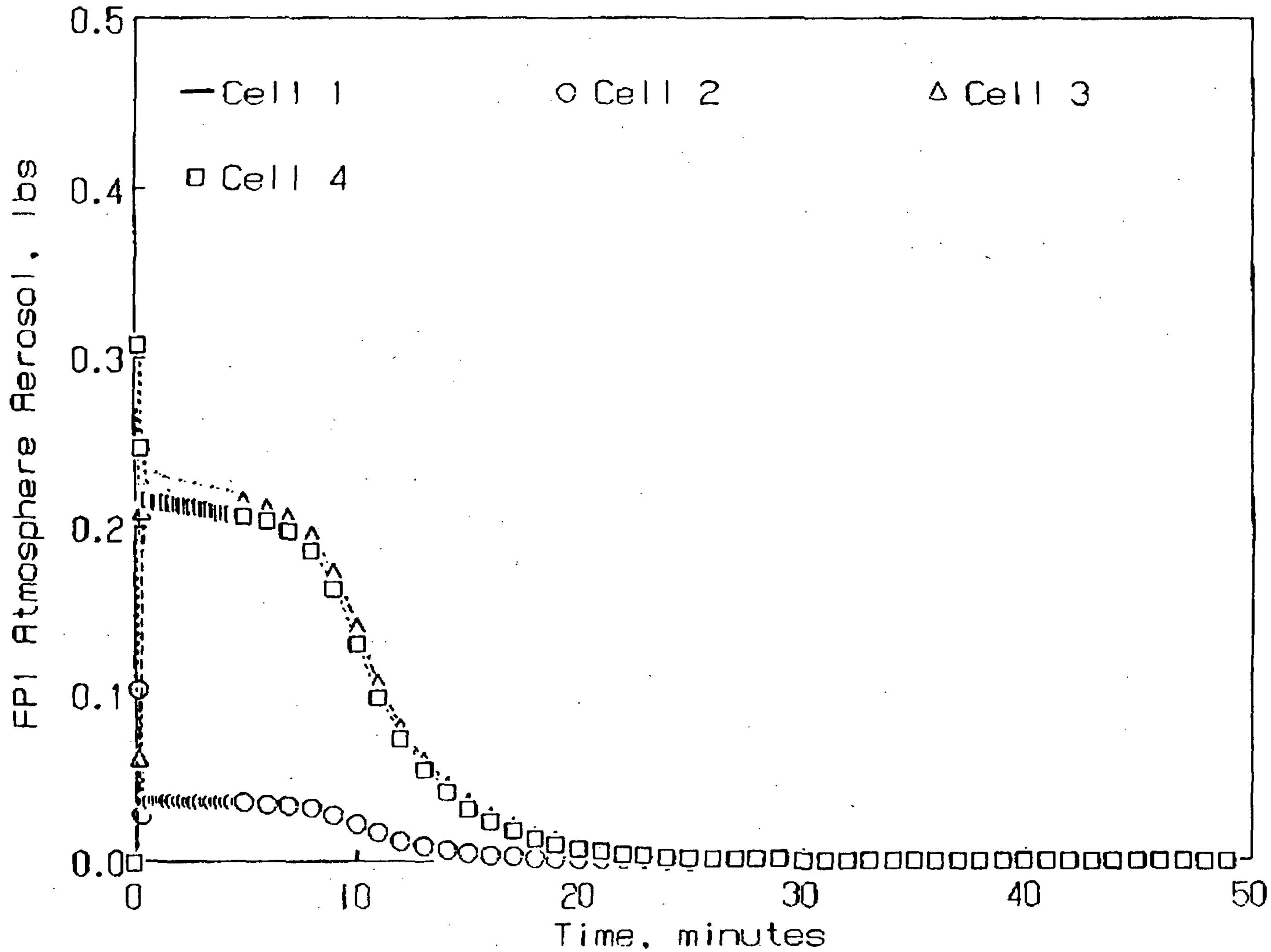


Figure G.4.1-18 - SUSPENDED AEROSOL vs TIME (without fire)

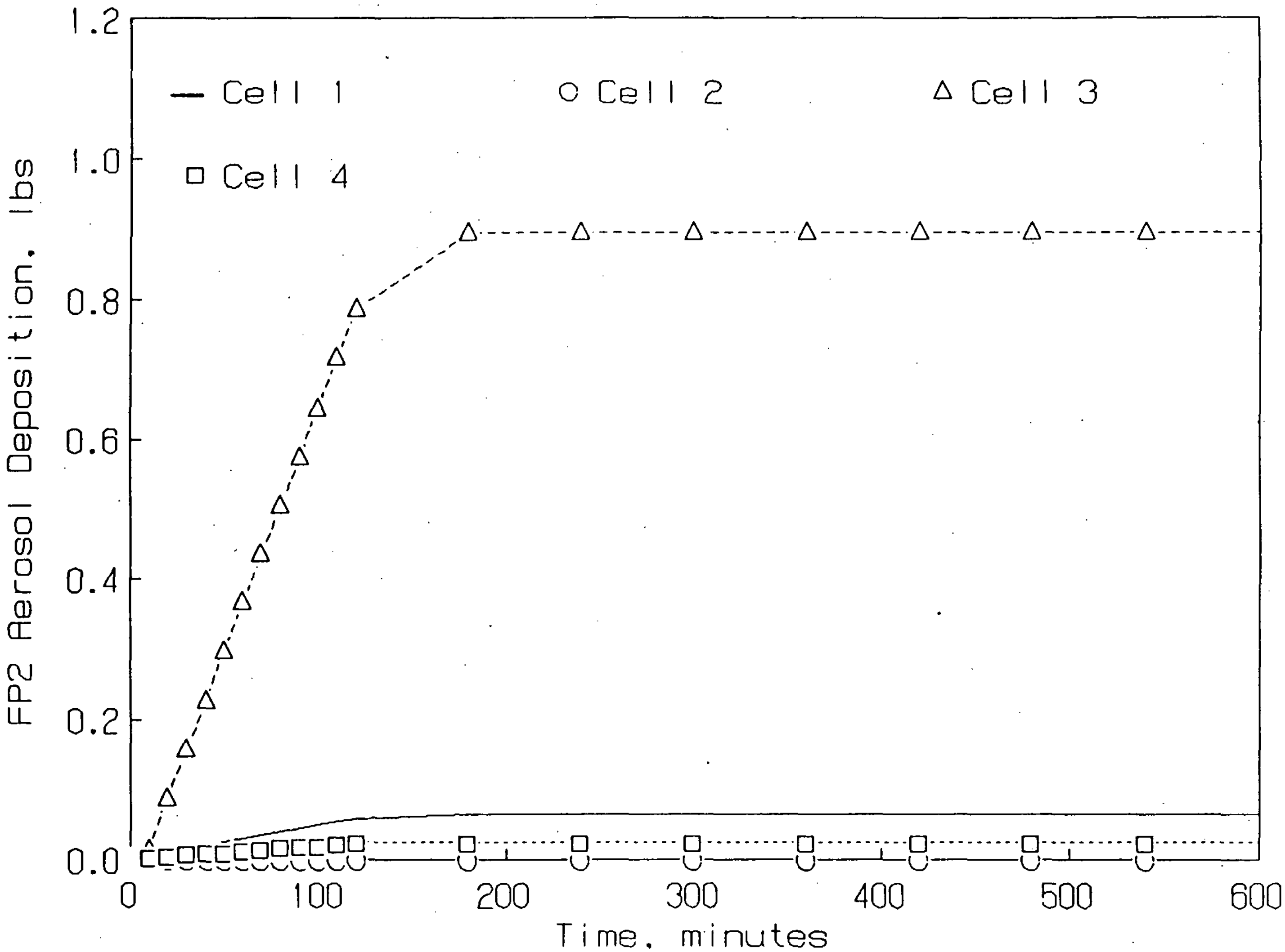


Figure G.4.1-19 - AEROSOL DEPOSITION vs TIME (with fire)

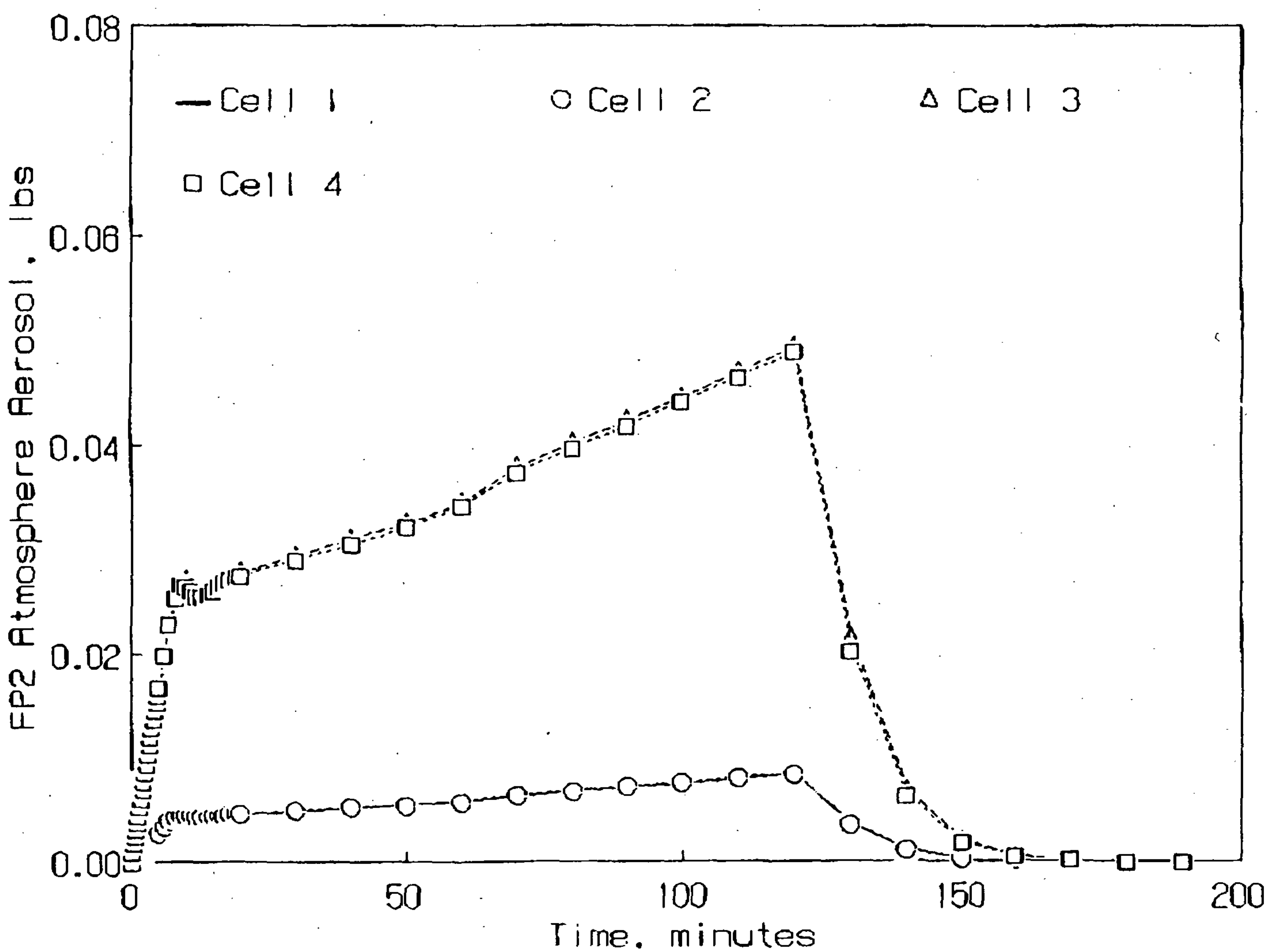


Figure G.4.1-20 - SUSPENDED AEROSOL vs TIME (with fire)

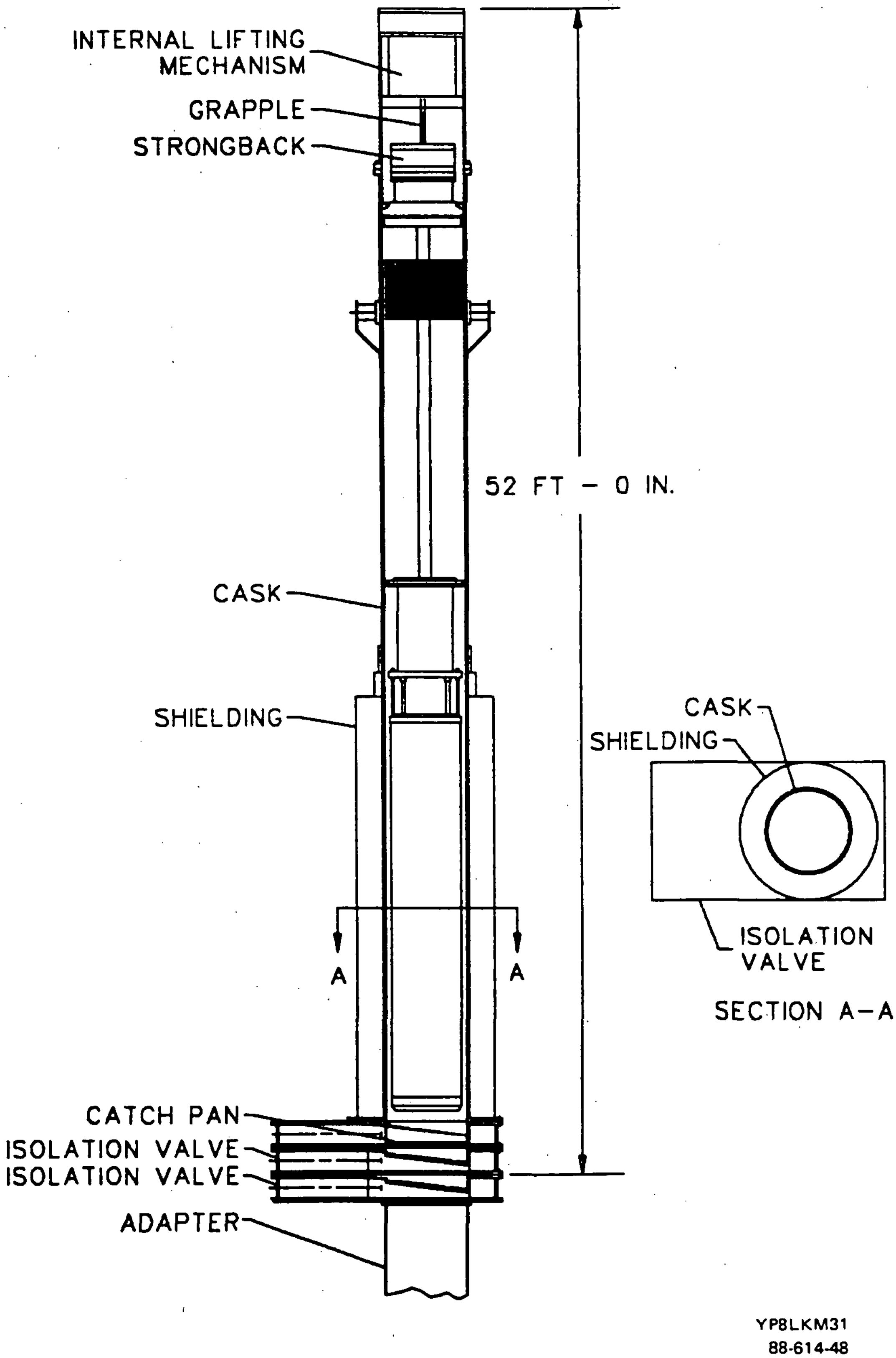


Figure G.4.1-21 - EM PUMP TRANSFER CASK

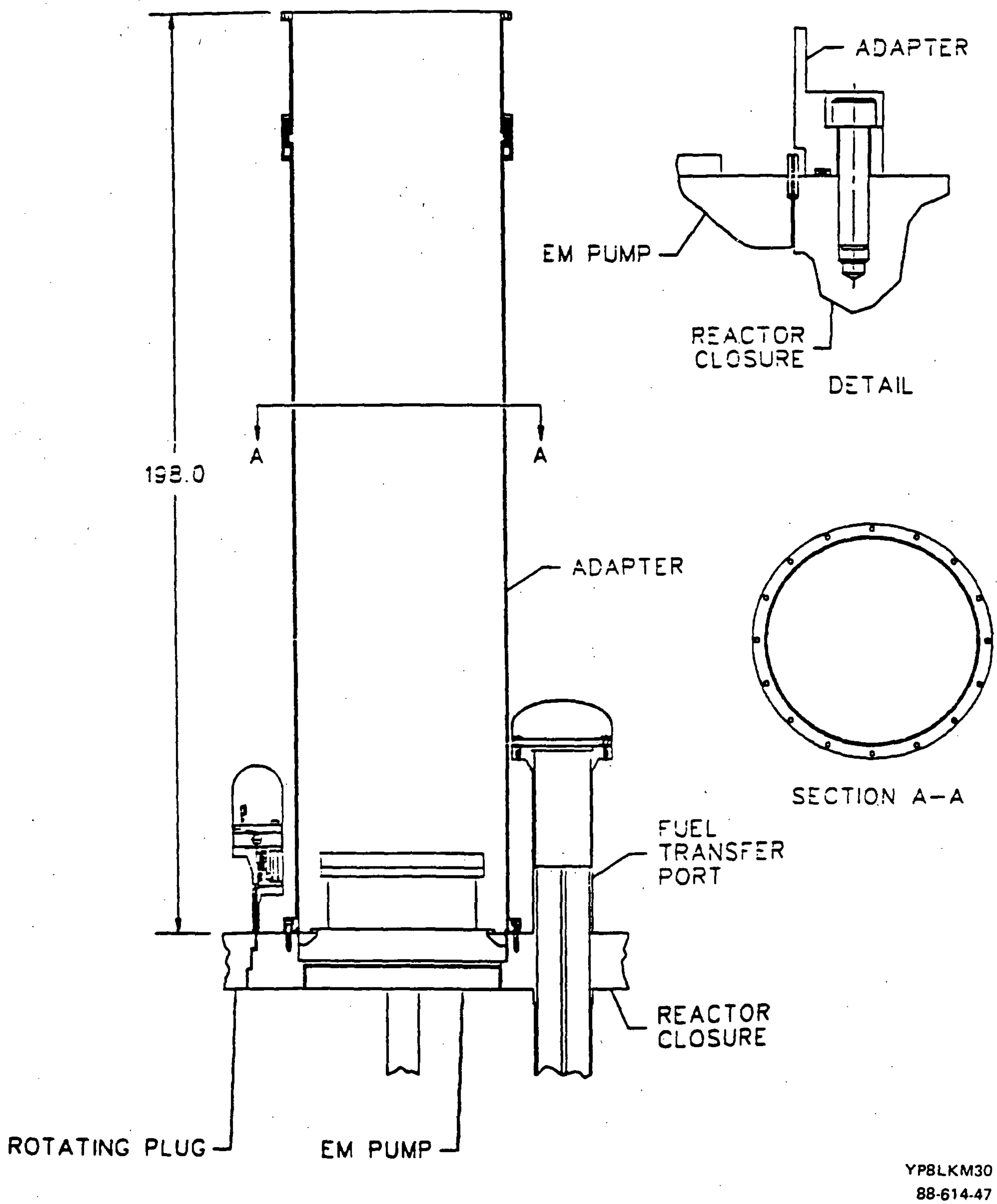


Figure G.4.1-22 - EM PUMP ADAPTER

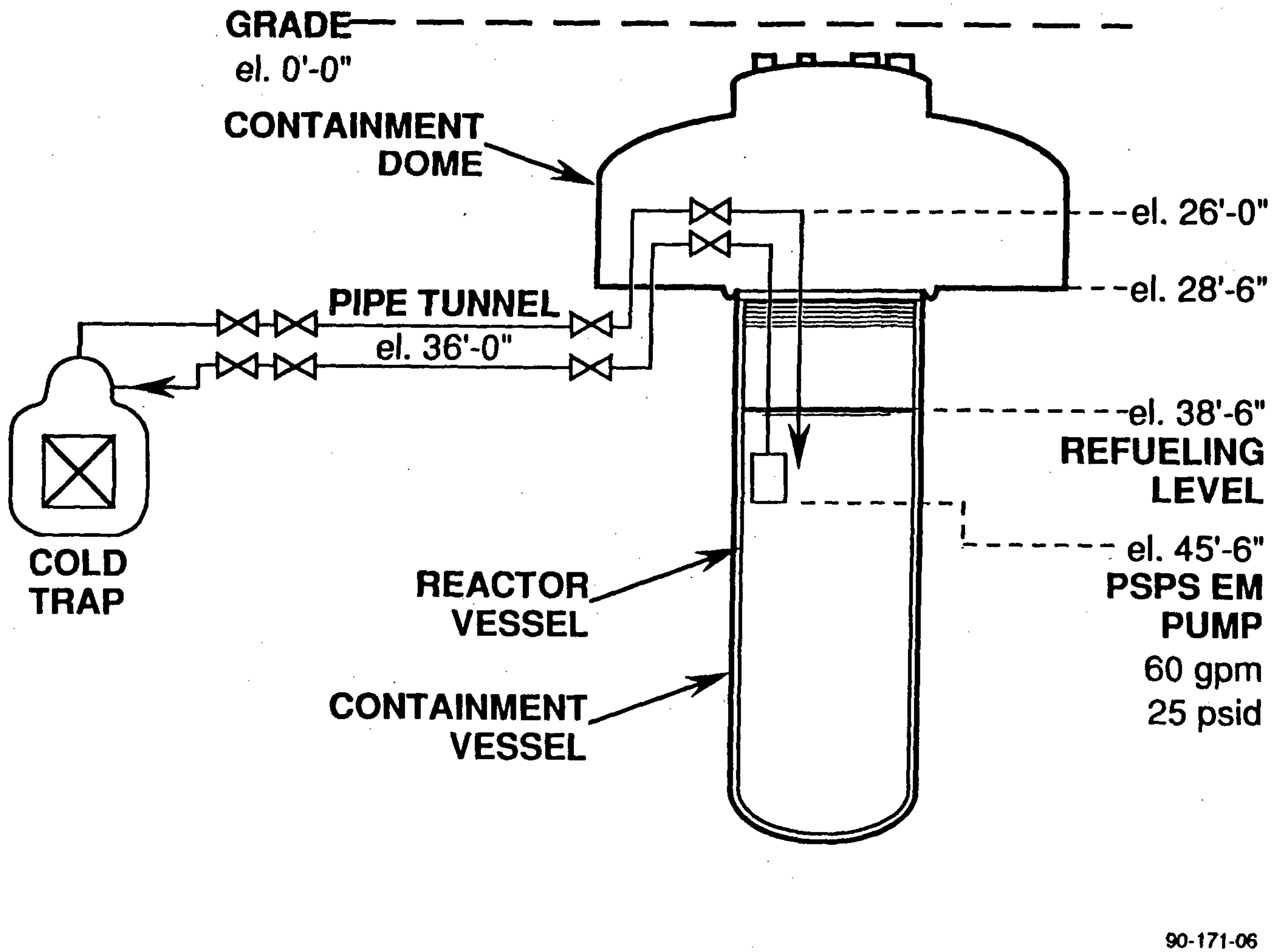


Figure G.4.1-23 - PRIMARY SODIUM PROCESSING SYSTEM HYDRAULIC PROFILE DURING REFUELING CONDITIONS

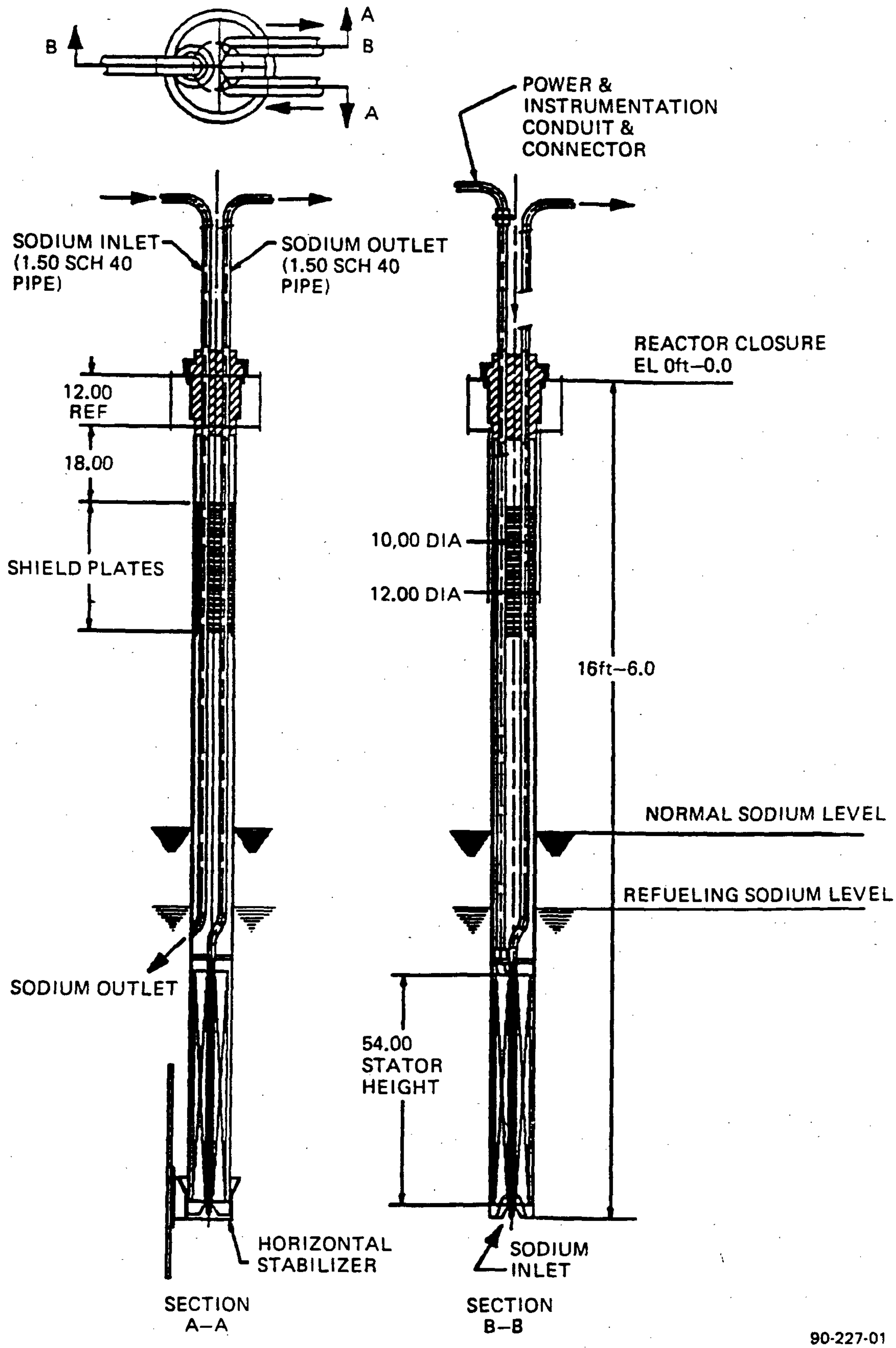
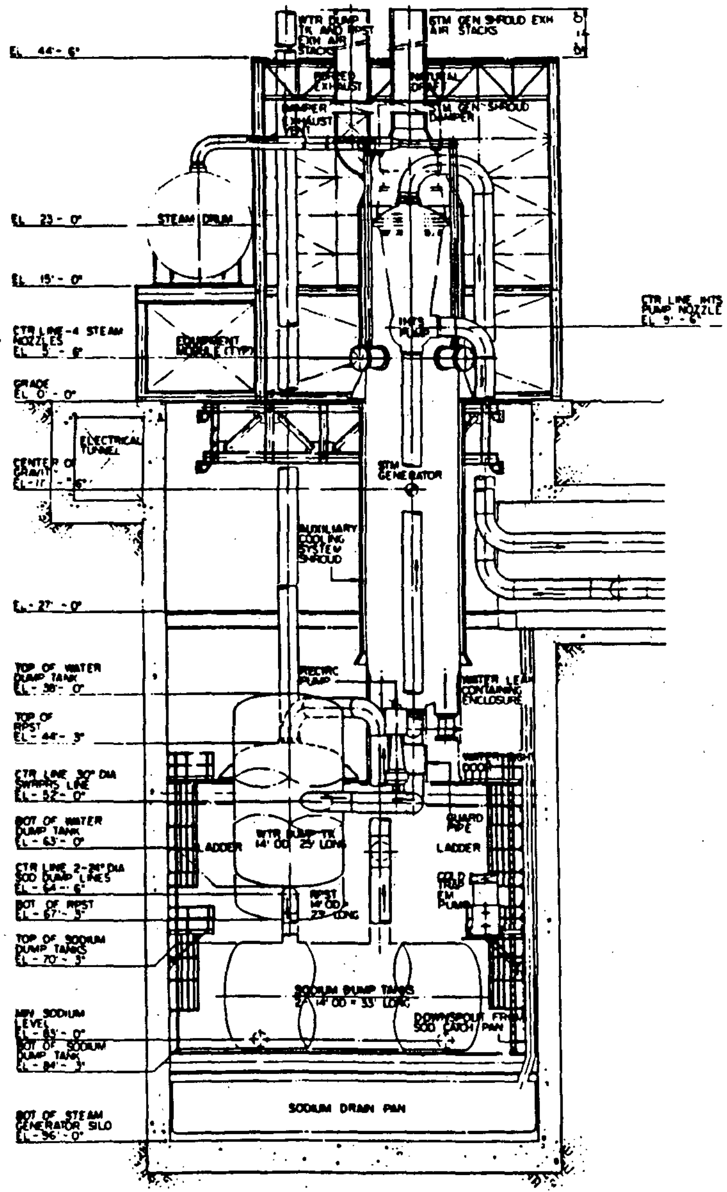
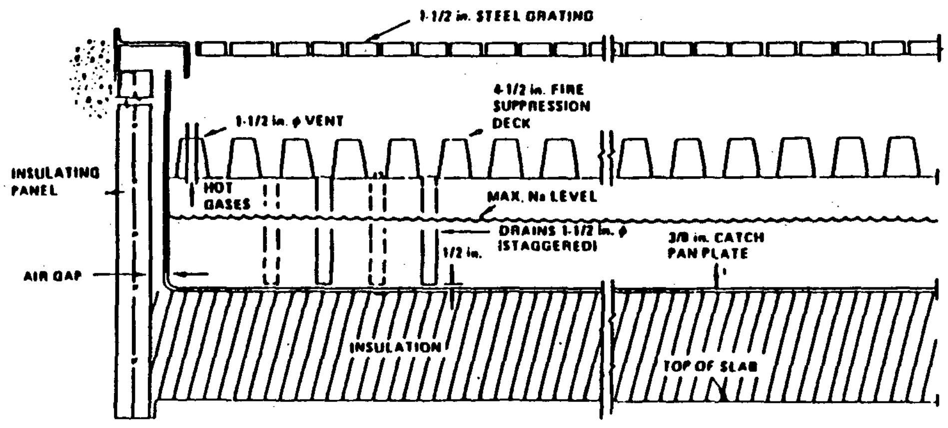


Figure G.4.1-24 - PRIMARY SODIUM PROCESSING SYSTEM EM PUMP





Typical Catch Pan Fire Suppression Deck Arrangement

Figure G.4.1-25 - SODIUM CATCH PAN ARRANGEMENT

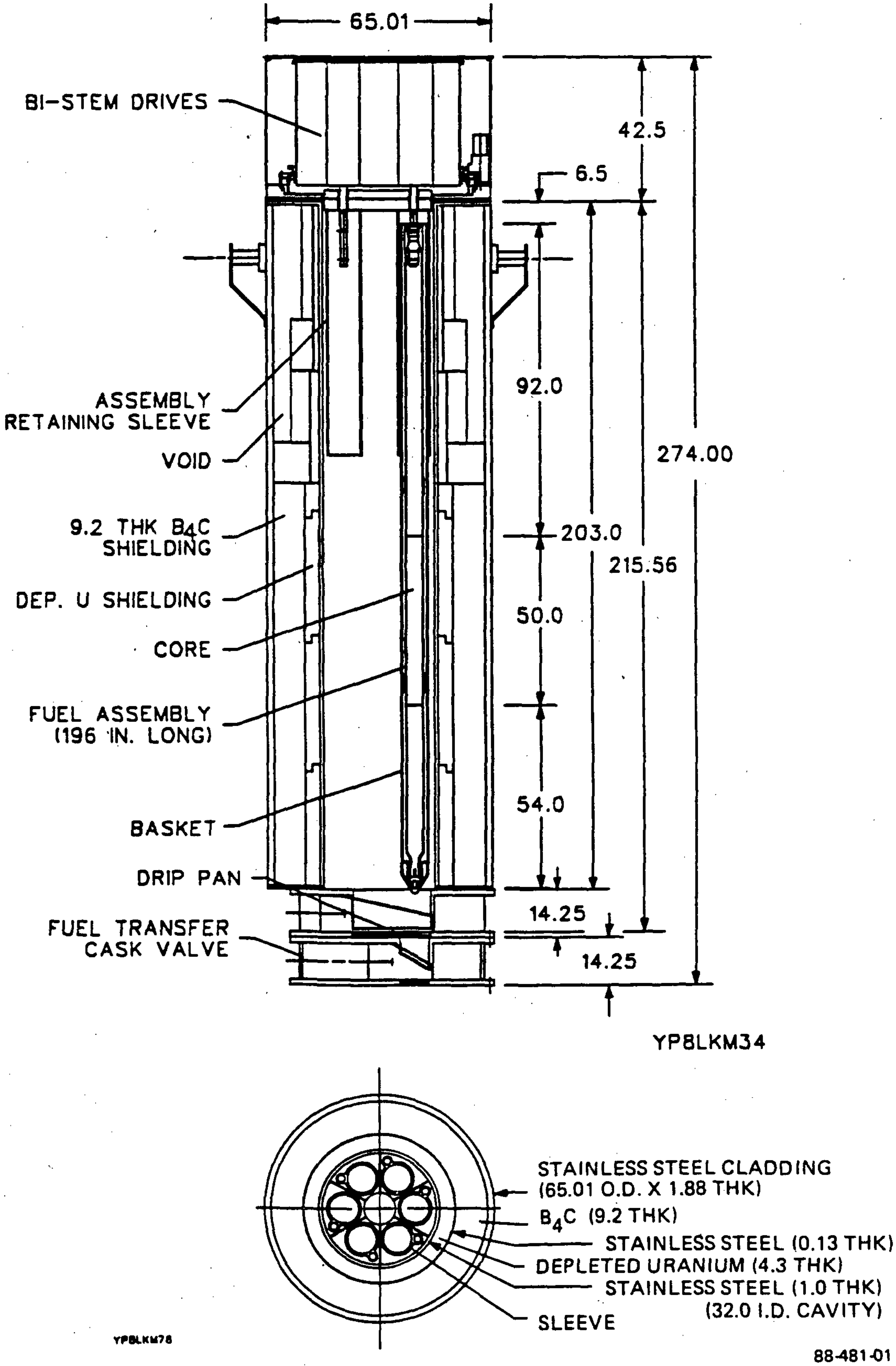


Figure G.4.1-26 - FUEL TRANSFER CASK

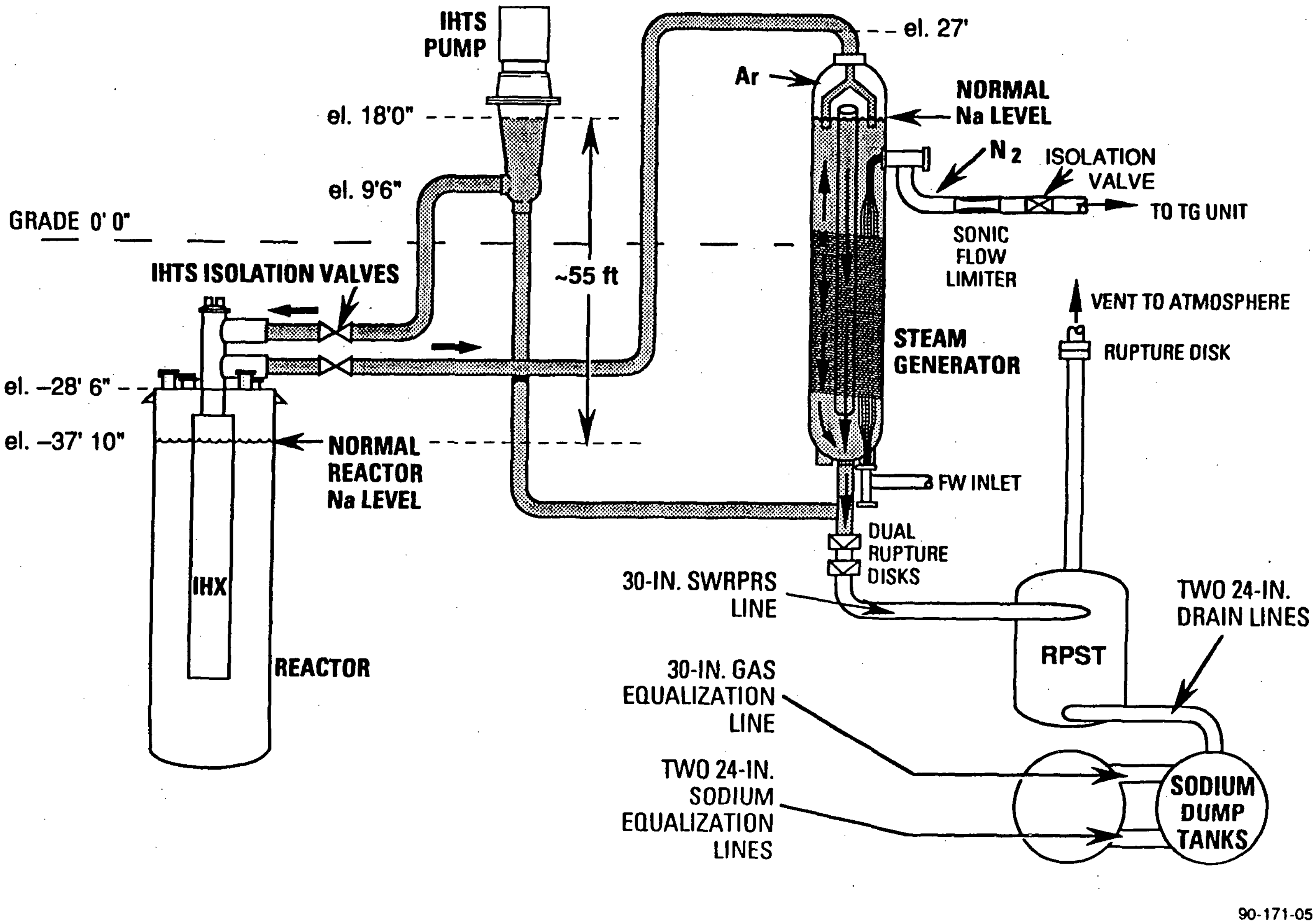


Figure G.4.1-27 - SODIUM LEVELS DURING NORMAL OPERATION

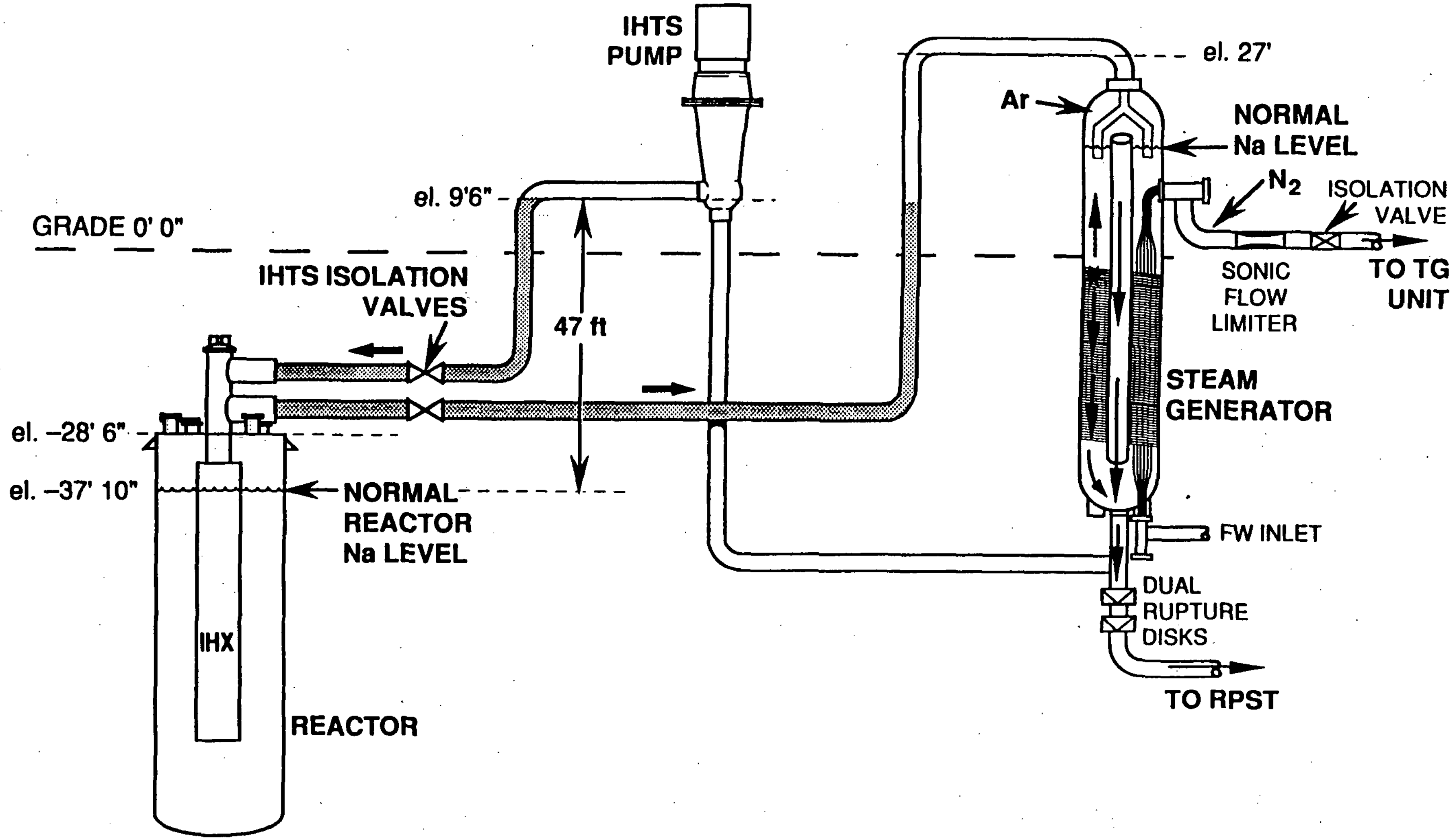


Figure G.4.1-28 - SODIUM LEVELS FOLLOWING STEAM GENERATOR LEAK

- o Drain IHTS Sodium
- o Purge IHTS Pipe With Argon Gas Through Vent/Purge Line Connections At IHX
- o Remove Piping Jacket and Insulation
- o Remove Plug and Insert Inflatable Bladder
- o Inflate Bladder To Isolate Sodium System
- o Cut Pipe and Install Welded Cap At Open End Of IHTS Pipe

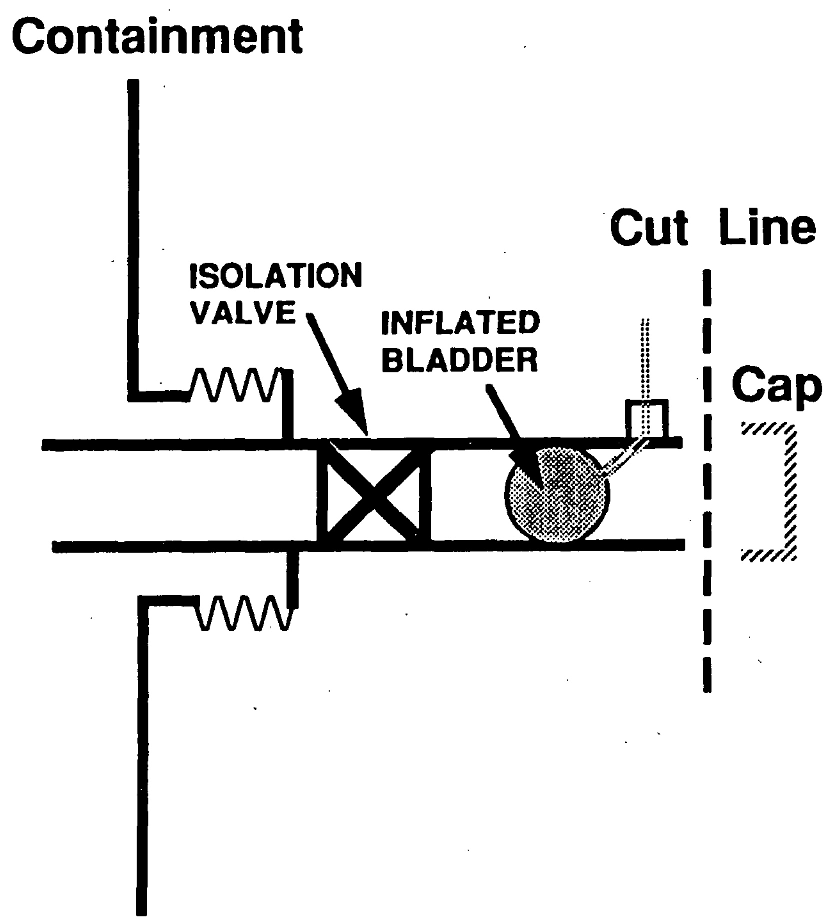


Figure G.4.1-29 - IHX BOUNDARY - RECOVERY PROCEDURE

G.4.2 Shutdown Systems

In the draft SER, five design and safety related issues on the ALMR shutdown systems are identified. These are summarized in section G.4.2.1. Descriptions of the reference shutdown systems are given in Section G.4.2.2. Responses to the SER positions are given in Section G.4.2.3.

G.4.2.1 SER Positions on Shutdown Systems

G.4.2.1.1 Two Independent Reactivity Control and Shutdown Means (SER Section 3.2)

"The requirement of GDC-26 for an independent, diverse engineered means of reactivity control is provided in PRISM by the inherent reactivity feedbacks of the design which, according to the designers, bring the reactor to zero power upon loss of flow or loss of a normal heat removal path, even if there is a failure to scram. This is acceptable to the staff as a means of meeting GDC-26 and the criteria discussed in Section 3.1.1.1 of the SER provided that certain conditions [discussed below] can be met (see SER Section 7.2.5.1). Adequacy of the proposed design to meet the purpose of the GDC through passive feedbacks will be demonstrated by prototype testing prior to certification of the design."

Section 3.1.1.1 of the SER defines the criteria for the shutdown system as follows:

"Two diverse, independent means of reactor shutdown, each of which is capable of shutting down the reactor assuming a single failure of active components and without dependence on support systems (electric power, instrument air, etc.). One of the systems must be capable of bringing the plant to cold shutdown indefinitely. The other system must be capable of bringing the plant to hot shutdown for an extended period of time."

The issues raised in Section 7.2.5.1 of the SER are summarized in Sections G.4.2.1.4 and G.4.2.1.5.

G.4.2.1.2 Positive Sodium Void Coefficient (SER Section 4.3.5)

"The positive sodium void coefficients result in certain EC-III events having the potential to lead to positive reactivity insertion events . . . . The positive sodium void reactivity coefficient is a concern to the staff and efforts should be made to reduce its magnitude, as much as practical, even if the likelihood of sodium boiling is reduced such that no events which could lead to sodium boiling are in EC-I through EC-III."

G.4.2.1.3 Core Performance Under Transient Conditions (SER Section 4.5.5)

SER Section 4.5.5 raises three concerns about transient performance: (1) achieving cold shutdown, (2) ability to meet criteria for bounding events, and (3) flotation of control assemblies.

Section 7.2.5.1 of SER raises concerns about a lack of means to bring the reactor subcritical following the action of the feedback and the need to develop a program to verify the adequacy of the feedbacks over the plant life. "Since there is currently some problem in this area (see Chapter 15 and Appendix B of the SER), this remains an open item."

Chapter 15 and Appendix B of the SER raises concerns about the ability of the PRISM design to meet the criteria for certain bounding events, uncertainties in the nature and adequacy of the feedbacks and the magnitude of the positive void worth.

"As noted in Section 4.4.5 of the SER, it must be shown that flotation of the control assemblies will not occur in the event of primary coolant pump startup while the control rods are delatched for refueling."

G.4.2.1.4 Adequacy of Reactivity Feedbacks (SER Sections 4.6.5, 7.2.5.1)

1) Is It A Shutdown System?

"Because the power runback that generally develops when the PRISM reactor 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 can accept passive shutdown characteristics as a diverse, independent means of reactor shutdown for PRISM provided certain conditions are met (see [SER] Section 7.2.5.1)." These conditions (stated in SER Section 7.2.5.1) are that suitable recovery actions are developed to achieve subcriticality in a reasonable time and an in-service testing program can be developed to verify over the life the plant that the magnitude and nature of the feedbacks are sufficient to respond to events in EC-I through EC-III without reliance on the RPS.

2) Is The Response Predictably Safe?

"For PRISM, the response to loss of flow, loss of heat sink, and transient overpower events appears to be quite acceptable, although the safety tests will be needed for confirmation. However, for events for which reduction in flow caused by a loss of pump coastdown occurs, sodium boiling could occur. Prevention and/or mitigation of this event needs further study."

3) 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 PRISM reactor 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 core-wide basis under failure to scram conditions, the reactor would be likely to experience a severe power excursion. Note, however, that sodium boiling is extremely unlikely. Certain events analyzed for PRISM have the potential to lead to sodium boiling and need further study before the acceptability of the PRISM design can be determined."

G.4.2.1.5 Feedback Verification and Cold Shutdown Method (SER Section 7.2.5.1)

"GE acknowledges a need for a highly reliable scram of the reactor. GE is relying on one shutdown system that indeed appears to be highly reliable; however, its susceptibility (with diverse means of rod insertion) to common cause-failure needs a thorough review at a later design stage. The staff believes that the diverse means of shutdown provided by the passive reactivity feedbacks is acceptable to meet the intent of GDCs 26 and 27, provided that suitable recovery actions are developed to achieve subcriticality in a reasonable time and an in-service testing program can be developed to verify over the life of the plant that the magnitude and nature of the feedbacks are sufficient to respond to events in EC-I through EC-III without reliance on the RPS."

G.4.2.1.6 SER Conclusion (SER Sections 4.5.6, 4.6.6, and 7.2.6)

"The use of a single active safety grade scram system is acceptable provided that this system, in conjunction with the passive shutdown characteristics, can protect the core (no melting or significant damage) under all EC-I through III events."

"The passive response of the PRISM reactor is not a true reactor shutdown mechanism, but it does accomplish the essential function by reducing the power generation to a level where 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, provided the provisions of Section 7.2.5.1 are met."

"Final acceptance of verification of the passive shutdown features will depend upon completion of additional R&D and satisfactory development of a means for in-service testing and measurement of the reactivity feedback mechanisms and recovery actions to achieve subcriticality."

G.4.2.2 Reference Shutdown Systems Description

The ALMR has multiple and diverse means for reactivity control and shutdown. These are, in their expected sequence of use, as follows:

- o Run-in of the six control rods by the PCS shim motor,
- o Gravity scram of the six control rods by the safety-grade RPS in response to an event developing too rapidly for the PCS,

- o Fast drive-in of the six control rods by the safety-grade RPS drive-in motors (initiated simultaneously with the gravity scram),

- o Inherent negative reactivity feedback with rising temperature in response to undercooling and overpower events without scram to bring the reactor to a safe, stable state,

- o Passive means of reactivity control,

  - GEMs to enhance the negative reactivity feedback during the loss of flow without scram event,

  - A rod stop system to limit reactivity addition during the all-rod withdrawal without scram event, and

- o Reactor shutdown to cold subcritical conditions by manually releasing neutron absorbing balls containing boron-10 into an assembly in the center of the core.

Extremely high reactivity control and shutdown reliability is achieved with these multiple and diverse means. Each ALMR reactor has six control rods, controlled by a triply redundant reactivity controller, which is part of a highly reliable plant control system (PCS). The reactor can be reliably shut down by PCS-directed rod run-in. In addition, the ALMR reactor has a Class 1E reactor protection system (RPS) which is totally separate from the PCS (separate sensors, separate electronics, and separate actuators), which utilizes quad redundant electronics, and which provides two diverse means for scramming each of the six rods. The worth of each rod is sufficiently high so that insertion of only one control rod will shut down the reactor. The RPS scrams each rod by releasing the rod latches for fast gravitational rod insertion and simultaneously activating the Class 1E rod drive-in motor for ensured motor rod insertion. The ALMR also has strong negative reactivity feedback with rising temperature that makes the reactor capable of passively withstanding severe undercooling and overpower accidents without scram. Finally, the ALMR has a manually actuated ultimate shutdown system (USS) which, when activated, permits boron-10 balls to fall by gravity from a container at the reactor closure into an assembly in the core center. For ATWS events, reactor inherency maintains the reactor in a safe, stable state until cold shutdown is achieved with the ultimate shutdown system.

The six control rods, each with enough worth to shut down the reactor, and each with three diverse insertion means (PCS shim motor run-in, RPS gravity scram, and RPS drive-in motor scram), provide a high level of redundancy and diversity for reactor shutdown. The estimated probability of the RPS failing to shut down the reactor is less than 10^{-6} per demand. The reactor inherency and the ultimate shutdown system provide additional means for reactor control and shutdown beyond the RPS.

The ALMR design provides multiple levels of protection against events that challenge plant safety. The first level of protection is provided by the PCS, which is a highly reliable digital control system with triply redundant controllers. The PCS maneuvers the rods, controls the primary and secondary pumps, and regulates the feedwater, turbine admission, and other BOP valves to provide optimum plant operation while protecting plant equipment for normal and most anticipated upset conditions during all operating modes (startup, power operation with load following, shutdown). Included in the PCS list of control strategies is the ability to run back the reactor to low or zero power to maintain the reactor in a safe operating state with margin.

The second level of protection is provided by the safety-grade reactor protection system (RPS). If an emergency develops too rapidly for PCS control and mitigation, and reactor safety limits are threatened, the safety-grade RPS located at each reactor module will automatically scram the reactor.

The third level of protection is provided by the core's strong inherent negative reactivity feedback with rising temperature. As the reactor temperature increases during an event, the negative feedbacks from the radial expansion, grid plate expansion, axial expansion, Doppler and control rod driveline expansion collectively generate a significant net negative reactivity for the core. This feature, combined with the passive RVACS heat removal capability, makes the ALMR capable of safely withstanding severe undercooling and overpower accidents without scram.

For example, one extremely low probability bounding event considered for the ALMR involves accidental withdrawal of all control rods at their maximum rate to the limit permitted by the control rod mechanical withdrawal stops (40¢ positive reactivity insertion at a peak rate of 2¢ per second) with simultaneous loss of all cooling by the intermediate heat transport system, initiated from a steady-state full power condition. The estimated probability of occurrence of this event is on the order of less than 10^{-9} per plant-year; it envelopes in severity several other extremely low probability events. The system responds to this event by bringing itself, totally passively, to a stable equilibrium state at a core outlet sodium temperature that is: (1) below the ASME Code long-term structural design limit for faulted conditions, (2) below the limit for incipient eutectic formation between the metal fuel and the ferritic fuel pin cladding (1300°F), and (3) is more than 470°F below the sodium boiling point. This condition can be safely accommodated until corrective action can be taken to bring the reactor to cold, subcritical shutdown. Based on results of metal fuel tests by ANL, it is expected that no major failures of the fuel pin cladding will occur during this event.

The fourth level of protection is provided by the ultimate shutdown system. Final shutdown can be achieved by activating the USS, which releases neutron absorbing balls containing boron-10 and allows them to fall by gravity from a container at the reactor head closure into an assembly in the center of the core. Substantial time (hours) is available for this action. The shutdown action itself can be completed in less than one minute once initiated.

The descriptions of the control rod system, inherent negative feedback, ultimate shutdown system, and the negative feedback enhancers (GEMs and rod stops) are given in the following sections.

G.4.2.2.1 Control Rod System

The reactor is equipped with six control rods containing pins loaded with boron-10. The control rods provide startup control, power control, burnup compensation, and absorber run-in ("runback") in response to demands from the plant control system (PCS). These PCS-directed control rod movements are accomplished with the rod shim motors. The system also provides rapid shutdown in response to demands from the reactor protection system (RPS). RPS-directed control rod scram is accomplished by releasing the rod latches for gravity fall, and by activating the rod drive-in motors. The conceptual design of the control rod system is shown in Figures G.4.2-1 through G.4.2-3. The locations of the six control assemblies in the core pattern are shown in Figure G.4.2-4.

Each control rod has diverse means of shutting down the reactor. The first is by unlatching the absorber bundle from the driveline allowing it to drop into the core due to gravity. The second is by a fast drive-in of the driveline by an irreversible motor powered by an uninterruptable power supply. Fast drive-in is initiated at the same time as latch release and can exert over 2000 pounds force, if necessary, to ensure absorber bundle insertion. The third is by slow drive-in using the PCS-activated shim motors. Each of the six rods has sufficient individual worth to shut down the reactor, thereby providing a one in six redundancy. The diverse means of rod insertion, plus the one in six redundancy, result in an estimated failure to scram probability of 3×10^{-7} per demand, which satisfies the 10^{-6} failures per demand design requirement.

As a design basis, the unlatch time is specified as 0.2 seconds and the full stroke rod insertion time as two seconds. The fast drive-in is at 120 inches per minute, with acceleration to full speed in 0.2 seconds, giving a full stroke insertion time of 18 seconds. The maximum shim speed is nine inches per minute, one rod at a time, which results in a maximum reactivity rate of 2¢ per second. This maximum reactivity rate is used for the PCS-initiated "runback".

The design and performance descriptions of the drive mechanism, driveline, and control assembly are given in the following sections.

Drive Mechanism

The drive mechanism for each control rod (shown in Figure G.4.2-2) is mounted on the rotatable plug and provides for the axial positioning of the absorber bundle in the core. The axial motions are normal shim withdrawal and insertion, fast drive-in, and scram. Shim motion and fast drive-in are produced by a gear driven ball-nut acting on a lead screw. The screw is restrained against rotation and the nut is restrained against axial motion. Rotation of the ball-nut raises and lowers the lead screw.

The ball-nut gear is driven by three motors: two stepper shim motors and an irreversible dc drive-in motor. One shim motor is designated the lead motor and the other a standby motor. If the lead motor were to fail, it would be de-energized and use of the standby motor would be initiated. A brake is provided to prevent rotation of the nut, with resultant control rod movement, when the shim motors are de-energized.

The motors are connected to the ball nut gear through a torque limiting clutch. The fast drive-in motor is sized so that it will meet the fast drive-in requirements when driving against the torque limiter. If energized, the drive-in motor is also powerful enough to overcome any drive-out by the shim motors in the event the torque limiter does not slip.

Attached to the top of the lead screw is a dual coil electromagnet. The electromagnet holds an armature to which is attached the tension tube that extends down through the hollow lead screw and driveline to the latch at the bottom end of the driveline. The latch attaches the absorber bundle to the driveline. Scram is accomplished by de-energizing the electromagnet. When de-energized, the electromagnet releases its hold on the armature, and the tension tube drops about 1/4 inch releasing the latch's grip on the absorber bundle coupling head as shown in Figure G.4.2-2. The bundle then drops into the core by gravity. Simultaneously, the fast drive-in motor is energized causing the driveline to follow the absorber and ensure that it inserts completely. In the event the latch does not release or the absorber fails to insert, insertion is accomplished by the driveline pushing the bundle into the core.

The dual coil design allows testing and maintenance of the RPS circuits without causing the control rod to drop and thereby shut down the reactor.

Driveline

The driveline connects the drive mechanism to the absorber bundle. It consists of three concentric shafts: an outer drive tube, a tension tube, and a position indicator rod. The lower extremity of the outer tube provides the cam surface for the absorber bundle latch. The tension tube is connected to the multi-fingered latch at the driveline lower end. The latch connects to the coupling head and supports the absorber bundle.

The control rod latch design is nearly identical to the one used in the Clinch River Breeder Reactor secondary control rod system. This latch was extensively evaluated and tested and found to operate correctly over a broad range of loading, temperature, and misalignment conditions.

The innermost driveline member is a position indicator rod. When the absorber bundle is latched, its lower extremity rests on top of the absorber bundle coupling head, and its upper extremity extends through the reactor closure head and drive mechanism to a point above the electromagnet where its elevation can be measured. The position indicator rod also senses when the coupling head of the absorber bundle enters and leaves the latch, thereby confirming engagement or disengagement of the absorber bundle with the driveline.

Control Assembly

The control assembly (see Figure G.4.2-3) consists of an absorber bundle contained within a channel or duct. Six control assemblies are located within the array of core assemblies as shown in Figure G.4.2-4. Movement of the absorber bundles in and out of the core region regulates reactivity.

The absorber bundle is a closely packed array of tubes containing compacted boron carbide pellets. The natural boron-10 enrichment is 20%. The tubes, referred to as "pins", are each helically wrapped with wire and bundled into a triangular pitch, hexagonal pattern as shown in Figure G.4.2-3. The wire wrap maintains the pin spacing so that coolant may circulate freely through the pin bundle. The bundle of pins is contained in a thin duct that channels flow through the bundle and protects the pins from damage as they slide within the outer fixed duct.

The control assembly outer duct is hexagonal, having the same external dimensions as the fuel and blanket ducts except for the nosepiece which has unique discrimination features to preclude inadvertent installation into an unassigned core matrix position. The duct directs coolant flow to the absorber bundle. For recoupling after the absorber bundle is released from the driveline, the opening at the top of the duct aligns and guides the driveline into re-engagement with the absorber bundle coupling head. The control system is designed to be operated with the absorber bundle partially inserted at all times.

The drive mechanism stroke is nominally 50 inches, so that the driveline may be withdrawn during shutdown when the absorber bundle is disconnected from the driveline and the driveline is parked far enough above the core to permit rotation of the closure plug for refueling.

<u>Control Assembly Performance</u>

The control system scram worth requirement is determined by the peak core excess reactivity, the temperature defect, the rod insertion pattern (how many scrammed), the shutdown margin requirement and uncertainties. To include core reactivity design uncertainties, the rod average scram worth requirements are determined based on the highest excess reactivity state allowed by the core reactivity design uncertainty tolerances. The average worth requirement is adjusted by the appropriate rod interaction factor and number of rods to determine the nominal system scram requirement. The boron worth uncertainty is then applied to calculate the design system worth requirement.

Table G.4.2-1 lists the excess reactivity contributors in the reference core, and combines them into the beginning and end of cycle reactivity states. For the ALMR core, the highest excess reactivity state occurs at the beginning of cycle as a result of burnup reactivity loss and fuel axial expansion. Based on a 3-sigma design basis, the core excess reactivity is 2.39$. Each rod suppresses one-sixth of the total excess reactivity at full power. The reactivity elements suppressed by a rod during a scram are: (1) the full power excess reactivity state suppressed by one rod, (2) the temperature defect and uncertainty divided by the number of scrammed rods, and (3) any required shutdown margin divided by the number of scrammed rods. The sum of these three elements provides the average rod scram requirement.

Table G.4.2-1

ALMR REFERENCE METAL CORE REACTIVITY CHARACTERISTICS

Reactivity Contributor	Mean ($)	Standard Deviation ($)
Fast Runback Margin	0.50	-----
Over Power Margin	0.12	-----
Cycle Criticality Margin	1.53	-----
Reload Uncertainty		
Fissile Variation	----	0.05
Calculational Uncertainty	----	0.06
Batch Reload Variation	----	0.00
Fuel Axial Expansion (5%)	-1.06	0.11
Burnup Swing	-0.04	0.05
Temperature Defect	1.20	0.02
Boron Worth Uncertainty	----	2%
Design Standard Deviations	3	-----

Core Operational State	Net Excess Reactivity ($)		
	-3 Sigma	Nominal	+3Sigma
Beginning of Cycle Full Power	1.91	2.15	2.39
End of Cycle Full Power	0.62	1.05	1.47
End of Cycle 110% Power	0.50*		

* Rod Physical Limit, A Fixed Design Parameter

The ALMR core design requirements specify two scram requirements: (1) a five of six rod scram requirement and (2) a one of six rod scram requirement. The five of six rod scram assumes one rod is stuck at the full power banked position. A shutdown margin of 2$ is also required in this scram event. Table G.4.2-2 details the calculation of the resulting scram worth requirements for the reference core.

Table G.4.2-2

FIVE OF SIX ROD SCRAM WORTH REQUIREMENT

Reactivity Element	Reactivity ($)
BOC Full Power Excess Reactivity Divided By 6 Rods	0.40
Temperature Defect Divided By 5 Rods Scrammed	0.24
Temperature Defect Uncertainty Divided By 5 Rods Scrammed	0.01
Scram Margin Divided By 5 Rods Scrammed	0.40
Average Rod Scram Requirement	1.05
Rod Worth Uncertainty Factor (2%)	0.02
Interaction Factor	0.96
Single Rod Worth Requirement	1.11
Six-Rod System Worth - Requirement	6.66
- Actual (Natural B_4C)	20.43

The single rod scram requirement specifies a nominal shutdown, interpreted to mean no shutdown margin is needed above the already included core reactivity design uncertainty factors. Table G.4.2-3 details the resulting scram requirement calculations.

In both scram cases, the boron worth uncertainty is included to ensure that assemblies of sufficient worth can be designed within the core envelope constraints.

As shown in Tables G.4.2-2 and G.4.2-3, these design requirements result in the single rod scram requirement (12$) governing control worth specification. Comparison of these requirements with the control system worth in each table shows that the requirements can be satisfied with natural enrichment boron carbide. The reference rod design with natural B_4C has about a 20$ scram worth.

Table G.4.2-3

ONE OF SIX ROD SCRAM WORTH REQUIREMENT

Reactivity Element	Reactivity ($)
BOC Full Power Excess Reactivity Divided By 6 Rods	0.40
Temperature Defect	1.20
Temperature Defect Uncertainty	0.06
Average Rod Scram Requirement	1.66
Rod Worth Uncertainty Factor (2%)	0.04
Interaction Factor	0.85
Single Rod Worth Requirement	2.00
Six-Rod System Worth - Requirement	12.00
- Actual (Natural B_4C)	20.43

Control Rod Scram Actuation

The quad-redundant reactor protection system (RPS) monitors the reactor performance parameters and automatically scrams the reactor if safety setpoints are exceeded. The RPS monitors the neutron flux, core coolant inlet temperature, core coolant outlet temperature, core coolant inlet pressure, and reactor coolant level (see Figure G.4.2-5). Exceeding setpoints in any one of these five reactor parameters causes the RPS to scram the six control rods. The RPS is a Class 1E system with one RPS provided for each reactor.

An RPS trip command results in the release of bundle absorber into the reactor core as shown in Figure G.4.2-6. Interruption of the electrical current to the coils of the electromagnet (a trip) opens the latch, releasing the absorber bundle and allowing it to drop into the core under its own weight. A unidirectional dc motor (much more powerful than the shim motor), when activated by the RPS as part of a trip sequence, drives each control assembly driveline to the bottom of its stroke to ensure complete insertion of the absorber bundle. The RPS has no control rod withdrawal capability. In addition to control rod insertion, the RPS initiates an EM pump primary sodium flow coastdown (after confirmation of rod insertion) by opening safety related circuit breakers between the PCS power conditioning unit and the EM pump.

Referring to Figure G.4.2-6, the RPS is divided into four identical divisions, each located within its own seismically isolated instrument vault. Each division is provided with a sensor for each measurement. Thus, there are four sensors for each monitored parameter. Each division consists of signal conditioning input circuitry; interdivisional data exchange; the central processing logic unit; actuator output circuitry; and independent, battery backed, uninterruptable, Class 1E power supplies. Provision is made to send all RPS readings to the Class 1E remote shutdown console in the remote shutdown facility (RSF), and to the PCS and the control room after Class 1E isolation.

The four divisions operate asynchronously, in parallel with data exchange, verification and validation as a single fault tolerant system. The four divisions share data via optically isolated interdivisional cables from all sensors. Each votes 2 out of 3 on the validity of the data and its analysis. Each division is output to two trip breakers. The breakers are arranged to form a hardwired 2 out of 4 fail-safe logic between the four divisions.

The ALMR RPS is designed to ensure that: (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required protection function.

All operations of the RPS are automated. There is no requirement for operator action at this time. The RPS is designed to manually execute a trip sequence from its own Class 1E scram buttons, an action that bypasses all electronics and interrupts power to the trip breakers directly. Manual scram may be initiated by a control operator from the main control room. A safety related, manually input scram command may be input to the RPS from the scram buttons located on the face of the Class 1E remote shutdown console in the remote shutdown facility. Once a reactor trip sequence is complete, operator action is required to initiate scram recovery and restart the reactor.

Normal Control Rod Positioning

Positioning of the control rods for reactor startup, power regulation, burnup compensation, normal shutdown and fast run-in is accomplished through the non-safety related plant control system (PCS). Each control rod drive mechanism has a shim motor, plus an in-place spare, controlled by the PCS for raising or lowering the absorber bundle to control reactor power.

The system used for rod control operation is shown in Figure G.4.2-7. The system monitors reactor power and control rod positions, and operates the individual rod shim control motors. It contains the necessary switching and selection logic to address individual rods.

The rods are withdrawn as a bank. To minimize the effects of inadvertent control rod withdrawal, the six rods are connected to the control rod power supply one at a time. To prevent unbalanced withdrawal of the rods, the motion of an individual rod is limited before the power source is switched to another control rod. Additionally, the position of an individual rod must be within a specified tolerance of the average control rod position, or power to rod that is terminated.

Measurements used by in the reactor controller include (1) primary sodium flow, (2) reactor module flux, (3) reactor module sodium temperatures, (4) rod positions, and (5) power source and drive motor conditions.

The rod bank position is adjusted to meet the reactor module power (flux) and primary hot leg sodium temperatures demanded by the power block's supervisory controller. Power changes during normal load following are kept below 1% per minute to minimize thermal transients. However for a 10% step load change, a 1% per second power change is allowed. For runback, the rods are run in one at a time at 9 inches per minute corresponding to a peak reactivity change of 2¢/sec.

The weighting of the control variables and the controller logic utilized assures that the controller does not interfere with the inherent safety responses of the reactor modules. Models of reactor power and other pertinent reactor state variables, fed by real-time plant data, are run continuously in real time within the PCS controller, and continuous on-line diagnostics are provided to automatically initiate investment protection actions for abnormal operating conditions.

G.4.2.2.2 Inherent Negative Reactivity Control

The ALMR reactor is designed to provide strong inherent negative reactivity feedback with rising temperature. This characteristic, combined with the RVACS heat removal capability, makes the ALMR capable of safely withstanding severe undercooling and overpower transient events without scram. An overview description of the negative feedbacks is presented in this section.

As the temperature increases during an event, the negative feedback from the radial core expansion, grid plate expansion, axial core assembly expansion, Doppler, and control rod driveline expansion are activated, and these generate a net negative reactivity for the core. This feedback responds according to the associated time constants, to overcome the positive reactivity from the sodium density effect and any external source. Because of the small Doppler feedback in metal fuel, and the correspondingly small temperature defect, the drop in power can be quite large. Each of the important reactivity feedbacks are discussed below.

Doppler feedback is generally the fastest acting feedback mechanism since it is almost instantly affected by core power level. Doppler removes reactivity from the system as the temperature rises and can thus help limit the extent of power increases. However, as the fuel temperature drops with a power reduction, Doppler adds reactivity which tends to limit the power decrease.

The fuel thermal expansion is a relatively fast acting feedback mechanism. The radial fuel slug thermal expansion is accommodated within the pin and does not affect the core reactivity significantly. Axial fuel expansion increases the core height as temperatures rise, and changes the reactivity of the system by increasing the neutron leakage. The result is a rapid negative feedback contribution from an increase in fuel temperature, or a rapid positive feedback in response to a decrease in fuel temperature.

Thermal expansion of the sodium results in a net positive reactivity feedback. The thermal expansion results in fewer sodium atoms within and surrounding the core. The reduced density surrounding the core results in fewer neutrons being scattered back into the core, and produces a small negative feedback effect by increasing the leakage around the periphery. However, the dominant effect is to reduce the collisions between neutrons and sodium atoms, which hardens the neutron energy spectrum and yields a net positive reactivity feedback effect.

Control rod driveline and vessel differential thermal expansion affect the axial position of the absorber bundles relative to the core. The absorber bundles are supported from the reactor head via the driveline. The core is located near the bottom of the reactor and supported from the reactor head via the reactor vessel. During power operation the driveline is at a temperature of 905°F, since it is located in the hot upper plenum. The average vessel temperature is about 700°F, since at the lower elevation it is in contact with the cold pool sodium (640°F) and at the upper elevation it is thermally isolated from the hot plenum by a liner and baffles. As the core coolant exit temperature increases, the driveline temperature (being above the core) rises and the drivelines elongate. This inserts the bundles into the core adding negative reactivity. During the initial portion of the temperature rise, the vessel temperature remains unchanged since the hotter coolant has not yet reached the vessel. Further in time, the vessel temperature increases due to the higher overall reactor temperature. The higher vessel temperature elongates the vessel and causes the core to be pulled away from the absorbers, adding positive reactivity.

The radial expansion of the core is a result of thermal expansion, as well as the design of the core and restraint system. The core assemblies are restrained at three locations: the inlet nozzle, the above core load pads (ACLP), and the top of the core load pads (TLP). The TLPs are restrained at the core edge by the core former ring. The ACLPs are not restrained at the core edge. The inlet nozzles are inserted into the inlet modules which are fixed by the inlet grid plate. This restraint system is called the "limited free bow" design.

There is an overall expansion at the ACLP plane due to the increased core temperatures. The duct region is thin and has a small heat capacity, causing the expansion feedback effect to respond within a few seconds. The effect of this growth in volume and outer surface area of the core is to increase the loss of neutrons from the core region through the surface area. This causes a reduction in core reactivity.

In addition, the radial power profile across the core results in a decreasing temperature gradient from center to periphery. The side of the assembly duct facing the core center is hotter than the side away from the core center, so that the differential thermal expansion of the duct tends to cause the assembly to take a shape that is convex to the core center line. The interaction between adjacent assemblies and core restraint system forces the core to deflect outward and reduces the neutronic efficiency of the core. This is because the assembly tries to "flower" outward but is constrained by the top load pads and top former ring to maintain its radial position at the top of the assembly. Core compaction would then result in the region of the active core if it were not for the above core load pads, which stop the inward movement at their elevation. The movement caused by the rigid ACLP produces a reverse deflection on the assembly, which results in outward bowing in the active core region as the temperatures are increased and, therefore, a negative bowing reactivity feedback.

The performance of the reactivity feedbacks in demonstrating severe accident accommodation capability is discussed in Section G.4.16, Safety Analysis.

G.4.2.2.3 Gas Expansion Modules

Gas expansion modules (GEMs) are devices designed to passively provide negative reactivity feedback during loss of primary flow (LOF) events. Their principle of operation is to control neutron radial leakage from the core with a gas and sodium filled cavity at the driver core perimeter that is connected hydraulically to the high pressure plenum. When the pumps are at full flow, the plenum pressure compresses the gas to the cavity top end above the core, producing neutron back scattering into the core by the sodium in the cavity. When the flow decreases, the trapped gas expands and replaces the sodium in the core elevation of the cavity. The gas scatters fewer neutrons back into the core and thus produces a negative reactivity feedback. The ALMR employs three GEMs for reactivity feedback enhancement during the loss of flow without scram events to help meet the clad-fuel eutectic temperature limit.

The ALMR GEMs are designed to satisfy the following requirements:

a) Provide less than 1$ of reactivity feedback.

b) Provide sufficient reactivity feedback to permit accommodation of ALMR ULOF events and Bounding Events within the EC-III damage limits. Based on ARIES analyses, the reactivity feedback worth of the GEMs in the reference metal core should be greater than 0.4$.

c) Be capable of in situ testing.

d) Accommodate expected core performance uncertainties and variations, such as the uncertainty in inlet plenum pressure at full flow.

Requirement "a" is determined by the desire to avoid prompt critical events during inadvertent pump start-ups with rods partially withdrawn.

Requirement "b" is determined by the feedback required to offset control withdrawal feedback resulting from driveline/vessel thermal expansion. This requirement will vary with core design.

Requirements "c" and "d" accommodate expected core operations.

A GEM is essentially an empty assembly duct, sealed at the top, open at the bottom and connected to the core high pressure inlet coolant plenum. Figure G.4.2-8 shows the key features of a GEM with core locations of the three GEMs shown in Figure G.4.2-4. The GEM upper section consists of a handling socket and a short shield section that also forms a sealing plug at the top of the assembly. A standard hexagonal cross section duct forms the body and a nosepiece completes the bottom end.

The lower shield block is Inconel to maximize the gas volume available below the core. Maximum gas volume is essential to maximizing the "stroke" of the GEM sodium level. The upper shielding is HT9. The length of the assembly above the sealing plug is ample for shielding with the less efficient HT9 neutron reflection.

At completion of insertion into the core by the in-vessel transfer machine, the gas bubble is compressed into the cavity by the static head of the sodium and by the low flow (~5%) primary EM pump head, such that the sodium level rises through the nosepiece to a level near the bottom of the active core. When the inlet plenum pressure increases at full pump flow, the sodium level in the cavity rises until the gas pressure balances the coolant plenum pressure. The elevation of the top of the gas cavity is set by the shield plug such that the sodium-gas interface is then above the active core. The performance of the GEMs for demonstrating LOF event capability is discussed in Section G.4.16, Safety Analysis.

G.4.2.2.4 Rod Stop System

A rod stop system (RSS) is provided to limit control rod withdrawal so as to bound the amount of reactivity that can be added to a core as a result of an uncontrolled rod withdrawal event. This feature makes possible the passive accommodation of events that are precipitated by one or more control rod runouts accompanied by a failure to scram (see Section G.4.16). The system is comprised of a motor driven, movable stop within each control rod drive mechanism and a computerized controller. The stop physically limits the withdrawal stroke of the control rod drives. An electronic controller computes the position to which the rod stop should be set, subject to plant operator permission for set changes, in order to accommodate reactivity changes over the operating cycle. It is expected that resetting the rod stop position will be required five to six times each fuel cycle (18 months).

<u>Functional Requirements</u>

The rod stop system is designed to satisfy the following requirements:

a) Limit to 0.40$ the reactivity insertion possible from all control rods being withdrawn from the normal full power banked position until stopped by the limiter.

b) Permit adjustment during the operating cycle.

c) Have a probability of failure on demand of less than 1×10^{-3}.

Requirement "a" is determined by the reactivity self-control provided by the core feedbacks and by the thermal margins provided by design. Analyses of transient overpower events indicate that the ALMR core can accommodate up to 0.40$ of reactivity insertion from full power without scram, and still meet EC-III limits.

Requirement "b" is a result of reactivity changes that can occur during a cycle in metal cores as a result of fuel axial expansion and the increased burnup swing from using LWR recycle fuel as feedstock for the startup core. This requirement limits the availability impact of the rod stop system.

Requirement "c" is selected to provide high assurance that a UTOP event will not cause major core damage.

Design Description

The RSS for the ALMR is comprised of (1) a redundant electronic controller for initial rod stop positioning and subsequent adjustment and (2) mechanical out-motion blocks located within each control rod drive mechanism (CRDM).

The conceptual design of the mechanical portion of the RSS is illustrated in Figure G.4.2-9. A stepper motor drives an Acme screw through a combination of spur gears. The rod stop is a block attached to and positioned by the Acme screw drive and sized to prevent the control rod from exceeding the stop position. The stop is guided on a track to maintain alignment with the control rod driveline. Redundant, absolute position sensors, attached to the rod stop or the screw drive, continuously monitor the stop position. Similar redundant absolute position sensors connected to the control rod drive provide continuous control rod position information.

The mechanical portion of the rod stop will be designed to prevent the control rod drive mechanism from moving the stop. The stop is an integral part of the Acme screw and nut. The Acme screw is a one-way device - turning the screw will move the stop but the stop cannot rotate the screw. This feature will ensure that the stop maintains its preset position in the event the control rod drive mechanism pushes against the stop.

Figure G.4.2-10 illustrates the key functions of the rod block system controller. The RSS controller, power breakers, power supply, stepper motor controller and distributor are located in the RPS electronics vaults adjacent to the head access area. The rod stops, associated adjustment motors, stop position sensors and hardware are located inside each CRDM. The console at which the operator receives rod stop positional data and reset requests, and from which he provides permission for the rod stop adjustment, is located in the RSF adjacent to the control building. Rod stop positions are computed automatically by the RSS controller, but an operator permissive is required before the stop adjustment can be made.

The rod stop adjustment stepper motors are powered by a power supply sized to permit only one motor to function at a time. Power is directed to the proper rod stop by an integral power distributor. This method of preventing multiple rod stop movements is similar to that employed by the PCS in preventing simultaneous control rod movements.

The rod stop system controller includes a redundant, 1E controller, rod stop drive selector, and limited capacity power supply which controls power to each of the six rod stop adjustment drive motors, one for each control rod. Absolute position sensors are used to determine control rod and stop positions. The rod stop system controller is separate from the reactor protection system (RPS) controller. The RSS obtains reactor power and absolute control rod position data from the redundant class 1E sensors through the RPS controller. The RSS is on (activated) at all times to monitor rod stop positions and reactor power. Stop adjustment stepper motor power is enabled only on operator permissive.

The level of redundancy needed by the computerized controller will be determined in a future study. The reliability requirements based on preventing the rod stop system from affecting availability during operation or testing, and on providing any additional investment protection, have not been investigated.

When reactivity loss from burnup swing and fuel axial expansion has caused the banked full power control rod positions to move close enough to the rod stops that normal operation may be affected, then the adjustment system is started by operator action. The controller is turned on and the operator instructs the controller to begin a rod stop readjustment cycle. The adjustment is an active process.

Rod Stop Adjustment Process

The rod stop adjustment process is initiated by the RSS controller. The monitoring and computation function of the RSS controller predicts at intervals the difference between the full power rod positions and the stops. When reactivity loss from burnup swing and fuel axial expansion has caused the margin to decrease to the extent that normal operation may be affected, the controller computes a new rod stop position, and issues a request to the operator for permission to adjust the rod stops.

The PCS simulation and advisory function predicts appropriate stop positions as support to operator judgment in granting permission. Assuming there are no administrative limits to operation, and that both the PCS advisory and the operator prediction of allowable rod stop settings are in agreement with the controller request, the operator grants permission for adjustment. The permissive must be granted from the RSF console, which has a Class 1E interface with the RSS controller. If there are administrative limits or the systems do not agree on predicted stop setting, the operator will withhold permission and the stops will not be moved. The operator will initiate diagnostic and corrective actions if problems are detected by the cross-checks.

The operator permissive requirement provides an administrative control which protects against controller errors. Allowing only RSS-controlled stop adjustments to take place prevents operator errors or sabotage from incapacitating the passive stopping function.

Since the reactor may not be exactly at full power during the stop reset calculations, the controller predicts the full power rod positions. The controller uses the current power and rod position data from the RPS and, if available, prior known full power positions. The accuracy of this prediction is determined largely by the deviation between the current reactor power and full power. An allowance is made in the prediction to account for uncertainties. The allowance increases with increasing deviation from full power.

If the permissive is granted, the RSS controller holds the current stop reset calculation results. It issues a command to the PCS to keep the rods stationary unless a runback is required. All automatic load changes are disabled during this time. The RPS scram function is not affected and if the RPS setpoints are exceeded while the stops are being adjusted, the rods will scram. The RSS controller enables the limited capacity power supply by connecting power through the breakers, directs the integral power distributor to select the appropriate control rod stop, and directs the stepper controller to output the appropriate stepper motor pulses for the stop motion required. It then proceeds to adjust all six stops. After completing the adjustments, the controller directs the power distributor to park in the standby position, and then removes power connections to the power supply. The controller monitors the feedback from the standby position of the distributor to verify removal of power from the adjustment motors.

During stop adjustment, the controller monitors the stop positions and terminates all stop adjustment if an error is detected. A warning to the operator is issued and the operator initiates corrective actions.

Figure G.4.2-11 illustrates the rod bank position during a typical equilibrium cycle at full power and the resulting stair-step stop adjustment sequence. The minimum gap allowed between the rod and stop is equivalent to 0.12$ to permit operation up to 110% power. With a maximum clearance between the rod and stop at full power of 0.30$, 0.18$ is available for core reactivity change between stop resets. A maximum clearance equivalent to 0.30$ is used to maintain ample margin against the 0.40$ limit. For comparison, note that a rod movement of 0.2 inch at the core midplane is equivalent to a system reactivity change of 0.18$.

During each cycle, burnup swing and axial expansion cause a core reactivity decrease of about 1$. Almost all this change occurs during the first six months of the cycle. The control rods are slowly withdrawn to maintain full power during this interval. Whenever the rod stops threaten to interfere with normal operation, (i.e., when the margin is reduced to 0.12$), the RSS requests a readjustment. With operator permission, the adjustment is made. This process is expected to occur between five and six times during the cycle, mostly in the first six months. Reactivity changes in startup cores using LWR recycle fuel feedstock are larger, and thus will require more frequent readjustments.

The rod stops do not move during load following and thus will maintain the prior setting of 0.30$ or less above the current full power critical rod position.

During rod drop tests conducted during shutdown, the RPS energizes one scram latch at a time and the RSS adjusts the appropriate rod stop to the fully raised position to permit full stroke drop tests. Since the shutdown/refueling mode signal from the RPS enables stop withdrawal, testing is not prevented by the rod stops.

G.4.2.2.5 Ultimate Shutdown System

The ultimate shutdown system (USS) provides a diverse, independent means of bringing the reactor to cold shutdown. During an anticipated transient without scram event (ULOF, ULOHS, or UTOP) strong inherent negative reactivity feedback with rising temperature maintains the reactor at a stable but hot equilibrium state. Final cold shutdown can be achieved with the USS by operator action to release neutron absorbing balls containing fully enriched boron-10 from a container at the closure head which fall by gravity into an open assembly in the center of the core.

For the control rods to fail to insert into the core, severe distortion of the core internals must occur. The B_4C balls are relatively small in size and have the characteristic of being able to follow a torturous path. So even under distorted conditions, it is reasonable to conclude that the balls will be able to move into the center core assembly and accomplish the cold shutdown function.

The USS is illustrated in Figure G.4.2-12. B_4C balls are stored in a dry canister within the reactor above the sodium. Upon actuation, the balls fall by gravity down a guide tube into an open thimble at the core center. The worth of absorber inserted into the core is sufficient to bring the reactor from 135% of full power to a cold shutdown.

The system is made up of four subassemblies which include the absorber storage canister, center shutdown absorber assembly, absorber guide tube, and guide tube drive mechanism. These subassemblies are described in the paragraphs following.

The absorber storage canister maintains the absorber material in a dry condition and out of the core position during normal reactor operating and shutdown periods. The absorber consists of approximately one cubic foot of 0.25-inch diameter fully enriched B_4C balls. The balls may be clad in a metallic jacket if testing shows a problem with the B_4C cracking.

The balls are retained in the canister by a hinged, sealed door secured by pull pins on each side of the hinge. The hinges form part of the absorber storage canister structure. The canister release mechanism is operated by a drive motor at the top of the assembly which pulls on a rod running the length of the canister. A rod guide tube is used to keep the balls from interfering with the actuator rod. Absorber release is accomplished by pulling on the actuating rod and disengaging the pull pins. The hinged bottom folds down allowing the balls to enter the absorber guide tube. A backup release is available by pushing down on the actuator rod and shearing the hinge support bar. To perform a release mechanism test without releasing the absorber material, the actuator rod is moved down a small distance (which does not shear the hinge), then up to its original position.

The guide tube extends from the closure to the core, and channels the B_4C balls from the absorber storage canister to the center core assembly. The tube is supported from the closure via the guide tube drive mechanism. During reactor operation the guide tube extends six inches into the center core assembly. During refueling the guide tube is raised 12 inches to permit plug rotation. The guide tube necks down above the core to provide clearance for the refueling machine.

The center core assembly maintains the B_4C absorber in the proper location once the ultimate shutdown system has been actuated. The assembly consists of a top handling and lifting socket, an outer hex duct, an inlet nozzle with a greatly reduced inlet area for draining and cooling for very low heat loads, and a small mesh support grid for the ball column. During reactor operation the center assembly is a sodium-filled hole in the core with an absorber guide tube inserted in the lifting socket. Removing the assembly, either at the end of life or after an absorber insertion, will be accomplished using fuel handling equipment.

A guide tube drive mechanism is provided. During refueling the drive mechanism lifts the guide tube 12 inches (six inches out of the core). The upper portion of the mechanism is removed during absorber canister inspection. The mechanism includes position indication instrumentation to confirm the location of the guide tube, and a load cell to confirm the absorber release.

The ultimate shutdown system is activated manually from a pair of ultimate shutdown buttons located in the remote shutdown facility (RSF), or from a similar pair of buttons in the RPS vaults. Each button passively connects two parallel contacts, which satisfies Class 1E criteria. The buttons are connected in series to prevent inadvertent activation, and to provide testability. There are a pair of buttons for each reactor module in the panel located in the RSF and one pair in the RPS vault of each reactor as shown in Figure G.4.2-13.

One of the alternatives being considered for the USS control system utilizes a fiber optic link between the RSF and the RPS vaults. For this concept, power for the system is provided by the Class 1E, redundant power supplies in the RSF and used to provide energy for redundant, Class 1E, light-emitting diodes (LED). The LEDs used are high-reliability, solid state devices, similar to those used for transatlantic and transpacific fiber optic communication systems, and are Class 1E qualified.

When the RSF USS buttons are pressed, light generated in the LEDs is sent down the fiber optic cables to the RPS instrument vault. In the RPS instrument vault, the light in the fiber optic cable is used to activate a switch in a hardware logic circuit. This switch closes a pair of contacts and provides the electric current to the actuator which opens the hinged doors and dumps the absorber balls into the center assembly in the reactor vessel. The hardware logic is configured such that light from two out of four parallel fiber optic cables is required to initiate the actuation and release the absorber into the center assembly.

In this concept design, there are four cables in parallel from the RSF to the hardware logic distributed throughout the four RPS instrument vaults. The hardware logic requires a signal from two out of four cables. This permits the system to tolerate a single failure and permits the system to tolerate taking one of the divisions out of service for maintenance and repair with no loss of USS function or reliability. Power is required for actuation to avoid accidental or inadvertent actuation.

The ALMR plant has four Class 1E ducts between the RSF and the reactor modules. The fiber optic cable from Division I at each reactor module will be carried by the first of the four Class 1E ducts. All fiber optic cables from Division II will be carried in the second, and so forth. This bringing together of all of the Division I cables in a single conduit running the length of the plant meets the criteria for a Class 1E system, gives divisional separation and isolation from non-Class 1E systems and conductors, and minimizes the cost of the installation of the system.

G.4.2.3 Response to SER Positions

This section responds to the SER positions on the shutdown systems described in Section G.4.2.1.

G.4.2.3.1 Two Independent Reactivity Control and Shutdown Means

The diversity requirement is satisfied by three insertion means (PCS shim motor run-in, RPS gravity scram, and RPS fast drive-in scram provided in each control rod). The redundancy requirement is provided by the single rod shutdown capability. The system is single active failure proof. If the absorber bundles fail to insert by gravity, they are forced in by the shim and fast drive-in motors. This internal diversity and redundancy are intended to satisfy GDC-26 and draft SER Sectin 3.1.1.1.

Inherent negative reactivity control and the ultimate shutdown system provide an extra means for reactor shutdown. These features give the ALMR another level of protection. In the event there is a loss of flow, loss of heat sink, or all-rod transient overpower all without scram, negative reactivity feedbacks keep the reactor in a safe, stable state (below the long term structure temperature limit, local sodium boiling and fuel-clad eutectic temperature) for an extended period. Cold shutdown can be achieved by manual actuation of the ultimate shutdown system.

G.4.2.3.2 Positive Sodium Void Coefficient

The positive sodium void coefficient of the ALMR core is acceptable because: (1) the probability of voiding a significant fraction of the core is extremely low in EC-IV and (2) core voiding, even if it were to occur, would not result in radiological release. For a detailed discussion of this issue, see Section G.4.5, Sodium Void.

G.4.2.3.3 Core Performance Under Transient Conditions

Achieving Cold Shutdown

The ultimate shutdown assembly has been added to the reference design in order to provide an independent, diverse means of bringing the reactor to cold shutdown from the safe, stable conditions maintained by the negative reactivity feedback. An in-service testing program will be developed which will periodically verify the adequacy of the feedbacks over the plant life.

Meeting Bounding Events

All bounding events are shown to meet the EC-III criteria. The detailed discussion of the bounding events, and the evaluations showing that the criteria are met, are given in Section G.4.16, Safety Analysis.

Absorber Bundle Flotation

Operation of the EM pumps during refueling will not result in absorber bundle ejection or flotation. The rods are designed to drop rapidly into the core against full flow when released from the driveline for reactor scram. The margin between operation and flotation conditions is large. Periodic rod drop testing will ensure absorber bundle drop against full flow. A detailed response to this issue is given in Section G.4.7, Electromagnetic Pumps.

G.4.2.3.4 Adequacy of Reactivity Feedbacks

The negative feedbacks maintain the reactor at a safe stable state at an elevated temperature but the reactor may still be critical. The ultimate shutdown system has been added to bring the reactor to cold shutdown. An in-service testing program will be identified which will periodically verify the adequacy of the feedbacks over the life of the plant.

The analysis of Bounding Event No. 4 given in Section G.4.16, Safety Analysis, shows that large margins to sodium boiling exist for events in which a loss of pump coastdown occurs.

The bounding events are discussed in Section G.4.16, Safety Analysis, and the specific issue of the positive void worth in Section G.4.5, Sodium Void. Sodium boiling is extremely unlikely. More importantly, even if sodium boiling and core voiding were to occur, there would be no radiological release as discussed in Section G.4.19, Mitigation of Severe Core Accidents, and Section G.4.1, Containment.

G.4.2.3.5 Feedback Verification and Cold Shutdown Method

Feedback Verification

A program will be developed to verify the adequacy of the inherent negative reactivity feedback. The magnitude and nature of the feedback will be verified during the ALMR prototype safety test and periodically, as appropriate, during the subsequent long-term power demonstration phase. An in-service testing program will be developed for the commercial ALMRs to verify the adequacy of the core feedback. The frequency for verifying the feedback in the commercial ALMR will be based on the final core design uncertainties and the experience gained in the ALMR prototype testing.

Argonne National Laboratory has developed a method by which the feedback can be measured on an operating liquid metal reactor (Reference G.4.2-1). These measurements can be made periodically over the reactor life to verify the magnitude and nature of the feedbacks. The feedback measurement technique will be initially demonstrated during the full size ALMR prototype safety test and used to verify periodically the adequacy of the feedbacks during the subsequent long term power operation phase.

The reactor core can be influenced by external events only through changes in the coolant inlet temperature, coolant flow rate, or through externally induced reactivity changes owing to control rod motion or seismically-induced core geometry changes. Of these communication paths, the balance of plant can influence the core only through coolant inlet temperature. These three all-encompassing paths by which external changes can influence the reactor are embodied in the three generic anticipated transients without scram (loss of flow without scram, loss of heat sink without scram, and rod runout transient overpower), plus two overcooling accidents (pump overspeed and chilled core inlet temperature).

Given the limited ways the core can be influenced by external events, a quasi-static reactivity balance can be written:

$$0 = \Delta\rho = (P-1)A + (P/F-1)B + \delta T_{in}C + \Delta\rho_{ext} \qquad (1)$$

where

P, F = normalized power and flow, respectively,

δT_{in} = change from normal coolant inlet temperature,

$\Delta\rho_{ext}$ = externally imposed reactivity,

A, B, C = integral reactivity parameters that are measurable on the operating plant via perturbations introduced through the communication paths,

C = inlet temperature coefficient of reactivity (¢/°C),

$(A+B)$ = reactivity decrement in cents experienced in going to full power and flow from zero-power isothermal at coolant inlet temperature,

B = power/flow coefficient (¢), and

A = net reactivity decrement (¢).

In transients which are slow enough to preclude nonequilibrium stored energy in the fuel pins and delayed neutron nonequilibrium, Equation (1) can be solved for the new power level after inherent adjustment of the reactor core to a new set of externally-controlled conditions of coolant flow, inlet temperature, and externally induced reactivity. The power adjusts up or down to compensate, through the power coefficient, any reactivity changes caused by external events. In this manner, any significant change in the inherent reactivity feedbacks can be measured.

FFTF Inherent Safety Tests (Reference G.4.2-2) performed during FFTF Cycle 8A, consisted of steady-state reactivity feedback measurements at various core power, flow, and temperature states. The state changes were chosen to isolate feedbacks into fuel, structural, and coolant temperature groups. Thus, the methodology proposed by ANL was not directly demonstrated. However, the test program did demonstrate the overall feasibility of such a program to quantify core reactivity feedbacks.

It is planned to demonstrate the effectiveness of the ANL (or similar) technique during the ALMR prototype safety tests and the subsequent long-term power operation. The technique can then be used periodically in commercial ALMRs.

Cold Shutdown Method

The ultimate shutdown system (USS) has been added to the reference ALMR design to provide a means of bringing the reactor to cold subcritical conditions following ATWS events. During an anticipated transient without scram event (ULOF, ULOHS, or UTOP), strong inherent negative reactivity feedback with rising temperature brings the reactor to a stable equilibrium state at a core outlet sodium temperature that is below ASME Code long-term structural design limit for faulted conditions, at a fuel temperature below eutectic formation, and at a local coolant temperature well below sodium boiling. Final shutdown can be achieved with the USS by releasing the neutron absorbing balls containing boron-10 from the container at the closure head, which fall by gravity into an open assembly in the center of the core. Substantial time is available for this action. The shutdown action itself can be completed in a few seconds once initiated. Based on results of metal fuel tests by ANL, it is expected that no major failures of the fuel pin cladding will occur during these events.

G.4.2.4 References

G.4.2-1 D. C. Wade and Y. I. Chang, "The Integral Fast Reactor (IFR) Concept: Physics of Operation and Safety, "Proceedings of the International Topical Meeting on Advances in Reactor Physics and Computation, Paris," France, April 1987.

G.4.2-2 B. J. Knutson and R. A. Harris, "FFTF Inherent Safety Tests: Results of Cycle 8A Steady-State Reactivity Feedback Measurements," WHC-EP-0177, July 1989.

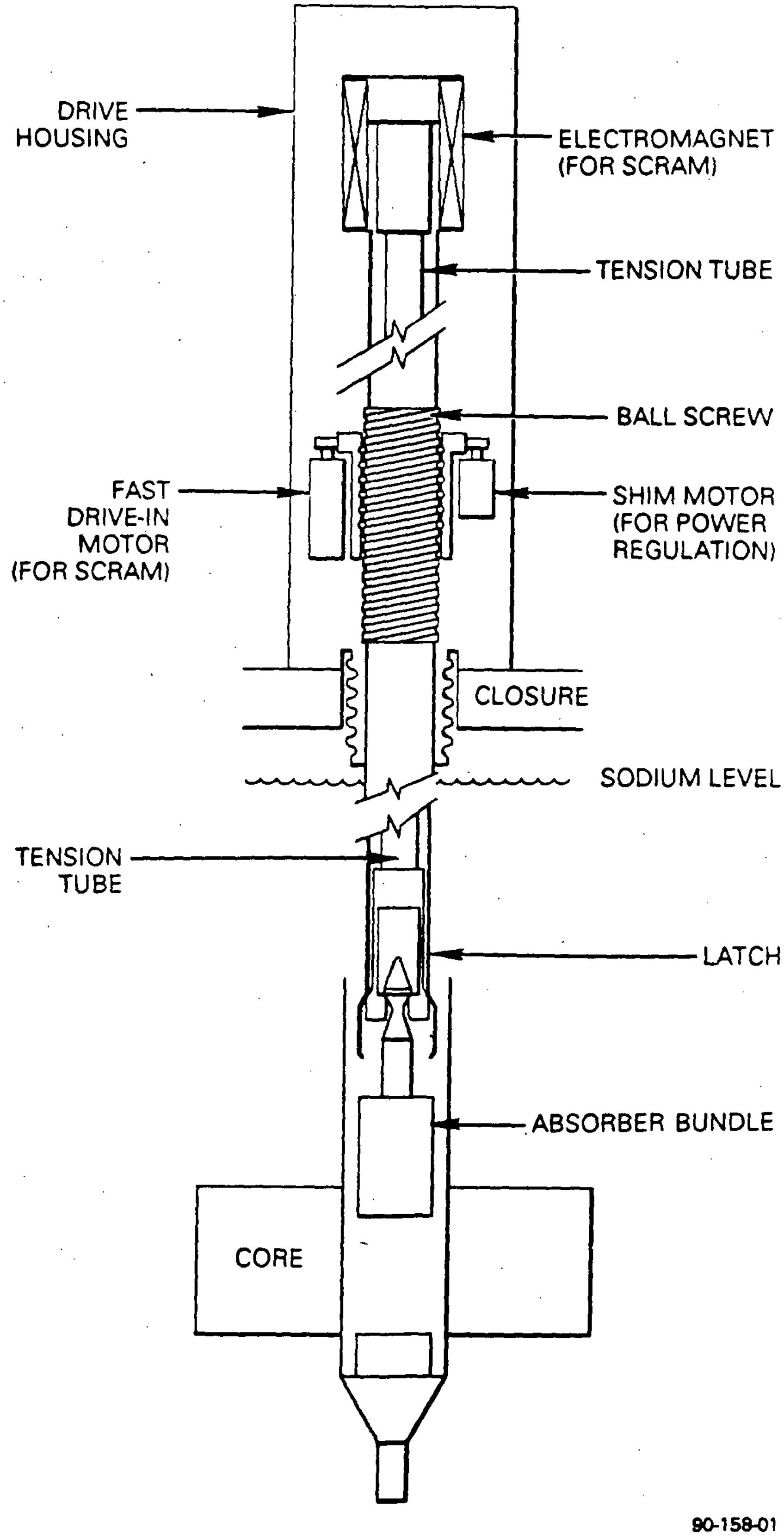


Figure G.4.2-1 CONTROL ROD SCRAM SYSTEM

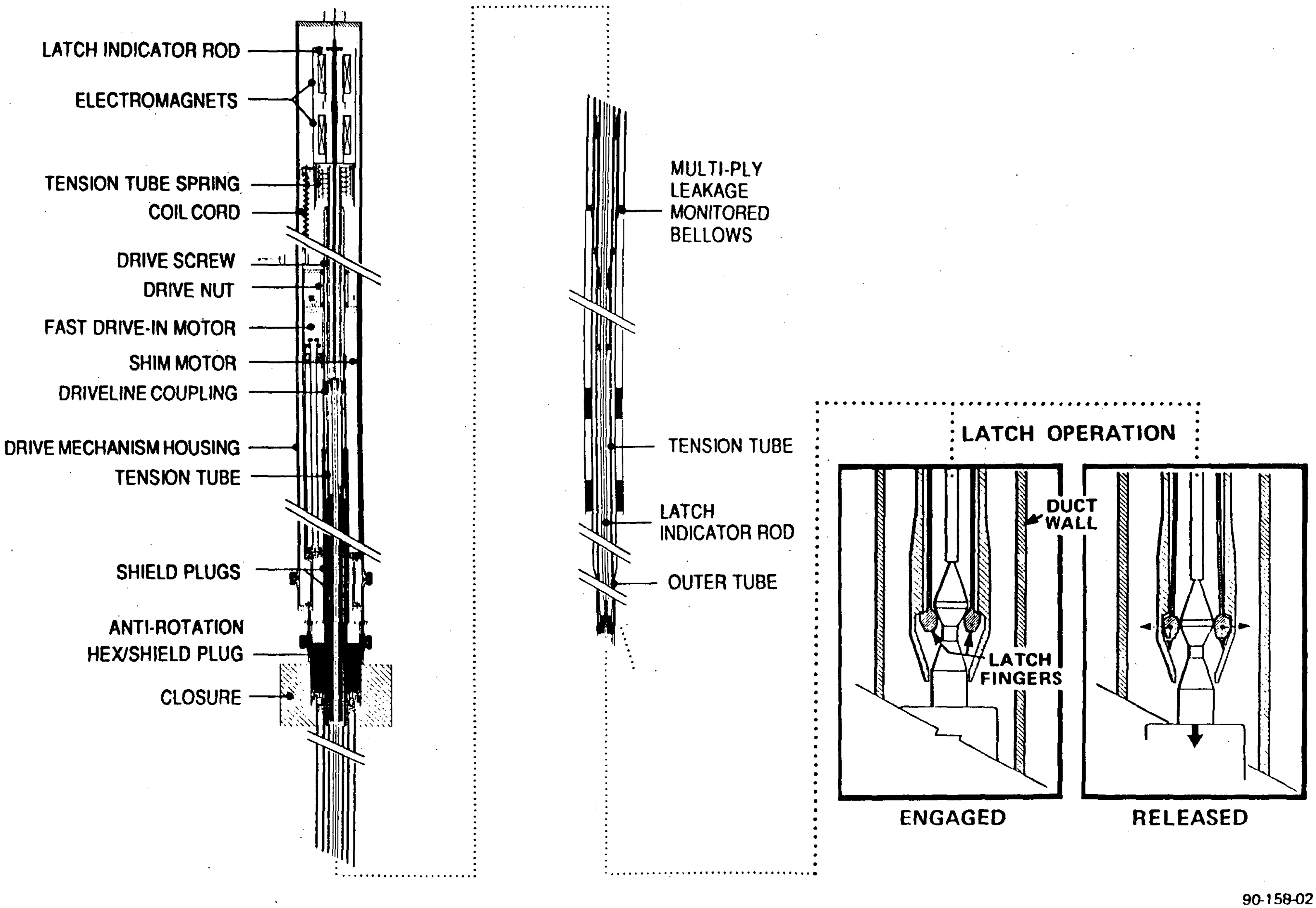


Figure G.4.2-2 CONTROL ROD DRIVE MECHANISM AND DRIVE LINE

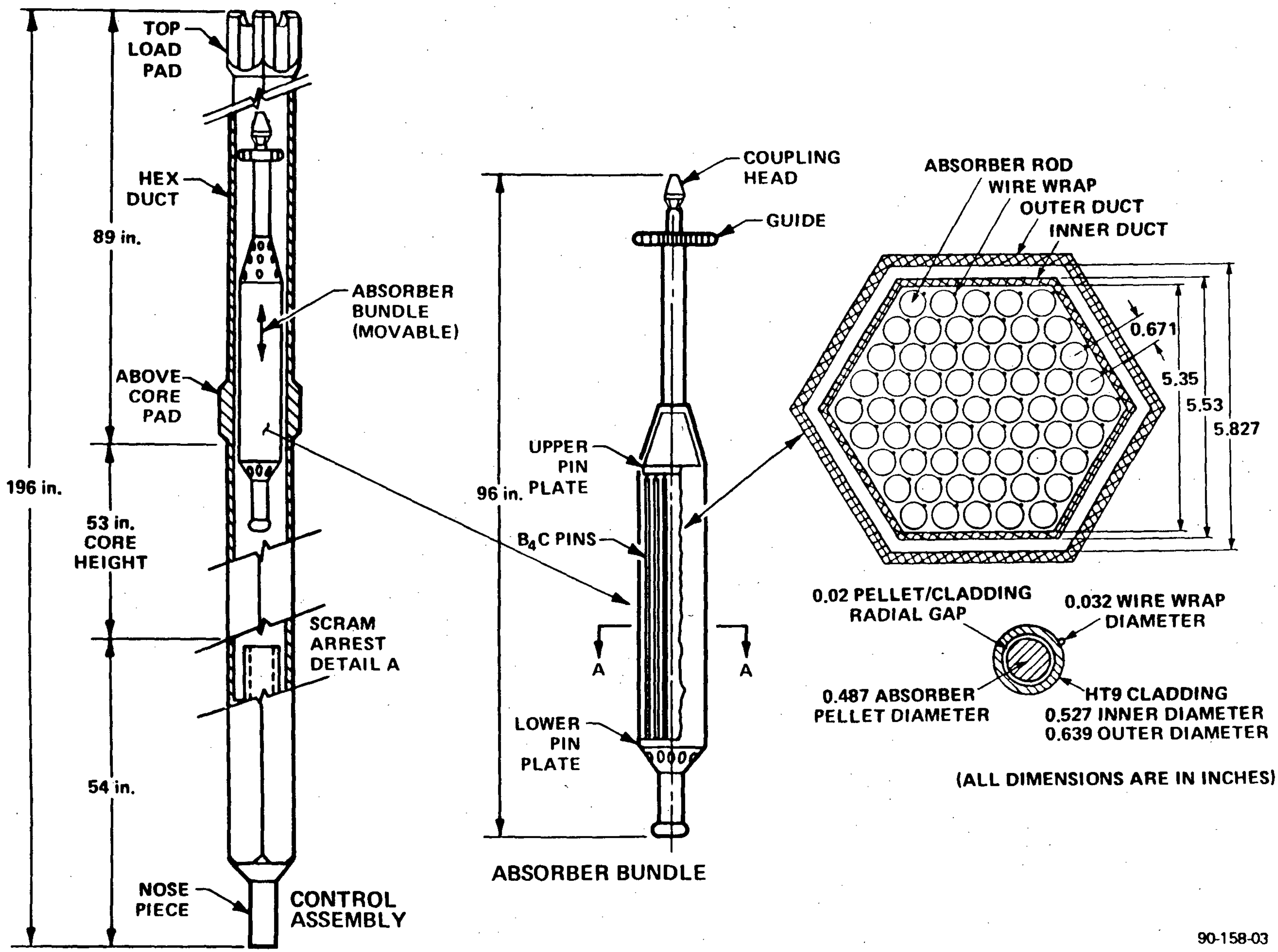


Figure G.4.2-3 CONTROL ASSEMBLY

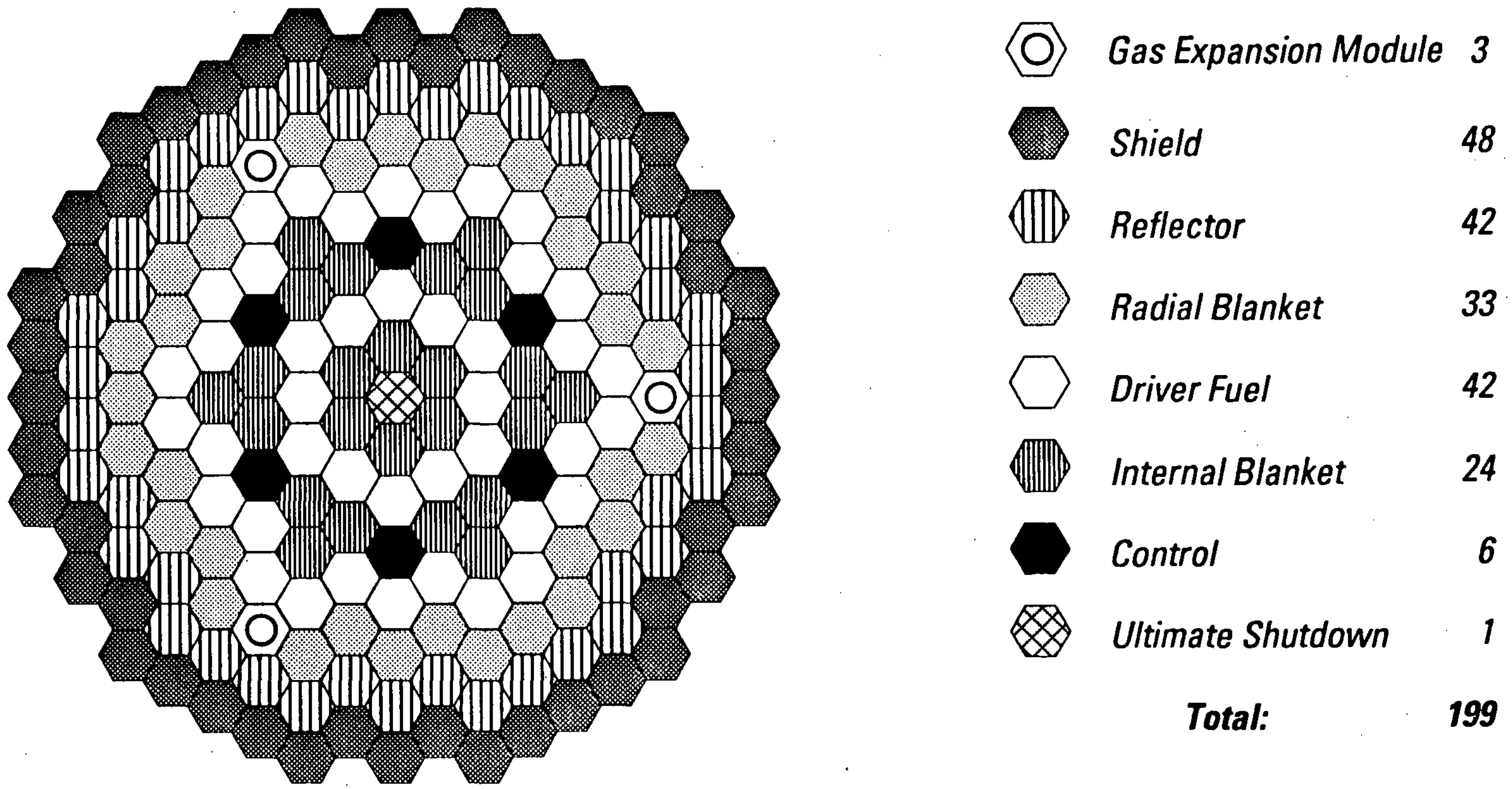


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Figure G.4.2–4 REFERENCE METAL CORE

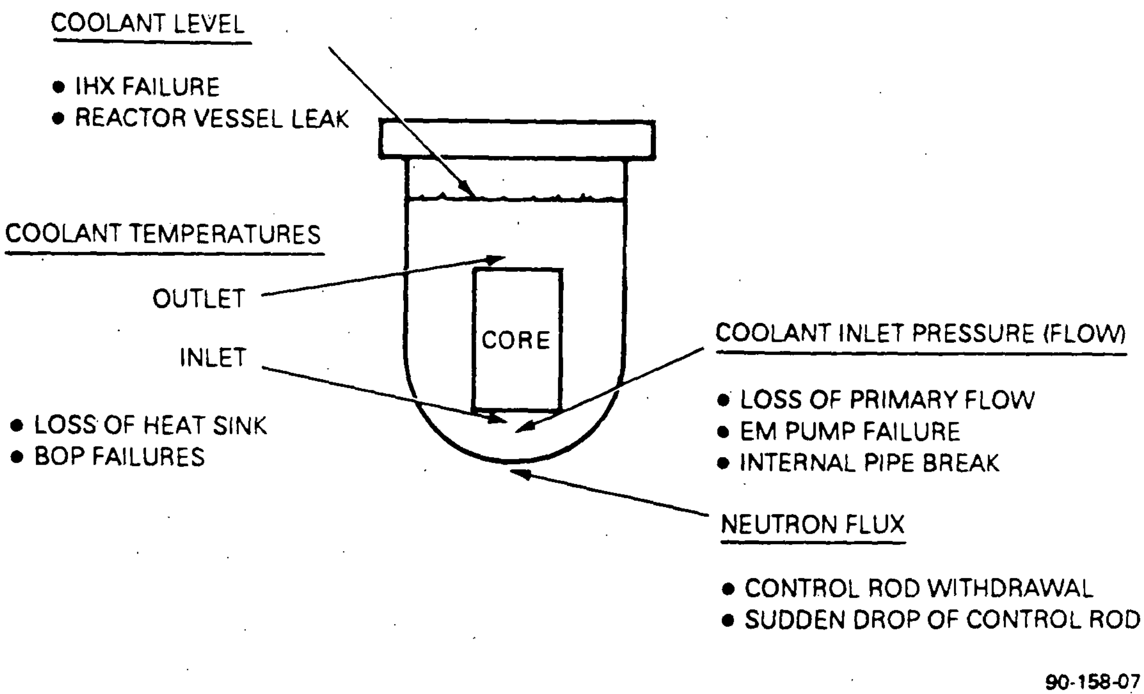


Figure G.4.2-5 RPS TRIP PARAMETERS

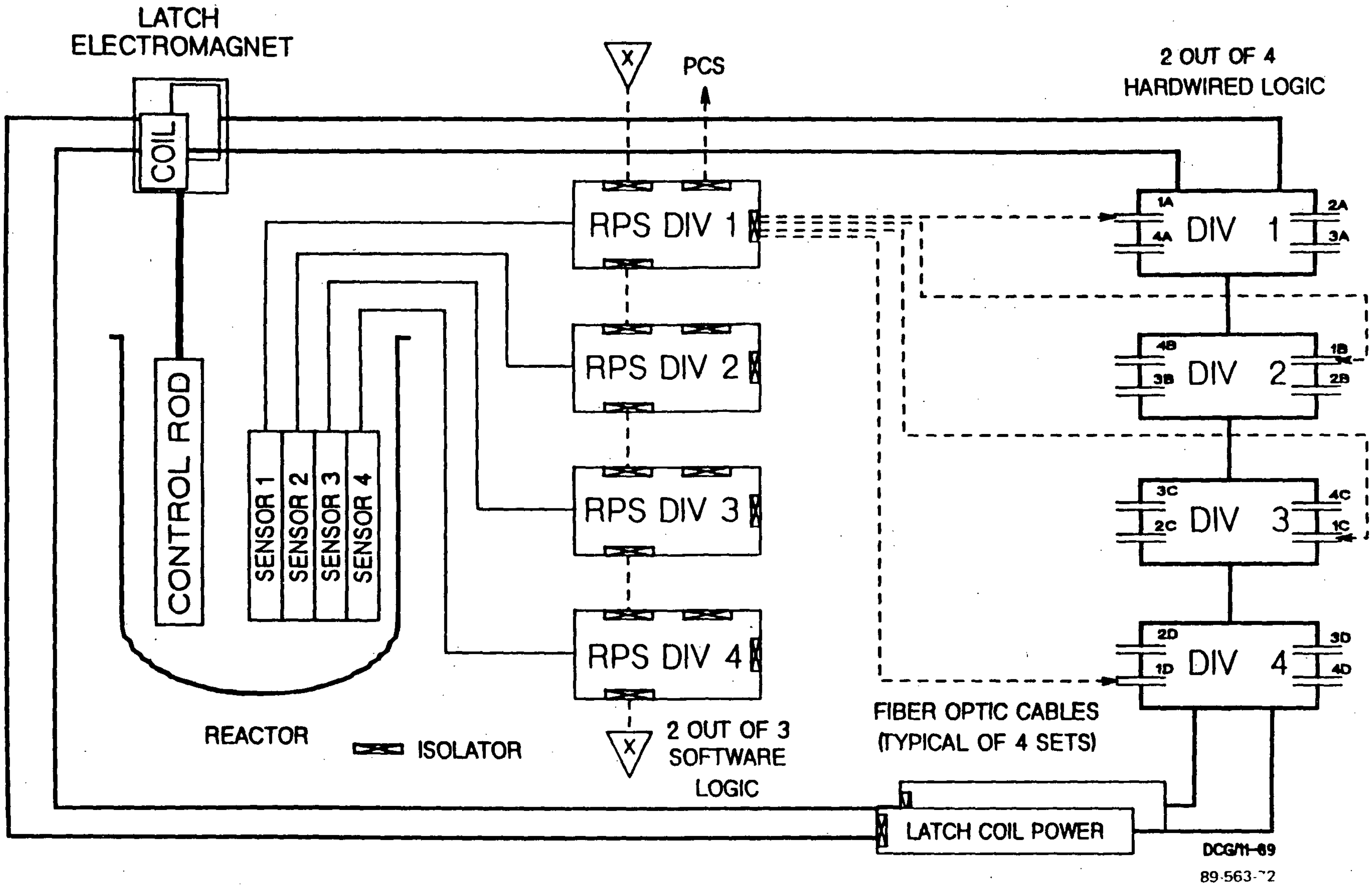


Figure G.4.2–6 RPS CONFIGURATION

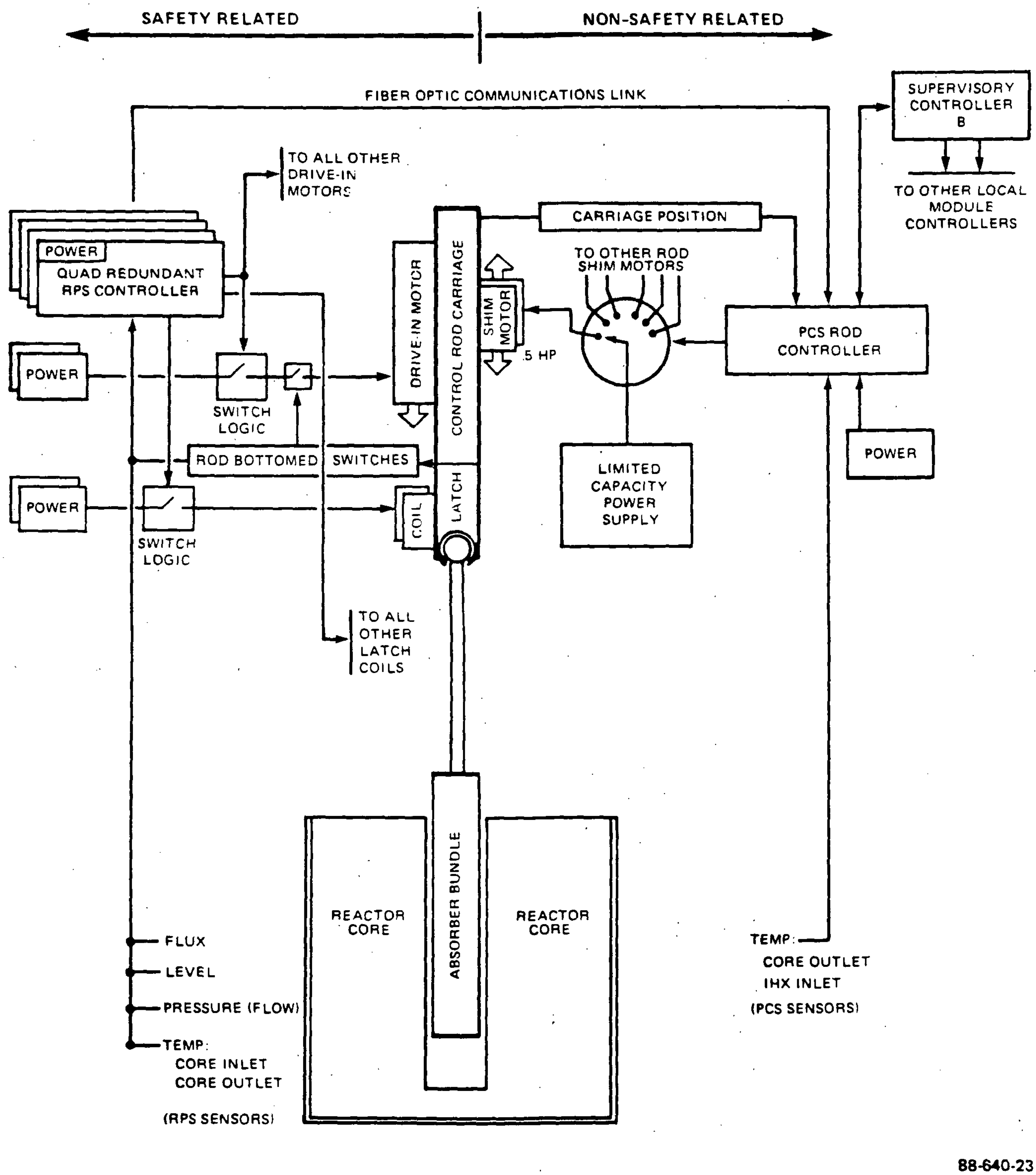


Figure G.4.2-7 ALMR ROD CONTROL

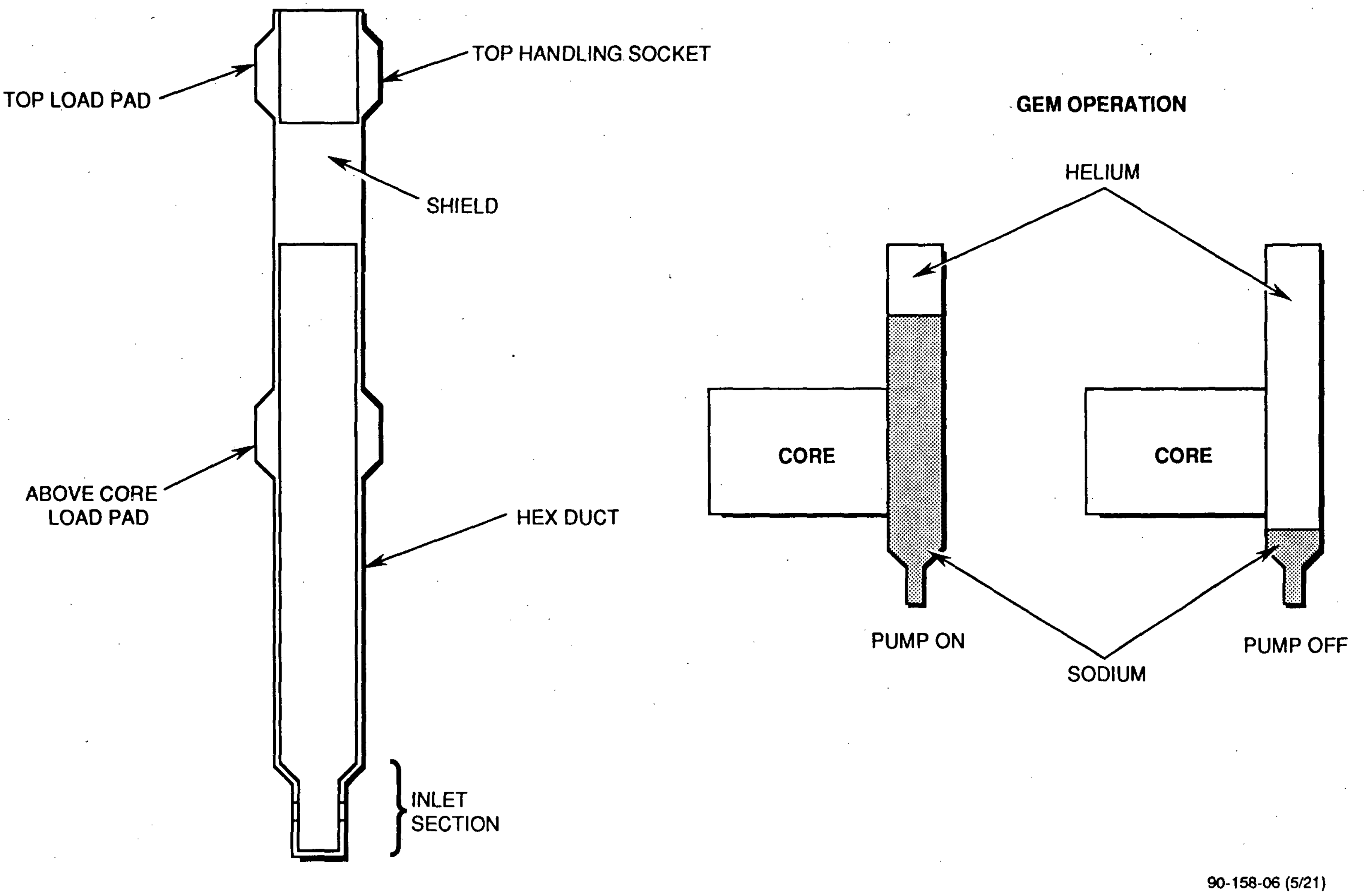


Figure G.4.2–8 GAS EXPANSION MODULE (GEM)

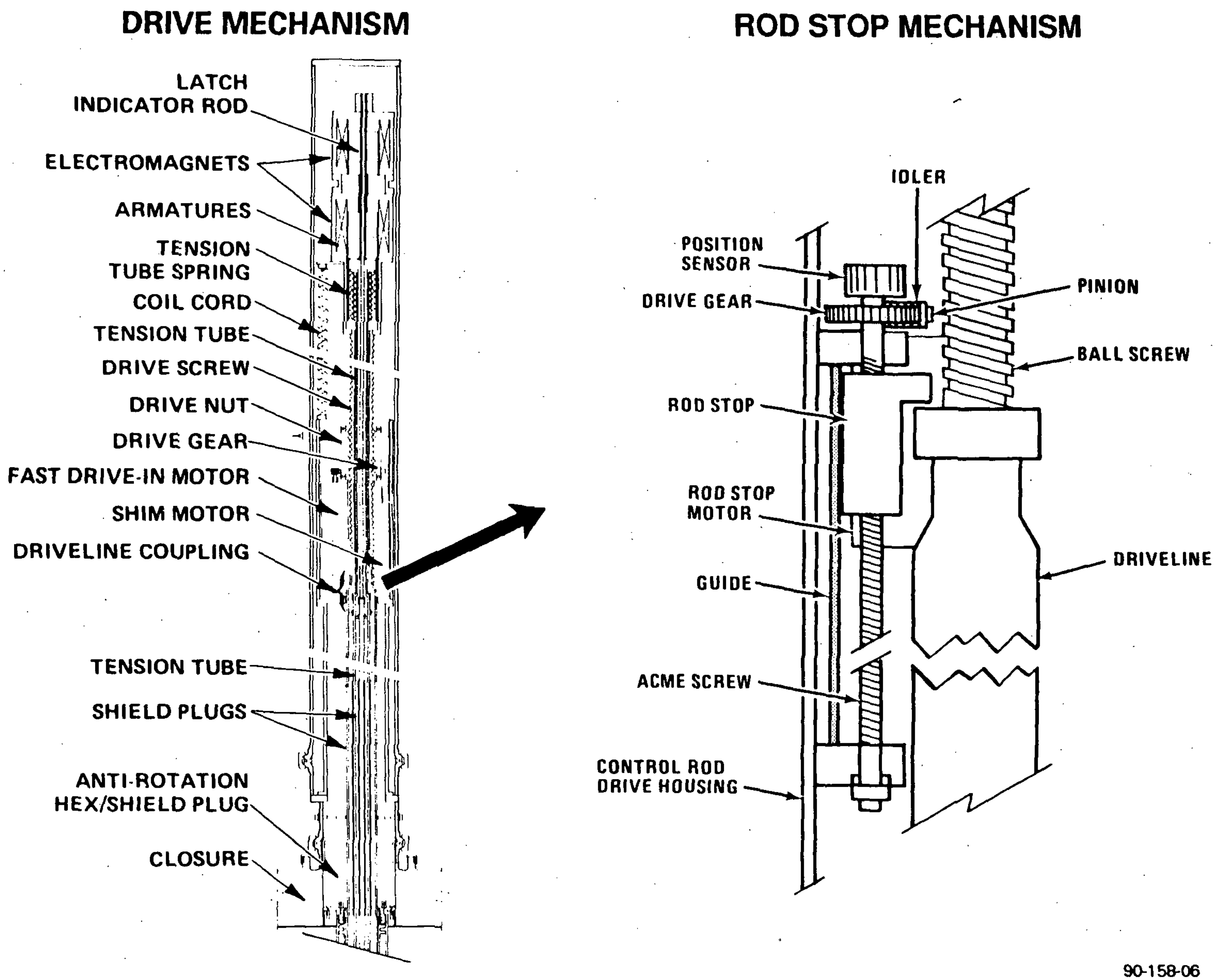


Figure G.4.2–9 ROD STOP MECHANISM

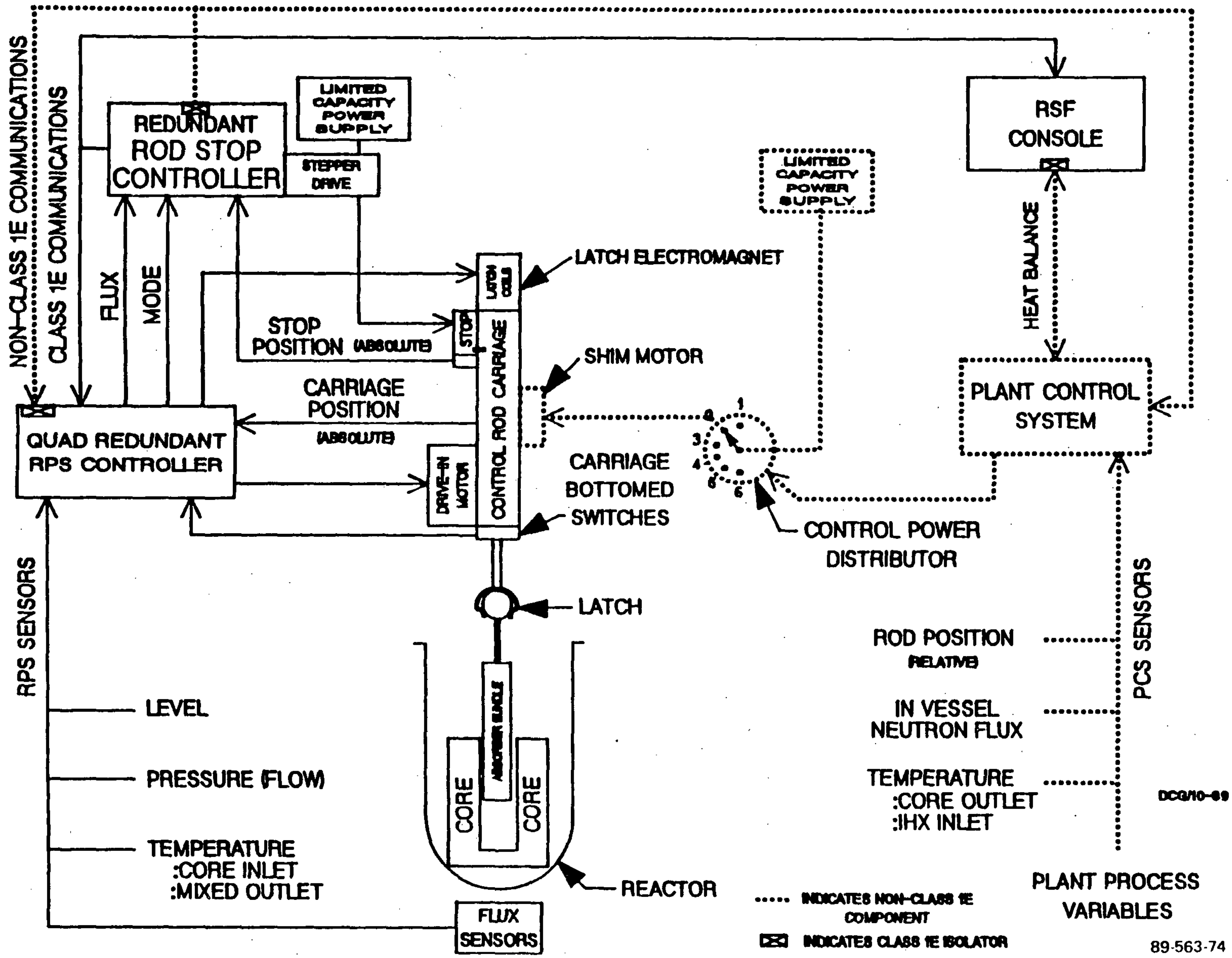


Figure G.4.2–10 ROD STOP SYSTEM CONTROL

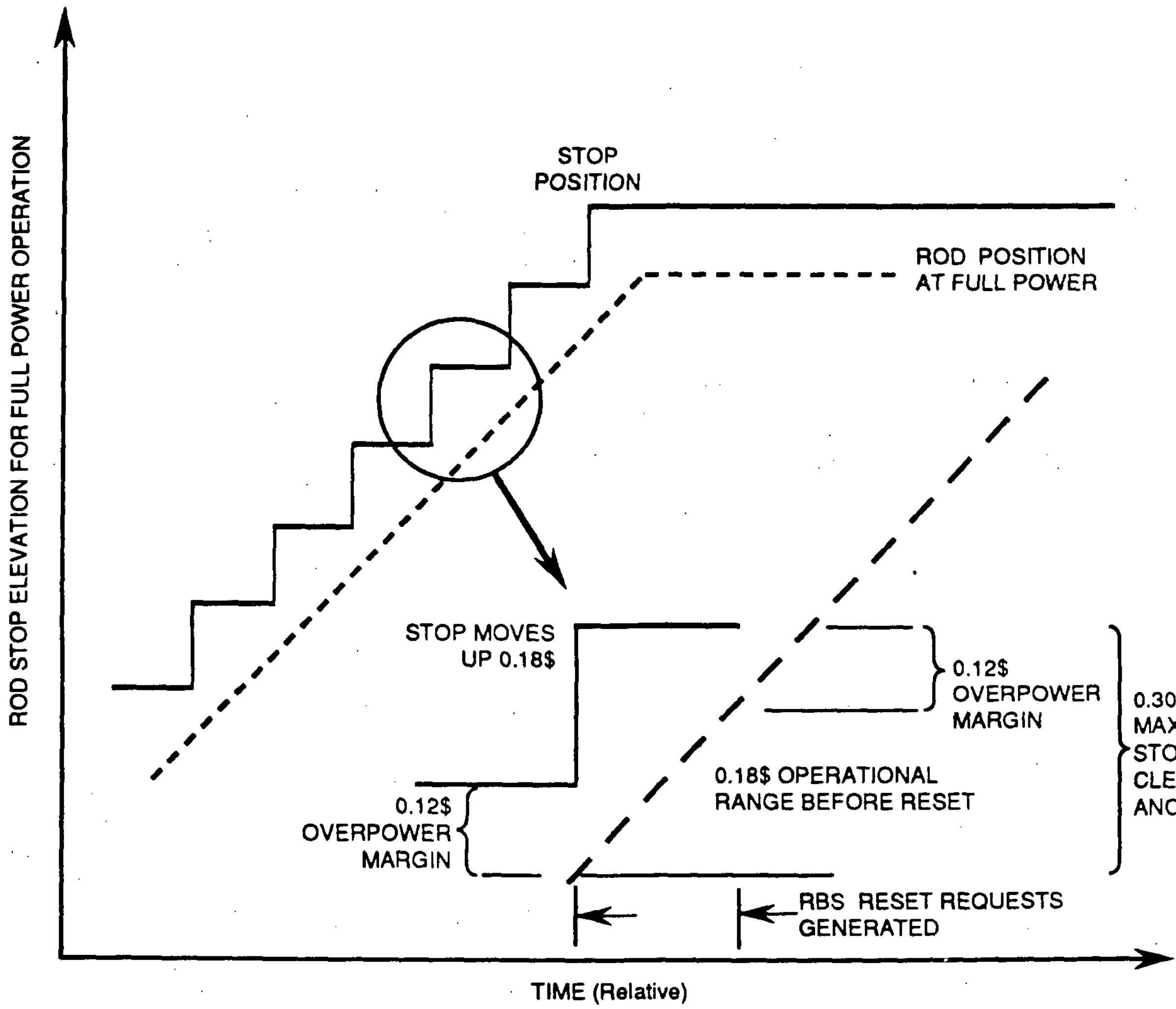
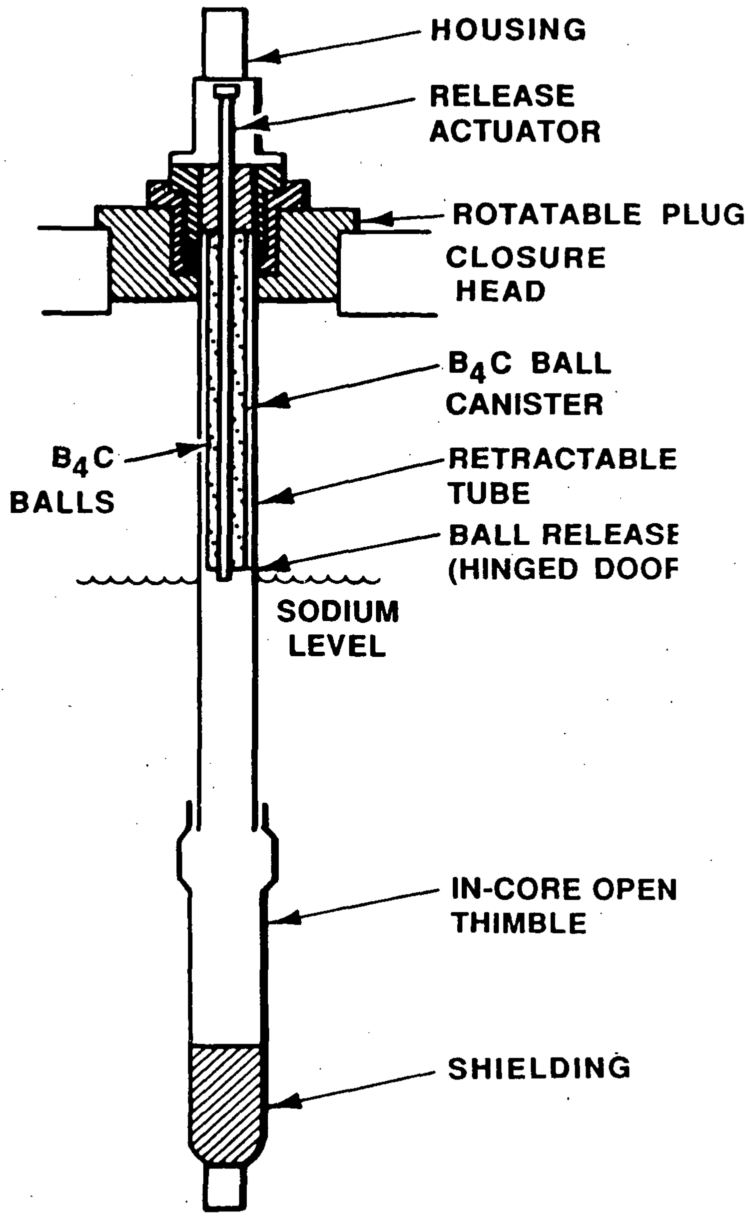


Figure G.4.2-11 ROD STOP SYSTEM NORMAL OPERATION



CHARACTERISTICS

SHUTDOWN:	METAL & OXIDE CORES
DIVERSITY:	FULLY
TESTABILITY:	TESTABLE
RESETABILITY:	OFF-LINE
ACTUATION:	MANUAL
	— CONTROL ROOM
	— REMOTE SHUTDOWN FACILITY (1E)

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Figure G.4.2–12 ULTIMATE SHUTDOWN ASSEMBLY

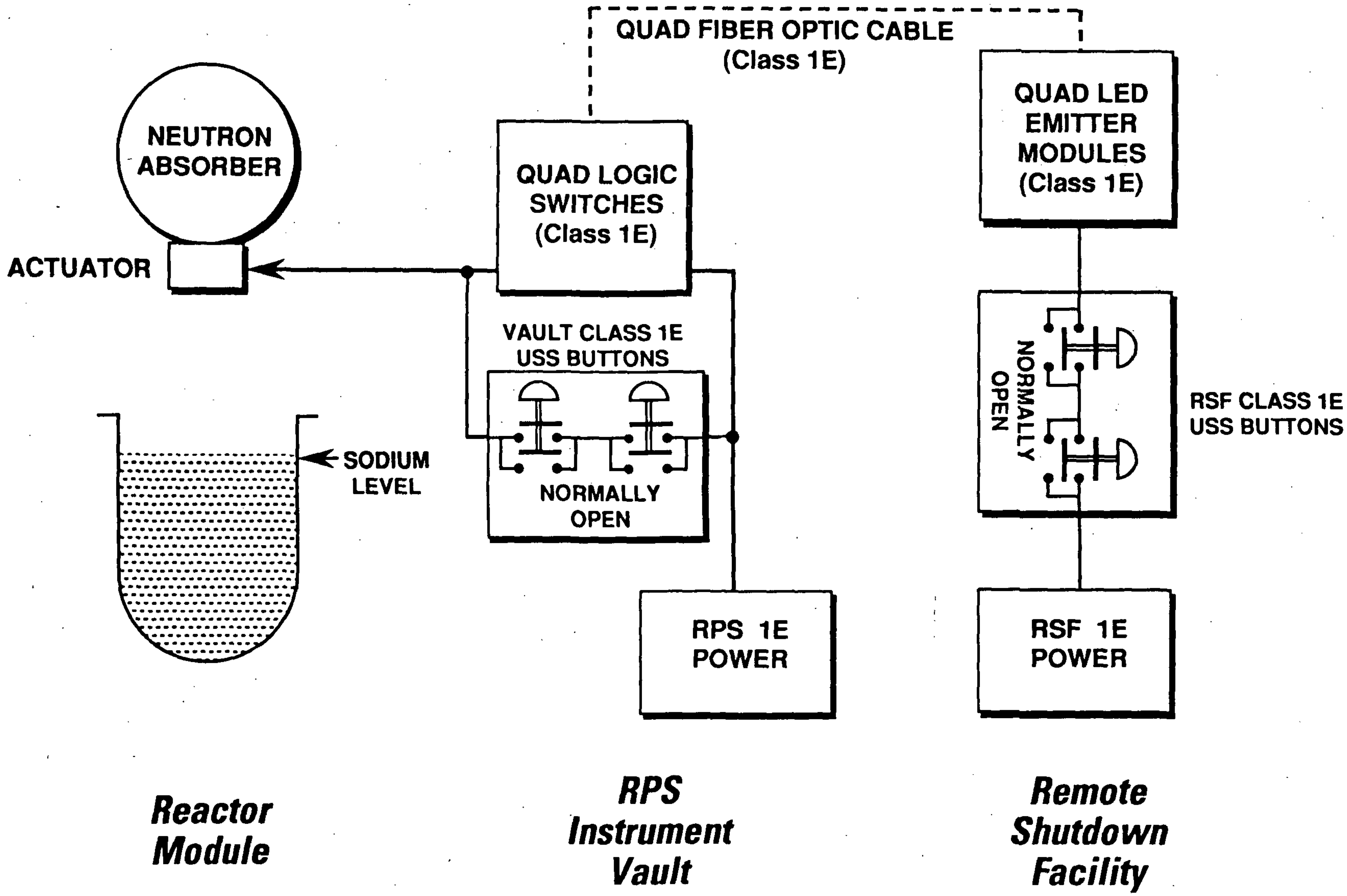

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Figure G.4.2–13 ULTIMATE SHUTDOWN SYSTEM ACTUATION

G.4.3 60-Year Plant Life

G.4.3.1 SER Position on Plant Life

Section 3.3.4 of the SER identifies the 60 year plant life as both an institutional issue and a technical issue. NRC is limited by existing legislation to licensing plants for 40 years. Sufficient information will have to be provided on materials degradation and aging to support designing a plant for a 60 year lifetime. Approval of a 60 year lifetime will require a thorough understanding of degradation and aging phenomena associated with the ALMR materials and operating environment. Special surveillance measurements and inspections will also be necessary to manage and track such phenomena. It is the designer's responsibility to identify the components and systems affected and to develop and provide information to support the 60 year lifetime request.

G.4.3.2 Reference Design Features and Approach For 60-Year Plant Life

The 60-year plant life design goal is consistent with other advanced reactors. Requirements for advanced light water reactors specify a 60-year minimum design life (Reference G.4.3-1). Advanced light water reactors are being designed with the expectation of obtaining a 60-year operating license. DOE has also specified a 60-year design life for the development of the advanced conceptual design for the ALMR. Since the advanced light water reactors are expected to be licensed before the ALMR, it is assumed that the institutional issue of 60-year licensing will be resolved and accepted for application to the ALMR.

The ALMR is designed to operate for 60 years without the need for a major, extended, refurbishment outage. The design philosophy adopted to meet the 60-year design requirement is summarized as follows:

a. Long term, life limiting problems will be eliminated by design. For example, materials will be selected consistent with the operating environment and features will be added as necessary to protect structures and components from damaging conditions.

b. Major components will be designed for the 60 year life and will also be designed for replacement or refurbishment as appropriate. Component replacement will be planned to be accomplished within the plant availability requirement.

c. The plant arrangement is designed to facilitate replacement of equipment, such as instrumentation and control components, which are likely to become obsolete or require replacement over the plant life.

d. Material performance is monitored through in-service inspection and periodic testing of material surveillance coupons to ascertain any degradation in material performance.

Specific ALMR design features which support a 60-year design life are:

a. Core shield and reflector assemblies are replaceable during normal refueling outages.

b. Replaceable shielding installed in the reactor core limits neutron exposure to the support cylinder, reactor structures, reactor vessel, and the containment vessel.

c. Permanent shielding installed in the reactor limits activation of intermediate sodium and protects the reactor structures and the reactor and containment vessels from neutron damage.

d. The number of reactor scram transients is reduced by use of a fast runback feature.

e. Plant startup and cooldown transients are controlled by plant control system.

f. Low primary system pressure (atmospheric pressure at full power).

g. Low operating temperatures limit thermal aging and creep effects (905°F core outlet temperature at full power).

h. Major components, such as EM pump, IHX, control rod drives, IVTM, are designed to be replaceable.

i. Upper internal structure can be replaced.

j. Instrumentation is installed in drywells for ease of replacement.

l. No penetrations in the reactor and containment vessels.

m. In-service inspections, material surveillance coupons, material condition monitoring, and monitoring of plant parameters provide information to support life extension.

n. Equipment is located to minimize exposure to adverse and life limiting conditions.

o. Material performance is monitored with surveillance coupons, in-service inspection and replacement capabilities, and periodic testing of seismic isolator bearings.

The design life of each removable reactor component, other than core assemblies, is listed in Table G.4.3-1.

Table G.4.3-1

DESIGN LIFE OF REMOVABLE REACTOR COMPONENTS

Component	Design Life
Intermediate Heat Exchanger (IHX)	60 years
In-Vessel Transfer Machine (IVTM)	30 years
Primary EM Pump	30 years
Control Rod Drive Mechanism (CRDM)	60 years
Control Rod Drive Line (CRDL)	20 years

G.4.3.3 Rationale Supporting 60-Year Design Life

The principal structural effects of extending the plant life to 60 years are: 1) the material degradation due to neutron exposure, 2) the degradation from exposure to flowing sodium, and 3) the degradation due to thermal aging. These effects are addressed in the design of the ALMR permanent components by: 1) maintaining large design margins to allow for uncertainties in extrapolating the design methods validated for a 40-year design life to the 60-year plant design, 2) minimizing the parameters conducive to material degradation, 3) a focused materials R&D program, and 4) monitoring the changes in the material behavior through in-service inspections and surveillance coupons.

Reactor Module

Exposure to neutron irradiation decreases the material ductility and fracture resistance. These effects are limited in the design by shielding the reactor structures from the core so that the materials retain residual total elongations (RTE) in excess of 10% for the load bearing structures and 5% for the non-load bearing structures. Table G.4.3-2 compares the 60-year neutron exposures estimated for the near-core load-bearing reactor structures with the material irradiation limits to ensure 10% RTE. The estimated neutron damage and damage limits in the table are in terms of displacements per atom (dpa) which account for the differences in the neutron energy levels in the data base and in the local energy spectra. As indicated in Table G.4.3-2, the calculated neutron damage is increased by 50% to 100% to allow for uncertainties in the neutron flux and energy estimates at different locations.

Table G.4.3-2

ALMR COMPONENT NEUTRON DISPLACEMENT DAMAGE ESTIMATES

Component	Material	Uncertainty In Damage Estimate, %	Calculated dpa Damage Including Uncertainty, dpa	Design Limit, dpa
Reactor Vessel	SS 316	50	0.00015	4.1
Support Cylinder	SS 304	50	0.77	2.4
Upper Grid Plate	SS 304	50	2.3	2.4
Inlet Plenum Plate Welds	SS 308	50	0.35	1.3
Support Cylinder Girth Weld	SS 308	100	0.01	1.3
Core Former Ring	HT-9	100	0.01	1.4

The effects of sodium exposure are surface removal by erosion and corrosion, which decrease the load-bearing thicknesses of the structures, and bulk effects which may decrease the material strength to greater depths by changing the alloy constitution. For the ALMR structures, these effects are estimated to produce insignificant loss of thickness by erosion or corrosion, carbon loss less than 0.02 inches of the surface layer, and insignificant reduction in the material creep rupture strength over the 60-year plant design life (Reference G.4.3-2).

The 10% residual total elongation limit was selected to ensure validity of the shakedown and limit load concepts used in developing the ASME Code primary and secondary stress limits, and to envelop the strain limits specified in the ASME high temperature Code Case (Code Case N-47). Thus, the RTE limit, together with the insignificant sodium effects listed above, permit use of the ASME Code criteria in the conventional manner. Still, to allow for extrapolation uncertainties, the structures are designed for large margins (>100%) against the primary loads, including the seismic loads, and insignificant creep-fatigue damage (<0.1 compared to the ASME Code limit of 1.0) (Reference G.4.3-2).

In addition to providing large design margins to account for uncertainties in extending the design correlations to 60 years, the uncertainties are reduced by minimizing the operating temperatures. The containment vessel, reactor vessel, closure, and the cold pool structures, including the core support, are maintained within the ASME Code low temperature design limits for Class I structures (700°F for 2-1/4 Cr-1 Mo, and 800°F for stainless steels) during the normal operation and anticipated scram transients. The high temperature exposure of these structures is limited to a few off-normal events of limited duration (<1000 hours, <0.05 creep damage). This exposure is irrelevant to 40-year and 60-year design lives governed by the low-temperature, time-independent rules of the ASME Code which are relatively insensitive to the aging effects.

The hot pool structures are exposed to the 905°F core-exit coolant which exceeds the ASME low temperature limit of 800°F. However, the stresses in these structures are maintained at levels where the design is again governed by the time-independent rather than time-dependent stress limits of the code as shown in Figure G.4.3-1. The figure shows the time independent stress limits (S_m) and the stress limits (S_t) for 500,000 hours (57 years) with the latter limits obtained by extrapolating the ASME Code data. The time-independent limits govern the design up to 900°F for the SS304 structures assuming the structures are designed to the code limit without any design margins. However, the actual stresses in the outlet plenum components are less than 5,000 psi (Table G.4.3-3) compared to the ~14,600 psi SS304 limit in Figure G.4.3-1 permitting considerably higher temperatures. This is substantiated by the low creep damage estimates shown in Table G.4.3-3.

Table G.4.3-3

OUTLET PLENUM COMPONENT OPERATING STRESSES AND CREEP DAMAGE (NORMAL OPERATION)

Reactor Structure	Material	Uncertainty In Damage Stress, psi	ASME Code Life Limit, hours(1)	Creep Damage, Life Fraction(2)
Support Cylinder	SS 304	1210	50,000,000	0.01
Reactor Liner	SS 304	2430	50,000,000	0.01
Baffle Plates	SS 304	4370	50,000,000	0.01
Upper Internals	SS 316	1350	50,000,000	0.01

(1) ASME Code life limit based on extrapolation for 905°F data.
(2) ASME Code limit = 1.0

The low operating temperatures also decrease the effects of sodium corrosion and interstitial transfer as well as sensitivity of the materials to these effects as shown in Figures G.4.3-2 through G.4.3-4. According to these figures, the effects of sodium on the structural thickness and material strength would be insignificant at temperatures below 905 °F.

Concrete Reactor Cavity

A cylindrical concrete reactor cavity surrounding each reactor module has an inside diameter of about 25.3 feet and extends from approximately 56 feet to 92 feet below grade. The concrete wall is about 2.5 feet thick and the bottom slab is about 5 feet thick. Influences which could cause degradation of the below grade concrete structure during its 60 year life include water ingress and corrosion of reinforcing steel and thermal effects (temperature and thermal gradients).

Small cracks in the concrete are acceptable because an oxide film is initially formed which protects the reinforcing steel (Reference G.4.3-3). But to prevent reinforcing steel damage, cracking needs to be controlled and a waterproofing system provided in areas of high groundwater. Contrary to the design of embedded missile silos, the ALMR reactor cavity is designed not to leak. At sites where ground water exists, bentonite panels will be installed in the excavated hole on the outside surfaces of the below grade concrete structures to provide continuous waterproofing. In areas of high groundwater, additional waterproofing membranes, such as high density polyethylene (HDPE), will be installed to provide a positive barrier which can bridge cracks, take the high hydrostatic pressure, and last the 60 year life of the plant. Waterstops are installed at all concrete construction joints below the groundwater table for additional protection against leakage. As a further precaution, a sump pump is provided to remove any water that might collect at the bottom of the cavity.

American Concrete Institute's ACI 349-85, "Code Requirements for Nuclear Safety Related Concrete Structures", governs the design of the ALMR safety related concrete structures. This code specifies load combinations which include both normal and accident temperature effects used in determining the required strength of the concrete structures. Appendix A of ACI 349-85 specifies a temperature limit of 150°F for normal operation or any other long term period, with temperatures not exceeding 350°F for accident or any other short term (generally less than 24 hours) period. Higher temperatures are permitted locally, such as areas around penetrations. At these temperatures, testing of the concrete is not required to evaluate reduction in strength due to thermal effects. ASME Code Section III, Division 2, Concrete Reactor Vessels and Containments, also specifies the same temperature limits. Under normal operation, 100°F ambient air will flow between the reactor cavity and the collector cylinder at 35 lbm/sec, which will maintain the concrete at less than 150°F. During RVACS events, the increased air flowrate (and velocity) of about 50 lbm/sec continues to maintain the concrete below 150°F. These temperatures are below any concrete limits for long term, continued operation.

As the reactor cavity is heated on its inside surface, a thermal gradient is created across the concrete wall, and tension results at the outside which is resisted by reinforcing steel. The reinforcing steel can be designed to control cracking to acceptable levels, and the design limits the maximum operating temperatures so excessive cracking is prevented. With exterior loadings of soil and, in most cases, water, the structure will be primarily under compressive loads which tend to further reduce cracking.

Other reinforced concrete structures on the plant site are accessible for inspection and repairs, should they be necessary. Loadings and environmental conditions are accounted for as part of the normal design process. NUREG/CR-4652 states that concrete will have infinite durability unless subjected to extreme external influences (overload, elevated temperatures, industrial liquids and gases, etc.) Under normal environmental conditions, aging of concrete does not have a detrimental effect on strength. These ALMR structures are expected to survive the 60-year plant life with substantial margin.

<u>Collector Cylinder</u>

The function of the collector cylinder is to direct air flow to the reactor module for decay heat removal by the RVACS. Incoming air flows downward between the collector cylinder and the reactor silo. At the bottom of the collector cylinder, which is about 3 feet above the silo bottom, the air enters the hot air riser annulus between the collector cylinder and the containment vessel where it is heated providing the natural draft head required to maintain system air flow. The collector cylinder is constructed of 2-1/4Cr-1Mo alloy steel. Its temperature ranges between 237°F and 260°F during normal operation for the bottom and top of the collector cylinder respectively, and 498°F and 716°F during decay heat removal with only the RVACS. The collector cylinder operates at a higher temperature than the air flow thereby preventing formation of condensate on the hot collector cylinder surface and accompanying surface corrosion. The collector cylinder is designed for the 60 year life of the plant. In-service inspections performed for the containment vessel, also 2-1/4Cr-1Mo and exposed to the same atmospheric air environment, and the collector cylinder will provide indications of any life-limiting material degradation. Since there are only dead weight loads on the collector cylinder, surface oxidation from exposure of the hot surface to air will have no impact on its function.

<u>Seismic Isolation Bearings</u>

The seismic bearings are located in a vault below the reactor facility platform. Natural rubber and steel bearing materials are selected for performance and durability for the 60 year plant life. A three-inch protective layer of natural rubber encloses the rubber and steel laminations of the bearing assembly. Protected from ozone and high temperature, natural rubber retains its physical characteristics for many years. Radiation shielding for the isolators is provided by the 2.5-foot thick cylindrical concrete wall. Accumulated radiation dose for the seismic isolators is estimated to be less than $2x10^6$ rads over the life of the plant, a factor of 10 below the level expected to result in the onset of embrittlement. Material surveillance coupons adjacent to the bearings are periodically removed for testing. A significant feature of the reactor facility design is the ability to replace individual bearings if necessary. Also, as noted in Section G.4.4, individual bearings are periodically removed (about every 12 years) for performance testing where any degradation due to aging will be detected.

<u>Instrumentation</u>

Instrument sensors are located within the reactor in drywells. Quad redundant sensors are used for safety grade monitoring functions. Sensor replacement can be performed during normal outages without breaching the reactor coolant boundary.

Summary

The 60-year design life of the structures is based on incorporating sufficiently large design margins to account for uncertainties in the extension from 40 years to 60 years, and sufficiently low temperatures to reduce such uncertainties to a minimum possible level in the absence of a long-life data base. Two additional measures are taken to address any residual uncertainties and possible unexpected failure modes: (1) R&D programs are proposed to provide a basis for extending the ASME Code high temperature design rules to a longer life, taking advantage of the stainless steel data becoming available for longer high temperature exposures, and (2) surveillance coupons are to be used to monitor the effects of operating environment in and around the reactor.

G.4.3.4 References

G.4.3-1 EPRI Advanced Light Water Reactor Utility Requirements Document, Chapter 1: Overall Requirements.

G.4.3-2 GEFR-00832, PRISM Reactor Structural Evaluation, GE Nuclear Energy, September 1988.

G.4.3-3 Bechtel National letter, BNI-089, Durability of Silo Concrete, December 22, 1989.

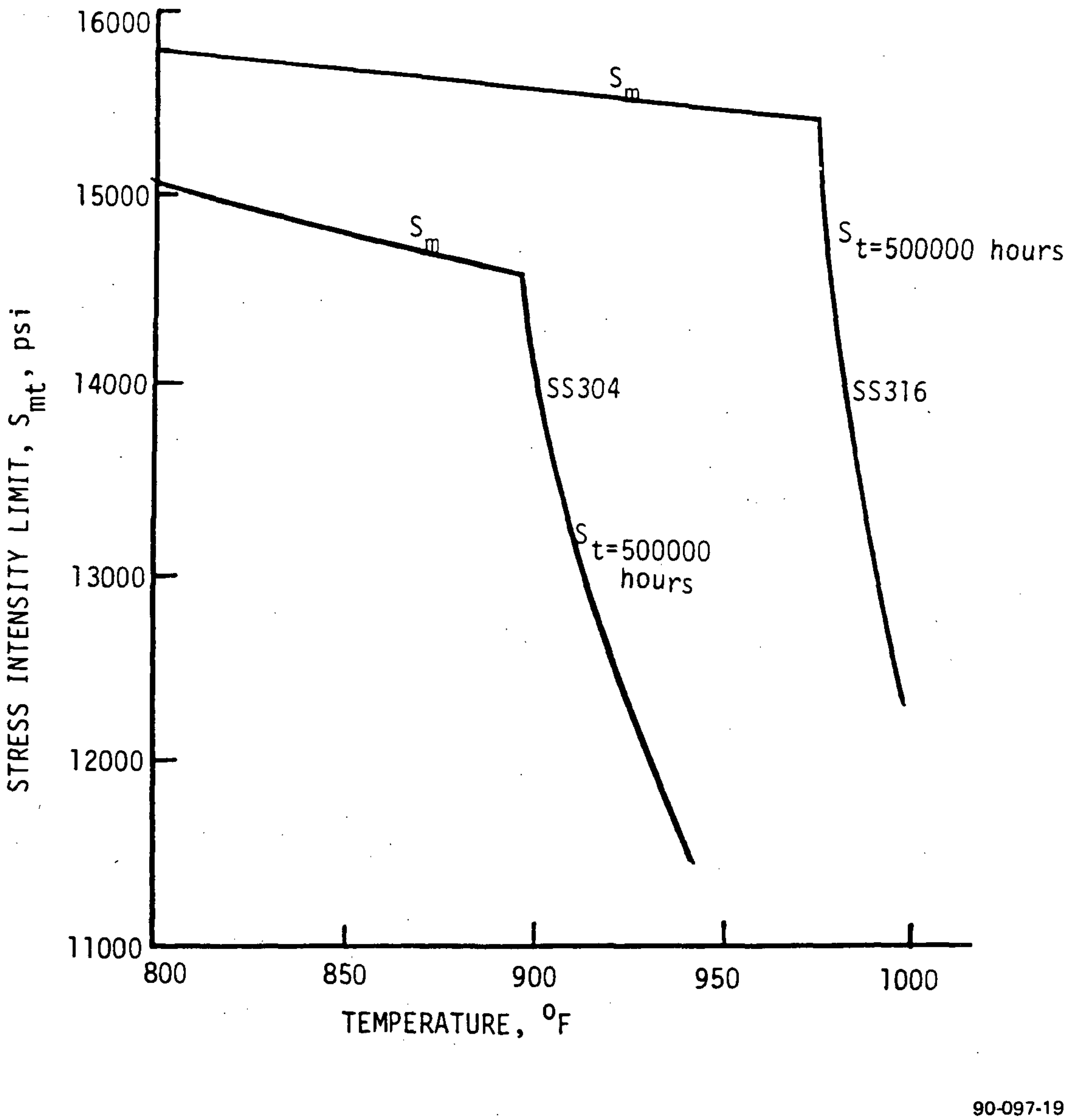

Figure G.4.3–1 ASME CODE ALLOWABLE STRESS INTENSITIES (S_{mt}) FOR 500,000 HOURS OPERATION FOR SS304, SS316

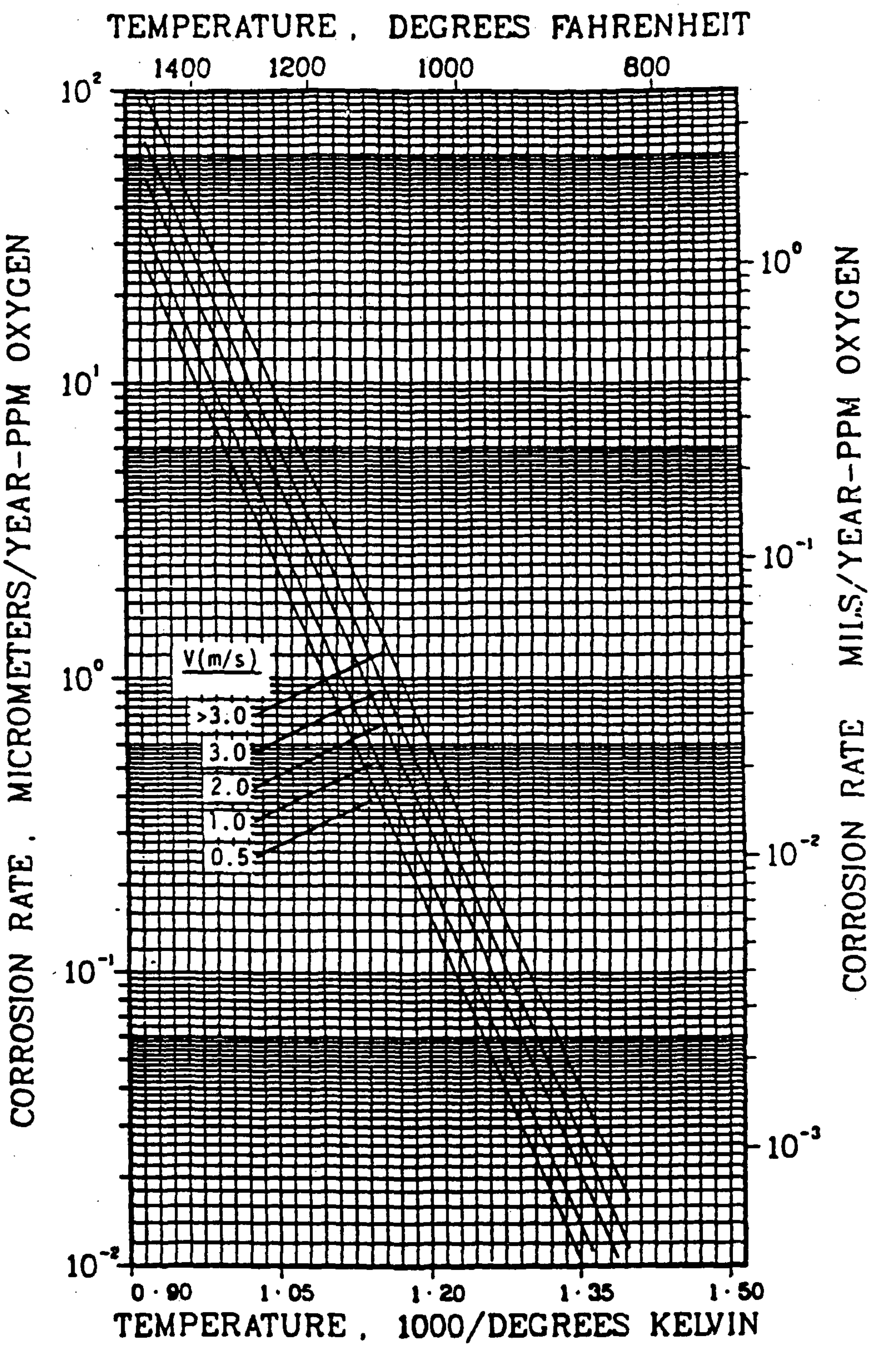

Figure G.4.3–2 TYPES 304 AND 316 STAINLESS STEEL CORROSION RATES IN SODIUM (Nuclear Systems Materials Handbook)

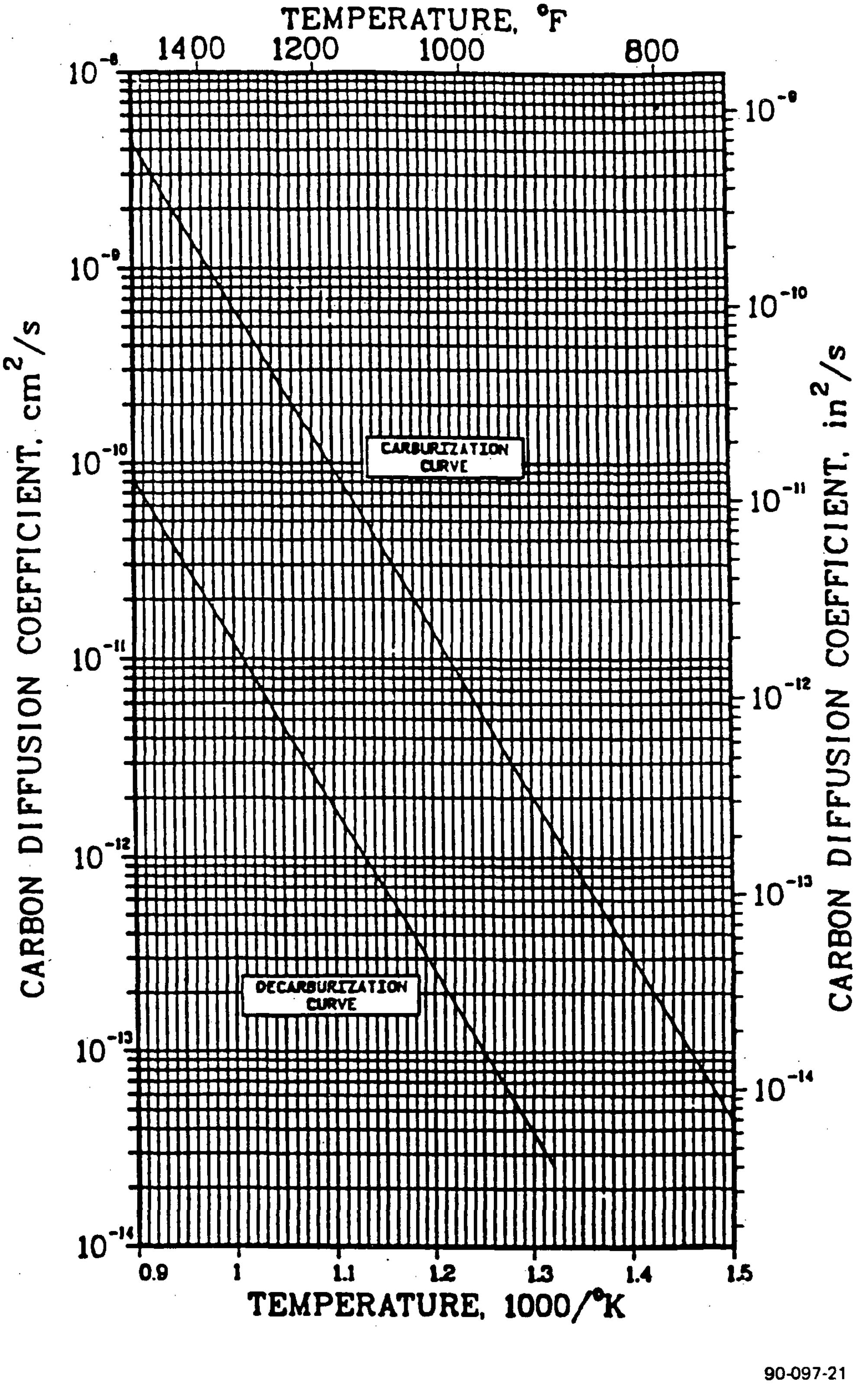

Figure G.4.3–3 TYPE 316 STAINLESS STEEL CARBON DIFFUSION COEFFICIENTS (Nuclear Systems Materials Handbook)

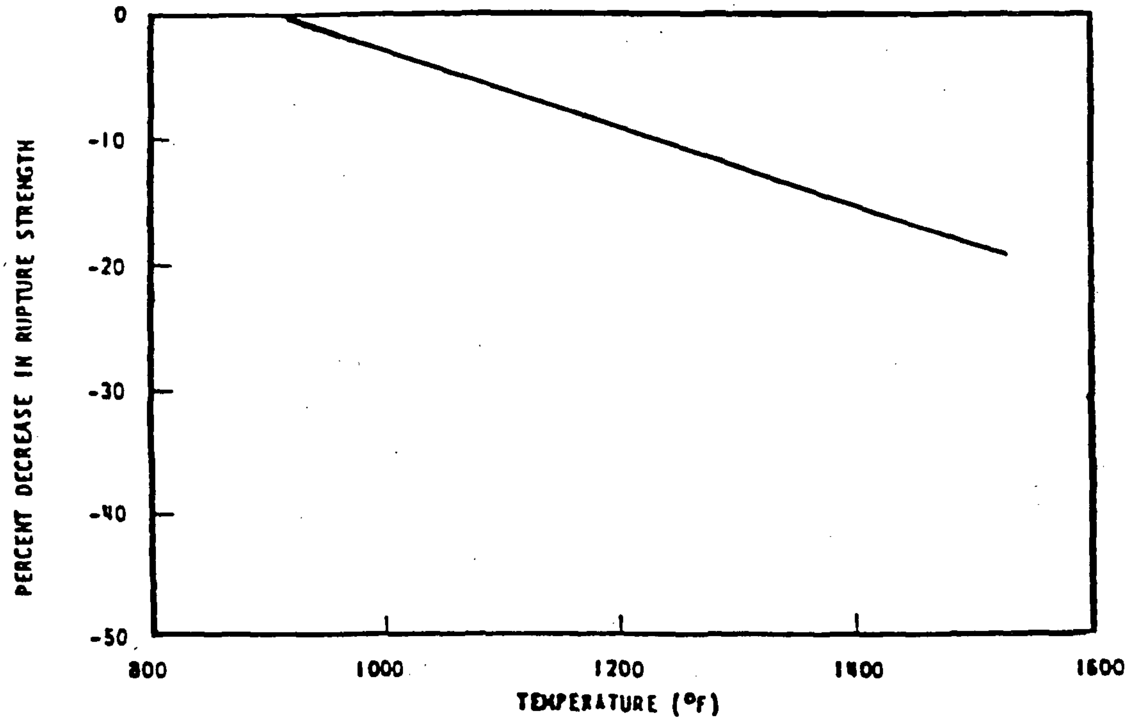

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Figure G.4.3–4 TYPE 316 STAINLESS STEEL RUPTURE STRENGTH REDUCTION IN SODIUM (CRBR UIS Equipment Specification)

G.4.4 Seismic Isolators

G.4.4.1 SER Position on Seismic Isolation Design Approach

The SER states that ALMR has proposed the use of "seismic isolators" for the nuclear island to reduce the magnitude of any horizontal ground acceleration transmitted to the safety grade nuclear island structures, systems, and components by a factor of about one-third. A test program is proposed to qualify and demonstrate the performance of the isolators and to provide sufficient data to support final design. The NRC staff supports continuation of this work and the effort to apply an innovative design feature to the ALMR to provide additional seismic margin; however, further evaluation of the acceptability of the isolator system will be dependent upon additional design information and the results of these tests. (Ref. SER Section 3.3.4)

G.4.4.2 Reference Seismic Isolation Design

The reference ALMR design uses horizontal seismic isolation for the reactor facility, the shutdown heat removal system (RVACS), the reactor shutdown systems, and the EM pump coastdown system. Seismic isolation has emerged during the past decade as a promising new technology which enhances structural margins of buildings and significantly contributes to protecting people and equipment in buildings during large magnitude seismic events. Seismically "isolated" structures transform the range of high energy seismic input waves into low frequency (for ALMR, about 0.75 Hz) response cycles with significantly reduced horizontal accelerations (factor of 3 or more) allowing for a rigid body response of the structures.

G.4.4.2.1 Seismic Isolation System

Seismically isolated equipment in the reference ALMR plant design includes the reactor module, containment, reactor vessel auxiliary cooling system (RVACS), head access area (HAA) components, the safety related reactor instrumentation, and EM pump synchronous machines, as shown in Figure G.4.4-1. The reference seismic isolation system supporting the seismically isolated platform consists of 31 seismic bearings arranged in a separate vault with access for inspection and maintenance. The seismic bearings are supported on a seven-foot thick basemat, arranged as shown in Figure G.4.4-2. The seismically isolated platform is 72 feet wide by 81 feet-6 inches long. Within the seismic bearing vault, a 2 foot-6 inch thick continuous circular shield wall located adjacent to the reactor module provides radiation shielding for the bearings.

High damping steel laminated elastomeric bearings, of the type described in Reference G.4.4-1 and shown in Figure G.4.4-3, are used for the reference seismic isolation system. The bearings are positioned below the major loads supported by the seismic platform and each bearing carries a vertical load of about 500 kips. With a diameter of 52 inches and a total height of 23.1 inches, and consisting of 30 layers of 1/2-inch thick elastomer and 29 steel shim plates 1/8-inch thick, the bearings were chosen to provide the loaded seismic platform with fundamental frequencies of 0.75 Hz in the horizontal direction and greater than 20 Hz in the vertical direction. One-inch thick steel plates form the top and bottom surfaces of the bearing and provide interfaces with connecting structures. A three-inch thick layer of elastomer is added to the circumferential surface area of the bearing to serve as a protective barrier for the bearing for potentially adverse environmental conditions. All steel and rubber layers are vulcanized together into a composite structure.

G.4.4.2.2 Seismic Isolation Performance

The seismic isolation system transforms the high energy horizontal ground motions into a low frequency response. Analytical results show that accelerations of the components in the isolated system are greatly reduced. For example, the reactor support spectral horizontal acceleration at the core natural frequency of 10 Hz is decreased from 2g to 0.5g with seismic isolation.

Table G.4.4-1 summarizes the performance characteristics of the ALMR seismic isolation system.

TABLE G.4.4-1

ALMR SEISMIC ISOLATION SYSTEM CHARACTERISTICS

Safe Shutdown Earthquake (SSE)	
Design Requirement	= 0.3g
Design Capability	= 0.5g
Operating Basis Earthquake (OBE)	
Design Capability	= 0.17g
Seismic Platform to Ground Relative Displacement	
At 0.3g	= 8.5 inches
At 0.5g	= 14 inches
At bearing limit(1)	= 30 inches
Seismic Platform Natural Frequencies	
Horizontal	= 0.75 Hz
Vertical	= >20 Hz
Reactor Horizontal Seismic Load Reduction Factor	
Horizontal	> 3
Vertical	none

(1) Based on test results reported in Reference G.4.4-3.

G.4.4.3 Rationale Supporting Reference Seismic Bearing Design

Seismic isolation has been developed and is successfully applied in large non-nuclear structures, including computer buildings, public and office buildings. A complementary qualification program was identified which will confirm the reliability required for nuclear application. In general, an extensive data base is available for the performance of seismic isolation bearings for design basis events. The additional characterization will focus on performance margins and accommodation of beyond the design basis conditions.

The elastomeric compound used in the bearings is formed from natural rubber filled with a damping material; References G.4.4-2 and G.4.4-3 describe the properties of this material. Damping is desired to provide energy absorption characteristics of the bearing and thus reduce the maximum relative displacement magnitudes. An additional benefit from damping is the minimal movement of the seismic platform during strong winds and small earthquakes.

Lateral displacement between the bearing's top and bottom plates results from the horizontal shear forces applied to the flexible rubber layers. The load is applied on the bearings through dowels which connect the top and bottom plates to the superstructure and the basemat, respectively. Under large relative displacements between the top and bottom plates, the dowels allow the top and bottom plates to bend and thus limit tensile stresses in the elastomer. With plate bending, some of the dowels move progressively out of their dowel holes. Even under these conditions, testing has shown that sufficient dowel engagement remains to transmit the horizontal forces. When the relative horizontal forces and displacement decrease or reverse, the dowels move back into the dowel holes.

Verification of the performance of the bearing is planned through a series of static and dynamic displacement tests. Results of the first series of these tests on half size ALMR seismic bearings are described in Reference G.4.4-3. These quasistatic tests, conducted at the University of California Earthquake Engineering Research Center, demonstrated large margins for accommodating relative horizontal displacements and vertical loads. For example, the bearings are designed for a shear strain (relative horizontal displacement divided by bearing height) of 50 percent for the maximum relative displacement due to a peak horizontal ground acceleration of 0.3g (a SSE event). While carrying a load of 420 kips, the bearings were able to sustain a relative displacement of 200 percent, four times the expected maximum value. At this relative displacement, the limit of the test rig, substantial warping of the bearing end plates and some disengagement of the dowels occurred but failure could not be induced. This was demonstrated by conducting subsequent tests with shear strains up to 50 percent and observing that the bearing load-deflection behavior was unchanged from initial tests up to 50 percent shear strain.

Evaluation of the load-deflection curves from these first tests identified another desirable characteristic of the bearing. At high strains the stiffness of the bearing increases due to stiffening of the elastomer even though yielding of the end plates results in lower stiffness than if the plates were rigid. The resulting benefit is a further limiting effect on relative displacements during extreme events.

In an attempt to determine the ultimate load carrying capability of the bearing during this first test series, one bearing was subjected to a very large vertical load while in its normal condition. The top and bottom plates were kept aligned during the loading cycle. Even though loaded to the maximum capacity of the testing machine, 4000 kips, the bearing sustained no apparent damage to either the elastomer or internal steel plates. Failure would be anticipated to occur by tensile failure of the steel plates under the ultimate vertical load. To determine ultimate failure with this test machine, a smaller scale bearing would be required. Of greater interest is the ability of the bearing to carry a vertical load while free to displace. This characteristic was measured by conducting a buckling test in which one bearing was placed on top of another and together the stack was subjected to a vertical load test. The bearing end plates at the center of the stack were free to move laterally as the vertical load was increased. The buckling load was reached when the load carrying capability began to decline and represented a margin of 28 times the design load.

The service lifetime of these bearings is expected to extend beyond the 60-year design life of the ALMR. Experience has shown that natural rubber retains its physical characteristics for many years when protected from ozone and high temperatures. Radiation effects are a concern in the ALMR application and radiation shielding has been provided as previously described. By reducing the accumulated radiation dose to less than 2×10^6 rads over the life of the plant, which the reference shielding does, the rubber material is expected to retain its properties (see Reference G.4.4-4). An in-service inspection program summarized in Table G.4.4-2, has been planned to frequently monitor the condition of the bearings. Note that the bearings are examined in place every refueling interval and that

Table G.4.4-2 - PLANNED IN-SERVICE INSPECTION PROGRAM - REACTOR FACILITY SEISMIC ISOLATION BEARINGS

Category	Frequency	Type of Inspection	Component Inspected/Tested	Inspection/Testing Activity	Number of Bearings Inspected/Tested
I	Every 18 Months	Visual	Bearing Rubber Cover	o Check for Obvious Surface Cracks of Tears o Check for Surface Bulges Which May Be Indicative of Bond Failure Between the Rubber and Steel Shim Plate	All 31 Bearings
			Bearing	o Verify Vertical Height(1)	
		Testing	Bearing Rubber Cover	o Measure Hardness (Indicative of Shear Modulus) at Six Points Using a Durometer	16 Bearings (2)
II	Every 4-1/2 Years	Additional Tests to Determine Aging Effects	Bearing Specimens(3)	o Perform Vertical Static Compression Tests to Determine Vertical Bearing Stiffness o Perform Horizontal Static Tests to Determine Horizontal Bearing Stiffness	Perform Vertical and Horizontal Tests on 5 Test Specimens
III	Every 12 Years	Additional Testing	Bearing	o Perform Vertical Static Compression Tests to Determine Vertical Bearing Stiffness o Perform Horizontal Static Tests to Determine Horizontal Bearing Stiffness	Replace and Test 2 Bearings.(4) Vertical and Horizontal Tests Performed on Both Bearings
IV	Following an OBE	Visual/ Testing	Same As Category I	o Repeat All Category I Inspections and Tests	Same as Category I
		Additional Visual	Bearing	o Verify no Permanent Horizontal Displacement of Bearings(5)	All 31 Bearings

NOTES: (1) Any vertical height reduction represents bearing shortening and its effect on continued bearing performance is evaluated against established limits.

(2) Different bearings are tested during each inspection until all 31 of the bearings have been tested; then the process is repeated.

(3) Five 1/4 scale (or smaller) bearing specimens subjected to equivalent vertical design loads are aged during storage in the seismic bearing vault. AT 4-1/2 year intervals all five bearing specimens are removed from storage and tested. After testing, the bearing specimens are returned to storage for further aging in the loaded condition. Any deterioration in bearing stiffness based on test results is used to evaluate degradation effects of all bearings due to aging.

(4) Select bearings for testing on a random basis; replacement bearings are qualified spares. After testing, tested bearings become qualified spares.

(5) Following an earthquake, the bearings are expected to return to their approximate horizontal starting position. The effects of any permanent displacement on continued bearing performance is evaluated against established limits.

every twelve years two bearings are removed for testing and replaced with qualified spares. Local jacking of the isolated platform to take the vertical load off of a bearing will permit it to be removed and replaced. If, during the annual inspection, any bearing's condition is found to be outside the prescribed limits, it will be replaced. Space is provided for equipment to transport the bearings within the bearing vault to a location below a shaft opened to grade by removing hatches in overhead floors.

A method other than dowels for installing bearings and transmitting horizontal forces is under consideration and is described in Reference G.4.4-5. The bearings are bolted to both the basemat and the isolated platform. One attraction of this design is the more positive connection between bearings and supported structures. Additional testing is planned to further assess the merits and possible advantages of this design and to assure that the higher tensile stresses can be accommodated.

G.4.4.4 Experience With Seismic Isolation of Structures

The practice of locating building structures on seismic isolation bearings is relatively new. However, this approach to protecting important structures from the effects of earthquakes is receiving considerable worldwide attention. Currently, 125 structures worldwide are seismically isolated. In the U.S., the first application was the use of base isolation for the Foothill Communities Law and Justice Center in San Bernadino, California (Reference G.4.4-2). Subsequent applications include the retrofitting of an important historical building and a computer manufacturing facility (References G.4.4-6 and G.4.4-7) in Salt Lake City, a hospital building under construction at the University of Southern California (Reference G.4.4-8), and a fire command building in Los Angeles (Reference G.4.4-9). The 1991 Uniform Building Code will include a section on earthquake regulations for seismic-isolated structures. This is expected to accelerate the application of seismic isolation in the U.S.

Application of this technology to nuclear power plants is described in Reference G.4.4-10. Seismic isolation systems have been installed in the two-unit Koeberg plant in South Africa and in the four-unit Cruas plant in

France. An extensive seven-year test program is underway in Japan to develop the information believed necessary to use seismic isolation of nuclear structures in that country.

G.4.4.5 Seismic Isolator Qualification Program

A technology development program has been specified to support the qualification of a seismic isolation system for the ALMR (see References G.4.4-11 and G.4.4-12).

The qualification program includes: (1) the testing of high damping rubber bearings, (2) the qualification of gimballed expansion joints for the secondary heat transfer system piping, (3) large building tests with prototype isolators, (4) scale model tests of reactor structure with isolators on a shake table, (5) the development of analytical models, (6) bearing material optimization and qualification, (7) the development of seismic isolation guidelines, and (8) seismic margin assessment.

Seismic Isolator Bearing Qualification

For the qualification of the seismic isolator bearings an experimental program was specified to determine the following performance characteristics:

- horizontal static and dynamic stiffness
- vertical stiffness
- damping
- margin to failure and failure modes

Static and dynamic tests were specified to determine the dependence of parameters on frequency, displacement and number of displacement cycles. Tests to failure were included to provide: a) insight in available margins to failure, and (b) a data base for the specification of pertinent acceptance tests to be conducted following the bearing fabrication.

To provide a sufficient data base for the specification of safety margins, a sufficiently large number of bearings will be tested to obtain statistical information on bearing performance parameters covering the range of expected fabrication variables. Bearings will be tested at various scales. Presently 1/4, 1/3, 1/2 and full scale tests are included in the test plan. The available facility capability will allow an adequate performance characterization of the bearings over the selected design range. However, tests to failure will be performed with half-size or smaller bearings.

Gimballed Expansion Joints

Programs have been conducted in the U.S. and Japan to evaluate the performance characteristics of flexible piping joints which could be included in the heat transfer system piping of a liquid metal reactor to accommodate differential thermal expansions and relative seismic motions. The work performed earlier led to the specification of ASME Code Case N-290-1 which provides guidelines for design analyses and required supplementary performance tests. The present experimental data base appears sufficiently advanced to allow a modification of the code case for design by analysis only rather than by analysis and testing.

The qualification of gimballed expansion joints for seismic applications may require supplementary tests to establish margins to failure.

Building Tests

As precursor to installation in nuclear power stations, seismic isolation systems installed in buildings with seismic instrumentation can provide useful information on response characteristics for a comparison with analytical predictions. Four types of tests will be conducted to verify the responses of large structures: (1) vibration tests with counter-rotating oscillators to provide uni-direction excitation, (2) static displacement tests to a maximum displacement, (3) tests of instantaneous releases from a maximum displacement, and (4) measurement of building responses to natural seismicity.

System Tests

Scale system tests are planned representing an approximate mass distribution of the reactor system modeled by a steel frame structure. The scale of the test will be compatible with the capability of existing shake tables. Adequate scaling of the system is achievable for the first and dominant eigen frequency, which is a rigid body mode. The system is mounted on four or more isolator bearings and may include flexible component substructures.

The scaling characteristics of bearings will have to be verified.

Analytical Models

As part of the seismic isolation technology development, computer programs are developed for the evaluation of both individual seismic isolator bearing responses and the response of overall seismic isolation systems subjected to earthquake motions. Specifically, this work includes development of the following:

1. Finite element methods for evaluation of individual isolator bearing response when subjected to static and dynamic (seismic) loads; and

2. A three-dimensional seismic isolation system evaluation code, including soil-structure interaction.

Bearing Materials Development

The major objective of this task is to optimize and standardize the bearing compound. The required materials performance parameters are:

- o Adequately high damping (>10% critical damping)
- o Acceptable temperature sensitivity of compound in the design range, including temperature dependence of shear modulus, etc.
- o Acceptably low creep for the high shape factor bearing

- o Consistent good bonding to steel plates with a bond strength greater than the rubber strength

Tests will be performed to establish the performance life and life limiting factors of the bearings. It appears feasible to correlate the compound decomposition or its effect on key materials parameters at various elevated temperatures (<200°F) and exposure times with an Arrhenius curve, and extrapolate to expected bearing lifetimes, which is expected to be in the range of 60 years.

The resistance of the bearing compound to gamma-radiation will be evaluated with coupon tests in EBR-II which started in March 1989. It is expected that no embrittlement affects will appear below 10^7 rads. The tests will confirm the adequacy of the shielding design.

To limit interactions with ozone, a diffusion barrier can be bonded to the rubber surface. Similar considerations apply to the improvement of the fire resistance of the bearings.

The key aspect of the rubber compound development is the demonstration of reliable bonding of the rubber to the steel layer. The bond strength, as well as the key material properties of the rubber will be determined, including the rupture strength, strain, bulk modulus, damping, and the effects of aging and temperature on these properties. Certain material optimization is required, consistent with the requirements for the long-term performance of large diameter bearings.

Quality Control will be performed for the compound testing and production batch control per ASTM Test Standards.

<u>Seismic Isolation Design Guidelines</u>

A proposal for seismic isolation design guidelines for seismically isolated nuclear power plants has been prepared by ENEA in cooperation with ISMES SpA, and GE Nuclear Energy. These guidelines were established for horizontal isolation systems using high damping steel-laminated elastomer bearings. The seismic isolation design guidelines considered the most recent information on seismic analysis of nuclear reactors and the state-of-the-art design of isolated structures. The release of the document by ENEA/GE for a broad review is intended. The qualification procedure specified for the isolator bearings may eventually lead to the definition of an industrial standard and potentially the use of standardized products for seismic isolator bearing design.

The following aspects were addressed in the guidelines document: (1) definition of ground motions, (2) design requirements for the isolated buildings and isolation support structure, the overall seismic isolation system and isolated structures, (3) design requirements for individual isolation devices, (4) qualification of seismic isolation bearings and isolation system, (5) acceptance testing of isolator bearings, (6) reliability and seismic safety margins, and (7) seismic monitoring and monitoring systems.

G.4.4.6 References

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G.4.4-12 Gluekler, E. L., Bigelow, C. C., DeVita, V., Kelly, J. M., Seidensticker, R., Tajirian, F. F., "Seismic Isolation Development for the U. S. Advanced Liquid Metal Reactor Program," First International Seminar in Seismic Base Isolation of Nuclear Power Facilities, San Francisco, California, August 21-22, 1989.

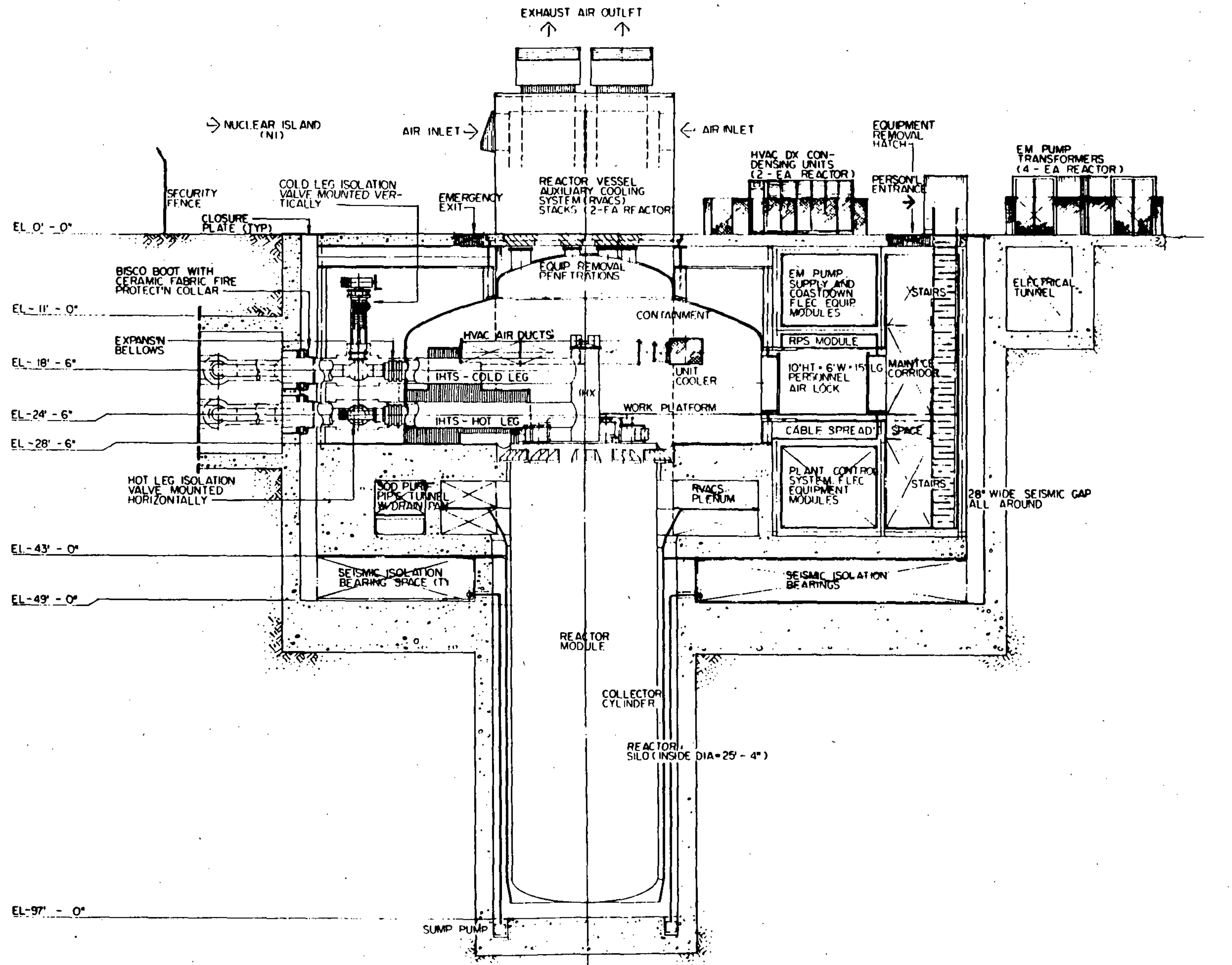

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Figure G.4.4–1 SEISMICALLY ISOLATED REACTOR STRUCTURE

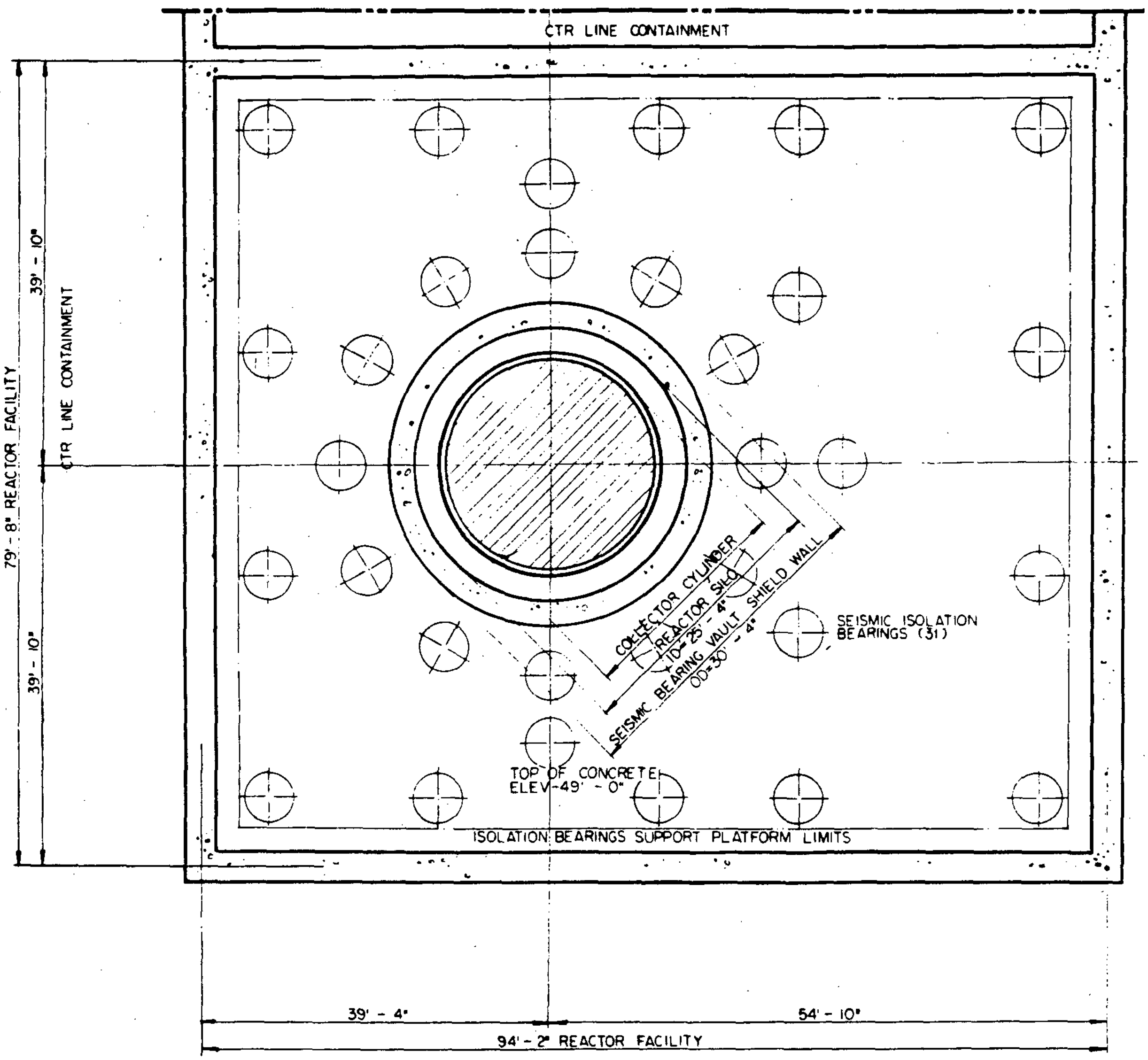

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Figure G.4.4–2 GENERAL ARRANGEMENT OF SEISMIC ISOLATORS

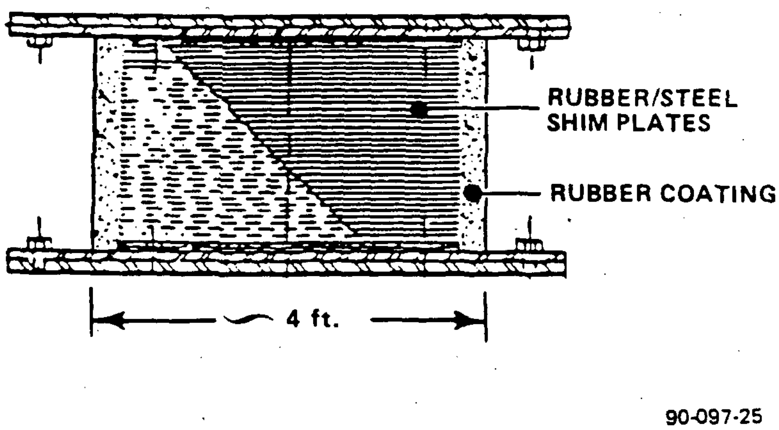

Figure G.4.4–3 REFERENCE ALMR SEISMIC ISOLATOR BEARING

G.4.5 Sodium Void

G.4.5.1 SER Position on Sodium Void Worth

SER Position

The SER position on sodium void worth is summarized in SER Section 4.3.5: "The positive sodium void reactivity coefficient is a concern to the staff and efforts should be made to reduce its magnitude, as much as practical, even if the likelihood of sodium boiling is reduced such that no events which could lead to sodium boiling are in Event Category III."

NOTE: The NRC Staff has identified four event categories as follows, where P is the probability per plant year of an event occurring:

EC-I	$P>10^{-2}$
EC-II	$10^{-2}>P>10^{-4}$
EC-III	$10^{-4}>P>10^{-7}$
EC-IV	$10^{-7}>P$

ACRS Position

The Advisory Committee on Reactor Safeguards (ACRS) has also identified the overall positive sodium void reactivity worth as a major safety issue that must be resolved before ALMR reactor designs can be licensed.

Quoting from Reference G.4.5-1, "[The ALMR] 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 [the] 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."

G.4.5.2 Sodium Void Worth of Reference ALMR Design

The reference ALMR has been designed to have as low a sodium void worth as reasonably attainable, consistent with meeting other safety and design criteria. The maximum sodium void worth in the current design, assuming only driver fuel and internal blanket assemblies void, is nominally 5.50$. If radial blanket assemblies are included, the sodium void worth is nominally 5.26$. The total sodium void worth, assuming complete core void, is nominally 1.42$. Void worths of this magnitude are acceptable if it can be shown that sodium voiding is highly improbable, and that the consequences are tolerable if sodium voiding were to occur.

For sodium voiding to occur, multiple failures of highly reliable, safety-grade, redundant, and diverse systems are required. For example, sodium boiling will <u>not</u> occur under the following conditions:

a. Loss of power to all primary pumps <u>and</u> complete failure to scram, assuming the primary pump coastdown system performs as designed for at least three out of four pumps.

<u>or</u>

b. Loss of power to all primary pumps <u>plus</u> complete coastdown failure of all pumps, assuming the scram system performs as designed.

<u>or</u>

c. Loss of power to all primary pumps <u>plus</u> complete coastdown failure of three out of four pumps <u>plus</u> loss of scram for five out of six control rods.

It should be noted that the above conclusions were reached prior to the addition of gas expansion modules (GEMs) to the reference ALMR core. The GEMs will further reduce the probability of boiling.

As will be shown in G.4.5.3, core voiding events of any kind are of extremely low probability. No event in EC-III, even including the bounding events, results in significant voiding, i.e., more than one assembly.

Because of the relatively low creep rupture strength of the reference HT-9 cladding, voiding is more likely to result from creep rupture and liquid phase cladding attack of high burnup pins followed by fission gas blanketing than it is from sodium boiling. The probability of cladding rupture in unprotected (unscrammed) loss of primary flow events is less than 0.001, and less than 0.005 for unprotected transient overpower events. The probability of failure to scram is 3×10^{-7} per demand (Reference G.4.5-2). Therefore, the probability of voiding from this cause is conservatively estimated as $<2 \times 10^{-9}$ per initiating event severe enough to demand RPS scram action. These events are clearly within EC-IV.

Core voiding from other causes is limited to extremely low probability, hypothetical EC-IV events. In such an EC-IV event, major core-wide voiding could add sufficient reactivity to cause significant fuel melting and the potential for fuel motion. However, the consequences of a core melt accident with the small ALMR metal core are tolerable, as shown in Section G.4.19. The ALMR reactor vessel and closure can sustain and safely accommodate hypothetical core disruptive accident (HCDA) loads resulting from energetics on the order of 500 MJ without loss of structural integrity, disengagement of the rotatable plug from the reactor closure, or expulsion of sodium. This level is more than an order of magnitude greater than the anticipated energetics from any conceivable HCDA. Therefore, any conceivable HCDA will not seriously challenge the primary coolant boundary, and will not release any fission products into the head access area.

Core design alternatives to reduce the sodium void worth have been investigated. These studies show that design changes made to reduce the void worth invariably impact the safety performance of other parameters, and the economics of power production, in an adverse manner. For instance, the void worth can be reduced by "pancaking" the core to have a small height and large diameter. However, such a change increases the burnup reactivity swing, which increases the amount of positive reactivity the control rods must hold down. Such a change also reduces the negative radial reactivity feedbacks. Therefore, decreasing the height to diameter ratio reduces the effectiveness of the inherent reactivity feedbacks to terminate power excursions during rod withdrawal accidents.

Reducing the height to diameter ratio also requires an increase in the linear power generation rate if economical power densities and reactor vessel diameters are to be maintained. However, the linear power generation rate cannot be increased too far or centerline fuel melting will occur, which is unacceptable. Therefore, the amount of reduction in void worth which can be achieved by pancaking the core is also limited by the requirements of maintaining centerline fuel temperatures below the melting point, and maintaining plant economics such that the ALMR is a competitive design.

As will be discussed in Section G.4.5.3, trade-offs such as the above limit the practical sodium void worth to a value only 25-35% below the current value of 5.50$. There is little safety benefit to modify the ALMR core design to achieve such a small reduction. Therefore, because (1) meaningful reductions in sodium void worth adversely impact other safety parameters, (2) meaningful reductions in sodium void worth require reactor designs which increase the cost of producing power, (3) it can be shown that sodium voiding is highly improbable, and (4) it can be shown that the consequences of sodium voiding are tolerable if it were to occur, no design changes have been made to reduce the sodium void worth.

Tables G.4.5-1 and G.4.5-2 summarize detailed dimensional information on the fuel assemblies and blanket assemblies, respectively. Additional configuration and basic dimensional data are discussed in Section G.2.1.

Table G.4.5-1

FUEL ASSEMBLY DATA

Duct Pitch (In.)	6.282
Duct Material (In.)	HT-9
Duct Gap (In.)	0.175
Duct Wall Thickness (In.)	0.140
Duct Outer Flat to Flat (In.)	6.107
Duct Inner Flat to Flat (In.)	5.827
Overall Assembly Length (In.)	196
Bundle Flow Area ($In.^2$)	10.74
Pins Per Assembly	331
Pin Spacer	Straight Start Wire Wrap
Pin Pitch/Diameter	1.1985
Fuel Height (In.)	53
Upper Gas Plenum Height (In.)	74
Upper End Plug (In.)	1
Lower End Plug & Shielding (In.)	40

PIN DATA:

Fuel Slug Type	U-27%Pu-10%Zr
Pin Overall Length (In.)	168.0
Pin Outer Diameter (In.)	0.263
Cladding Material	HT-9
Cladding Thickness (In.)	0.020
Slug Diameter (In.)	0.193
Slug Cladding Gap (In.)	0.030
Slug Fabrication Density (%TD)	100.0
Slug Smear Density (In.)	75.0
Wire Wrap Diameter (In.0	0.051
Wire Wrap Pitch (In.)	12.0
Bond	Na

VOLUME FRACTION DATA:

Fuel	- BOL Casting	0.2833
	- BOL Bond	0.0949
	- Smeared	0.3783
Na	- Coolant	0.3693
	- Coolant + Bond	0.4643
Structure		0.2524

Table G.4.5-2

BLANKET ASSEMBLY DATA

Duct Pitch (In.)	6.282
Duct Material (In.)	HT-9
Duct Gap (In.)	0.175
Duct Wall Thickness (In.)	0.140
Duct Outer Flat to Flat (In.)	6.107
Duct Inner Flat to Flat (In.)	5.827
Overall Assembly Length (In.)	196
Bundle Flow Area ($In.^2$)	6.70
Pins Per Assembly	271
Pin Spacer	Straight Start Wire Wrap
Pin Pitch/Diameter	1.0685
Blanket Height (In.)	53
Upper Plenum Height (In.)	74
Upper End Plug (In.)	1
Lower End Plug & Shielding (In.)	40
PIN DATA:	
Blanket Slug Type	Depleted U-10%Zr
Pin Overall Length (In.)	168.0
Pin Outer Diameter (In.)	0.476
Cladding Material	HT-9
Cladding Thickness (In.)	0.022
Slug Diameter (In.)	0.3983
Slug Cladding Gap (In.)	0.0337
Slug Fabrication Density (%TD)	100.0
Slug Smear Density (In.)	85.0
Wire Wrap Diameter (In.0	0.032
Wire Wrap Pitch (In.)	12.0
Bond	Na
VOLUME FRACTION DATA:	
Fuel - BOL Casting	0.4630
- BOL Bond	0.0817
- Smeared	0.5447
Na - Coolant	0.2510
- Coolant + Bond	0.3327
Structure	0.2043

G.4.5.3 Rationale Supporting Current Sodium Void Worth

G.4.5.3.1 Void Worth Distribution and Values

The detailed distribution of void worth within the ALMR reference metal equilibrium cycle core has been determined. Values for the startup core are expected to be similar. The calculations were performed using the DIF3D neutronics code for the flux solutions, which were carried out with 22 neutron energy groups, three-dimensional triangular-z geometry in a 60-degree core layout, and utilizing three-dimensional perturbation calculations by the VARI3D code. The detailed three-dimensional reactivity map was obtained by exact perturbation calculations with the VARI3D code. The reactivity effect from total core (full length of core assemblies) voiding at the end of equilibrium cycle conditions is summarized in Table G.4.5-3.

Table G.4.5-3

ALMR SODIUM VOID WORTH, BY CORE REGIONS

Void Region	Worth ($)	
Driver Fuel	3.12	
Internal Blankets	2.38	
Radial Blankets	-0.24	
Driver & Blankets		5.26
Control	-2.71	
Ultimate Shutdown	-0.22	
Reflector & Shield	-0.21	
Gas Expansion Module	-0.69	
Other		-3.83
Total		1.43

The contributors to positive void worth are the interior assemblies (fuel and internal blankets). The peripheral assemblies such as radial blankets, reflectors, and shield assemblies have a small net negative void worth, as the negative effect from increased neutron leakage near the core edge becomes pronounced and overrides the positive spectral effect from sodium voiding.

Figure G.4.5-1 shows the void worth per assembly in a 60-degree core sector. The maximum void worth is about 0.15$ for an inner fuel assembly. The void worth becomes less positive for those assemblies that are some distance from the core center and eventually becomes negative for the radial blanket and shield assemblies. In the axial traverse, the void worth is negative at the upper and lower ends of the fuel columns due to the neutron leakage effect, although the net effect of complete axial voiding is positive for the ternary fuel. The axial distribution of void worth is shown in Figure G.4.5-2 for the average fuel, internal blanket and radial blanket assemblies, and for control, reflector and shield assemblies in Figures G.4.5-3. Figures G.4.5-4 and G.4.5-5 show the cumulative void worth when the voiding occurs progressively from the top of the fuel column. The results in Figure G.4.5-4 indicate that the cumulative void worth does not become positive until about the upper 30% of the active fuel region is voided.

G.4.5.3.2 Probability of Voiding

G.4.5.3.2.1 Introduction

The ALMR reactor design incorporates many design features to enhance inherent reactivity feedbacks during off-normal events to prevent damage to the core and to minimize the potential for radiological release. The reactivity feedback mechanisms in the ALMR are present in all liquid metal reactor concepts, but specific features have been selected in the ALMR design to minimize the positive feedback mechanisms and enhance the negative feedback mechanisms. These optimized feedback mechanisms, combined with the inherent properties of metal fuel, permit the ALMR reactor to withstand a wide range of events, including all EC-I, II and III events, without significant risk to the public.

The principal reactivity feedback mechanisms in the ALMR are:

- o Doppler
- o Sodium expansion
- o Axial fuel expansion
- o Control rod driveline expansion
- o Core radial expansion
- o Gas expansion modules (GEMs)

The core radial expansion feedback consists of three related effects: thermal expansion of the above core load pads, thermal expansion of the gridplate, and bowing (bending) of the fuel assembly ducts. The effects of uncertainties in these feedback mechanisms have been analyzed for two key anticipated transients without scram (ATWS) events:

- o Unprotected Transient Overpower (UTOP)
- o Unprotected Loss of Flow and IHTS Cooling (ULOF/LOHS)

The third standard ATWS event, unprotected loss of heat sink (ULOHS), has been shown to be similar to, but less severe than, the ULOF/LOHS event; therefore, it has not been analyzed in this study. Margins to voiding for the unprotected loss of heat sink event will significantly exceed those for the ULOF/LOHS event.

Margins to voiding have been calculated by considering the reactivity feedback uncertainties plus one other key uncertainty for each event. For the unprotected transient overpower event, this is the uncertainty in the control rod worth added during the transient as the six control rods drive out to the rod stop limits. For the unprotected loss of flow and IHTS cooling transient, the additional degradation mechanism considered is the failure of a pump synchronous machine resulting in loss of coastdown flow from one of the four pumps. It should be noted that this study was done prior to the addition of GEMs to the reference ALMR core. The GEMs will further reduce the probability of voiding for the ULOF/LOHS event.

G.4.5.3.2.2 Analysis of Reactivity Feedback Uncertainties

The uncertainty values for each feedback mechanism are summarized in Table G.4.5-4. The one-sigma uncertainties are taken directly from Mueller and Wade (Reference G.4.5-3). Linear extrapolations are made to two-sigma and three-sigma uncertainties levels for each mechanism except for duct bowing. The duct bowing term is typically 20-30% of the above-core load pad radial expansion feedback, and the uncertainty in the combined core radial expansion has therefore been taken as somewhat larger than that for the load pad and gridplate.

Table G.4.5-4

ALMR REACTIVITY FEEDBACK UNCERTAINTIES

	Uncertainty (%)		
	One-Sigma	Two-Sigma	Three-Sigma
Doppler	20	40	60
Sodium Density	20	40	60
Fuel Axial Expansion	30	60	90
Control Rod Expansion	20	40	60
Radial Expansion of Core			
Above-core Load Pads	20	40	60
Core Grid Plate	20	40	60
Duct Bowing	50	90	*
Combined Radial Expansion	30	60	90

* Duct bowing assumed to be zero

The effects of uncertainties in the reactivity feedbacks on the design margins during the two ATWS events were calculated using the ARIES transient computer code. The ARIES code has been used for the safety analysis of the ALMR reactor for both Event Category II and III transients. ARIES has been shown to give results in excellent agreement with the national standard safety code developed by Argonne National Laboratory, SASSYS, which has been validated by comparison to EBR-II and FFTF integral test data. ARIES has also been shown to be in excellent agreement with the SSC-PRISM code developed by Brookhaven National Laboratory in support of the NRC (Reference G.4.5-4). The ANL single assembly bowing model is used as part of ARIES to determine the core radial expansion reactivity feedback.

The reactor is modeled by four assembly types - an average driver fuel assembly, an internal blanket assembly, an outer radial blanket assembly and a peak driver fuel assembly. The peak driver fuel assembly is modeled as a fresh fuel assembly located at the peak radial power location (the peak driver assembly type is not used for the computation of reactivity feedback effects). An additional coolant channel is used to model the bypass flow, and the flow through the control and radial shield assemblies. Nine axial nodes are used to model the active region of the fuel.

Three series of cases have been analyzed for the two key ATWS events (UTOP and ULOF/LOHS) using the ARIES code in which a consistent (one-, two- or three-sigma) uncertainty was applied to each reactivity feedback mechanism separately. It is difficult to combine the feedback uncertainties in a statistically rigorous manner in ARIES because the feedback mechanisms are very different. However, as a reasonable approximation to a combined uncertainty (or confidence) level, the differences in peak temperatures between the nominal transient and the transients calculated for each individual uncertainty are combined as the square root of the sum of the squares.

The calculated peak fuel, cladding and coolant temperatures at the one-sigma, two-sigma and three-sigma uncertainty levels are summarized in Tables G.4.5-5, G.4.5-6 and G.4.5-7 for an unprotected transient overpower (UTOP) event of 0.36$ insertion at a constant 0.02$/sec rate corresponding to withdrawal of all six control rods, and in Tables G.4.5-8, G.4.5-9 and G.4.5-10 for the unprotected loss of flow with loss of heat sink (ULOF/LOHS) event.

Table G.4.5-5

ALL-RODS UTOP PEAK FUEL, CLADDING AND COOLANT TEMPERATURES AT ONE-SIGMA UNCERTAINTY LEVEL IN REACTIVITY FEEDBACKS

	Peak Power(%)	Peak Temperatures (°F) Fuel	Fuel/Clad Interface	Cladding	Coolant	Mixed Mean Core Outlet
Nominal Feedbacks	172	1935	1350	1321	1291	1110
Reactivity Term						
Control Drive	174	1946	1358	1330	1299	1117
Sodium Expansion	175	1955	1365	1336	1305	1120
Doppler	177	1966	1372	1343	1312	1125
Fuel Expansion	178	1966	1373	1344	1313	1126
Core Radial Exp	177	1964	1371	1342	1311	1125
Combined Reactivities	182	1994	1393	1363	1331	1139

Table G.4.5-6

ALL-RODS UTOP PEAK FUEL, CLADDING AND COOLANT TEMPERATURES AT TWO-SIGMA UNCERTAINTY LEVEL IN REACTIVITY FEEDBACKS

	Peak Power(%)	Peak Temperatures (°F) Fuel	Fuel/Clad Interface	Cladding	Coolant	Mixed Mean Core Outlet
Nominal Feedbacks	172	1935	1350	1321	1291	1110
Reactivity Term						
Control Drive	176	1956	1367	1338	1307	1125
Sodium Expansion	179	1973	1378	1348	1317	1129
Doppler	184	2001	1399	1368	1336	1143
Fuel Expansion	184	2001	1399	1369	1337	1143
Core Radial Exp	183	1997	1396	1366	1334	1141
Combined Reactivities	193	2055	1439	1408	1374	1171

Table G.4.5-7

ALL-RODS UTOP PEAK FUEL, CLADDING AND COOLANT TEMPERATURES AT THREE-SIGMA UNCERTAINTY LEVEL IN REACTIVITY FEEDBACKS

		Peak Temperatures (°F)				
	Peak Power(%)	Fuel	Fuel/Clad Interface	Cladding	Coolant	Mixed Mean Core Outlet
Nominal Feedbacks	172	1935	1350	1321	1291	1110
Reactivity Term						
Control Drive	177	1969	1379	1350	1319	1136
Sodium Expansion	182	1992	1393	1363	1331	1139
Doppler	191	2041	1430	1398	1365	1163
Fuel Expansion	191	2040	1429	1397	1364	1163
Core Radial Exp	190	2035	1425	1374	1361	1160
Combined Reactivities	206	2127	1495	1461	1425	1208

Table G.4.5-8

ULOF/LOHS PEAK FUEL, CLADDING AND COOLANT TEMPERATURES AT ONE-SIGMA UNCERTAINTY LEVEL IN REACTIVITY FEEDBACKS

	Peak Temperatures (°F)				
	Fuel	Fuel/Clad Interface	Cladding	Coolant	Mixed Mean Core Outlet
Nominal Feedbacks	1630	1437	1436	1434	1227
Reactivity Term					
Control Drive	1634	1440	1439	1438	1229
Sodium Expansion	1644	1452	1451	1449	1239
Doppler	1635	1439	1437	1436	1230
Fuel Expansion	1632	1439	1437	1436	1228
Core Radial Exp	1635	1461	1459	1458	1245
Combined Reactivities	1647	1466	1465	1463	1248

Table G.4.5-9

ULOF/LOHS PEAK FUEL, CLADDING AND COOLANT TEMPERATURES AT TWO-SIGMA UNCERTAINTY LEVEL IN REACTIVITY FEEDBACKS

	Peak Temperatures (°F)				
	Fuel	Fuel/Clad Interface	Cladding	Coolant	Mixed Mean Core Outlet
Nominal Feedbacks	1630	1437	1436	1434	1227
Reactivity Term					
Control Drive	1636	1442	1441	1439	1225
Sodium Expansion	1657	1467	1466	1464	1257
Doppler	1642	1442	1440	1438	1231
Fuel Expansion	1633	1440	1439	1437	1233
Core Radial Exp	1639	1487	1486	1484	1276
Combined Reactivities	1661	1496	1495	1493	1284

Table G.4.5-10

ULOF/LOHS PEAK FUEL, CLADDING AND COOLANT TEMPERATURES AT THREE-SIGMA UNCERTAINTY LEVEL IN REACTIVITY FEEDBACKS

	Peak Temperatures (°F)				
	Fuel	Fuel/Clad Interface	Cladding	Coolant	Mixed Mean Core Outlet
Nominal Feedbacks	1630	1437	1436	1434	1227
Reactivity Term					
Control Drive	1638	1444	1443	1441	1224
Sodium Expansion	1670	1483	1482	1480	1255
Doppler	1648	1445	1443	1441	1230
Fuel Expansion	1635	1442	1440	1439	1228
Core Radial Exp	1644	1518	1517	1515	1305
Combined Reactivities	1677	1531	1530	1528	1310

For each transient, the margin to sodium boiling can be determined by calculating the maximum local peak coolant temperature for various confidence levels, and comparing these maximums to the sodium saturation temperature of approximately 1760°F for a loss-of-primary-flow event and approximately 1960°F for an overpower event at full flow. This is shown for the two ATWS transients in Figure G.4.5-6. It can be seen that there is significantly more than a three-sigma margin to sodium boiling for both events, as the three-sigma peak coolant temperatures are only 1425°F and 1528°F for the all-rods UTOP and ULOF/LOHS, respectively.

The margin to fuel pin creep rupture, which could cause voiding by fission gas blanketing, is somewhat more difficult to calculate, as creep rupture is a function of time at temperature. Inspection of Figure G.4.5-7 indicates that rupture of the HT-9 cladding occurs within about 300 seconds at 1500°F and within 120 second at 1550°F for end-of-life (third cycle) fuel pins. Below the minimum fuel-clad liquid phase formation temperature of 1300°F, creep rupture times are expressed in months. Slightly above the minimum liquid phase formation temperature, for instance at 1320°F, creep rupture requires about 10 hours. The specific calculations of margin to fuel pin creep rupture for the UTOP and ULOF/LOHS events are discussed in the following section.

G.4.5.3.2.3 Calculated Margins to Voiding

<u>Unprotected Transient Overpower</u>

The peak and long-term steady state fuel/cladding interface temperatures for a 0.36$ UTOP with different levels of reactivity degradation are plotted in Figure G.4.5-8. Inspection of the ARIES transient plots indicates that clad failures occur in severe UTOP events only after the power and temperatures have returned to a new, elevated steady-state level. At the three-sigma confidence level, the long-term peak fuel/clad interface temperatures are approximately 1280°F, slightly below the minimum fuel-clad liquid phase formation temperature of 1300°F. Based on reactivity feedback uncertainties alone, cladding failures would be predicted to begin, in the hottest fuel assemblies, at about the 3.5-sigma confidence level, or in less than 0.03% of the UTOP cases.

However, the reactivity feedbacks are not the only sources of uncertainty in UTOP events. Of more importance in setting the margin to voiding is the uncertainty in the control rod worth which can be added during an all-rods UTOP. The reference reactor design utilizes electronically positioned mechanical control rod stops set to provide a 3-sigma confidence level that a UTOP potential greater than 0.40$ does not exist at any time during the reactor operating cycle. That is, there is less than 0.14% chance that an all-rods UTOP, if it were to occur, would exceed 0.40$ and be damaging. If the UTOP event adds less than 0.40$ reactivity, the 1300°F threshold temperature for fuel-clad liquid phase formation is not reached and fuel pin failures are not expected during the transient.

Based on the uncertainties in the UTOP reactivity addition potential alone, cladding failures would be predicted to begin after several hours in the hottest fuel assemblies, (the four inner ring third cycle fuel assemblies) at about the 3-sigma confidence level; that is, in no more than 0.14% of the UTOP cases. If one statistically combines the uncertainty in the UTOP potential with those of the reactivity feedbacks by calculating the increase in the long term, quasi-equilibrium fuel-clad interface temperature, the 1300°F liquid phase threshold is reached at a 2.6-sigma confidence level. In other words, considering combined uncertainties, less than 0.5% of UTOP events would result in some long-term cladding failures and release of fission gas. The probability of voiding an assembly is significantly less than this because of the strong likelihood of sweepout of the fission gas due to high flow velocities in an UTOP event. Therefore, equating the probability of cladding failure to the probability of voiding is conservative.

It is noted that, for UTOP events with reactivity additions of 0.40$ to 0.50$, there is significant time to achieve cold shutdown before cladding failures occur. For example, a 0.50$ UTOP event, corresponding to a 4-sigma confidence level (0.0032%), results in fuel-clad interface temperatures reaching a long term, quasi-equilibrium temperature of somewhat less than 1400°F. Corresponding creep rupture failure of the highest burnup fuel pins will take about 45 minutes. If cold shutdown is achieved in less than this time, there will be essentially no possibility of voiding. No credit has been taken in this study for the ultimate cold shutdown assembly in the core center position.

Unprotected Loss of Flow

Peak fuel and fuel/clad interface temperatures during the ULOF/LOHS transients, considering various levels of reactivity feedback uncertainties, are plotted in Figure G.4.5-9. It can be seen that a temperature of 1485°F is exceeded for 100 seconds at the three-sigma confidence level. This time-temperature history is not sufficient to cause cladding rupture. This can be estimated from Figure G.4.5-7, but was actually calculated by integrating the HT-9 cladding strain damage under the temperature history curve. The total integrated strain damage fraction was 0.43. Based on reactivity feedback uncertainties only, cladding failures would be predicted to begin, in the hottest fuel assemblies, at about the 4-sigma confidence level, or in less than 0.004% of the ULOF/LOHS cases.

An additional degradation mechanism that must be considered for ULOF/LOHS transients is the failure of a pump synchronous machine resulting in loss of coastdown flow on one of the four pumps. This is estimated (Reference G.4.5-2) to have a probability of 8.7×10^{-7} per demand. ARIES analyses have shown (Table G.4.5-11) that loss of coastdown flow on one pump increases peak temperatures during a ULOF/LOHS event no more than 35°F.

Table G.4.5-11

PEAK TEMPERATURES REACHED DURING ULOF EVENTS, WITH NORMAL COASTDOWN AND WITH LOSS OF ONE PUMP COASTDOWN

	Peak Temperature (°F)			
	Fuel Centerline	Fuel-Clad Interface	Local Coolant	Core Avg Outlet
Nominal ULOF/LOHS	1630	1437	1434	1227
ULOF with loss of one pump coastdown	1665	1472	1451	1237

If this 35°F increase in fuel-clad interface temperature is added directly to the temperatures plotted in Figure G.4.5-9, the margin to cladding failure, given failure to scram, is reduced from approximately 4-sigma (4×10^{-5}) to 3.5-sigma (2.3×10^{-4}). It is appropriate for ULOF events to equate the probability of cladding failure to that of voiding since, at low flow rates approaching natural circulation conditions, there is insufficient flow velocity to sweep the fission gas bubble out of the assembly before it expands to void the entire active core region.

The probabilities of losing flow coastdown on two or more pumps simultaneously are so low ($<10^{-8}$ for loss of two pumps) they need not be considered here. It is worth noting that voiding is not calculated to occur even with loss of coastdown on two pumps, using nominal reactivity feedbacks.

G.4.5.3.2.4 Conclusions on Probability of Voiding

The calculated margins to voiding show that, if voiding occurs at all, it is more likely to result from creep rupture and liquid phase cladding attack of high burnup pins rather than by sodium boiling. The original PRISM PRA (Reference G.4.5-2) states the nominal probability of failure to scram to be 3×10^{-7} per demand. Cladding rupture would only occur at significantly greater than three-sigma degradation (<0.001) for ULOF/LOHS events, and at greater than 2.6-sigma degradation (<0.005) for UTOP events. Therefore, the probability of cladding rupture which could lead to voiding is conservatively estimated as $<2 \times 10^{-9}$ per initiating event severe enough to demand RPS action (i.e., scram). These probability values are extremely low and are definitely in the residual risk Event Category IV.

G.4.5.3.3 Consequences of Voiding

As shown in the previous subsection, voiding events are of extremely low probability. No core voiding occurs in any EC-III event, including the NRC bounding events. (See also Section G.4.16, Safety Analysis.) Core voiding is limited to extremely low probability, hypothetical EC-IV events. In such an EC-IV event, major core-wide voiding could add sufficient reactivity to cause significant fuel melting, with the potential for fuel motion. It is shown in Section G.4.19 that the consequences of a full core melt accident, or a hypothetical core disruptive accident (HCDA), are tolerable with the small ALMR metal core. The ALMR reactor vessel and closure can safely accommodate HCDA loads resulting from energetics on the order of 500 MJ without loss of structural integrity, disengagement of the rotatable plug from the reactor closure, or expulsion of sodium. This level is more than an order of magnitude greater than the anticipated energetics from any conceivable HCDA. Therefore, any conceivable HCDA will not seriously challenge the primary coolant boundary, and will not release any fission products into the head access area. Therefore, it is concluded that the consequences of voiding, should it occur, are tolerable.

G.4.5.3.4 Core Design Alternatives to Reduce Void Worth

Studies have been conducted by GE, Westinghouse, and Argonne National Laboratory on possible core design changes to reduce the positive sodium worth of the central region of the reference ALMR core.

G.4.5.3.4.1 Criteria for Alternative Designs

For an alternative core design with reduced void worth to be acceptable, the following criteria must be met:

a. <u>The total positive sodium void reactivity worth of the core must be reduced to less than 0.50$</u>. This is the maximum void reactivity worth for which assurance can be maintained that any voiding event would not, of itself, result in substantial core melting. This is so because the magnitude of core energetics resulting from voiding are determined by the sodium void worth value and the rate of voiding. A low void worth ensures low energetics upon voiding. Analyses of unprotected transient overpower events with 0.50$ or more reactivity addition have shown that these transients, after an initial power and temperature overshoot, stabilize at a power level sufficiently high that the peak fuel-clad interface temperatures are greater than 1300°F, the minimum fuel-clad liquid phase formation temperature. In these cases, it is at least theoretically possible for significant fuel melting to occur.

b. <u>The impact of the design change on the passive safety performance characteristics of the core must be acceptable</u>. The 2-sigma peak centerline fuel temperature at 110% steady-state power must maintain a margin to 1675°F, the solidus temperature of Zr-depleted fuel, to ensure no molten fuel during design basis transient operation. This is because irradiation causes radial redistribution of the uranium and zirconium, producing a zone near the pin centerline in which the zirconium is depleted to approximately 2%; the 2-sigma lower bound on the solidus (melt) temperature of U-26Pu-2Zr is 1675°F. As will be shown below, the steady-state no-melt criterion is a major limitation to reducing core height for practical power density cores. (Reducing core height and "pancaking" the core is the usual way to reduce sodium void worth in liquid metal reactors).

A second passive safety performance characteristic which is adversely impacted by decreased core height is the low burnup reactivity swing. A low burnup reactivity swing has been intentionally designed into the ALMR core to reduce the amount of positive reactivity which can be inserted by unprotected transient overpower (UTOP) events. Reducing core height increases the burnup reactivity swing, which degrades the ability of inherent negative reactivity feedbacks to terminate UTOP events. Of course, large burnup reactivity swings can be accommodated up to a point by the introduction of control rod stops. However, the greater the amount of burnup reactivity swing, the more reactivity worth must be designed into the rods and the more often the rod stops must be adjusted, introducing increased opportunity for adjustment errors and adverse safety consequences. There is therefore a practical limit to how much burnup reactivity swing can be accommodated by this method.

c. <u>The impact on the economics of power production must be acceptable</u>. Many changes which can be made to reduce the void worth adversely affect the economics of power production. Some changes are so adverse that they might jeopardize the viability of the ALMR concept. Therefore, the selection of design changes to reduce void worth must consider the economic impact.

G.4.5.3.4.2 Core Height Study (GE)

A study was carried out to assess the impact of changing the core height (fuel column length) on the overall core performance and sodium void worth in the ALMR reference core at equilibrium conditions. The study consisted of three parts:

a. Evaluation of core performance for reduction in the core height from the reference case of 53 inches to 46 and 36 inches while holding all other core parameters constant.

b. Evaluation of core performance at core heights of 53, 46 and 36 inches, while adding a 6.5-inch axial blanket segment to the top and bottom of the fuel column.

c. Evaluation of core performance at core heights of 40 and 30 inches, while allowing an increase in the assembly pitch (and, consequently, the core radius), thus effectively keeping the core volume (and fuel volume) constant. However, the number of pins in each bundle was varied to maintain constant average linear power.

The results of the study are summarized in Figures G.4.5-10, G.4.5-11 and G.4.5-12. The solid line in each figure represents the reference core case, the dashed line represents the reference core with axial blankets added, and the dashed-dot line represents the case of constant core volume (radius increases as height decreases) without axial blankets.

As an example from the study, consider the constant core diameter case. Peak linear power increases as the peak centerline fuel temperature increases. The Zr-depleted solidus of 1675°F is reached at 11.2 kW/ft for a 2-sigma margin. Therefore, as shown in Figure G.4.5-10, the core height cannot be decreased below 42 inches for the reference core diameter. At these dimensions the fuel + blanket void worth is reduced to a minimum of 4.20 - 4.50$ (Figure G.4.5-12). This is only a 15 - 20 % decrease from the reference 5.26$. In addition, the burnup reactivity swing, excluding that due to fuel axial expansion, is increased to a value between 2$ and 3$ (Figure G.4.5-11).

In general, the study shows the relationship of some of the key parameters, and the difficulty of achieving a significant reduction in void worth for a practical core. It quantifies the limited void worth reduction possible from shortening the core within the ALMR core diameter restraints. Relaxing the core diameter constraint would eliminate the linear power generation rate constraint by permitting more fuel and fuel pins in the shorter core. However, this approach would degrade radial reactivity feedback characteristics important to passive reactivity control. Also, relaxing the constraint on core diameter and designing cores with short heights and large diameters would introduce burnup reactivity swings much greater than the 2$ - 3$ calculated in the study. As mentioned earlier, large burnup reactivity swings can be accommodated up to a point by the introduction of control rod stops. However, the greater the amount of burnup reactivity swing, the more reactivity worth must be designed into the rods and the more often the rod stops must be adjusted, introducing increased opportunity for adjustment errors and adverse safety consequences. There is therefore a practical limit to how much burnup reactivity swing can be accommodated by this method.

In addition to the reactivity effects discussed above, other studies show that relaxing the core diameter constraint to achieve low height to diameter ratios introduces severe economic penalties by reducing breeding ratios and increasing fissile inventory requirements (References G.4.5-6 and G.4.5-7). The large diameter reactor vessels required for such a core would further degrade the economics of power production. Therefore, making significant reductions in void worth by reducing core height is not practical for the ALMR.

G.4.5.3.4.3 General Studies of Core Design Changes to Reduce Sodium Void Worth

A more general study of means of reducing sodium void worth was conducted by Argonne National Laboratory (Reference G.4.5-6). This study addressed metal fueled reactors of two sizes: 450 MWt and 900 MWt. The goal was to quantify the trade-offs among sodium void worth reduction and resultant changes in burnup reactivity swing, breeding gain, fissile inventory, core diameter and core volume. Three classes of design changes were evaluated: (1) composition changes at fixed core layout encompassing changes in steel, sodium and void volume fractions and the addition of BeO and B_4C; (2) changes in height to diameter ratio at fixed assembly design; and (3) changes in core layout encompassing axial heterogeneous, radial heterogeneous, homogeneous, annular and coupled cores.

The conclusions of this study are consistent with those of the more core specific GE study:

a. Sodium void worth can be reduced to near zero or even made negative, but the result will be an unfavorable change in one or more of the other performance parameters considered.
b. There is no universal best way to reduce sodium void worth because the relative importance of the several other performance changes will depend upon the specific design criteria.

The earlier Westinghouse study (Reference G.4.5-7) reached essentially the same conclusions. None of the core design options investigated to reduce sodium void worth in the ALMR core had both a low sodium void worth and low burnup reactivity swing. Low sodium void worth cores are achievable in a number of ways, but are always associated with a large additional burnup swing, typically above 5$. In addition, most of the design options would result in significant increases in core radial dimensions and decreased power densities.

G.4.5.3.5 Conclusions

Several independent studies show that design changes required to reduce the sodium void worth in liquid metal cooled reactors adversely impact other safety and economic performance parameters. Application of the results of these studies to the ALMR show that these adverse impacts become significant at sodium void worths only 25-35% below the current maximum value of 5.50$. There is little safety benefit to modify the ALMR core design to achieve such a small reduction. Therefore, because (1) significant reductions in sodium void worth adversely impact other safety parameters, (2) significant reductions in sodium void worth require reactor designs which increase the cost of producing power, (3) it can be shown that sodium voiding is highly improbable, and (4) it can be shown that the consequences of sodium voiding are tolerable if it were to occur, no design changes have been made to reduce the sodium void worth.

G.4.5.4 References

G.4.5-1 Letter, Forrest J. Remick (Chairman, ACRS) to Lando W. Zech, Jr (Chairman, NRC), "Safety Evaluation Report for the Sodium Advanced Fast Reactor (SAFR) Design," January 19, 1989.

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G.4.5-3 Mueller, C. J., and D. C. Wade, "Probabilities of Successful Inherent Shutdown of Unprotected Accidents in Innovative Liquid Metal Reactors," ANS Topical Meeting on Safety of Next Generation Power Reactors, Seattle, WA, April 1-5, 1988.

G.4.5-4 Van Tuyle, G. J.. et al., "Summary of Advanced LMR Evaluations - PRISM and AFR," NUREG/CR-5364, October 1989.

G.4.5-5 Herzog, J.P., and K. J. Miles, "Analysis of Fission Gas Release from IFR Subassembly X421," ANL Intra-laboratory Memo, March 6, 1989.

G.4.5-6 Kahlil, N., and R.N. Hill, "An Evaluation of LMR Design Options for Reduction of the Sodium Void Worth," ANL-IFR Memo, to be issued.

G.4.5-7 Gundy, L.M., "Low Sodium Void Core Designs for PRISM and SAFR," BTCM-88-21, October 1988

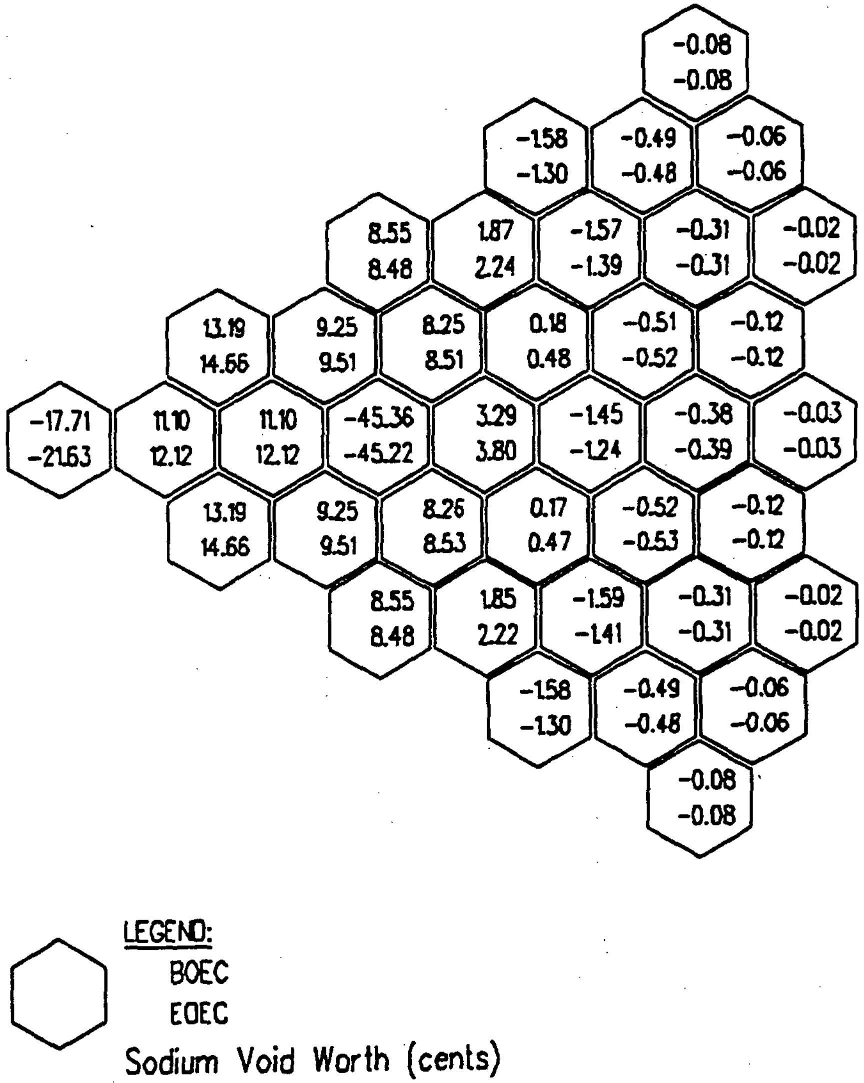

Figure G.4.5-1 VOID WORTH BY ASSEMBLY IN ALMR CORE

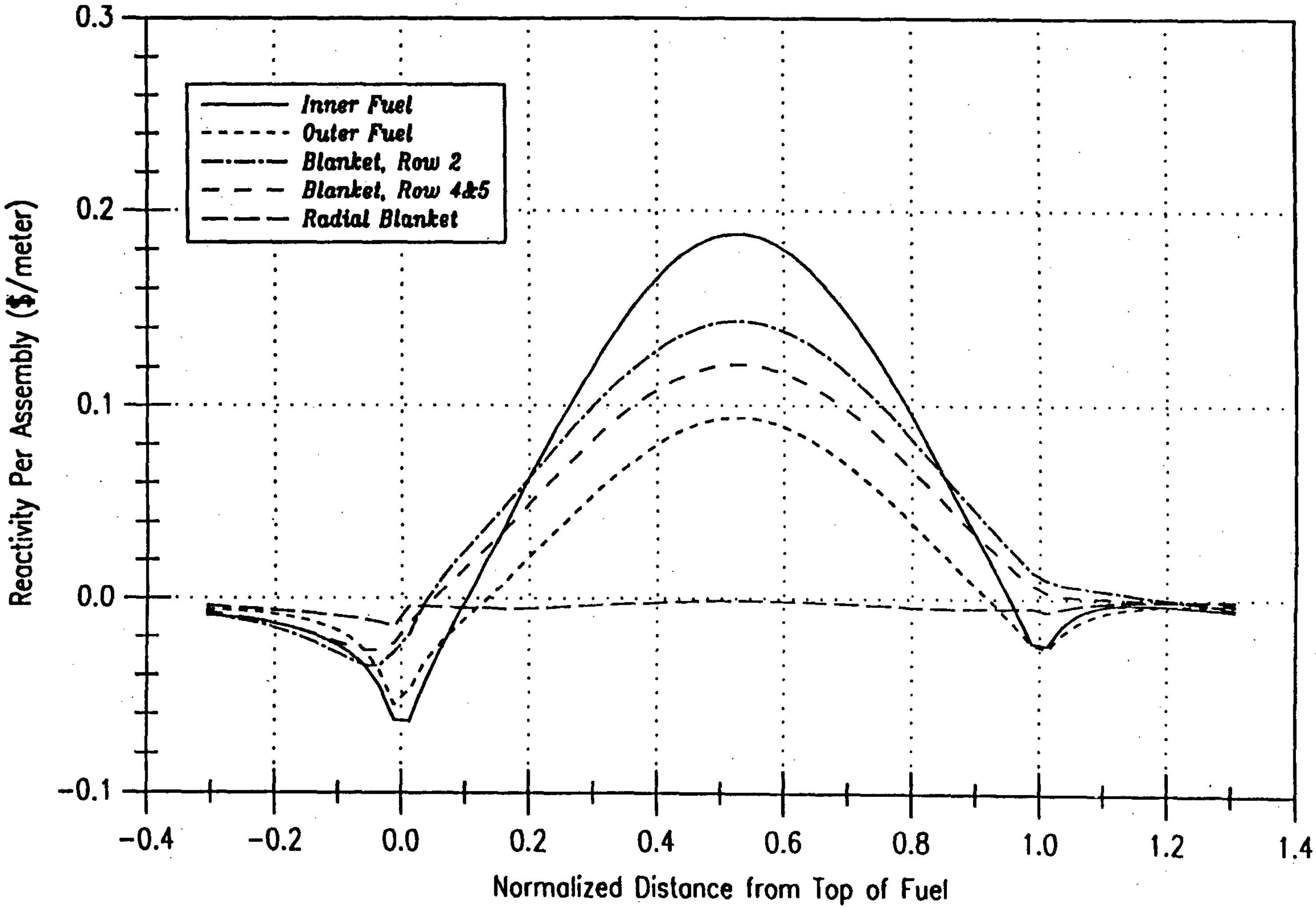

Figure G.4.5-2 AXIAL DISTRIBUTIONS OF VOID WORTH BY REGIONS, FUEL AND BLANKETS

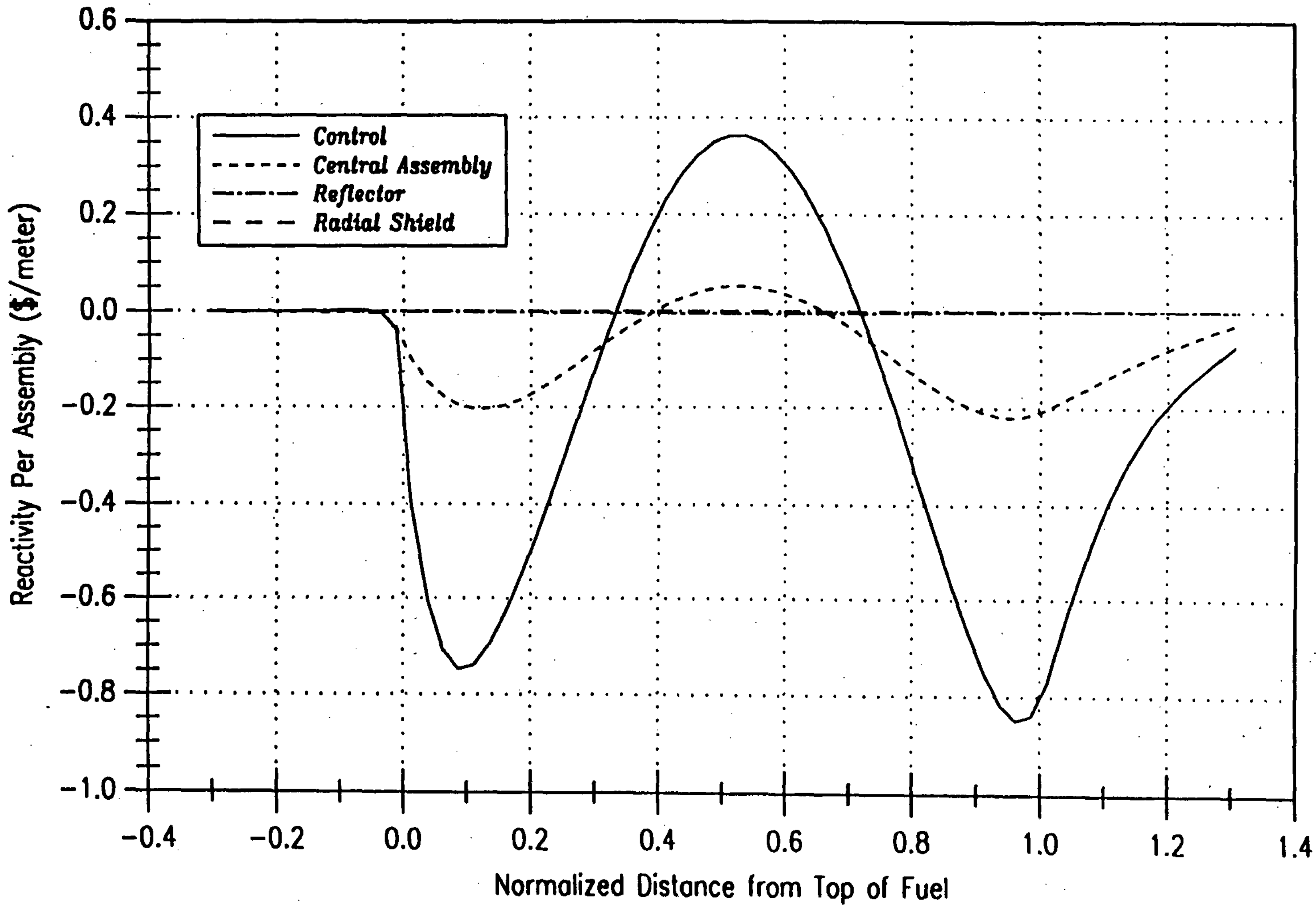

Figure G.4.5-3 AXIAL DISTRIBUTIONS OF VOID WORTH BY REGIONS, OTHER ASSEMBLIES

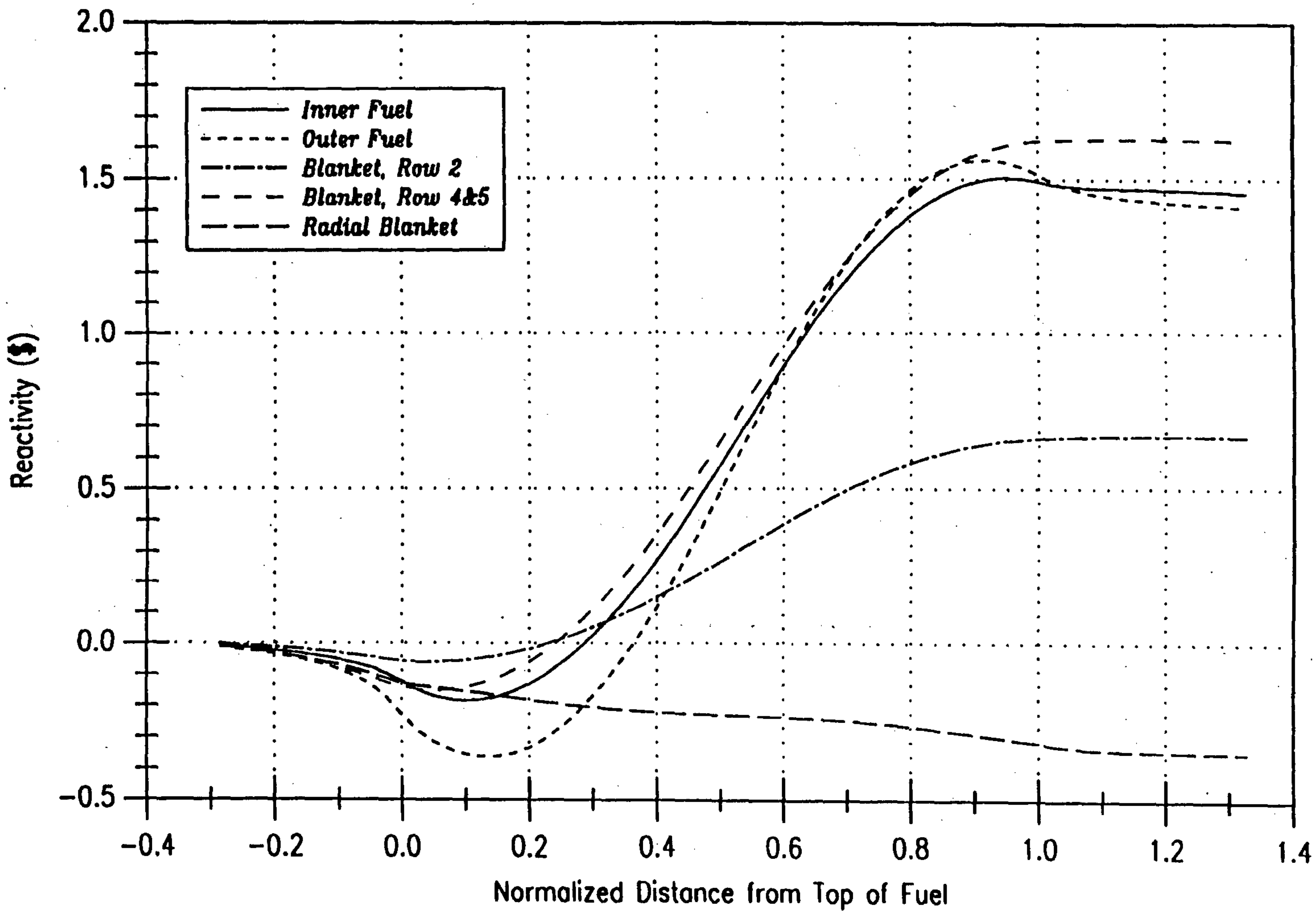

Figure G.4.5-4 AXIALLY CUMULATIVE VOID WORTH, FUEL AND BLANKETS

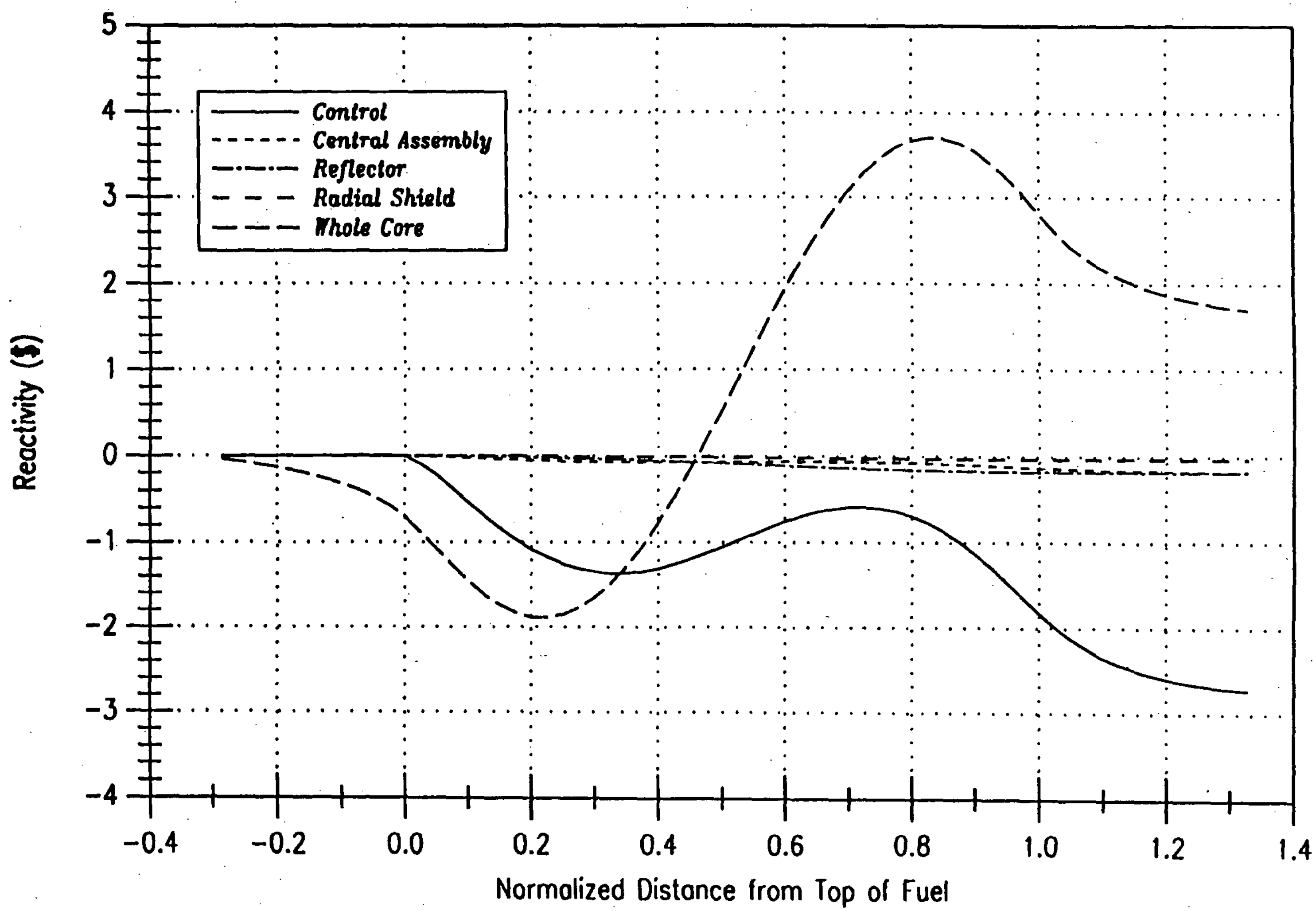

Figure G.4.5-5 AXIALLY CUMULATIVE VOID WORTH, OTHER ASSEBMLIES

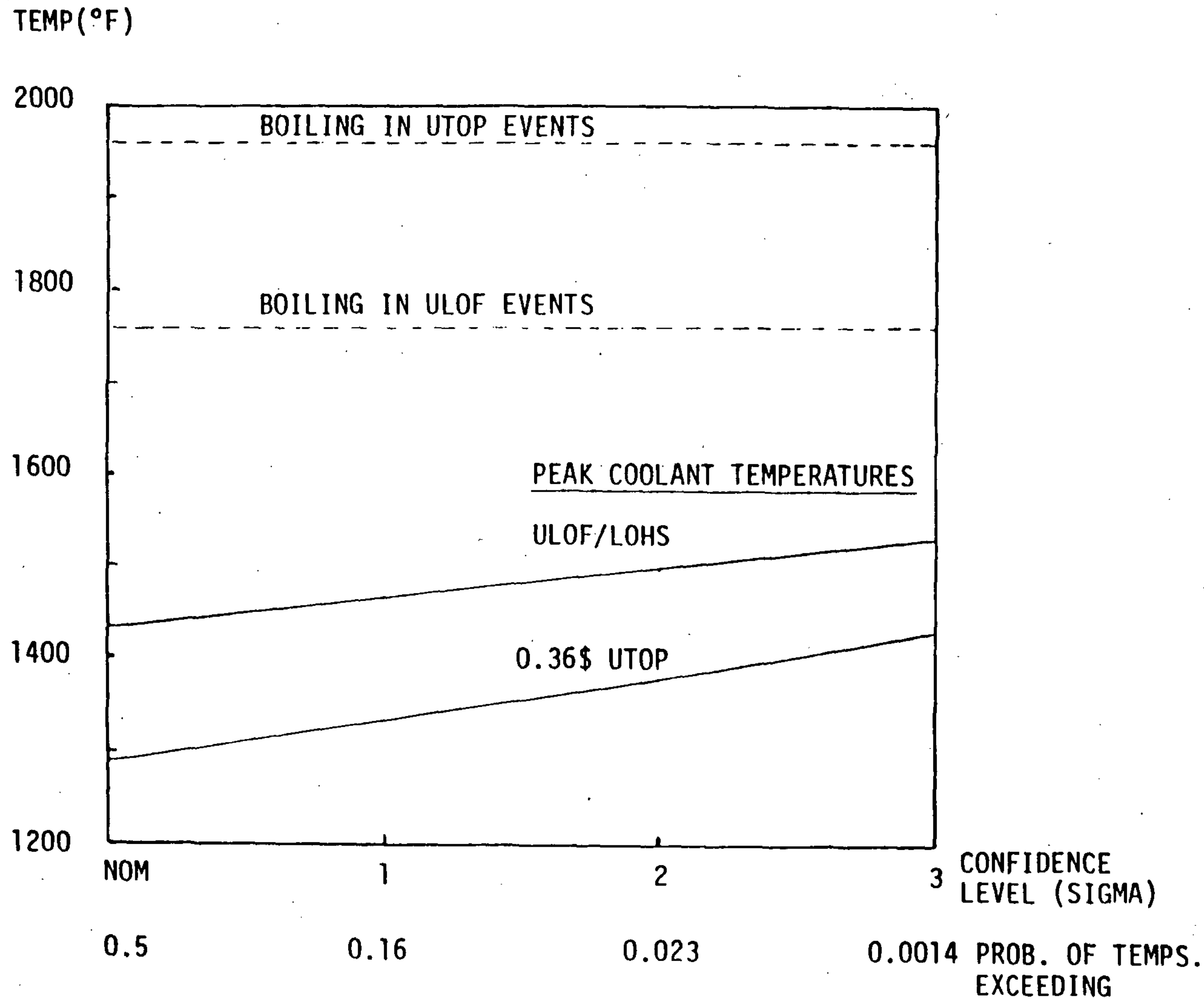

Figure G.4.5-6 PEAK COOLANT TEMPERATURES AS FUNCTIONS OF REACTIVITY FEEDBACK UNCERTAINTY LEVELS

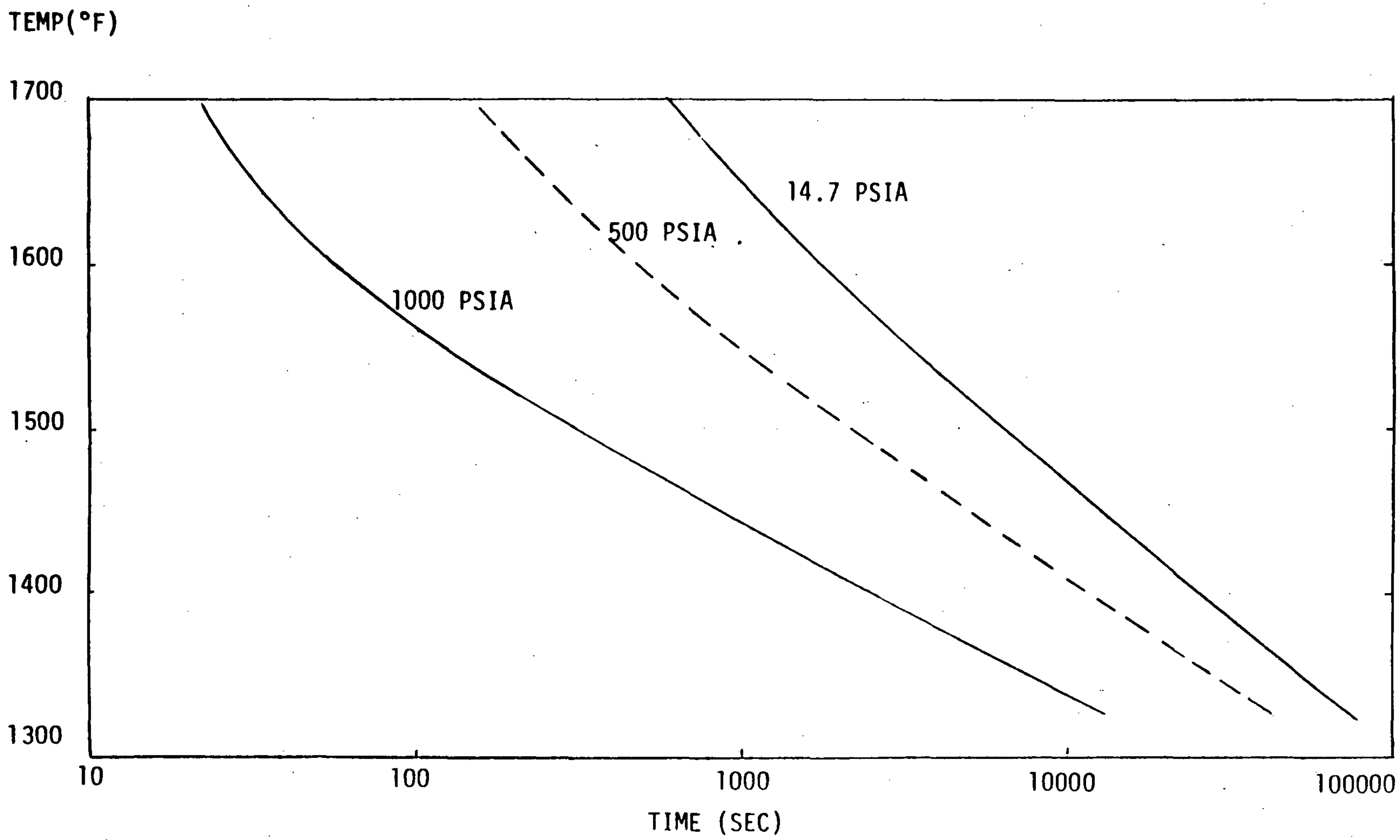

Figure G.4.5-7 TIME TO CLADDING FAILURE FOR INTERNAL PIN PRESSURES OF 14.7, 500 AND 1000 PSIA

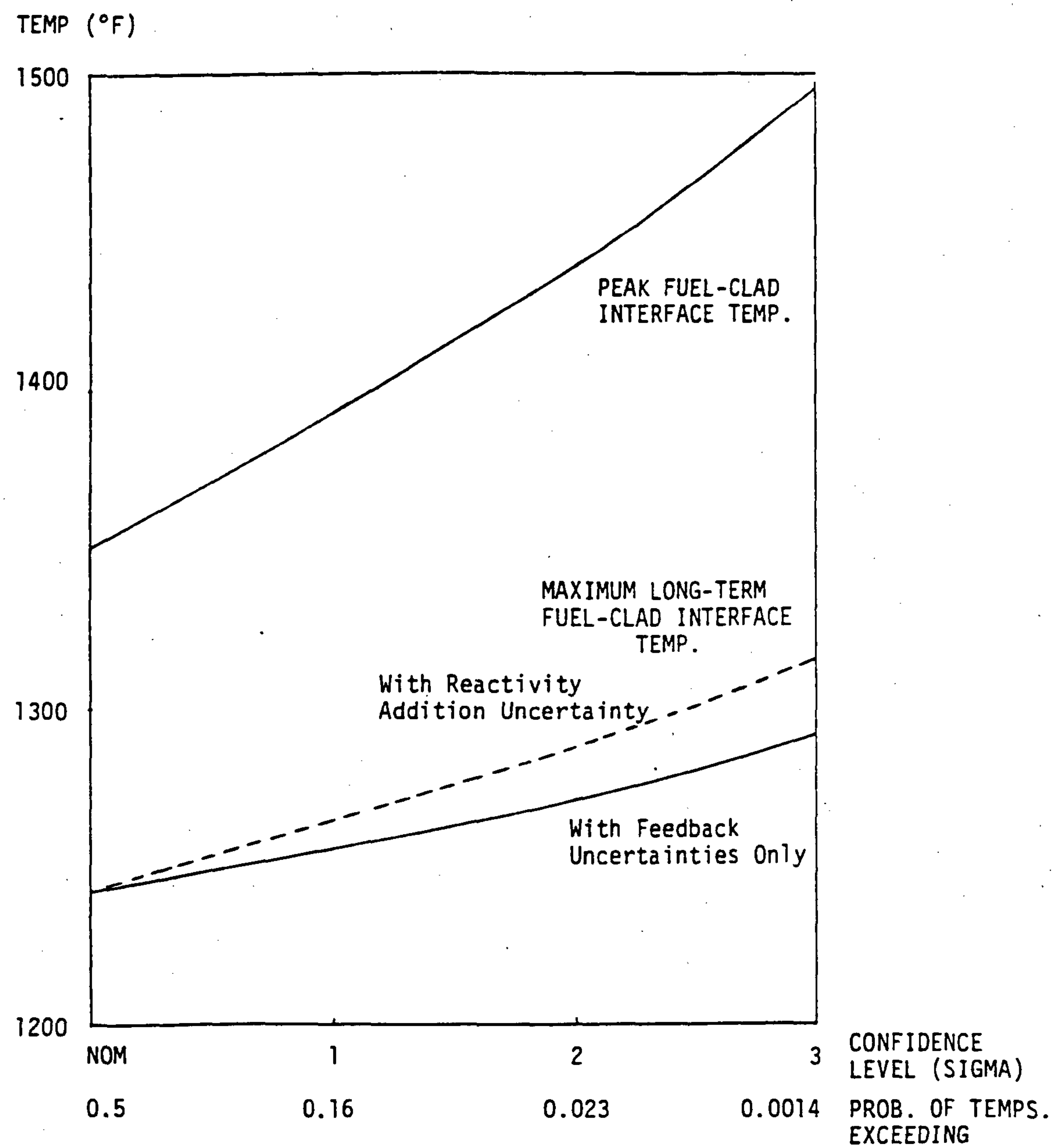

Figure G.4.5-8 PEAK FUEL/CLAD INTERFACE TEMPERATURES AS FUNCTIONS OF REACTIVITY FEEDBACK UNCERTAINTY LEVELS FOR UTOP EVENTS

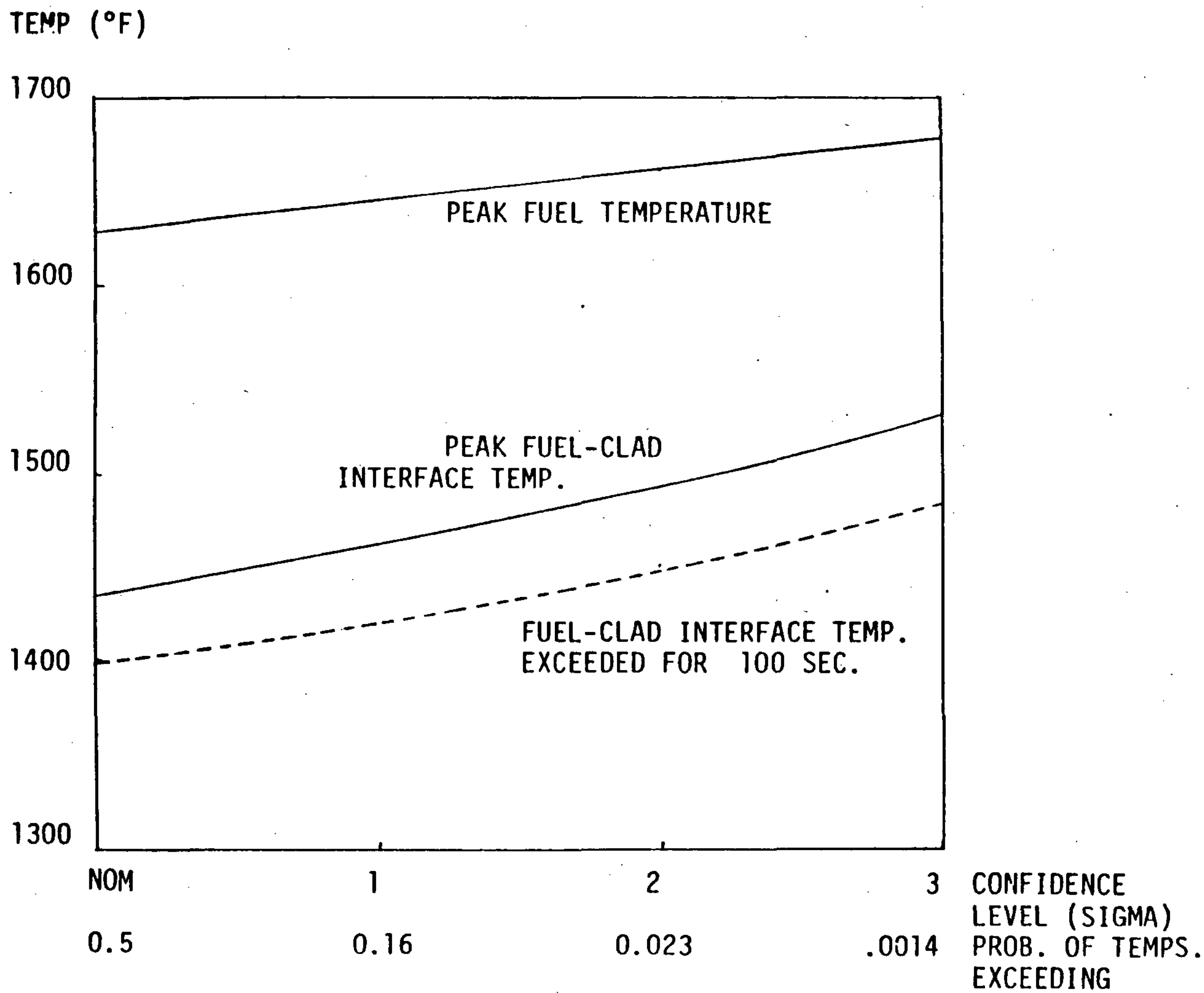

Figure G.4.5-9 PEAK FUEL/CLAD INTERFACE TEMPERATURES AS FUNCTIONS OF REACTIVITY FEEDBACK UNCERTAINTY LEVELS FOR ULOF/LOHS EVENTS

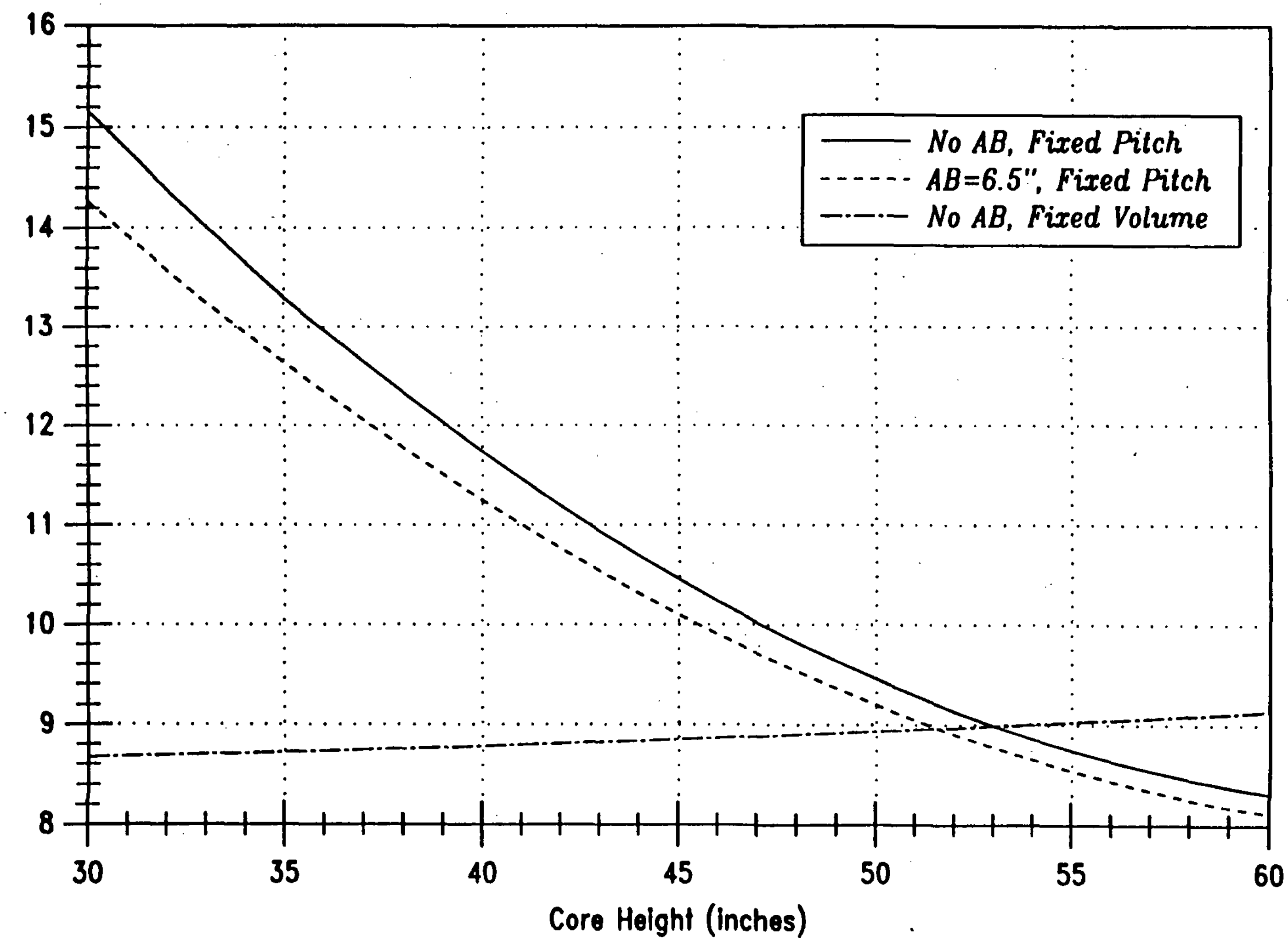

Figure G.4.5-10 PEAK LINEAR POWER (KW/FT)

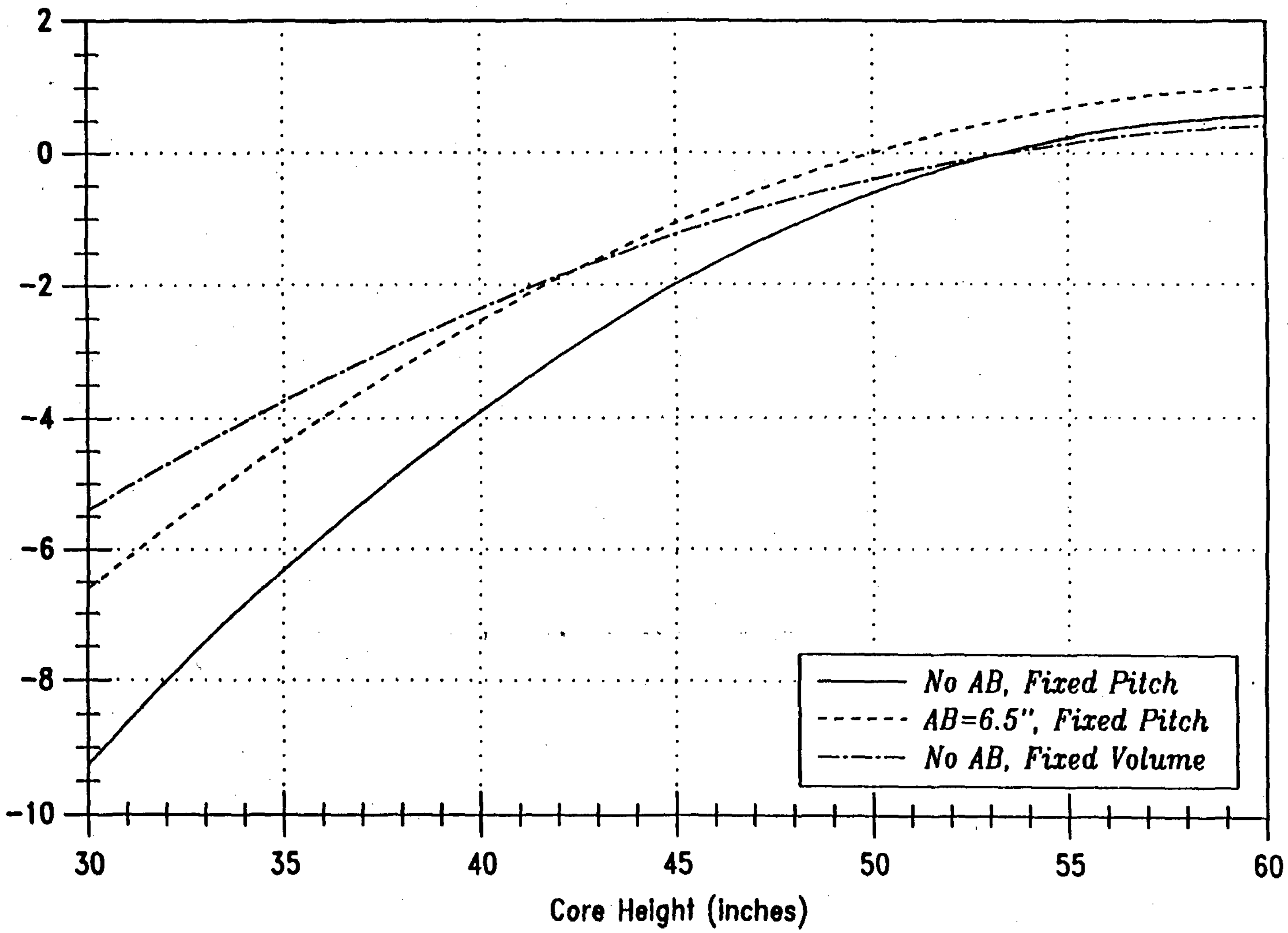

Figure G.4.5-11 BURNUP REACTIVITY ($)

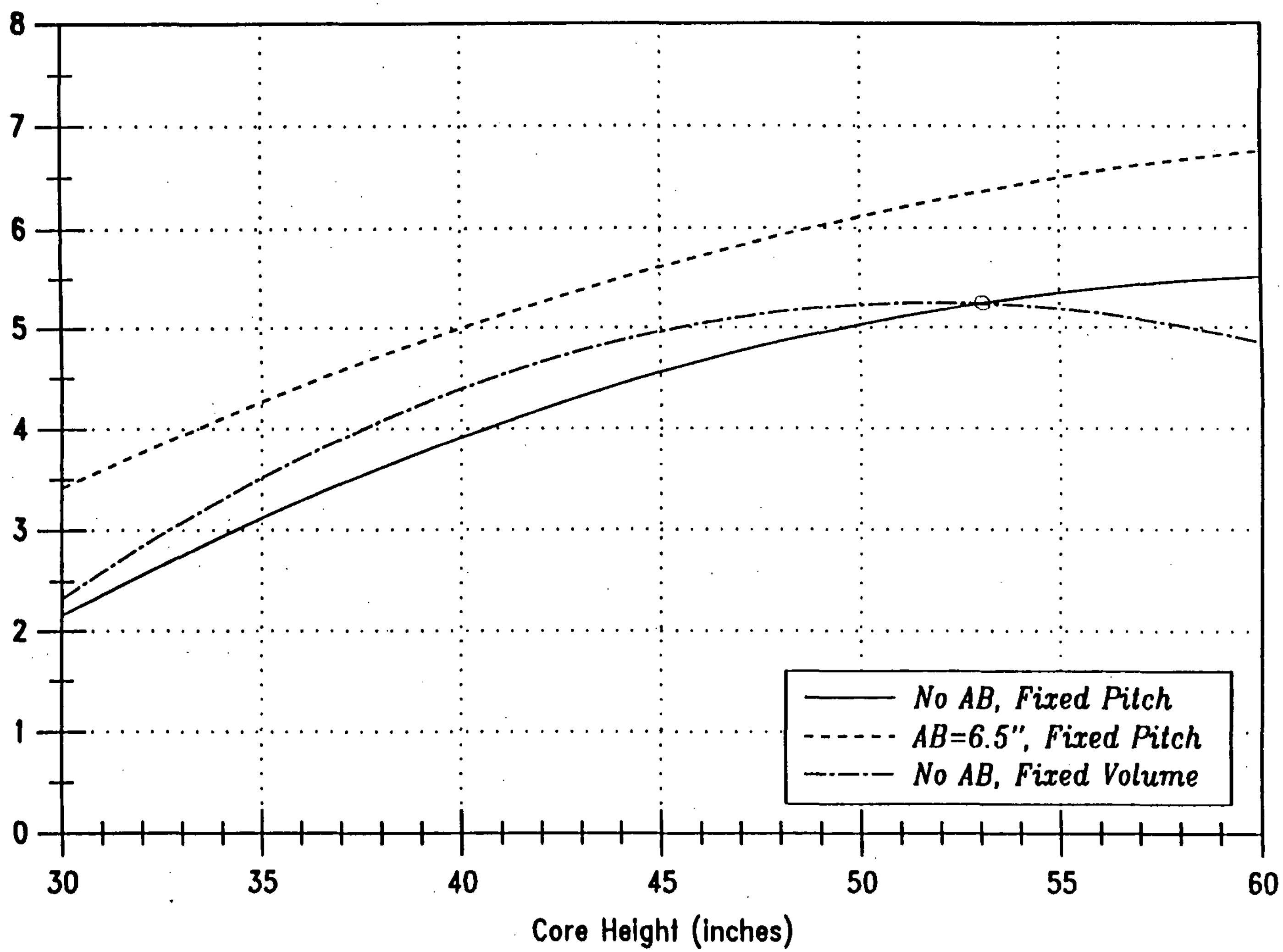

Figure G.4.5-12 FUEL + BLANKET VOID WORTH ($)

G.4.6 Flow Blockage

G.4.6.1 SER Position on Single Assembly Flow Blockage

The draft SER (Section 3.1.2.3) defines the following acceptance criteria: If the ALMR design is to be accepted for NRC certification of a design without a containment building, specific measures must be taken to ensure that no core melt accidents, no accidents with significant positive reactivity feedback, or other accidents with potential for a large radiological release are in the EC-I, EC-II, or EC-III spectrum.

Bounding Event No. 7 (flow blockage of a single fuel assembly) was not addressed by GE in the PSID. In Section 15.10.2 of the draft SER, the Staff judged this event to have the potential for sodium boiling and possible energetics. The Staff's concern is not related to blockages which might develop during power operation (e.g., assembly inlet blockage, assembly outlet blockage, local blockage within core region), but to fabrication errors which could result in a totally blocked assembly being inserted into the reactor (Section 4.4.5 of draft SER). As stated in draft SER Section 4.4.6, "... prevention/detection of assembly flow blockage due to fabrication errors remains an open issue."

The following discussion focuses on the specific issue of single assembly flow blockage. The other bounding events which were judged in the draft SER not to meet the EC-III acceptance criteria are discussed in Section G.4.16, Safety Analysis.

G.4.6.2 Summary of Flow Blockage Event and Its Consequences

A preliminary assessment of the consequences of a total flow blockage of a single core assembly at startup indicates it will be possible to show that the event meets the EC-III criteria.

Reactor startup is initiated from a bulk primary sodium temperature of 550°F with full core flow. The power is increased to 25% full power in no less than 30 minutes. Sodium boiling and centerline fuel melting occur in the interior pins of a totally blocked fuel assembly when the power reaches about 4% to 8% of full power.

The maximum possible reactivity additions due to assembly voiding and subsequent fuel slumping, at the low power levels at which they occur, will not result in propagation to other assemblies. As in the blockage accident in the Fermi fast reactor, molten fuel movement is expected to generate a net reduction in reactivity. The event is expected to be terminated by operator action either (1) after observation of an unexpected reactivity change, (2) on response to delayed neutron (DN) signals resulting from refluxing and repeated expulsion of sodium, with failed fuel particles, from the blocked assembly, or (3) in the worst case, in response to a DN signal resulting from penetration of molten fuel-clad alloy into edge channels of an adjacent assembly with flowing sodium. Given operator action, the event terminates in a benign configuration, with minimal core damage.

It is important to note that even in the extremely unlikely case that the event propagates rapidly to other assemblies and generates a mild energetics event, the primary coolant boundary can withstand an energetic event greatly in excess of the maximum possible with the ALMR metal fueled core. (See Section G.4.19, Mitigation of Severe Core Accidents.)

Additional analysis and experimental tests are required to confirm the tentative conclusions reached herein. Such studies are planned at Argonne National Laboratory as part of Phase III of the Integral Fast Reactor (IFR) metal fuel development program.

G.4.6.3 Analysis of Assembly Flow Blockages

Five subjects will be discussed:

- The probability of a major flow blockage developing during reactor operation
- The probability of a total flow blockage due to a fabrication error
- Comparison to the Fermi reactor flow blockage incident (flow blockage during operation)

- o The consequences of a total flow blockage of an assembly due to a fabrication error (flow blockage at startup)

- o Relevant studies planned for the Integral Fast Reactor Program, Phase III

G.4.6.3.1 Probability of Flow Blockage During Operation

Flow blockages during power operation could occur in one of three regions of a core assembly: inlet, outlet and within the active core. The inlet region of each assembly, and the associated inlet module which feeds the assembly, are designed following the philosophy of the Clinch River LMR design with multiple holes and flow paths. The probability of a flow blockage in this region is estimated as less than 10^{-8}/plant-yr. The assembly outlets (Figure G.4.6-1) are designed with flow blockage bypass ports, which prohibit total flow blockage even by a flat plate completely covering the exit of the assembly. The probability of a flow blockage in this region is estimated as less than 10^{-8}/plant-yr. Due to the excellent compatibility of the ALMR metal fuel with sodium, the likelihood of local flow blockages in the core region of an assembly, due to reaction products, is less than 10^{-7}/plant-yr.

It is believed that a flow blockage during operation is a very low probability event (in EC-IV). The Staff has not requested in the draft SER that a flow blockage during operation be investigated as a bounding event. Rather, Staff concerns appear to be focused on blockages due to fabrication error. As stated in Section 4.4.6 of the draft SER, "... prevention/detection of assembly flow blockage due to fabrication errors remains an open issue."

G.4.6.3.2 Probability of Fabrication Error Leading to Flow Blockage at Startup

It is believed that the probability of a fabrication error leading to total blockage of an assembly upon insertion in the core is also in the residual risk category EC-IV (less than 10^{-7}/plant-yr), for the following reasons:

- The driver fuel and blanket assembly designs virtually eliminate the potential for total blockage. The only assembly internals are the pin bundle and bundle supports. There are no orifice plates (with drilled flow holes), as all orificing is done in the separate inlet modules. There is no shielding block (with drilled flow holes), as the shielding in each assembly is accomplished using a 40-inch long solid steel extension at the bottom of each fuel pin. The only holes drilled for flow passages are the inlet holes in the assembly nose piece, which are on the outside of the assembly and easily visible. In addition, gas flow tests are planned to be run on each assembly prior to insertion in the reactor, to verify that the flow path is open.

- Flow through the inlet modules will be ensured by design, fabrication procedures and tests prior to the first loading of fuel into the reactor. The inlet modules, pictured in Figure G.4.6-2, are inserted into the inlet plenum, and locked into the grid plate. They are designed to last the lifetime of the reactor, but can be removed if necessary. The inlet modules provide multiple sets of holes and flow paths. Each module will be flow tested prior to insertion in the new reactor. In addition, methods to verify in-reactor flow and orificing for the inlet modules are being investigated.

G.4.6.3.3 Comparison to Fermi Reactor Blockage Incident

It is instructive to compare the predicted behavior of the ALMR during an assumed assembly flow blockage event with that of the Fermi reactor during its blockage incident.

__Description of Incident__

Fermi was a metal fueled, sodium cooled, three-loop fast reactor; its first core was rated at 200 MWt. The Fermi fuel elements were made of U-10Mo alloy with metallurgically bonded Zr cladding. Previous irradiation tests of the fuel had demonstrated that the burnup limit under Fermi operating conditions would be 0.8 at%. The lead fuel was at approximately 0.4 at% burnup at the time of the blockage event.

The following summary of the Fermi incident is taken from Reference G.4.6-1.

"At about 2:20 p.m. [on October 5, 1966], the rise in power was again begun [from 2 MWt] and continued until 8 MWt was reached, where there was a brief hold to put the reactor on automatic control. The power rise then continued on automatic control to 13 MWt when, at 2:45 p.m., there was another brief hold to place the boiler feedpump on automatic control. The automatic increase in power level was then resumed; at about 3:00 p.m. with the power at 20 MWt, the reactor operator observed variations in the automatic control system in which the rate of change of neutron level, the dn/dt signal, became erratic. The problem had been experienced in the past at about the same power level and was thought to be a noise pickup in the control system. Although there was no indication that control had been affected, the reactor was put on manual control and the dn/dt signal observed until the apparent noise disappeared. When there was no indication of instability or of a nuclear transient, the reactor was once again put on automatic control and the increase in power resumed. At approximately 3:05 p.m., the feed water flow control system was put on automatic. The reactor operator once again observed variations on the dn/dt indicator.

"At this time it was noted by the licensed staff member in charge of the operation that the control rods appeared to be withdrawn further than normal for this power level; both the shim and regulating rods were withdrawn approximately nine inches. It has since been determined that their normal withdrawn elevation for the existing inlet temperature, flow rate, and power level would have been about six inches, assuming an equal withdrawal of the rods. Because of the seemingly abnormal rod positions, the power increase was interrupted and the core outlet temperature instruments were checked. Abnormally high sodium outlet temperatures were being indicated by thermocouples over subassemblies M-140 and M-098. At this time, the inlet and bulk outlet sodium temperatures were 535°F and 600°F, respectively.

"Reactor power at this time was 31 MWt, as indicated by a calibrated neutron detector. At 3:09 p.m., there were alarms from the radiation monitors in the upper reactor building ventilation exhaust ducts. The containment building was automatically isolated; there was no one inside it at this time. A Class I radiation emergency, a class of radiation emergency which is lowest in severity and is restricted to a specific locality of the plant, was announced. The detector in the fission product detector building also exceeded its set point, isolating the fission product detector system.

"When the radiation monitor alarms were received, a power reduction was begun in accordance with operating procedures. By 3:20 p.m., assessment of the reactor information was completed, and the reactor was manually scrammed. Analysis of the calibrated neutron detector trace showed a prompt drop from 26 MWt to 3.3 MWt, demonstrating that all of the six safety rods scrammed properly."

The cause of melting was a blockage of subassemblies by a loose zirconium liner which had become detached from the sodium inlet flow guides, apparently due to vibration. The blockage resulted in the flow in adjacent assemblies M-098 and M-127 being reduced to 0 to 3% of nominal, the flow in assembly M-122 to somewhat less than 10% of nominal, and the flow in assembly M-140 to about 30% of nominal (Reference G.4.6-2). The Fermi core assembly flow inlet was simply an open hole in the bottom of each assembly, such that a flat plate could completely block multiple assemblies. Subsequent LMRs, e.g., EBR-II, FFTF, CRBRP, and ALMR, have adopted a different inlet design with multiple side holes that can not be blocked in this manner.

Post-test analysis showed that fuel melting started at a reactor power level of 9 to 18 MWt and cladding failure occurred at temperatures of 2100°F to 2600°F (Reference G.4.16-2). Significant fuel melting occurred in the two highly blocked assemblies, but some pin geometries were maintained at many elevations and there were no blockages at the assembly exits. There was no melting in the partially blocked assemblies or any other assemblies. Holes were found in adjacent faces of the two highly blocked assemblies. A hole was found in a second duct face of one of these assemblies; however, the duct of the adjacent unblocked assembly was not penetrated. Upon cooldown, the two highly blocked assemblies fused together.

Reference G.4.6-2 concludes that "if more assemblies had been blocked initially, it is not expected that the results would have been qualitatively different from those which occurred, other than that more assemblies would have melted. If the fuel had been irradiated more, it is expected that the fission gas would have been released from molten fuel after a short time, and the behavior would then have been the same as for low-burnup fuel."

Lessons Learned for ALMR

Key aspects of the Fermi blockage accident pertinent to the postulated ALMR assembly blockage are:

(1) The Fermi blockage occurred during a startup and resulted in initial fuel melting at a power level of 9 - 18 MWt. This corresponds to the 4 - 8 % power level (19 - 38 MWt) predicted for fuel melting in a totally blocked assembly in the ALMR. However, there are significant differences in power density and fuel form between the two reactors.

(2) Although flow blockage of two assemblies was essentially complete (less than 3% nominal flow), the event progressed slowly. More than nine minutes elapsed between initiation of fuel melting and the radiation monitor alarm, and more than 20 minutes between fuel melting and reactor scram. Even in the two totally blocked assemblies, the geometry of some pins was maintained over most cross-sections, and there were no blockages at the assembly exits.

(3) Fuel melting resulted in a less reactive configuration, not a more reactive state.

(4) The event did not propagate to other, unblocked assemblies. At least 20 minutes elapsed between fuel melting and reactor scram in the Fermi accident. During this time, fuel melting was confined to the two highly blocked assemblies. There was no indication that the event was progressing to other assemblies.

The Fermi flow blockage incident, in its initiation, development, termination and final configuration, strongly supports the ALMR single assembly flow blockage preliminary analysis presented herein and the basic conclusion that the occurrence of such a hypothetical event would be benign and satisfy the EC-III criteria.

G.4.6.3.4 Evaluation of Flow Blockage Event at Startup

The key assumptions of this analysis are:

- The flow blockage of the affected assembly is total.

- The blockage is due to a fabrication error that was not detected, and the assembly is blocked when it is placed in the core.

It follows that the fuel in the affected assembly is unirradiated, and the critical time to be investigated is reactor startup and rise to power.

Reactor startup consists of a number of steps. After closing and sealing the reactor, checkout tests are conducted at 400°F at low flow (about 5%). The primary pumps are then turned on to full flow and the reactor is heated to 550°F by the heat generated within the primary pumps. The control rods are withdrawn and the power slowly increased to 25%, over a minimum of 30 minutes. After a short hold time, the power is ramped to 100% at a maximum rate of 1%/minute.

A preliminary bounding analysis of the thermal response within a blocked assembly during this reactor startup sequence has been made by Argonne National Laboratory. For a totally blocked, inner ring fuel assembly, sodium boiling will begin in the central flow channels of the pin bundle at about 4% to 8% full power (based on uncertainties in reflux cooling and radial heat transport), followed shortly by cladding failures at the core midplane. At this time, from 20% to 40% of the total fuel in the central rings of the assembly could exist as molten fuel-clad alloy. The impact of boiling heat transfer and the degree of fuel-cladding contact resulting in eutectic penetration represent additional uncertainties that could raise the power at which failure occurs.

The maximum reactivity addition due to complete sodium voiding of any single assembly is 0.15$. (See Section G.4.5, Void Worth.) This reactivity addition is negligible in terms of its effect on other assemblies. No pin failures are predicted in any other assembly. It is noted that a 0.40$ UTOP bounding event is analyzed in Section G.4.16, Safety Analysis, and shown to satisfy the EC-III criteria.

A reactivity addition of greater than about 0.08$ would produce an overpower transient of sufficient magnitude to cause the reactor to be scrammed. However, the reactivity addition due to voiding of a single assembly can be essentially zero, depending upon the assembly's core location, and therefore cannot be relied upon to cause a scram that terminates the event. If some sodium flows through the blocked assembly, or if the initial expulsion of sodium from the assembly carries some molten fuel-clad alloy fragments into the primary sodium circuit, the delayed neutron monitors in the IHXs would alarm, followed by operator action to shut down the reactor. The fission gas monitors are ineffective as detectors of blockage, because of their long time constants (about 1 hr) and the assumption that the fuel in the blocked assembly is unirradiated.

Concurrent with sodium boiling and cladding rupture, the fuel in the central pins of the totally blocked assembly melts, beginning at the core midplane and extending vertically with time. It is anticipated that the net fuel movement will be away from the core midplane, with an associated reduction in reactivity. It is possible, however, that the net fuel slumping could be towards the core midplane, adding reactivity to the core. The limiting case is that all fuel within the assembly becomes molten and compacts about the core midplane. This will add less than 0.90$ reactivity for the maximum worth fuel assembly. A sudden addition of 0.90$ reactivity would result in about a ten-fold increase in power, conservatively assuming sudden fuel movement and a "prompt jump" in reactivity, without time for temperature-related negative reactivity feedbacks to occur. With the core power level at less than 10% at the time of melting and slumping, the transient power level will be limited to less than nominal full power. With full flow in all other core assemblies, no pin failures are anticipated in other assemblies.

If the fuel slumping in the blocked assembly generates a sufficiently large reactivity addition (>0.08$), the reactor will scram on the power increase and the event will be terminated. This is unlikely, as the molten fuel movement will probably result in a less reactive core state. It is anticipated that reactor startup will continue with molten fuel contained within the blocked assembly. The event may be terminated by operator action upon observation of an unexpected reactivity change. If not, natural refluxing and repeated expulsions of sodium from the blocked assembly are expected to carry fuel particles and associated delayed neutrons to the DN monitors causing an alarm and subsequent operator action. In the limit, molten fuel-clad alloy will progress through the duct walls of the blocked assembly and its neighbor, resulting in flowing sodium (in the adjacent assembly) carrying fuel particles with delayed neutrons to the DN monitors. The DN monitors will generate an alarm, resulting in subsequent operator action to terminate the event.

Even in the extremely unlikely case that the event were to propagate rapidly to other assemblies and generate a mild energetics event, the primary coolant boundary can withstand an energetic event greatly in excess of the maximum possible for the ALMR metal fueled core. (See Section G.4.19, Mitigation of Severe Core Accidents.)

A partially blocked assembly, with varying amounts of flow through the assembly, has also been considered. The event proceeds slower and the consequences are believed to be less severe than those for total flow blockage. For example, with 5% flow, fuel pin failure is calculated to occur at 20 - 35% full power and will be at the end of the active core region, rather than near the core midplane. The partial assembly flow will be effective in carrying particles to the DN monitors thus generating an alarm and operator action to scram the reactor, terminating the event before significant fuel motion.

Significant experimental and analytic work are required to confirm the tentative conclusions reached here. Such studies are planned at Argonne National Laboratory as part of Phase III of the IFR metal fuel development program.

G.4.6.3.5 Relevant Studies Planned for IFR Phase III

Significant related experimental and analytic work is planned for Phase III (1991-1995) of the Integral Fast Reactor (IFR) Program (Reference G.4.6-3). WBS 310, In-Reactor Experiments, will establish a database for validation of fuel disruption analysis capability for both transient overpower and loss of flow sequences by running multi-pin bundle transient tests in TREAT. WBS 320, Safety Analysis and Model Development, will complete development of models of metallic fuel response to severe accident conditions. WBS 330, Ex-Reactor Experiments, will investigate core melt phenomena in detail, including melt relocation, behavior of fission gas in molten fuel, effect of iron in melt composition, and fuel dispersal. The response of metal fuel to local faults will be studied in WBS 342, Local Faults.

G.4.6.4 References

G.4.6-1 McCarthy, J. F., "Monthly Report No. 3 - October 1966, Compilation of Current Technical Experience at Enrico Fermi Atomic Power Plant," APDA-CFE-3 (issued May 1967).

G.4.6-2 Friedland, A. J., "Thermal-Hydraulic Analysis of the October 5, 1966 Fuel Melting Incident in the Enrico Fermi Reactor," ADPA-LA-5, July 1968.

G.4.6-3 Argonne National Laboratory, "Integral Fast Reactor Program, Program Plan," September 1989 (draft)

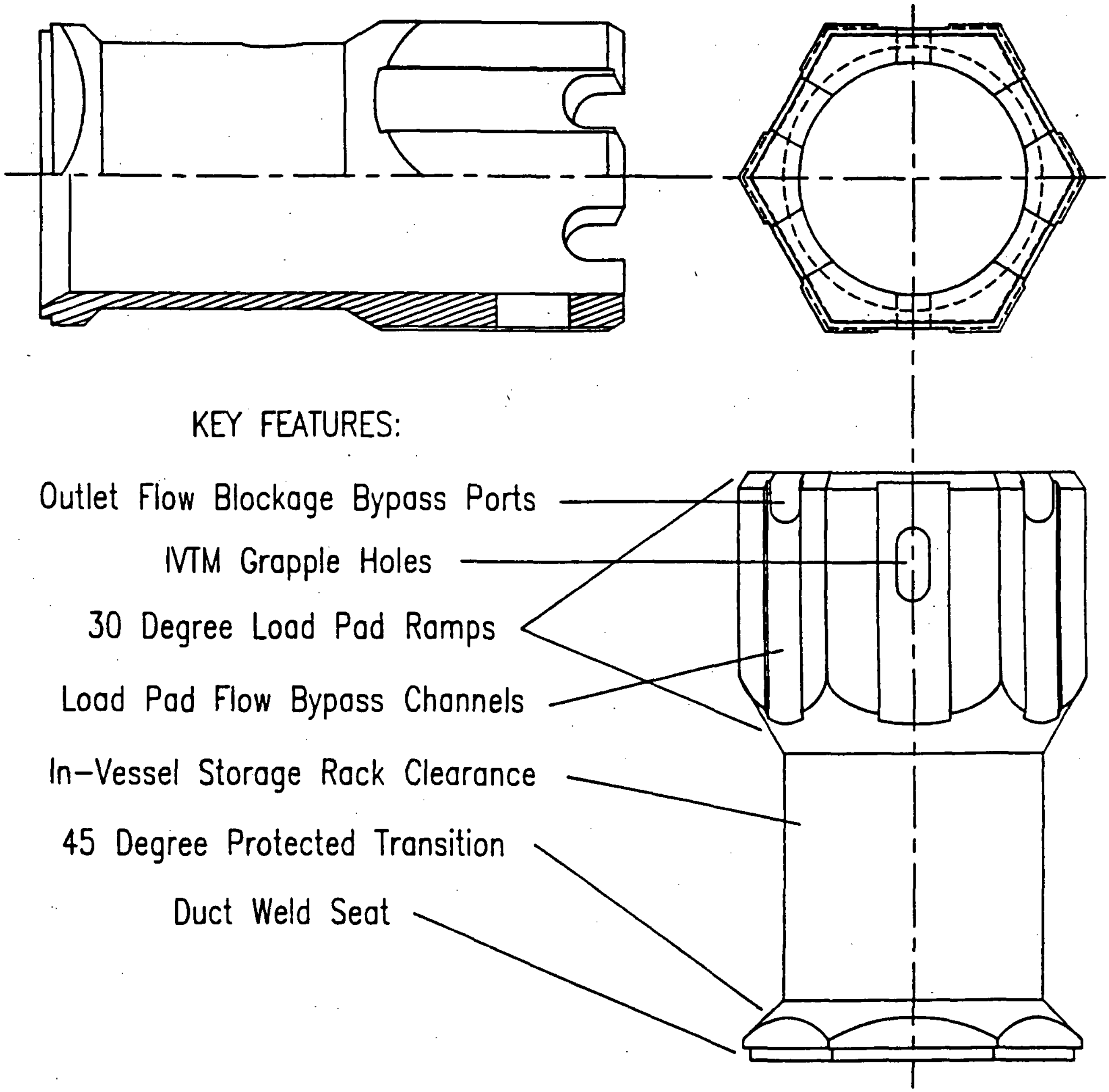

Figure G.4.6-1 - ALMR CORE ASSEMBLY OUTLET DETAIL

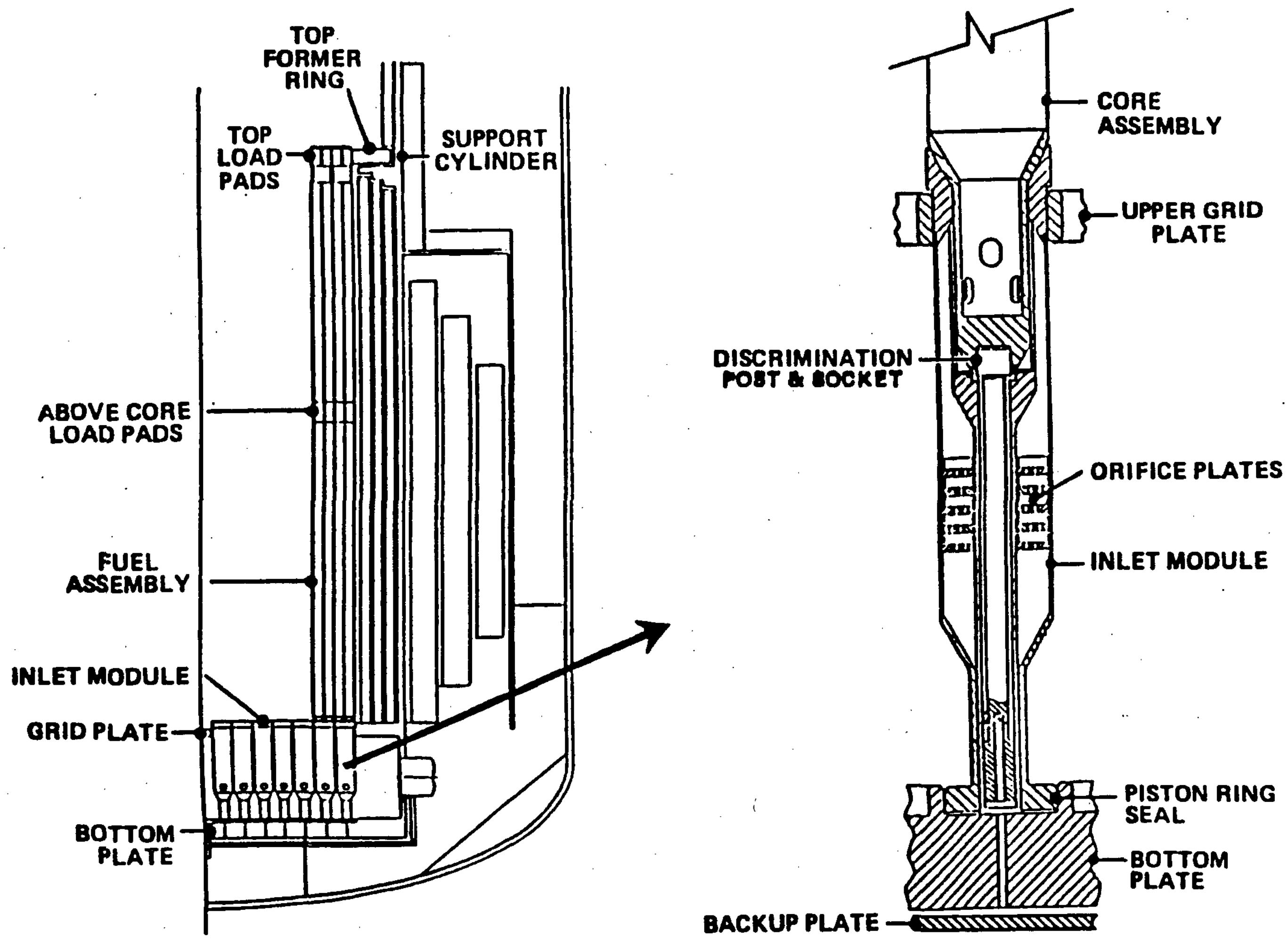

Figure G.4.6-2 - ALMR INLET MODULE

G.4.7 Electromagnetic Pumps

G.4.7.1 SER Position on EM Pump Issues

In the draft SER, six design and safety related issues on the electromagnetic (EM) pump are identified. These issues are summarized below. In some cases further elaboration of the issue is made to convey GE's understanding of the issue.

G.4.7.1.1 Absorber Bundle Flotation (SER Section 4.4.5)

The absorber bundles must be designed to avoid absorber bundle ejection or flotation at the maximum core flow. This is of particular concern if the EM pumps were operated at full flow during refueling when the drivelines are withdrawn above the absorber bundle assemblies.

G.4.7.1.2 Seismic Isolation of the Synchronous Machines (SER Section 5.4.1.3)

In the 1986-1987 PRISM design, the synchronous machines are located in non-seismically isolated equipment vaults. Lack of seismic isolation of the synchronous machines requires flexing of the cables running from the seismically isolated pump. The Staff has concerns on how these cables would be run, and the seismic effects on coastdown performance.

G.4.7.1.3 Adequacy of Coastdown Performance (SER Section 5.4.5.1)

It is essential that adequate coastdown performance of the EM pumps and synchronous machines be defined and verified through a test program. This stems from a concern that loss of one of the four EM pumps may have an adverse effect on reactor operation. If operation of one EM pump is lost due to a cable break or one downcomer pipe break (part of reactor internals), the core flow reduces quickly. If there is also a failure to scram, it is important to obtain coastdown of the three remaining EM pumps to avoid sodium boiling during the ULOF transient.

G.4.7.1.4 Control Rod Insertion Indication Before EM Pump Trip (SER Section 7.2.5.3)

Due to the importance of maintaining forced coolant flow while at power, the EM pump circuitry should be modified to include the prevention of pump trip until indication of control rod insertion has been received following a reactor trip signal.

G.4.7.1.5 Synchronous Machine Performance Monitoring (SER Sections 8.3.1 and 8.3.2)

The performance of the synchronous machine needs to be monitored during power operation to ensure adequate performance during coastdown. The ability to monitor the necessary parameters and provide for electrical disconnection under all potential loss-of-power conditions with safety-related equipment appears to be critical.

G.4.7.1.6 Pump Failure and Risk Estimates (SER Section A.4.2)

Failure of one pump coastdown has potential to lead to sodium boiling if the other three pumps do not coast down normally. Reliance on an external electrical source for coastdown appears to be a design weakness. Justification for the low probability of pump failure cannot be made because of lack of data and details. Failure rate estimates and risks (PRA) need to be substantiated. Several areas needing further study include:

a. EM pump, synchronous machine, and power supply interactions
b. Environmental interactions
c. Effects of aging on coastdown system
d. Effect of periodic maintenance, testing, human error
e. Ability to test and monitor system status during normal operation

G.4.7.1.7 EM Pump Performance Under Extreme Conditions

Several concerns and questions were raised during a NRC Staff - GE meeting in January 1990. These are:

a. What are the failure mechanisms of the EM pumps?

b. Do the pumps trip out at the Curie point on the ULOHS transient or are they tripped by the breakers?

c. Do the pumps coast down if tripped at the Curie point?

d. What happens if sodium leaks through the pump casing and contacts the electrical leads inside?

G.4.7.2 Reference EM Pump Design Features

This section describes the EM pump and its auxiliary equipment. The auxiliary equipment is comprised of the synchronous coastdown machine, the ground fault detection system, power conditioning unit (PCU) and the flow controller. An overall schematic of the pump and the auxiliary equipment is shown in Figure G.4.7-1.

There are four EM pumps in the reactor for circulating primary sodium through the core and the two in-vessel intermediate heat exchangers. Each EM pump is connected to its own auxiliary equipment. The four EM pumps are located in the reactor vessel, and the auxiliary equipment is located in the seismically isolated equipment vaults adjacent to the reactor. Thus, the pumps, and both the safety related auxiliary equipment (synchronous coastdown machines, EM pump circuit breakers, overcurrent protectors), and the non-safety related auxiliary equipment (ground fault detection system, PCUs, PCU circuit breakers, flow controllers) are all located in the seismically isolated portion of the reactor facility.

G.4.7.2.1 EM Pump

The EM pump is shown in Figure G.4.7-2. Each pump is approximately 40 inches in diameter, 41 feet long, and weighs about 22.5 tons. The pumps are installed through penetrations in the fixed portion of the reactor closure, and are located in the annular volume above the core shared with the intermediate heat exchangers. Primary sodium coolant is drawn from an inlet plenum beneath the pump. This plenum is filled with cold sodium from the IHX which has passed through the fixed core radial shield region.

As depicted on Figure G.4.7-2, sodium enters through a large annular opening at the bottom of the pump. Within the pump, the sodium converges to the tapered inlet section of the pump duct where the velocity increases from approximately 30 fps to the design velocity of approximately 50 fps through the remaining 2/3 of the pump duct. The sodium discharge at the top of the pump passes radially outward into a plenum from which it is piped to the core inlet structure. There are three reactor internal structure seal plate interfaces for the piston ring seals of the pump - one seal plate at the pump inlet and two seal plates near the top of the pump forming part of the discharge plenum.

The EM pump is seal welded to the reactor closure to eliminate leaks and is secured by bolts. The EM pump assembly is removable from the reactor by unbolting holddown segments and cutting the seal weld.

The pump is self-cooled in that the heat generated by electrical losses in the stator is transferred to the surrounding sodium. Most of this heat energy is transferred through the duct wall into the pumped sodium since this path provides the best thermal coupling to the heat source. A smaller portion is transferred radially outward through the stator support cylinder. Since all heat losses are transferred into the primary sodium coolant, the adverse effect of heat loss on overall plant efficiency is reduced.

The pump is comprised of the stator assembly, center iron assembly, stator housing, and riser section. The stator assembly is comprised of copper coils separated by magnetic iron lamination rings which, respectively, generate and conduct the electromagnetic force for pumping the sodium. The center iron assembly, comprised of magnetic iron material, completes the magnetic path. The stator housing provides a sealed enclosure for the stator coils and lamination rings. The riser section connects the lower stator portion of the pump to the closure penetration, and houses the power and instrumentation cables running from the stator to the reactor closure. The EM pump closure forms part of the primary reactor boundary.

Within the volume of the stator assembly are the stator iron and coils. The basic unit of the stator iron is the lamination. Laminations are stamped in two sizes, the larger size being used for the phase break laminations. Material with high magnetic saturation induction characteristics and a high Curie point temperature is used for the laminations. The laminations are treated to minimize electrical conduction and eddy-current losses between adjacent plates.

Coil electrical insulation consists of conductor-to-conductor insulation and coil-to-ground insulation. Conductor-to-conductor insulation is provided by dry-wrapped amber mica tape. Coil-to-ground insulation is provided by either amber or white mica tape, with Secon No. 5 potting compound applied as a binder between tape layers.

The pump stator is located radially outward from the pump duct. It is in an inert gas-filled enclosure formed by the outer pump duct wall, the external stator support cylinder, and the end forgings to which these cylindrical sections are welded. The electrical power leads are routed from the stator enclosure through conduits across the pump outlet plenum, and into the lifting/handling structure which extends upward through the reactor vessel closure.

The center iron assembly, which provides a magnetic boundary for the "air-gap" (flow annulus) flux, is also in an inert gas-filled enclosure. It is composed of rings of magnetic steel laminations, the principal plane of which is oriented radially and parallel to the centerline of the pump. The enclosure is formed by the inner pump duct wall and an internal support cylinder. The center iron assembly is installed in the center region of the pump near the end of the fabrication sequence, and thereby is an integral part of the pump as installed in the reactor vessel.

The EM pump is equipped with instrumentation to monitor its condition and performance. This instrumentation and the information usage are listed in Table G.4.7-1.

Table G.4.7-1

EM PUMP INSTRUMENTATION

Measured Parameter	Information Usage
Pump Discharge Sodium Pressure	Control performance and diagnostics
Insulation and Lamination Temperatures	Relate to coil and magnetic core material properties for the detection of impending failures, comparison to allowable limits
Duct Temperature	Performance analysis, comparison to analytical predictions
Sodium Leakage	Detection of internal sodium leakage (failure of seal welds).
Stator Internal Gas Pressure	Loss of inert gas from the stator cavity. (leak monitoring)

The pump characteristic curves of head versus volume flow, at various voltages and system pressures, are plotted in Figure G.4.7-2. The flowrate is determined by the driving frequency. The frequency can be adjusted along with the voltage to provide an operating condition which is both efficient and stable at the design point of 115 psi and 11,500 gpm (46,000 gpm for all four pumps). The EM pump operating parameters are summarized in Table G.4.7-2.

G.4.7.2.2 Synchronous Coastdown Machine

The synchronous coastdown machine is a three phase, electric machine connected in parallel with the windings of the EM pump. The synchronous machine provides reactive power to the EM pump for power factor correction during normal operation. Following loss of power to the EM pump, the synchronous machine converts the kinetic energy of the spinning rotor and flywheel into electrical energy required by the EM pump to provide primary flow coastdown as shown in Figure G.4.7-1.

Table G.4.7-2

PRIMARY SODIUM EM PUMP PARAMETERS

Parameter	Value
Flow Rate (gpm)	11,500
Developed Head (ft Na)	304
(psi)	115
Sodium Inlet Pressure (psia)	$\geq$11.9
Sodium Inlet Temperature (°F)	637
Line Voltage (Volts, rms)	627
Phase Current (amps)	1247
Frequency (Hz)	15.6
Power In (kW)	1186
Efficiency (%)	48
Stator Poles	8
Stator Coils	96
Coils/Pole/Phase	4

The synchronous machine is shown schematically in Figure G.4.7-3. The two main components of the synchronous machine are the rotor and the stator. The rotor consists of an even number of magnetic poles, each with a field coil of alternating polarity assembled around a central rotating shaft. Each pole has a field coil. The stator consists of windings placed in equidistant slots in the stator surface such that the coil sides are one pole division apart. A DC current, called the excitation current, is fed to the rotor windings which creates a magnetic flux around the rotor. As the rotor spins, the flux sweeps by the stator windings. The changing flux interacts with the stator winding coils and generates an output voltage.

The DC current to the rotor field windings is provided by a DC to AC coupling with the exciter which in turn is AC coupled to the pilot exciter. The pilot exciter has a set of stator coils that sense the changing flux from a rotating permanent magnet driven by the same shaft as the synchronous machine flywheel. These coils develop an AC current proportional to the frequency of the rotation, and through an internal regulation circuit provide AC current to the exciter stator coils. The exciter rotor, also driven by the synchronous machine shaft, develops an AC current related to the exciter stator current and rotor frequency. The AC current is then rectified and fed to the main synchronous machine rotor as the excitation current. The synchronous machine is self-excited since once the machine has started, the excitation current is generated through its own rotational motion without need for external power.

During normal operation, the synchronous machine generates a back electromotive force (EMF) that provides power factor correction and enables proper operation of the PCU. The components of the synchronous machine are selected so that it functions as a synchronous condenser, providing a capacitive load to the PCU for improved power factor correction over the operating range.

At equilibrium, the synchronous machine rotates at a frequency equal to the PCU frequency, and uses only a small fraction of the input energy to overcome frictional losses and maintain this frequency. Thus, the synchronous machine provides rotational energy for coastdown at virtually no power cost.

If PCU power to the EM pump is lost due to loss of site power, failure of the PCU, opening of the PCU or EM pump circuit breakers, or opening of the overcurrent protection device, the synchronous machine supplies electrical power to the EM pump and allows the pump to coast down. In this mode, conversion of rotational energy to EM pump voltage and frequency is done passively, without adjustments for pumping efficiency optimization and power factor correction. Simulation results show that even with less than optimum pump operation, there is sufficient coastdown flow to mitigate all EC-III events, including the bounding events.

The synchronous coastdown machine is equipped with instrumentation to monitor its condition and performance. A list of this instrumentation and the purpose for each measured parameter are listed in Table G.4.7-3.

Table G.4.7-3

EM PUMP SYNCHRONOUS MACHINE INSTRUMENTATION

Parameter	Purpose
Input/Output Voltage Input/Output Current	Determine load and control, for protective and diagnostic purposes.
Output Power	Control and waveform analysis for performance monitoring, diagnostics, maintenance, and the evaluation of power factor correction, switching transients, etc.
Output Frequency (Shaft Speed)	Performance, diagnostics, and maintenance
Shaft Torque	Performance, diagnostics, and maintenance
Rotor Electrical Voltage Current	Measure output of the synchronous machine's pilot exciter and regulator circuitry for control and diagnostic purposes
Vibration	Performance, diagnostics, and maintenance
Bearing Temperatures	Performance, diagnostics, and maintenance
Winding Temperatures	Performance, diagnostics, and maintenance

G.4.7.2.3 Ground Fault Detection and Limitation System

A ground fault detection and limitation system (schematically illustrated by Figure G.4.7-4) consists of a grounding scheme and a current measurement/limiting resistor. This system performs the requisite protective functions, and supplies the needed diagnostics for the on-line, continuous evaluation of the electrical insulation and detection of any possible deterioration.

All of the elements from the secondary of the isolation transformer through the EM pump and the synchronous machine are electrically isolated from ground. The only ground point in the electrical power system for the primary heat transport system is a grounding resistor located in the PCU. A separate safety ground wire connects the frame of the synchronous machine, the magnetic core laminations and Faraday shield of the input transformer, the housing of the EM pump, and the metal enclosure of the PCU to the facility electrical ground at the PCU. This safety ground conductor provides personnel and equipment protection.

If a fault to ground occurs within the EM pump or the synchronous coastdown machine, the current must flow through the fault into the grounding circuit, then through the grounding resistor to complete the circuit back to the source. The value of the grounding resistor is selected such that the maximum current available from the controller will be limited to a specified value (for example, 5 amperes) such that any single ground fault will be unable to provide sufficient energy for burnthrough of the EM pump duct.

The voltage drop across the grounding resistor is continuously monitored. Operator alarm and pump trip setpoints are selected such that protective action is taken before damage can occur. The protective action taken may be to shut down the reactor. If the ground leakage current is large enough, the PCS automatically opens the PCU circuit breakers allowing the EM pump to coast down.

If a scram should occur during a ground fault condition, the synchronous machine and EM pump will continue to provide coastdown. Once the EM pump circuit breakers are opened by the RPS to interrupt current from the PCU, the synchronous machine is the only source of power for continued operation of the EM pump. Since the breakers have removed any communication with the PCU and the ground resistor, the ground fault is now isolated and the current through the fault does not have a return path to the synchronous machine.

The alarm, shutdown, and diagnostic circuitry is based upon measuring the voltage across the ground resistor due to current flow through the resistor. The amplitude of the voltage is a direct measure of the leakage currents to ground in the circuitry of the EM pump and the synchronous machine. Any ground fault current from any of the three phases will flow through the single resistor. Thus, the total voltage can be utilized for the generation of an alarm or shutdown signal. Since the voltage is a direct measure of the leakage currents to ground, voltage levels below the setpoint may be utilized as a diagnostic measure of the insulation integrity.

G.4.7.2.4 Power Conditioning Unit

The power conditioning unit (PCU), illustrated in Figure G.4.7-5, is a solid state device comprised of three stages. The first stage is an AC to DC convertor, which takes the three phase power and rectifies it to a DC voltage. Rectifiers in the input converter stage of the PCU are turned on for only a portion of each 60 Hz sine wave. The period for which they are turned on determines the DC operating voltage. The second stage filters the DC current and makes it available as a current source to the output section of the power conditioning unit. The third stage of the PCU (the DC to AC inverter) consists of a set of solid state switches that sequentially switch the DC current source to provide three phase power to the EM pump and synchronous coastdown machine.

This type of inverter makes no attempt to establish a voltage waveform on the output terminals. Instead, it forces a three-phase square wave current to flow in the load windings. Since the synchronous machine is a rotating device, it will exhibit a sinusoidal counter electromotive force which will help to establish a nearly sinusoidal voltage waveform.

The PCU controller shown in Figure G.4.7-5 receives a demanded frequency setpoint from the primary flow controller. Through use of a voltage controlled oscillator (analog controller approach) or a frequency controller (digital controller approach), it sets the switching frequency of the DC to AC inverter section of the PCU, and hence establishes the frequency of the PCU output voltage. The PCU controller also receives a demanded PCU output voltage setpoint from the primary flow controller. Through the use of a dynamically compensated outer PCU output voltage control loop and an inner dynamically compensated PCU input current control loop, the time that the rectifiers in the PCU input converter stage are turned on is varied. This turn-on time period is varied until the PCU output voltage equals the voltage setpoint.

Sensors are provided as part of the PCU to permit the parameters shown in Table G.4.7-4 to be monitored.

Table G.4.7-4

PCU PERFORMANCE PARAMETERS MONITORED

Parameter	Purpose
Output Voltage Output Current Output Power	Determine load and control, for protective and diagnostic purposes.
Output Frequency	Control and waveform analysis for performance monitoring, diagnostics and maintenance.
Filter, Capacitor and Semiconductor Temperatures	Detection of impending failure and panel overtemperatures.
Ground Fault Current	Measure insulation performance and detect - Failure of insulation system - Output for a ground fault trip - Identification of the phase a ground fault is associated with.

G.4.7.2.5 Primary Flow Controller

A flow controller controls the EM pump sodium flowrate. The controller maintains desired steady sodium flowrates during normal power operation, or follows specified flow versus time profiles for flow reductions or fast runback operations. The flow control system also provides circuit breaker trips for protection of the PCU, EM pump, and synchronous machine. A block diagram of the sodium flow controller is shown in Figure G.4.7-5.

The controller uses process pressure and temperature feedback measurements to compute the reactor primary flow, and continuously adjusts the PCU frequency and voltage setpoints to provide the required flow. The controller also receives diagnostic inputs for operator information and control of the PCU breakers.

The flow controller interfaces with the control room operator's console and permits the operator to start up and shut down the EM pump and synchronous machine. The operator interface provides displays of process and equipment instrumentation readings, and indications when alarm settings have been exceeded.

The controller has the following capabilities:

- Adjust the PCU voltage and frequency setpoints, based on flow feedback and supervisory controller demands, in order to maintain maximum efficiency during operation.

- Limit the maximum voltage and frequency demands to the PCU to prevent EM pump and synchronous machine operating limits from being exceeded.

- Monitor EM pump, synchronous machine, and PCU performance by running performance diagnostic models fed by real time sensor data.

G.4.7.2.6 Equipment Locations

The locations of the EM pump and its auxiliary equipment within the reactor facility are shown in Figure G.4.7-6. The reactor, the head access area (HAA) with containment dome, RVACS structures, and equipment vaults are supported on a common platform which is mounted on seismic bearings. This provides horizontal seismic isolation for these structures and the equipment contained therein.

The electrical and instrument equipment for the EM pumps is housed in below grade, reinforced concrete, tornado hardened Seismic Category I vaults integral with and located adjacent to HAA enclosure walls as shown in Figure G.4.7-6. These seismically isolated equipment vaults house, structurally support, and environmentally protect the EM pump flow controllers, power conditioning units, the safety-grade synchronous coastdown machines, and related equipment.

G.4.7.3 Response to SER Positions

This section responds to the EM pump design and safety related issues summarized in Section G.4.7.1.

G.4.7.3.1 Absorber Bundle Flotation

Operation of the EM pumps during refueling will not result in absorber bundle ejection or flotation. An absorber bundle design requirement is that the bundle not be lifted (floated) by hydraulic forces when the driveline is disconnected and the pumps are operated at full flow, and also that the bundle be able to fall into the core in a few seconds against full flow following a scram signal. The inadvertent pump startup accident is most likely to happen during refueling if the operator accidentally starts the pumps. The calculated value of the pressure drop across the absorber bundle necessary to lift the bundle is 6.6 psi, which is over nine times the design operating value of 0.72 psi at full flow. The value of the bundle flowrate corresponding to the 6.6 psi pressure drop is 85,900 lb/hr. Calculated values are for nominal conditions, and the effect of uncertainties must be included. Uncertainties to be considered are those affecting the estimated hydraulic resistances, pump flow, bundle weight, plus any uncertainty associated with the analytical modeling. The margin between operating and floatation conditions, however, is so large that when accounting for uncertainties, the "no flotation" criterion can still be satisfied for bundle flowrates approximately double the 25,000 lb/hr value currently used. Periodic scram testing will assure absorber bundle drop against full flow.

G.4.7.3.2 Seismic Isolation of the Synchronous Machines

The reference ALMR design has been modified to seismically isolate the synchronous machines. The seismically isolated platform supporting the reactor and auxiliary equipment has been enlarged to provide space for the EM pump synchronous machines. This improves the reliability of the design since there is no relative seismic induced motion between the EM pumps and their synchronous machines.

G.4.7.3.3 Adequacy of Coastdown Performance

During normal operation the synchronous machine is in parallel with the EM pump (Figure G.4.7-1). When power to the EM pump is interrupted by opening the circuit breaker between the pump and the power conditioning unit (the power source), the synchronous machine becomes in series with the EM pump and acts as a generator. The kinetic energy of the spinning motor (flywheel) is converted to electric power and supplied to the EM pump for flow coastdown.

The synchronous machine flywheel and rotor are designed to provide at least as much coastdown flow as defined in Figure G.4.7-1. This flow versus time profile maintains the required flow-to-power ratio during core shutdown to minimize thermal shock, and provides sufficient flow coastdown to prevent overtemperature challenges during loss-of-flow events. To match the core power reduction, flow is reduced rapidly to about 60% of full flow, then decreased more gradually. With this approach, flow coastdown is sustained over a relatively long period of time because there is a low pressure drop throughout the reactor primary circuit, and the inertia in the synchronous machine is not quickly dissipated.

The planned test program for the EM pump includes component testing of a full size EM pump and synchronous machine in a separate test facility. Testing will be in sodium over a range of prototypic and extreme conditions. Included in the test program will be coastdown tests to verify the performance of the synchronous machine. Coastdown tests will be run covering the range of expected reactor sodium flow and temperature conditions. The EM pump and coastdown machine will also be tested as part of the ALMR prototype test. Finally, the synchronous machine is a Class 1E component and will be qualified in accordance with these requirements.

G.4.7.3.4 Control Rod Insertion Indication Before EM Pump Trip

Trip logic in the RPS circuitry delays pump trip following scram until indication of control rod insertion is received. The reactor scram sequence logic is shown in Figure G.4.7-7. The logic includes confirmation of rod insertion before cutting power to the EM pump and initiating flow coastdown. Decreasing flux is used as the measurement to verify absorber bundle insertion. Once the RPS senses that the core flux is rapidly decreasing, indicating that the rods are inserting, it sends a signal to open the EM pump circuit breakers. Opening of the breakers shuts off power to the EM pumps, which initiates the coastdown.

G.4.7.3.5 Synchronous Machine Performance Monitoring

As described in Section G.4.7.2.2 and listed in Table G.4.7-3, an extensive set of diagnostic sensors is provided with each synchronous machine to continuously monitor and detect problems before they can influence performance. This automatic and continuous monitoring is performed by the diagnostic and maintenance function of the plant control system. Any sensor whose failure could impact the safety performance of the synchronous machine is Class 1E, and is continuously monitored by the RPS.

The most sensitive monitor of synchronous machine performance is the EM pump. The EM pump pressure responds "instantaneously" to changes in the frequency, amplitude, and phase of the power supplied both from the PCU and the synchronous machine. The EM pump pressure is monitored continuously with quad redundant Class 1E sensors during normal operation and during and following transient conditions.

Performance models of each EM pump, synchronous machine, and PCU are stored in the PCS primary flow controller. The model parameters are determined through extensive factory and startup testing. Data from these models are continuously compared against actual operating data. Significant differences or trends are automatically flagged to indicate incipient or actual component failure. For example, if the magnetic properties of an EM pump deteriorate, a larger than normal EM pump electrical current will be required to produce the required primary coolant flow and pump outlet pressure. At the same time, the synchronous machine and PCU models would show no problems. Therefore, that EM pump would be identified as the faulty component.

If a major bearing failure occurred in a synchronous machine, the failure would be detected by elevated bearing temperatures, increased vibration, and reduced power factor correction. The EM pump and PCU models would show the proper terminal voltage waveform relationships and magnitude ratios needed for operation, and would correctly diagnose the EM pump as operating properly. The synchronous machine would therefore be identified as the faulty component. For another example, if a fault occurs in a synchronous machine (e.g., shorted turns in the field winding) or a power conditioning unit (e.g., an open capacitor), the currents and voltages at the terminals of these components would change. However, the model would show that the correct current and voltage phase relationships and magnitude ratios still exist in the EM pump. Therefore, the pump would be diagnosed to be operating properly, and the synchronous machine or power conditioning unit would be identified as the faulty component.

Potential failure modes have been identified for the synchronous machine. For each failure mode, sensors are provided for timely recognition of the fault. The potential failures and the detection parameters are summarized in Table G.4.7-5.

G.4.7.3.6 Pump Failure and Risk Estimates (SER Section A.4.2)

The basis for the EM pump failure rate and risk estimates in the PRA have been further evaluated and these results will be presented in Section G.4.18 of Appendix G of the PSID. The essence of these evaluations are summarized below for effects of system interactions, environmental interactions, aging, maintenance, and performance monitoring.

a. In the event of power interruption to the EM pumps, a coastdown of three EM pumps is required to prevent reactor core temperatures from exceeding acceptable limits following a highly unlikely scram failure event. It should be noted that this conclusion was reached prior to the addition of gas expansion modules (GEMs) to the ALMR core. The GEMs provide additional margin for this event. The EM pumps require power during a coastdown for about two minutes. A separate synchronous machine is used to supply coastdown power to each EM pump.

Table G.4.7-5

EM PUMP SYNCHRONOUS MACHINE FAILURE MODES

Failure	Information Usage
o Winding fails open	Input current, voltage and loop pressure/flow
o Winding turn-to-turn failure	Input current, voltage and loop pressure/flow
o Winding shorts to ground (ground fault	Ground fault detection and diagnostics
o General insulation degradation	Ground fault detection and diagnostics
o Rotor diode failure	Current, voltage and loop pressure/flow
o Regulator failure	Current, voltage and loop pressure/flow
o Pilot exciter fails to provide proper voltage and current	Current, voltage and loop pressure/flow
o Bearing fails	Bearing temperature, vibration, rotor speed
o Excessive vibration	Bearing vibration
o Shaft torque incorrect	Shaft torque
o Shaft rotational speed improper	Shaft speed, current, voltage and loop pressure/flow

There is no system interaction among the four EM pump systems other than obtaining power from the same site power supply system. The successful coastdown of each EM pump is fully dependent upon the successful operation of that EM pump and its synchronous machine, and its safety grade Class 1E breakers which open to disconnect the system from the normal power supply system. Backing up each circuit breaker are individual Class 1E current overprotection devices.

b. The only credible common cause failure which fails two or more coastdown systems simultaneously is a very large magnitude earthquake. Since all of the coastdown equipment is seismically isolated, the effect of a large magnitude earthquake is considerably mitigated.

Fire, smoke, and loss of HVAC are not postulated to be major common cause risk factors due to the separation and three-hour fire barriers of the EM pump auxiliary equipment vaults, and the short coastdown requirement (two minutes). The plant design provides significant protection against internal flooding.

c. The aging of EM pump auxiliary equipment is not expected to be a major risk factor for the following reasons:

(1) The EM pump auxiliary equipment will have the same design life as the rest of the reactor components.

(2) The synchronous machines will be operating continuously, and any effects due to aging will be readily detected. Preventive measures will then be taken. Reactor trip coupled with one synchronous machine failure may affect the plant capacity factor, but has little effect on reactor safety unless the remaining three pumps fail to coast down at the same time, and the control rods fail to scram. This combination of failures is highly unlikely. As mentioned earlier, the addition of GEMs provides additional margin for this event.

(3) The safety grade auxiliary support equipment will be maintained periodically during reactor refueling outages.

d. Plant operation and maintenance requirements will include protection to prevent common mode failure due to human actions of testing, calibration, and maintenance.

e. A comprehensive test and monitoring system has been designed for the synchronous machines to determine status during normal operation. The continuously monitored parameters are:

(1) Input/Output Voltage, Current, Power
(2) Output Frequency
(3) Shaft Torque and Rotational Speed
(4) Electrical Output
(5) Vibration
(6) Bearing Temperature
(7) Winding Temperature

G.4.7.3.7 EM Pump Performance Under Extreme Conditions

This section contains responses to concerns and questions raised during a NRC Staff - GE meeting in January 1990 concerning failure mechanisms, pump trip on ULOHS transient, coastdown on Curie point trip, and sodium leaks.

a. The failure mechanism of primary concern is an electrical fault in the pump stator or power feeds. Such a failure could result from a breakdown in the electrical insulation system due to excessive temperature or mechanical abrasion, or the leakage of sodium into the stator housing. The windings are arranged so that the fault would first occur to ground. A ground fault detection system is provided to alert the operator to take protective action, and if the fault is large enough to open the PCU circuit breakers to prevent serious damage to the pump. If the operator fails to act, the PCS will automatically open the PCU circuit breakers. To back up the PCU circuit breakers, a Class 1E overcurrent protector has been added. This is a passive device, such as a fuse, which breaks the connection between the power supply and the EM pump.

   The synchronous machine remains connected to the pump to provide the desired coastdown, and since the ground loop is also disconnected by opening the breakers, the fault will not impede the transfer of energy from the synchronous machine to the pump as long as the fault remains a simple fault to ground. Evaluations are being performed to determine what happens if the synchronous machine continues to supply power to the EM pump when the pump has an electrical fault. It is not expected that an unsafe level of damage will occur since the amount of power that the synchronous machine can supply is limited. In addition, a possible second passive overcurrent protection device, which could be incorporated into the synchronous machine, will also be evaluated.

A second failure mechanism of concern is a leak in the stator housing which allows sodium to enter the stator cavity, and gas in the stator cavity to enter the reactor coolant. Sodium leakage into the stator cavity will cause electrical shorts as discussed above. Also, the discharge of a significant quantity of gas into the reactor coolant is detrimental since, if it enters the core, it has the potential to cause positive reactivity insertion and a resultant power transient. To prevent gas from entering the coolant, the gas pressure within the stator cavity will be maintained below the sodium pressures in the vicinity of the pump.

b. The purpose of tripping the EM pumps during the ULOHS transient is to eliminate them as a source of heat to the reactor. Since the 1986-1987 design, a separate Class 1E thermal shutoff system (TSS), that backs up the RPS, has been added which automatically opens Class 1E EM pump circuit breakers when the cold pool sodium reaches a temperature of 1000°F. The pumps are tripped at 1000°F to ensure that they have sufficient electrical integrity to provide coastdown.

The thermal shutoff system utilizes a separate Class 1E thermocouple and temperature measuring electronic chassis for each EM pump. Each thermocouple measures pump outlet sodium temperature, which is within 5°F of the pump inlet sodium temperature. If the temperature is above the setpoint, each temperature measuring chassis sends a signal to a shutoff logic voting circuit for each pump which controls the Class 1E circuit breakers in the EM pump power supply. Each 2/4 logic circuit receives signals from all four thermocoules and opens the EM pump breakers if two of the four thermocouples exceed the setpoint.

Since the thermal shutoff system is separate from the RPS, the chance that it also fails for a ULOHS event is remote and well into the EC-IV range.

In the unlikely event that the pumps are not disconnected from their power supplies by the EM pump circuit breakers, adequate time (hours) exists for operator action to manually shut off the pump(s) before excessive sodium temperatures are reached due to pump heating.

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Also, it is likely that the pumps will shut themselves down due to loss of electrical integrity long before system temperatures reach 1200°F, and peak pump electrical insulation temperatures reach 1600°F. In addition, passive overcurrent protectors, such as fuses, have been added to the power supply lines from the PCU to add further assurance of ultimate pump shutdown.

c. If the EM pumps reach their Curie point, they begin to lose their ability to pump. The rate at which this loss of pumping capability occurs would provide some degree of coastdown. However, predicting the trip point or the shape of the coastdown curve obtained by this means would depend on the interaction between inherent pump characteristics and the rate of temperature rise. A coastdown curve based on these parameters is not reliable, and for that reason will not be used. Instead, the EM pumps will be intentionally tripped during the ULOHS transient, in order to provide a pre-designed coastdown. The EM pumps will be shut off by a signal from the separate Class 1E thermal shutoff system when the cold pool temperature reaches 1000°F, well below the Curie point temperature. This signal opens the EM pump circuit breakers, disconnecting normal PCU power to the pumps, but not disconnecting the synchronous machines. The kinetic energy of the spinning rotors of the synchronous machines is then used to provide coastdown to the pumps.

d. The pump casing will be monitored for sodium leakage and the pump will be shut down by operator action if such leakage is detected. If it is not detected, the PCS-controlled PCU circuit breakers will be tripped by the ground fault system to prevent serious pump damage. Also, the EM pump circuit breakers will be tripped by the RPS on a flux to flow mismatch if EM pump flow decreases due to shorting. Since the synchronous machine remains connected to the pump, a method for limiting the current from the synchronous machine will be evaluated to prevent serious damage. (See also the response in Section G.4.7.3.7.a).

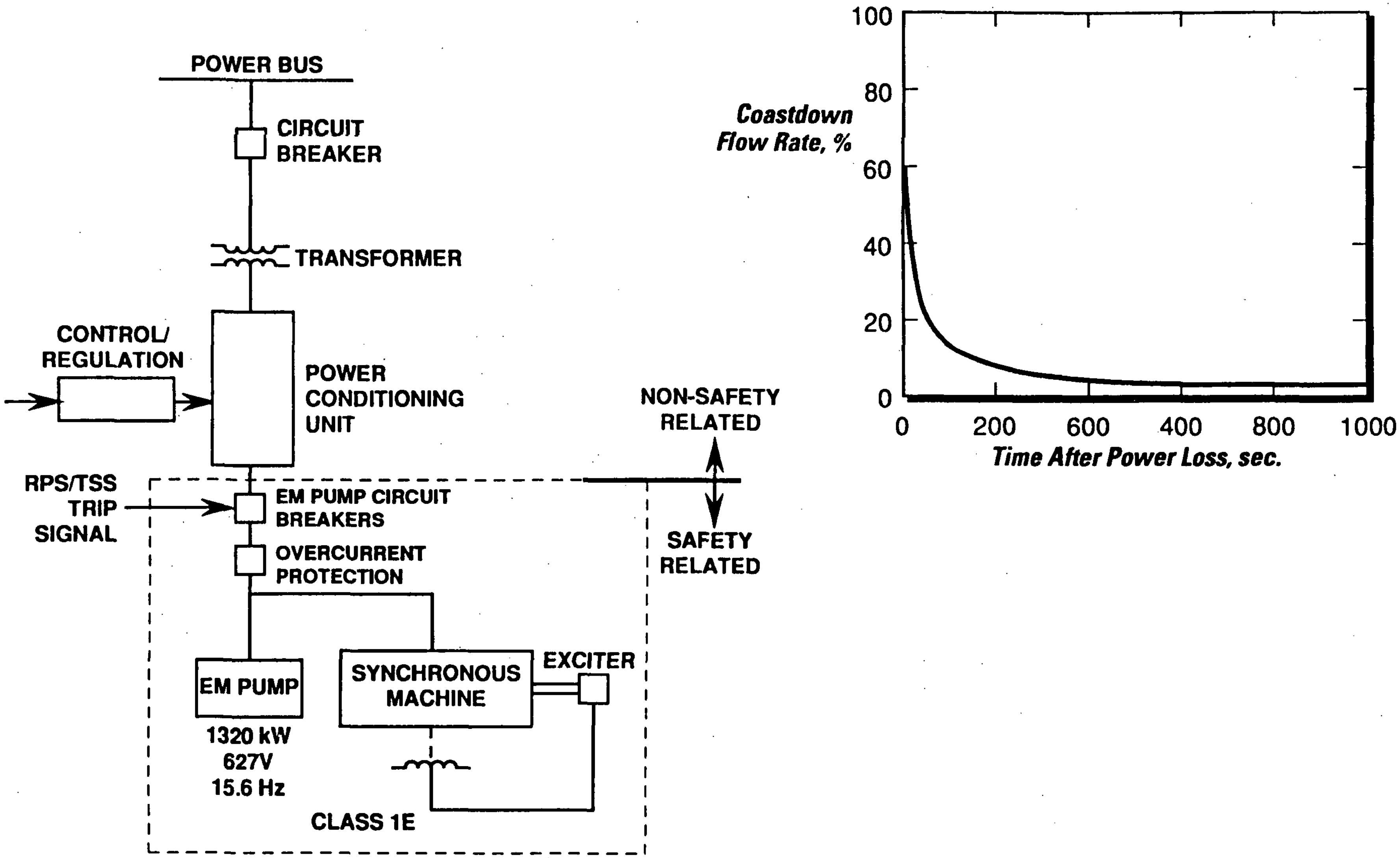

Figure G.4.7–1 EM PUMP POWER SUPPLY SCHEMATIC & COASTDOWN PERFORMANCE

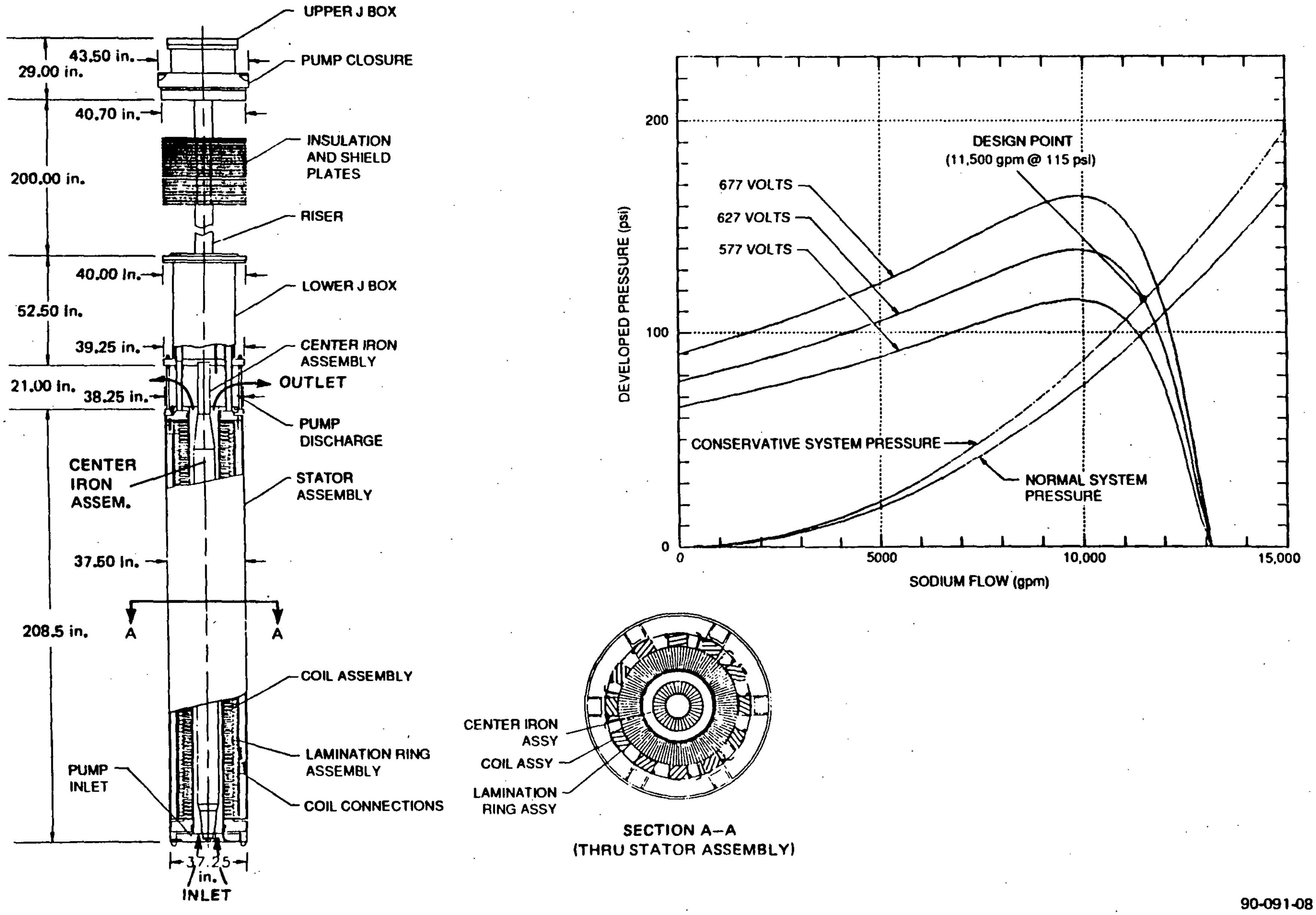

Figure G.4.7–2 EM PUMP DESCRIPTION AND PERFORMANCE

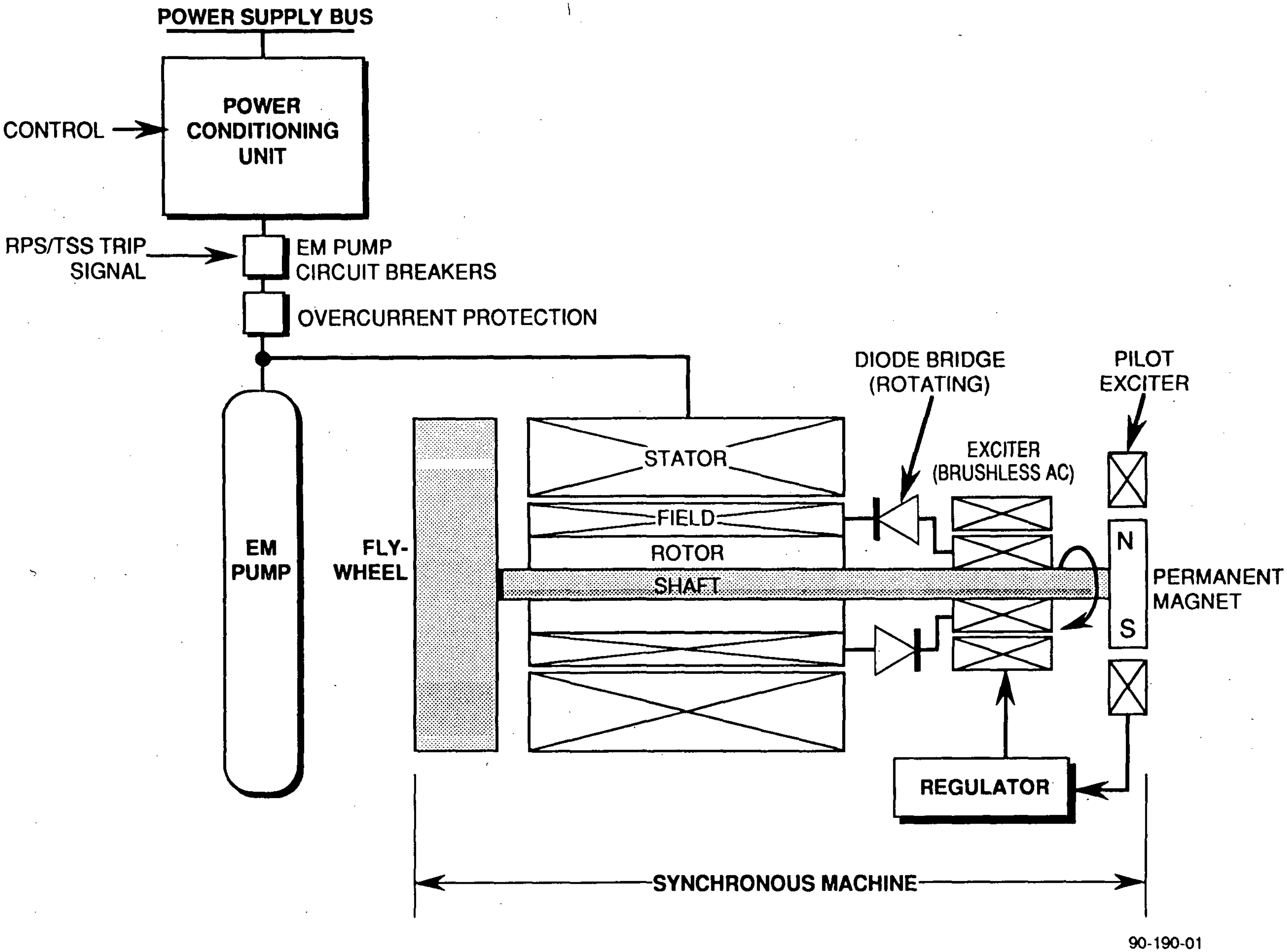

Figure G.4.7–3 SYNCHRONOUS COASTDOWN MACHINE

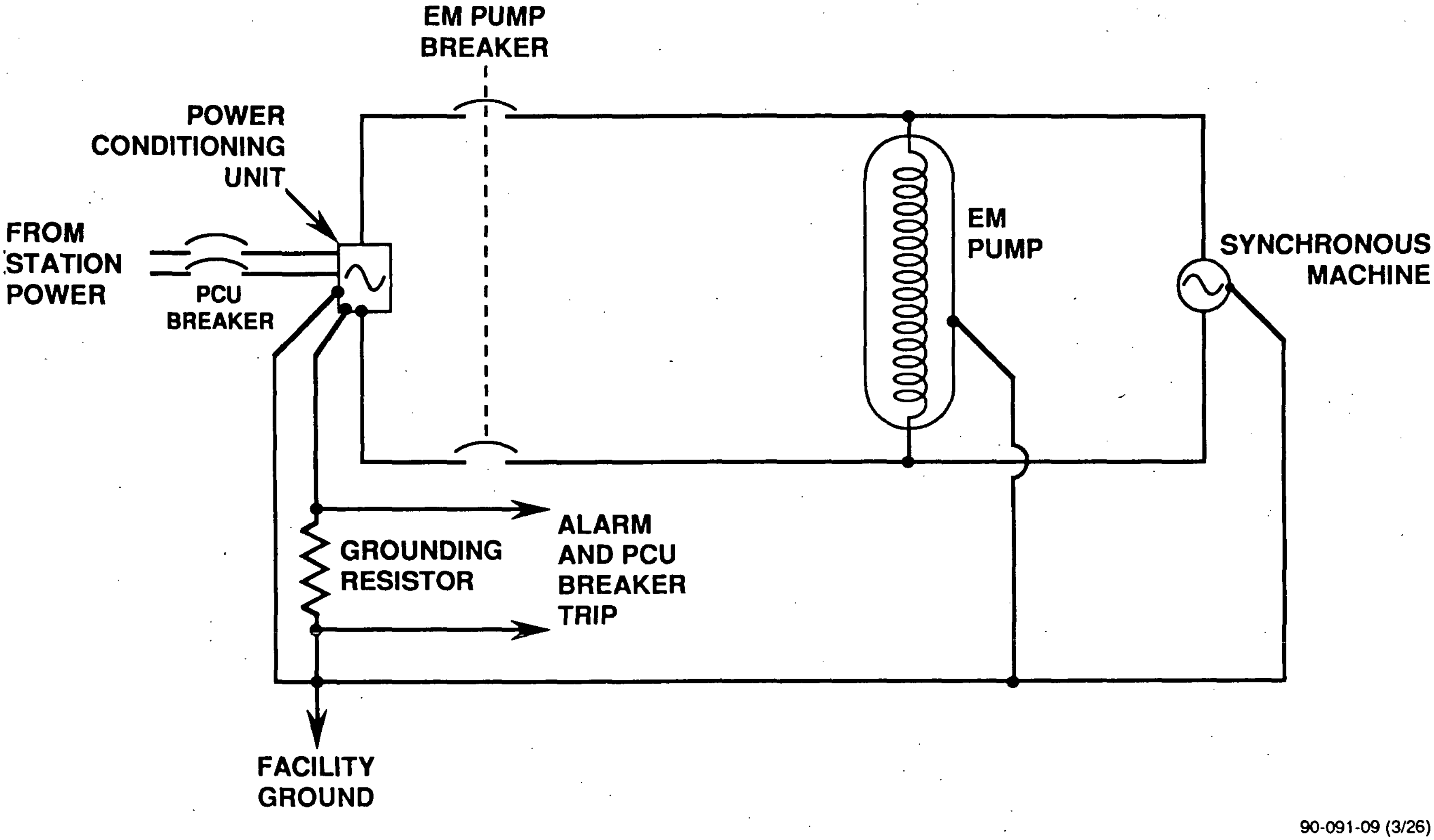

Figure G.4.7–4 GROUND FAULT DETECTION & CURRENT LIMITATION SCHEMATIC

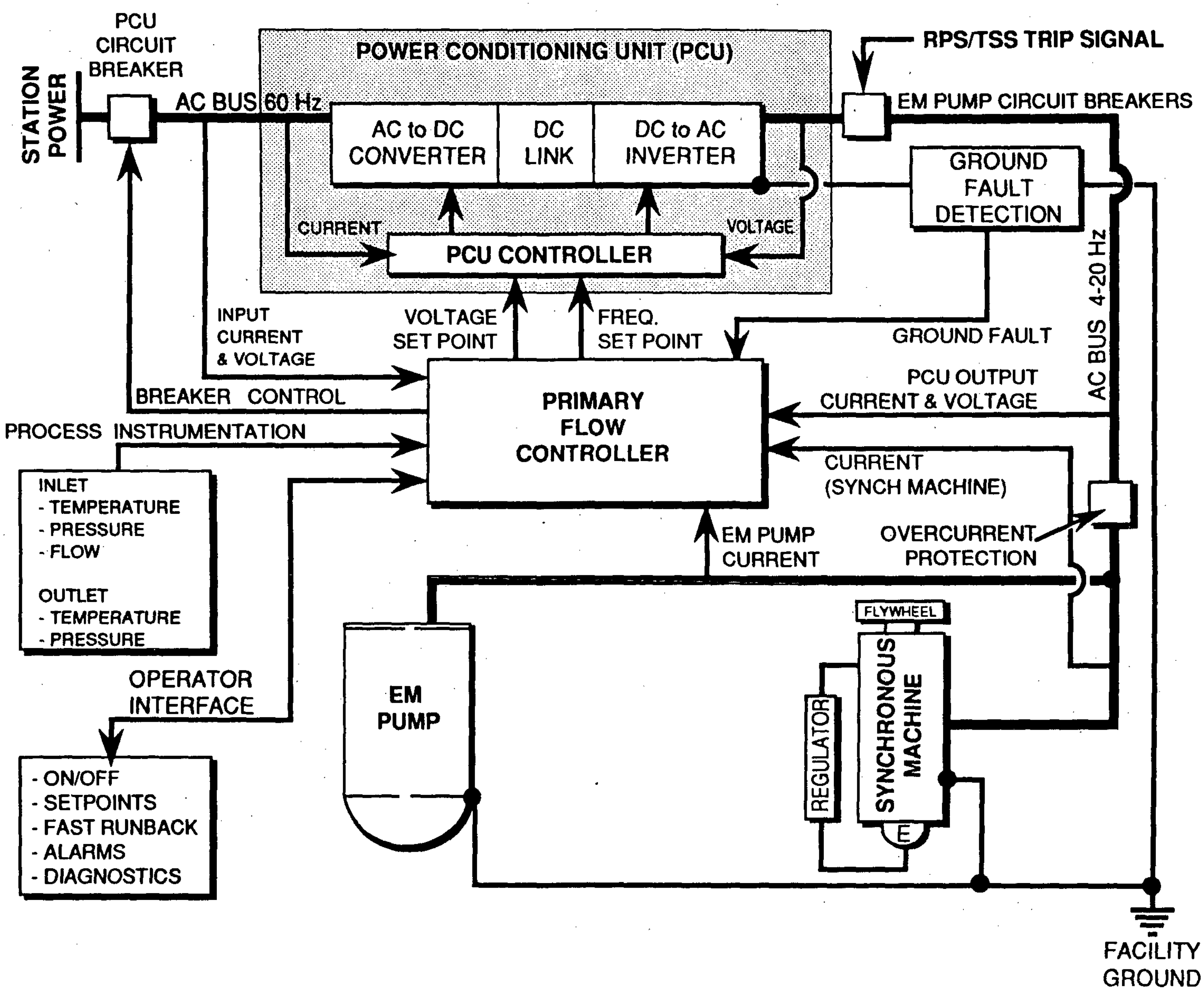

Figure G.4.7–5 EM PUMP CONTROL SCHEMATIC

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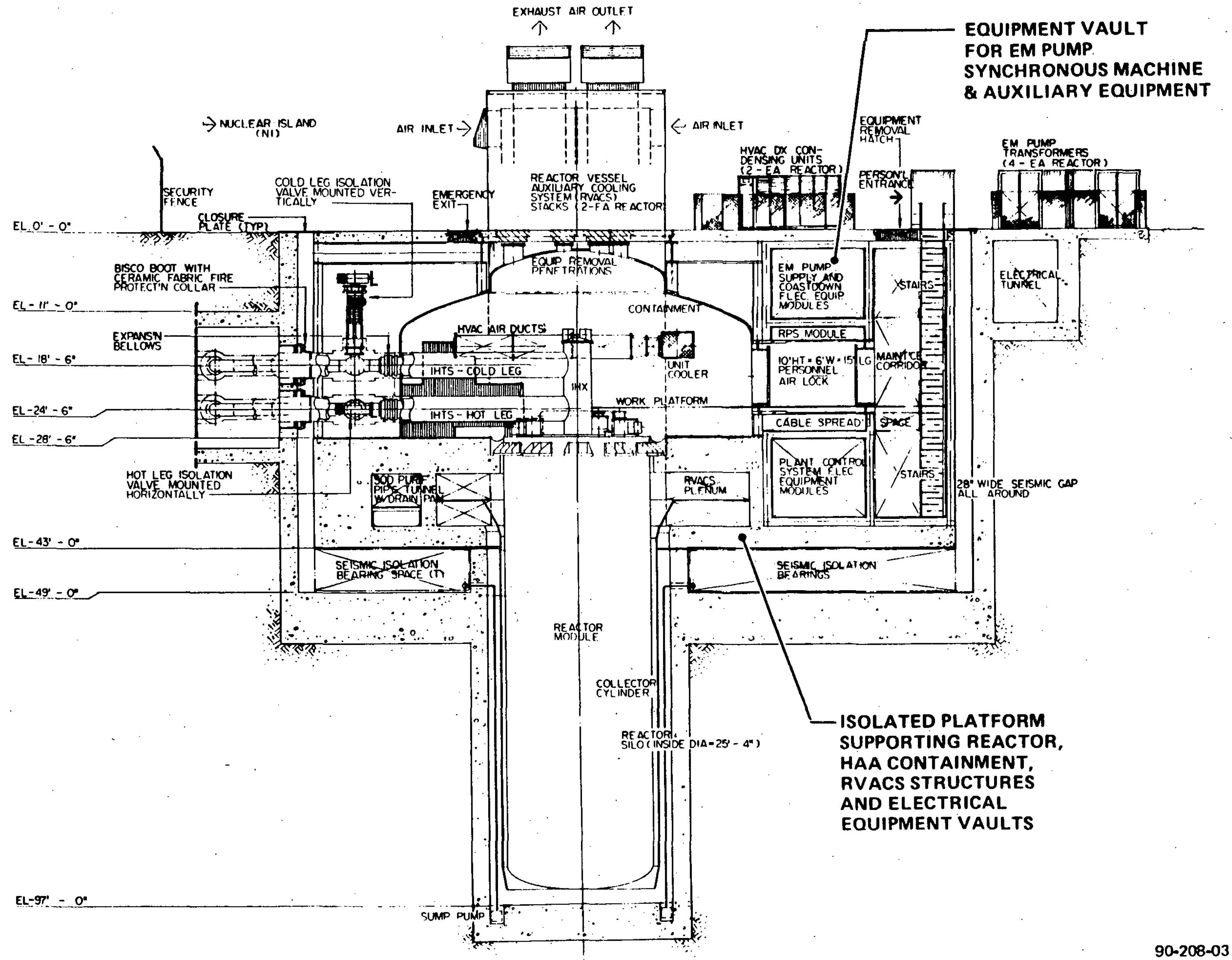


Figure G.4.7–6 REACTOR FACILITY

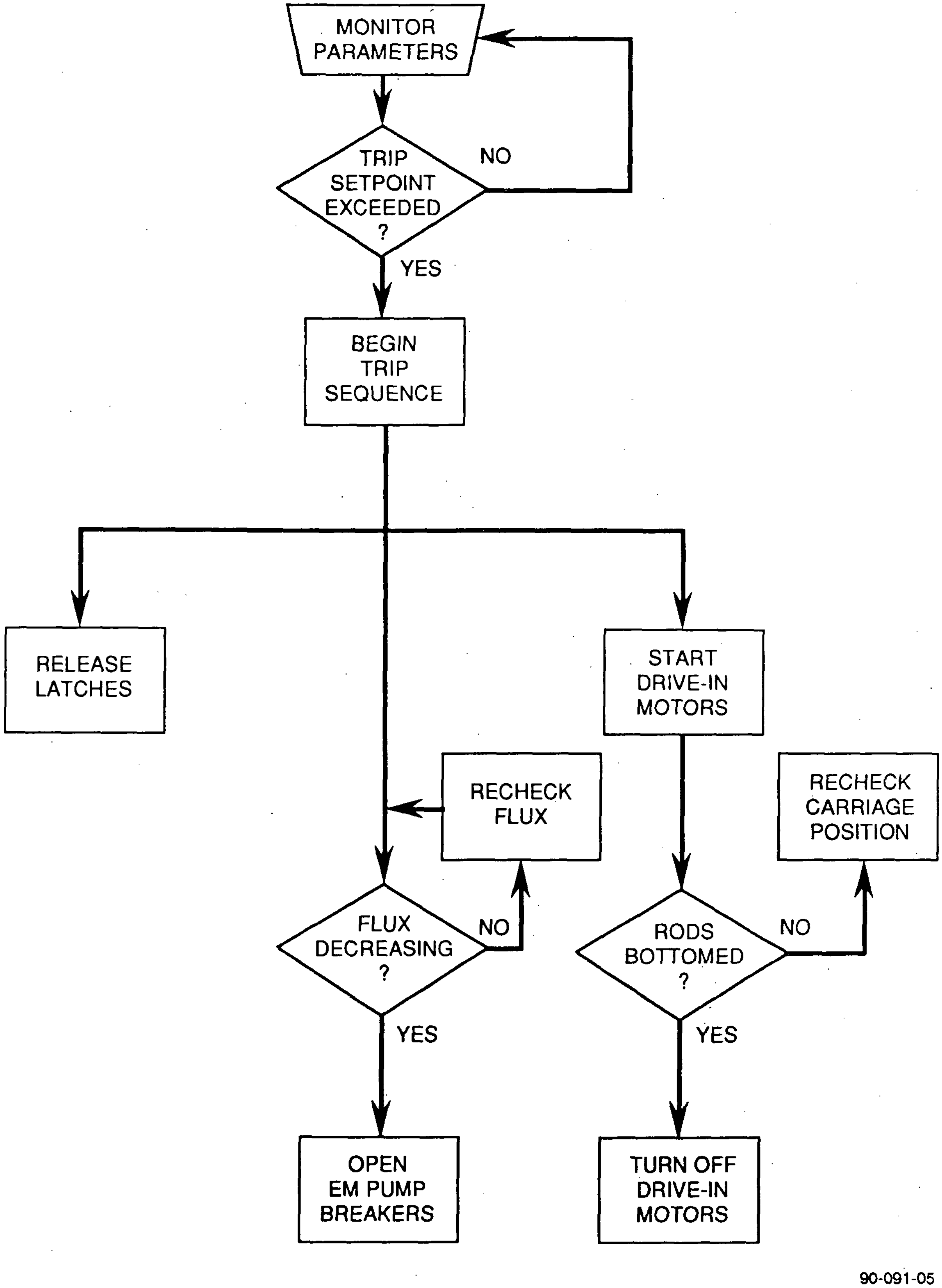


Figure G.4.7-7 REACTOR SCRAM SEQUENCE DIAGRAM

G.4.8 Sodium/Water Reaction Pressure Relief System

G.4.8.1 SER Position on SWRPRS

In Sections 5.5.5 and 5.6.5.2 of the draft SER, the staff concludes that the PRISM design provides a strong defense against the sodium-water reaction. There are independent systems for dumping the fluids from both the water side and the sodium side of the steam generator. And the IHX tubes are designed to take a 1000 psi pulse, which is equal to the steam system pressure. Thus, a sodium-water reaction event is very unlikely to damage the integrity of the IHX. However, such a conclusion is based upon ensuring highly reliable operation of the SWRPRS and water/steam dump system. Therefore, both SWRPRS and the water/steam dump system should be classified as safety grade, and designed with redundancy in active components.

G.4.8.2 Current Reference ALMR Design

The sodium water reaction pressure relief subsystem (SWRPRS) is located in a tornado-hardened Seismic Category II steam generator building designed to protect the SWRPRS system. The SG building is designed such that it cannot fail in a manner which will impact the integrity and operability of the SWRPRS. The SWRPRS rupture disks are safety grade in order to ensure overpressure protection of the IHTS and IHXs. In the event of a steam generator tube leak, the SWRPRS dumps the IHTS sodium and simultaneously initiates a rapid water-side blowdown of the steam generator system by means of the water dump subsystem.

While the SWRPRS and the water-side isolation and dump system are not safety grade (except for the rupture disks), they are designed to high reliability and with redundancy in the active components. For added protection of the safety grade reactor system components, isolation valves have been added, one on each IHTS pipe just outside the penetration through the containment dome.

As shown in Figure G.4.8-1, the SWRPRS consists of two safety-grade 28-inch rupture disks, a reaction products separation tank (RPST), two sodium dump tanks, a vent stack and a hydrogen ignitor. The SWRPRS has the capability to accommodate the sodium-water reaction products, steam, and sodium flows associated with guillotine size leaks of all the steam generator tubes with a back pressure in the IHTS of less than 700 psi. The flow path for reaction products is from the nozzle in the lower head of the steam generator through a 30-inch SWRPRS line, with dual rupture disks, to the RPST. The RPST is a vertically oriented 14-foot diameter and 23-foot high tank of SA-533 low alloy steel. The liquid and solid reaction products and displaced sodium are separated from gaseous reaction products within the RPST and drained into one of the horizontally oriented sodium dump tanks through two 24-inch drain lines.

The two 14-foot diameter, 33-foot long carbon steel sodium dump tanks are interconnected by two 24-inch sodium equalization lines and one 30-inch gas equalization line, allowing the two tanks to operate as a single unit. One tank is located directly below the RPST and water dump tank, while the other is directly below the steam generator and at the same elevation as its sister tank. The gaseous products are released and burned through the stack and flare tip ignitor into the atmosphere.

The intermediate heat transport system (IHTS), steam generator system (SGS), and steam generator building are not safety grade systems, but are designed such that their failure cannot cause a failure of a safety grade system, such as the primary boundary at the IHX or the SWRPRS rupture disks. To assure that the IHX barrier is maintained during a sodium water reaction, the IHX unit and the IHTS piping are designed to withstand the 1000 psia steam pressure under faulted Level D conditions. In addition, the rupture disks are safety grade to assure that they will rupture and relieve the steam pressure at ~325 psi, well below the 1000 psi value.

G.4.8.3 Evaluation of Severe Steam Generator Failure

The Staff defined Bounding Event No. 5 to be a steam generator tube rupture with failure to isolate or dump water from the steam generator.

Amendment 11 to the PSID indicated that the probability of this severe event damaging the primary system is well below the level of the safety goals, $<10^{-8}$ per rector module per year. The addition of the isolation valves on the IHTS pipes at the containment boundary now provide increased protection. An updated evaluation of Bounding Event No. 5 is presented here indicating that the primary system is sufficiently protected even if all active systems in the water-side isolation and dump system fail to perform their function, and even if the IHTS pipe isolation valves are not closed.

To evaluate the operation of SWRPRS under severe accident conditions, a worst case SG leak scenario was defined. The event assumes that all the active protective systems fail to perform their function. This event was then used to evaluate the integrity of the IHTS and containment boundary at the IHX. IHX integrity can be shown for this accident condition by demonstrating that the IHX will not be subjected to excessive pressures or that the sodium will not be displaced by the reaction forces generated during this worst case sodium water reaction. It is assumed for the purposes of this analysis that the IHTS isolation valves remain open during the event.

The accident scenario is initiated by a failure in the steam generator tubes, resulting in a pressure buildup on the shell side (sodium side) of the steam generator. The pressure in the IHTS builds to the point at which the SWRPRS dual, 28-inch diameter, 325 psi, safety grade rupture discs fail. The normal progression of this event continues with initiation of a rapid drain of the IHTS following failure of the SWRPRS rupture disks. The sodium and reaction products are dumped from the bottom of the steam generator into the reaction products separation tank through a 30-inch pipe. The gaseous reaction products are separated from the liquid and solid products. The latter are drained into two sodium dump tanks and the former are released to the atmosphere through a stack and the hydrogen is burned with a flare tip ignitor.

Rupture disk failure will initiate rector scram and closure of the IHTS pipe isolation valves. Additionally, the rupture disc failure is designed to activate the SGS water dump subsystem which is designed to initiate a rapid water-side blowdown by opening two parallel 10-inch water dump valves located at the inlet to the steam generator and opening the power relief valves located at the steam drum. In addition to dumping the water/steam in the SGS, the water dump subsystem isolates the SGS (steam side isolation and feedwater isolation) from the unaffected reactor modules in the power block, as well as from the BOP. The system is designed to reduce the SGS pressure from 1000 to 300 psig in less than 60 seconds. The two water dump valves direct the water/steam mixture to a water dump tank for temporary storage and the flashed steam is vented to the atmosphere.

An innovative feature of the helical coil steam generator is the manner in which the inner shroud provides an alternate route (from the tube bundle) for the water/steam mixture to travel during accident conditions, helping to equalize the pressure differential between the sodium inlet and outlet nozzles. This differential pressure is the driving force behind the displacement of the sodium/water interface towards the IHX.

To ensure that the sodium is not displaced from the IHX and that the free surface remains outside the IHX during a sodium water reaction, the maximum allowable pressure difference between the IHTS hot leg and cold leg corresponds to the height difference between the piping high point at the pump discharge and the elevation of the hot leg piping at the top of the reactor head. Based on the piping elevation layout, this elevation difference is approximately 38 feet and corresponds to a limiting differential pressure across the IHX of approximately 14 psid. Thus it is necessary to limit the pressure drop across the tube bundle to 14 psid to prevent the free surface, and therefore any potential for steam or feedwater, from entering the IHX (see Figure G.4.8-2). If the differential pressure across the IHX exceeds approximately 27 psid, the free surface could be forced to the bottom of the tube bundle. This condition would result in forcing sodium out of the IHX and therefore permit steam and feedwater to flow through the IHX. However, the hydraulic design of the IHTS, which will prevent the ingress of steam into the IHTS, is backed up by the IHTS isolation valves.

For the worst case event, the non-safety grade water dump system is assumed to fail. This system failure results in failure to open two water dump valves and the power operated relief valves, as well as in failure of dual steam and feedwater isolation valves to close. Thus, failure of the water dump subsystem permits both steam and feedwater from the BOP to continue into the affected steam generator. Therefore, the event is not immediately terminated, as designed, but continues unmitigated as a result of the continuous supply of steam and feedwater to the break. It is also assumed that the event results in the failure of additional tubes in the steam generator. For a worst case evaluation, it is assumed that all of the tubes in the steam generator fail. Additionally, to provide the maximum steam/feedwater flow to the break, the analysis assumes that the other two reactors in the power block are not tripped, but continue to operate and supply steam to the failed SG rather than to the turbine.

The worst location for a steam generator tube rupture, from a IHX protection standpoint, is near the top of the tube bundle. Tube failures at this location result in the maximum pressure drop across the steam generator tube bundle, and therefore the maximum differential pressure across the IHX. The steam and feedwater enter the steam generator shell side at the break and flow across the tube bundle and through the steam bypass channel to reach the exit from the steam generator to the SWRPRS. The pressure drop of the steam and feedwater flowing from the tube rupture site to the exit of the steam generator defines the driving force available to displace sodium in the IHX.

In order for the maximum pressure differential across the steam generator to be defined, it is further assumed that a failure in the exit piping from the steam generator occurs. This exposes the reaction product flow path exit to the atmospheric pressure near the steam generator, thus minimizing the pressure at the exit to the steam generator tube bundle. It is expected that the IHX will remain full of intermediate sodium during any sodium water reaction, and thus assure protection for the IHX against the effects of an unmitigated sodium water reaction event.

A flow restrictor in the steam line with a throat to inlet diameter ratio of 0.6., results in 3.62 x 10^6 lb/hr of saturated steam entering the steam generator through the tube rupture under choked flow conditions. Another 6x10^6 lb/hr of feedwater also enters the SG from the steam drum, and 10% flashes into steam. The flow restrictor and steam bypass channel cause the total flow across the tube bundle to be 4.2x10^6 lb/hr, with a steam generator pressure of 130 psia. This results in a steam generator tube bundle pressure drop of approximately 2 psid or about 5 feet of sodium, well below the limit of 14 psid. Therefore the sodium/water interface is not forced back into the IHX. In contrast, without the steam bypass channel, the pressure drop is raised to approximately 7 psid.

Another possible accident scenario to consider might be to assume the rupture of 30-inch IHTS cold leg piping at the sodium/steam interface (downstream of the mechanical pump). A large break in this line would reduce the pressure on the sodium side of the cold leg to atmospheric conditions, creating a situation where steam is allowed to flow through the IHTS and IHX. This event is considered extremely unlikely since the feedwater supply to the two non-affected loops cannot be maintained for more than 15 minutes before the water inventory is exhausted and it will take several hours for the steam/sodium reaction to penetrate the 1/2-inch thick IHTS piping.

In summary, the IHTS and SG systems have been designed in a manner which provides passive protection of the interfacing primary system boundary at the IHX. That is, a failure of the active protective system, such as failure of the redundant steam and feedwater isolation valves to close and terminate the event as designed, will not result in an IHX failure. For added protection, IHTS pipe isolation valves, as described in Section G.4.1, have been included in the reference ALMR design.

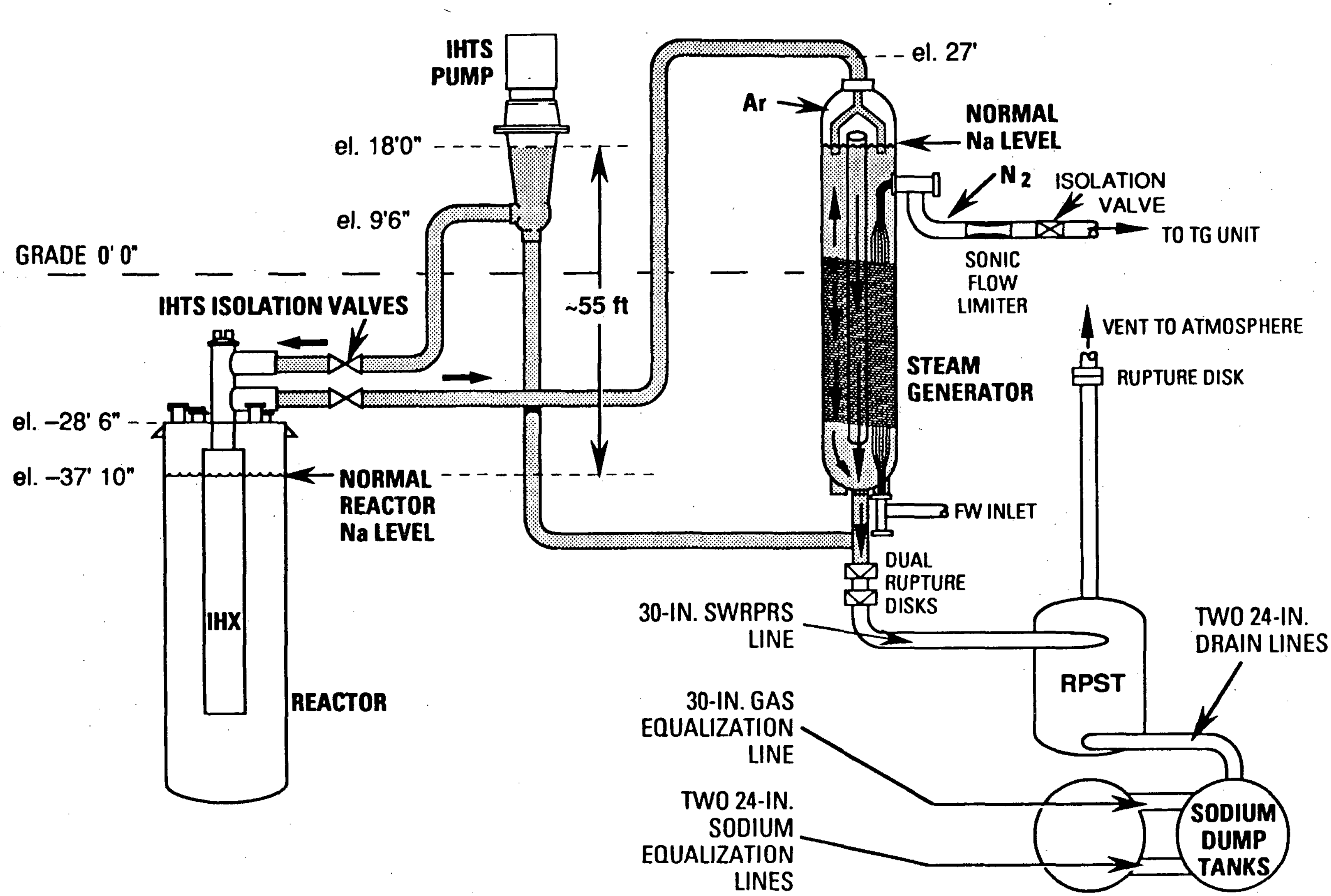


Figure G.4.8-1 - NORMAL SODIUM LEVELS

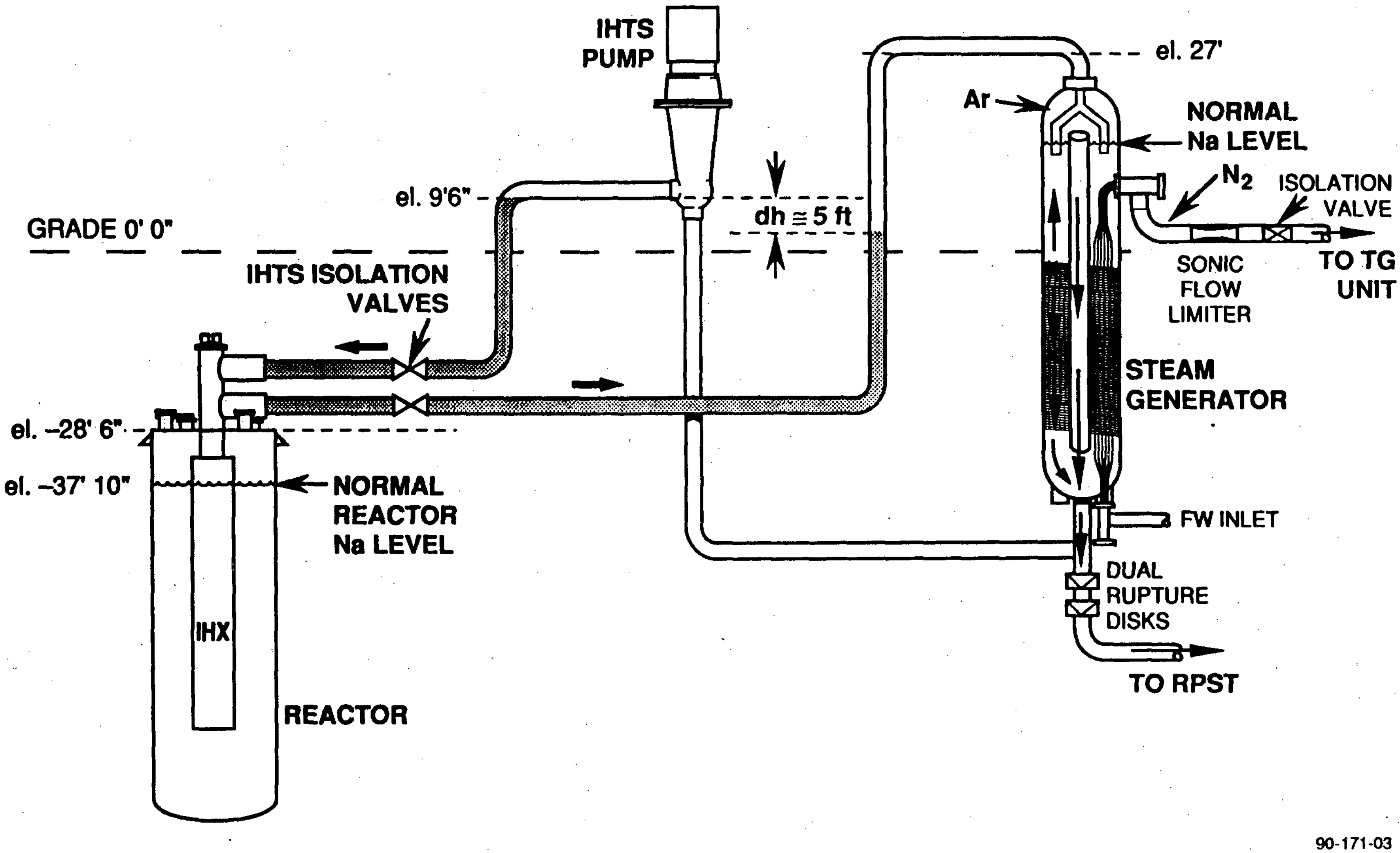


Figure G.4.8-2- SODIUM LEVELS FOLLOWING A SG LEAK WITH FAILURE TO ISOLATE

G.4.9 Reactor Vessel Auxiliary Cooling System

G.4.9.1 SER Position on RVACS Design Features/Approach

The draft SER states in Section 5.7.5:

"Two major safety issues remain on RVACS performance. The first is the response of the plant to Bounding Event #3 - loss of all decay heat removal for 36 hours. It is not clear that core integrity can be maintained under this condition... The second is the high temperature (~1200°F) to which the primary system is raised when removing decay heat on RVACS only. Subjecting the primary system to such a temperature may cause permanent damage affecting the ability of the plant to resume operation. In addition, since the only safety grade decay heat removal system is RVACS, the frequency at which the primary system is subjected to such elevated temperatures (and determining the acceptability of resuming operation) are of concern.... Also, recovery actions from such operation need to be developed to avoid excessive thermal shocking of the primary system on recovery of normal decay heat removal.

"Based on the above, the staff is of the opinion that a system to map reactor vessel temperature should be provided (to facilitate restart decisions) and the PRISM design should ensure that the frequency of high temperature challenges to the primary system is no greater than that for LWRs, (i.e., equivalent margin to RV challenges)."

G.4.9.2 ALMR RVACS Design

The PRISM power plant is equipped with three methods for shutdown heat removal: (1) condenser cooling in conjunction with intermediate sodium and steam generator systems, (2) an auxiliary cooling system (ACS) which removes heat from the steam generator by natural convection of air and transport of heat from the core by natural convection in the primary and intermediate systems, and (3) a safety-grade reactor vessel auxiliary cooling system (RVACS), which removes heat passively from the reactor containment vessel by natural convection of air. The combination of one active and two passive systems provides a highly reliable and economical shutdown heat removal system.

Significant analytical and experimental work has been carried out to demonstrate the excellent thermal performance of the safety-grade RVACS. Results show that RVACS will perform its function very well under expected conditions and under extremely unusual and severe conditions including complete blockage of the air inlet or outlet passages. A summary of the RVACS thermal performance for expected operating conditions and postulated accident events are presented following a brief description of the system.

G.4.9.2.1 Design Description

The RVACS can dissipate all of the reactor decay heat through the reactor and containment vessel walls to the ambient air heat sink by the inherent processes of natural convection in fluids, heat conduction in solids, thermal radiation heat transfer, and convective heat transfer. The reactor module size and design features were selected to provide a low cost, high performance RVACS without the need for heat transfer enhancement devices (e.g., fins) to maintain maximum temperatures below acceptable structural limits.

The RVACS operates continuously but functions at its intended high heat removal rate only when the normal and ACS decay heat removal systems are inoperative. The RVACS does not require any human or mechanical action to be put into full operation. The thermal performance is self-regulating and depends solely on the reactor temperature. The heat removal rate is low during normal operation conditions (0.7 MWt), as desired, and increases to about 2.7 MWt at peak performance.

Operation of RVACS is explained using the diagram of Figure G.4.9-1. Heat is removed from the core and transported to the reactor vessel wall by natural convection of primary sodium. Two alternative sodium flow paths exist in the vessel during most of the decay heat removal period. Initially, the sodium flow path is the same as that during normal reactor power operation; i.e., from the core upwards to the hot pool, then down through the two IHXs to the bottom of the vessel and then upward into the pump inlet plenum structure and through the pump duct. The sodium then enters eight inlet pipes which lead to the core inlet plenum.

An alternative, second sodium flow path becomes available after sodium temperatures have increased and the corresponding sodium volume expansion has resulted in overflow through slots provided in the reactor vessel liner. This alternative, slightly more efficient overflow path is downward through the annular gap between the reactor vessel and its liner where a portion of the sodium flow gives up some of its heat directly to the reactor vessel wall prior to exiting at an elevation near the IHX outlets. The remainder of the sodium follows the flow path used during normal operation.

Heat transport through the reactor and containment vessels is by conduction, while the reactor vessel to the containment vessel heat transport is mainly by thermal radiation. Only three percent is by natural convection in the argon-filled space between the two vessels. Thermal radiation heat transfer is promoted by providing a high thermal emissivity coating on the heat transfer surfaces. The surface coating consists of an oxide layer generated during heat treatment by air oxidation on the external surfaces.

Naturally convecting air removes heat from the containment vessel and collector cylinder which is heated by radiation from the containment vessel as indicated schematically in Figure G.4.9-1. Atmospheric air enters the RVACS through four inlet openings in the tornado hardened concrete chimney structures protruding about 15 feet above grade level. It is directed downward into the lower of two horizontal plena and from there into the annular region between the concrete reactor silo and the collector cylinder (cold air downcomer). This incoming air turns around at the bottom of the silo and enters the annular gap between the containment vessel and the collector cylinder (hot air riser) where it is heated by the hotter, surrounding steel structures. The air heating provides the natural draft needed to maintain air flow in this loop. The heated air flows into the outlet plenum and from there it is discharged to the atmosphere through four outlet stacks as indicated in Figure G.4.9-1.

The inlet and outlet air openings are protected by heavy steel screens or gratings with openings small enough to prevent large objects from entering. The openings are also protected to limit harmful amounts of rain and snow from entering the RVACS. As an additional precaution, a sump pump is available at the bottom of the reactor silo (not shown in Figure G.4.9-1) to remove any water that might enter by seepage, floods, etc., in such quantities that it is not evaporated by the air stream and the hot steel structures located in the cavity. Every reasonable effort has been made in the design to reduce form and frictional hydraulic losses in the air flow path to enhance the air flow rate.

Both the containment vessel and the collector cylinder are fabricated from 2-1/4Cr-1Mo steel which is not susceptible to stress-corrosion cracking, particularly when exposed to a coastal air atmosphere. High emissivity coatings are created on both surfaces of the containment vessel and on the inner surface of the collector cylinder by oxidation in air at elevated temperatures during manufacture, similar to the process used for the stainless steel reactor vessel.

G.4.9.2.2 Design Basis Performance

The analysis of the design basis RVACS event conservatively assumes that the normal and auxiliary heat removal systems, as well as the Intermediate Heat Transport System (IHTS) sodium, are lost immediately following reactor and primary pump trips. The passive RVACS only is available to remove reactor decay and sensible heat.

Transient analysis results for the RVACS design basis case with nominal expected analysis assumptions and with clean heat transfer surfaces are given in Figure G.4.9-2 (RVACS only). The curve represents the average core sodium outlet temperature during the transient. The sodium temperature reached for this case is 1125°F at about 30 hours. The upper 2-sigma limit is also indicated for the RVACS-only case to be 1195°F. This temperature was determined by calculating the increase in average sodium temperature that results when uncertainties are included for each of the major parameters separately. The 2-sigma temperature was then calculated by taking the square root of the sum of the squares of individual temperature increases. In particular, the parameters, and amounts of uncertainty representing a 95% confidence level, included were: 1) a 10% increase in decay heat generation, 2) a 3% decrease in the thermal emissivity of stainless steel surfaces, 3) a 6% decrease in emissivity of 2 1/4Cr-1Mo steel surface emissivities, and 4) an 18% decrease in the air-side convective heat transfer coefficient. It was demonstrated earlier (Reference G.4.9-1) that the RVACS performance is not sensitive to uncertainties in other parameters such as the air inlet temperature and the air and core flow resistances.

The calculated maximum sodium temperature is below the structural design temperature limit for ASME Level C service. Most reactor structures are at temperatures lower than the average sodium outlet temperature, except for the core outlet region which will have some temperature variation due to the different decay power levels in the core assemblies. The spent fuel assemblies will also be at a somewhat higher temperature. For example, a conservative analysis assuming only conductive heat transfer (i.e. no flow through) indicates that the metal fuel temperature is 38°F higher than the average sodium temperature at peak temperature conditions.

The slight discontinuity in the sodium outlet temperature at about four hours noted in Figure G.4.9-2 occurs when overflow starts at a hot pool temperature of 1000°F. This indicates that the RVACS performance does not change significantly when overflow starts and that overflow is not essential to RVACS operation. The excellent performance without overflow is the result of the long and slender design configuration with correspondingly large thermal heads (on the order of 28 ft) for either sodium flow path through the IHX or the overflow gap, the close proximity of the IHX to the reactor vessel liner, and the large IHX shell surface area.

It is estimated that RVACS will be called upon to remove decay heat only four times during the 60-year plant life: one time as a result of shutdown from full power and three times during steam generator maintenance (tube cleaning) operations. Thus, the reactor will experience a temperature transient similar to the RVACS only curve presented in Figure G.4.9-2 one time, whereas in the other three times the plant will be cooled down by the normal heat removal system prior to initiating IHTS draining and steam generator maintenance. The maximum temperature reached for those transients is much lower and depends on what time after reactor shutdown maintenance operations are initiated.

The time required to cool the reactor down to hot standby temperature of 550°F using RVACS only is about 80 days unless the normal heat removal system is restored. The ACS was included in the plant to reduce the number of RVACS transients. This system is available to cool the plant passively along with RVACS whenever there is sodium in the IHTS. The temperature transient that results when RVACS and the ACS operate is also given in Figure G.4.9-2. A heat loss of 0.5 MWt from the IHTS piping and cold trap was included. The maximum sodium temperature reached in this case is 920°F which is just slightly above the normal operating temperature of 905°F, and the cooldown time to hot standby conditions is about five days.

G.4.9.2.3 Off-Normal RVACS Performance

A number of postulated events and scenarios considered beyond the design basis have been evaluated to determine the capability of RVACS to cool the plant safely under unusual and unexpected conditions (References G.4.9-1 and G.4.9-2). In one series of scenarios various degrees of flooding of the reactor cavity with water were analyzed. Partial flooding by seepage of water through the concrete silo wall at rates of up to 1.6 gpm could be accommodated without affecting RVACS performance. At much higher seepage rates RVACS air flow would be impeded, but evaporation of water in direct contact with the containment vessel would safely cool the reactor.

Additional analysis was performed for the extremely unlikely event of an instantaneous, catastrophic complete flooding of the reactor silo. The results showed that the containment vessel and the collector cylinder will experience thermal shocks as they are quenched by 70°F water. However, the calculated maximum rate of change of temperature for the containment vessel was only 1.2°F/sec which is entirely acceptable from a structural point of view. The reactor vessel is insulated by the gas gap between the vessels and experiences no noticeable temperature change from the postulated flooding event.

Various postulated air flow path blockages at the inlets and outlets have also been considered. The results show that minor blockages have little effect on RVACS performance. The more severe cases included: (1) complete blockage of all air inlets, (2) complete blockage of all air outlets, and (3) complete blockage of all air inlets and outlets. Results of these evaluations indicate RVACS will perform its function for Cases 1 and 2 without difficulty although experimental verification will be required for confirmation of assumptions made in the analysis. The results also showed that complete blockage (Case 3) can be accommodated for a limited period of time.

Two specific but extremely low probability cases of RVACS blockage were evaluated in detail (see Section G.4.16). The first case, referred to as redefined Bounding Event No. 3, assumes complete loss of air flow through three of the four RVACS stacks (inlets and outlets) for an indefinite period of time. Temperatures calculated for this event are given in Figure G.4.9-3. These temperatures are entirely acceptable within the normal design basis. The maximum concrete temperature is about 170°F, which is also acceptable.

The second case, referred to as alternative Bounding Event No. 3, assumes complete blockage of the RVACS air flow path for 12 hr and a 25% unblocking thereafter. Results for this event are given in Figures G.4.9-4 and G.4.9-5. The structural temperatures for this case are higher than for the first BE-3 event, but the structures and the reactor remain safe following this event.

G.4.9.3 Rationale Supporting Acceptability of RVACS

GE agrees with the staff that the plant response to Bounding Event No. 3, as originally stated, is such that plant integrity cannot be assured in the long term. Results of the GE evaluation show that major core melting and concrete silo structural damage would result, but that the reactor would be in a safe state without any radiological release in the short term (several days). However, since the 1986-1987 PSID, the ALMR design has been changed to accommodate energetic and core melt events (Section G.4.19). In addition, a low leakage pressure-retaining containment dome has been added to the design (Section G.4.1). On the basis of these design changes, a redefinition of BE-3 to allow 25% flow is considered appropriate.

A second issue raised by the staff concerns the high temperature (~1200°F) to which the primary system is raised when removing decay heat by RVACS only. Structural evaluations presented in Section G.4.17.3 show that the one expected RVACS transient during the plant lifetime does not reduce the life of the major structures significantly compared to the normal duty cycle events. In addition, because of the expected very low frequency of its use (one time to maximum temperature and three times to a significantly reduced temperature) operation of RVACS is not of concern in the sense that it contributes to significantly reducing the life of plant. However, temperature measurements will be provided in the prototype to map temperatures on major structures, particularly the reactor vessel, to determine how they behave during the safety tests which will include an RVACS only transient. Such temperature measurements will provide valuable information. However, temperature problems are not expected to occur. In the ALMR axial variations in sodium temperature during an RVACS transient is expected to be relatively small because of mixing in the hot and cold pools caused by the low pump suction inlet and the long and slender reactor vessel configuration. Core temperature differences are typically only about 100°F, and overall structural temperature differences are not expected to exceed this value substantially.

G.4.9.4 References

G.4.9-1 A. HUNSBEDT and P.M. MAGEE, "Design and Performance of the PRISM Natural Convection Decay Heat Removal System," International Topical Meeting on Safety of Next Generation Power Reactors, May 1-5, 1988, Seattle, WA.

G.4.9-2 P.M. MAGEE and A. HUNSBEDT, "Performance of the PRISM Reactor's Passive Decay Heat Removal System," Transactions ANS Winter Meeting, San Francisco, CA, November 26-30, 1989.

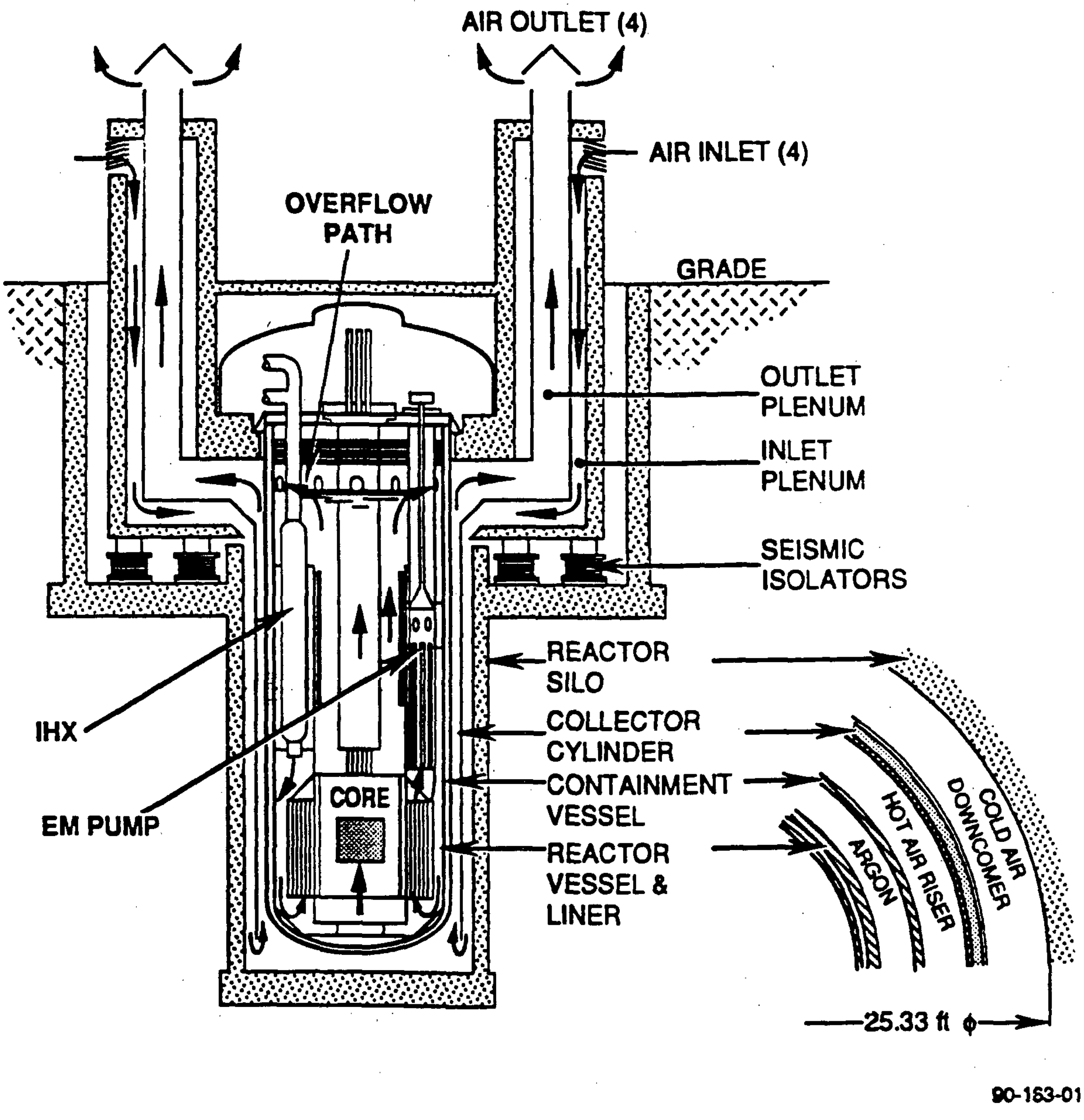


Figure G.4.9-1 - REACTOR VESSEL AND RVACS

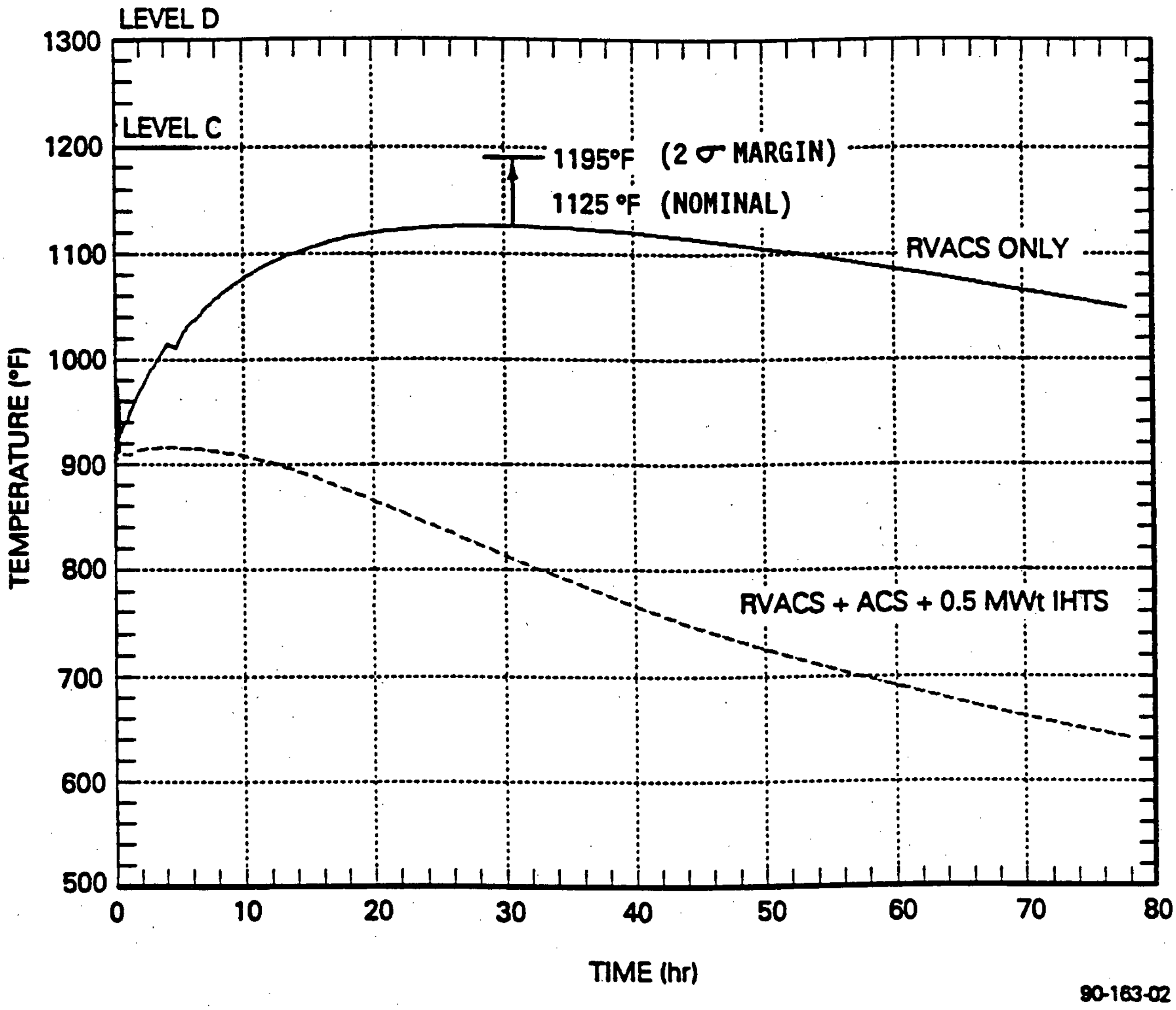


Figure G.4.9-2 - AVERAGE CORE SODIUM OUTLET TEMPERATURES

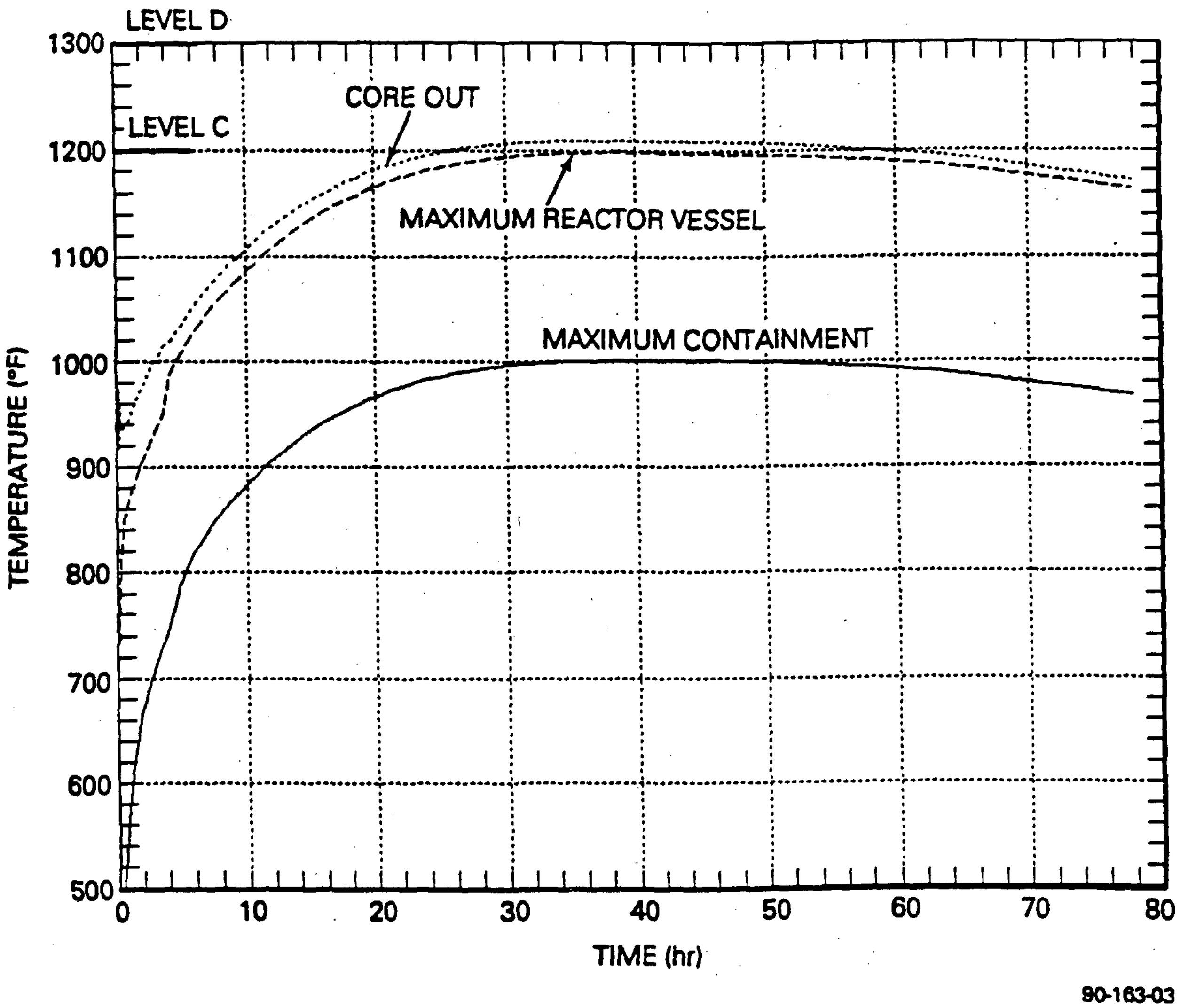


Figure G.4.9-3 - TEMPERATURE TRANSIENTS FOR REDEFINED BOUNDING EVENT 3 (RVACS 75% BLOCKED INDEFINITELY)

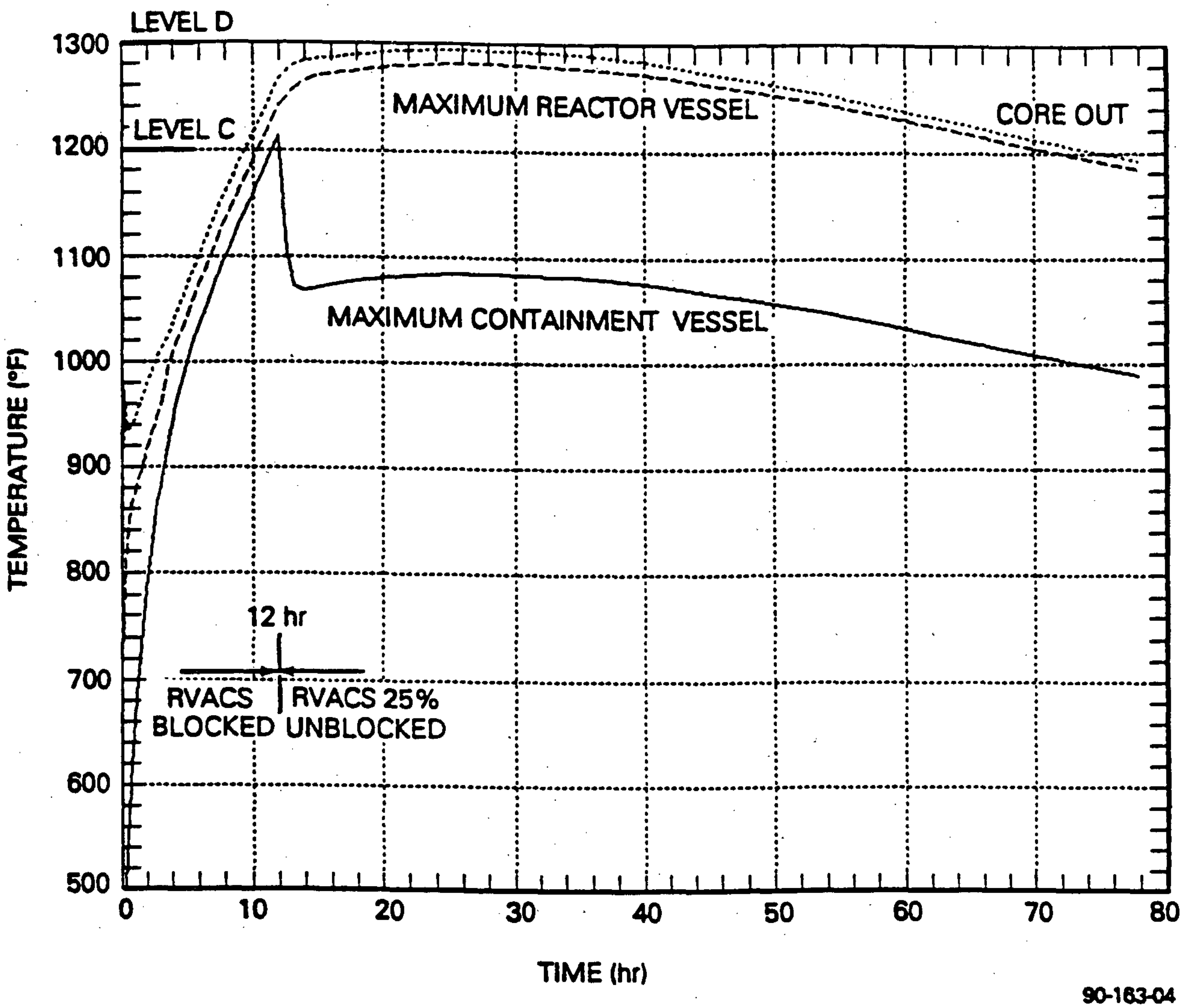


Figure G.4.9-4 - TEMPERATURES FOR ALTERNATIVE BOUNDING EVENT 3 (RVACS BLOCKED FOR 12 HOURS AND 25% UNBLOCKED THEREAFTER)

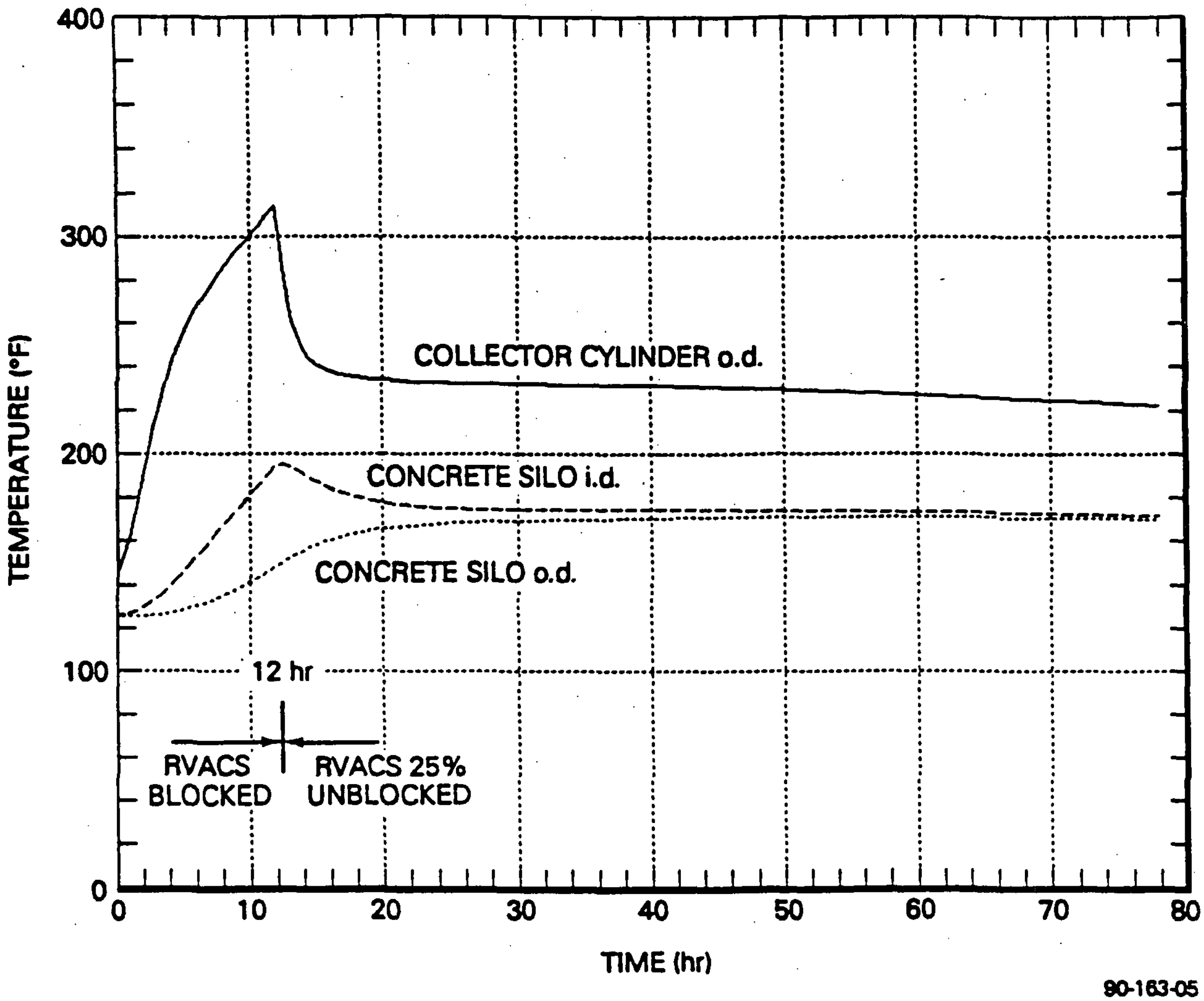


Figure G.4.9-5 - MAXIMUM COLLECTOR CYLINDER AND CONCRETE SILO TEMPERATURES FOR ALTERNATIVE BOUNDING EVENT 3

G.4.10 Control Room

G.4.10.1 SER Position on Control Room

The control room and the information displays presented to the operator in the control room should be safety grade. In addition the displays in the remote shutdown area should be safety related. (Reference SER Section 7.3.3)

G.4.10.2 Reference Control Room Design Features

The control room (CR) and remote shutdown facility (RSF) have been upgraded significantly since the 1986-1987 PSID. The control room has been upgraded for operator habitability and contains a non-safety related operator interface for optimum plant operation. A separate Class 1E remote shutdown facility, with safety-grade electronics and displays, is in close proximity to the control room for safety-grade operator monitoring and control interactions with plant safety systems.

Key design features of the reference control room, remote shutdown facility, and the safety-grade electrical equipment vaults are summarized in this section.

<u>Control Room</u>

The CR (along with the technical support center, information management center and associated communication room) is located inside the nuclear island protected area security fence and is constructed as a Seismic Category II, tornado hardened facility with upgraded operator habitability features. The CR provides improved security and sabotage protection and improved personnel protection from site hazards such as tornadoes, smoke, and hazardous chemical release.

The CR is designed to optimize plant operation. It contains operator consoles with electronics and displays which provide highly processed and well integrated plant information to the operator through a highly interactive, user-friendly, man-machine interface.

The information supplied to the CR operator console for each power block is comprehensive enough to enable one operator to follow the operation of three reactors and one turbine-generator system under all operational situations and to assist the operator in achieving high plant availability while protecting plant equipment. Extensive operator aids, with diagnostics and alarm management, are provided for the operator during both automatic and manual control modes for efficient plant operation. Real-time color-graphic displays are used for effective information transfer to the operator. Provision is made for the operator to interact with real-time predictive plant models to determine the plant state.

To perform these functions, the CR console electronics and displays are driven by a redundant array of computationally powerful high speed plant process computers, each containing real-time operating and database management system software, and a significant amount of application software. Battery back up of CR electronics is provided for eight hours to avoid disruption of CR activities following power failure.

The CR electronics, displays, and process computers are part of the plant control system (PCS) and are not safety related. All plant data, including reactor protection system (RPS) and post-accident monitoring (PAM) data, is sent to the CR so that the operator has benefit of all non-safety and safety related data. The safety related data are isolated by Class 1E isolators and are sent to the operator's console through the plant process computers. Therefore, these data are not Class 1E when displayed on the operator's consoles. A manual scram can be initiated from the CR operator's consoles. This "scram request" signal enters the RPS through a Class 1E isolator, but is also sent through the plant process computers and is not Class 1E.

<u>Remote Shutdown Facility</u>

The RSF is a Seismic Category I, tornado hardened structure located in the radwaste building about 40 feet from the control building. Operator access to the RSF is provided through a Seismic Category II, tornado hardened underground tunnel connected to the control building. The distance from the CR to the RSF is less than 120 feet; operator transit time is less than a minute. A safety grade heating, ventilating and air conditioning (HVAC) system with an emergency outside air filtration system and the capability of isolation during high toxic gas release, is provided for improved operator habitability. Uninterruptible backup power is provided for the RSF electrical systems using sealed batteries with a 36-hour capacity.

All functions which involve direct operator interface with plant safety systems are performed from the safety-grade RSF which is equipped with Class 1E electronics required to perform these functions. For the ALMR, direct operator interaction with safety systems is seldom required because the safety systems are simple and highly reliable, and the safety actions are fully automatic. However, Class 1E microprocessor based equipment is provided in the RSF to directly interface with the safety systems when required. These safety system interface functions are simple and do not require the use of plant process computers (which are not Class 1E). The following functions are performed from the RSF:

a. Class 1E manual scram and post-scram Class 1E safety parameter monitoring of each (or all) reactor(s) in the plant.
b. Class 1E plant post-accident monitoring.
c. Class 1E permissive for updating the neutron flux detector calibration factors.
d. Class 1E permissive for adjusting the electronically positioned mechanical rod stops.
e. Class 1E initiation of the ultimate shutdown system (USS).

A schematic diagram of the RSF console and electronics is shown in Figure G.4.10-1. The RSF is a Class 1E extension of the RPS man-machine interfaces provided in the vaults. All RPS and PAM data, after being processed by the RPS electronics located in the vaults adjacent to each reactor, are sent over a Class 1E fiber-optic communication system to the RSF, which also carries the scram initiation and periodic neutron flux detector calibration permissive signals from the RSF to the RPS electronics. The RSF also provides a similar interface to the rod stop system (RSS) electronics located in the RPS vaults, with all the RSS data sent to the RSF and the periodic rod stop adjustment permissive signals sent from the RSF to the RSS electronics.

Processing of the RPS and RSS sensor inputs, and generation of the RPS trip and RSS actuator signals, are performed by the RPS and RSS electronics in the reactor vaults automatically. The RSF provides a central location from which an operator can read the RPS and RSS data, give permissives, and initiate manual action from these systems. The RSF also contains a panel with Class 1E switches from where the ultimate shutdown system can be activated.

<u>RPS Instrument Vaults</u>

The safety grade man-machine interface electronics in the RSF are backed up by equivalent safety grade electronics in the RPS instrument vaults. The RPS vaults for each reactor contain safety grade processing electronics and safety grade operator interfaces for the RPS, PAM, RSS, and USS for that reactor. All the functions performed centrally from the RSF for all nine plant reactors, can be performed separately for each reactor from its RPS vault area. The RPS vaults are Seismic Category I, tornado hardened structures located on the seismically isolated platform of the reactor facility. However, they do not have the same HVAC capability and hence the same degree of operator environmental protection and habitability as the RSF.

G.4.10.3 Rationale Supporting Reference Control Room Design

The ALMR incorporates a number of safety features based on passive design principles and inherent physical processes. These safety features are such that even for Event Category III ATWS events (such as total loss of heat sink or loss of flow without scram, or uncontrolled withdrawal of all rods without scram), negative reactivity feedbacks and passive natural circulation cooling keep the reactor at a stable power condition and at temperatures below the Level D limits. The operator, therefore, has abundant time to bring the reactor to cold shutdown by either correcting and activating the scram system or, if necessary, by activating the ultimate shutdown system from the RSF. Thus even for such severe events, operator action is not required to maintain the plant within safety limits.

The quad redundant RPS, with a failure probability of about 3×10^{-7} per demand, automatically mitigates all challenges to plant safety through the RPS trip action. Operator action is required only for post accident monitoring, communication with outside authorities, and initiation of recovery actions. The ALMR's passive and inherent safety features, together with its automatic and highly reliable RPS, allow the ALMR operator to have such a passive safety role.

For normal operation, the PCS and the CR operator control the reactor and prevent challenges to the RPS. If an event occurs and the PCS is not able to mitigate it by running the reactor back, the RPS automatically scrams the reactor and prevents challenges to reactor safety. The operator can initiate a scram manually if he wants the scram to occur before RPS setpoints are reached, or if he feels that the automatic scram function has malfunctioned. Normally he would request scram from the PCS operator's console, but if that has failed he can request scram from the Class 1E console in the RSF.

Thus for ALMR, the only functions that require the operator to interface directly with Class 1E safety related systems, are the following:

a. Manual scram and post-scram safety parameter monitoring of each reactor in the plant.
b. Plant post-accident monitoring.
c. Permissive for updating the neutron flux detector calibration factors.
d. Permissive for adjusting the rod stops.
e. Initiation of the ultimate shutdown system.

These Class 1E functions require no complicated computations or man-machine interactions, and there is no need to use the CR process computers and console to perform them. Moreover, because of the ALMR's passive and inherent safety features, Functions a, b and e, which represent operator actions in response to plant transients, are performed infrequently, and the operator has a long time to initiate them.

Hence, it is prudent and justified to physically and electrically separate the non-safety related operator interface in the CR from the safety related operator interface in the RSF, with the CR consoles connected to the PCS computers and the RSF console connected to RPS electronics. This separation allows the control room electronics to be designed with computationally powerful computers for optimum plant operation without being burdened with safety functions that are not required for the operation function while providing a separate Class 1E facility, in close proximity to the CR, from which all operator monitoring and control interactions with the plant safety systems can be performed.

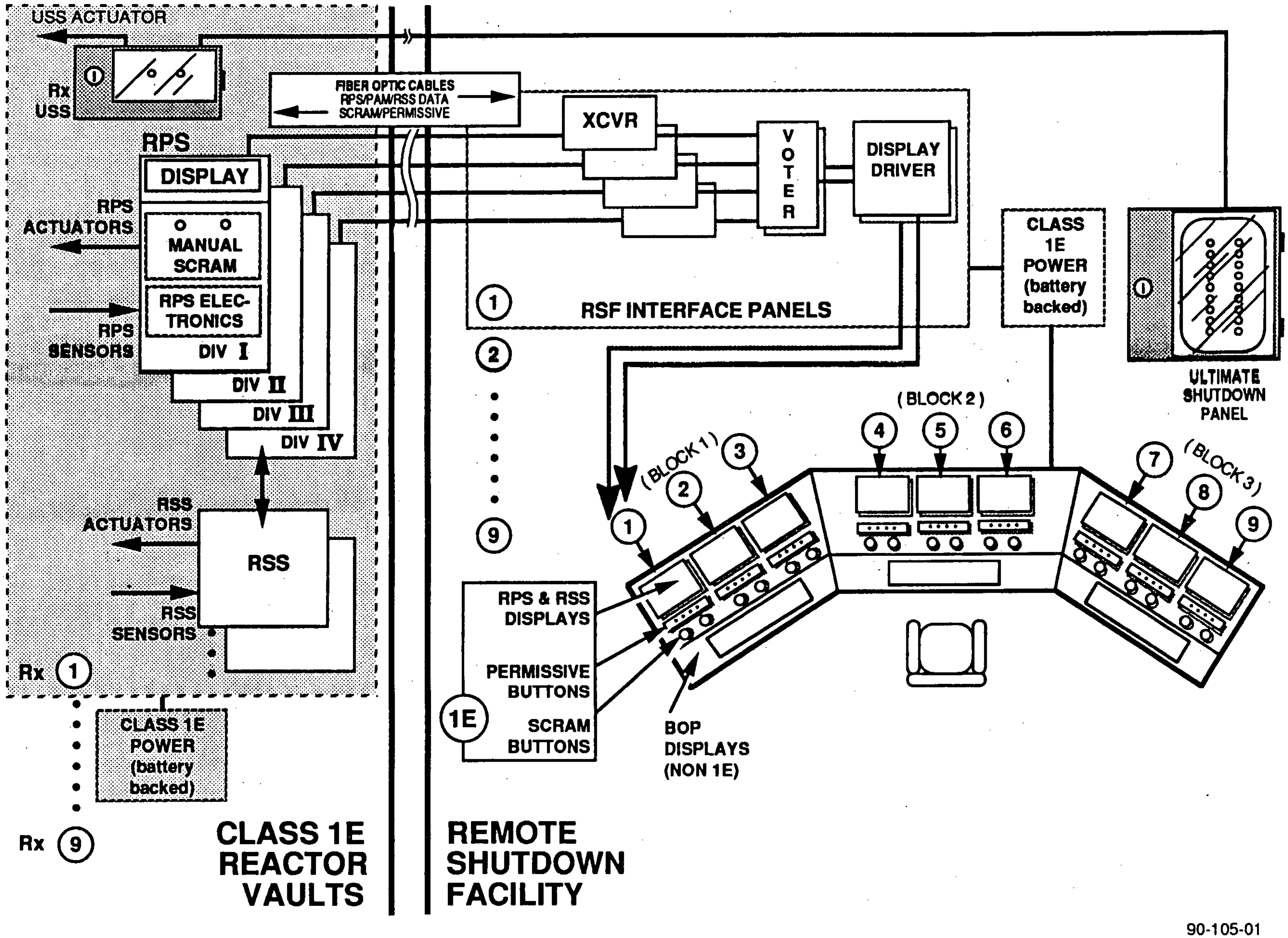


Figure G.4.10–1 CLASS 1E REMOTE SHUTDOWN FACILITY CONCEPT

G.4.11 Emergency Preparedness

G.4.11.1 SER Position on Emergency Preparedness

Sections 3.1.2.4, 13.1.1 and 13.1.4 of the draft SER (NUREG 1368) address the issue of off-site emergency planning. In section 3.1.2.4, the Staff states:

> "In the past, the Commission has not required off-site emergency planning in those situations where the lower level PAGs (Protective Action Guidelines) are not expected to be exceeded. For example, emergency planning for research reactors is restricted to the area around the reactor where the lower level PAGs are expected to be exceeded. This is usually within the owner-controlled area. For fuel cycle facilities, the final rule on emergency preparedness exempts those facilities where the lower level PAGs will not be reached outside the owner-controlled areas. Therefore, there is a precedent for not requiring off-site emergency planning beyond simple notifications, where warranted by operation."
>
> "Specifically, the Staff proposes that PRISM meet the following criteria if traditional off-site emergency planning (other than simple notification) is not provided:
>
> > An off-site emergency plan should be prepared, however, such a plan would not have to include early notification, detailed evacuation planning, and provisions for exercising the plan if:
> >
> > o the lower level PAGs are not predicted to be exceeded at the site boundary within the first 36 hours following any event in categories EC-I, II and III and
> >
> > o a PRA for the plant that includes at least all events in categories EC-I through EC-IV and indicates that the cumulative mean value frequency of exceeding the lower level PAGs at the site boundary within the first 36 hours does not exceed approximately 10^{-6}/year."

In Section 13.1.1, the Staff notes that "...the current policy of the Commission is that off-site emergency planning is a requirement for the licensing and operation of a nuclear power plant..." Therefore, application of the above criteria to a power plant requires a change in Commission policy.

In Section 13.1.4, the Staff states it "...cannot conclude that PRISM has the potential to meet the above criteria. In particular, our concerns are with the predicted PRISM response to certain of the bounding EC-III events as discussed in Chapter 6." Referral to Section 6.2.6 and Table 3-1 reveals that the bounding events of concern are:

BE-1b Inadvertent withdrawal of all control rods without scram for 36 hours, with RVACS cooling only

BE-3 Loss of forced cooling plus loss of ACS/RVACS, with 25 percent of RVACS unblocked after 36 hours

BE-4 Instantaneous loss of flow from one primary pump with failure to scram, and coastdown flow for other pumps

BE-7 Blockage of flow to or from one fuel assembly

G.4.11.2 Emergency Preparedness Approach for Reference ALMR

A design objective of the reference ALMR is to meet the criteria on lower level PAG levels specified by the Staff in Section 3.1.2.4 of the draft SER such that formal off-site emergency planning involving early notification, detailed evacuation planning, and provisions for exercise of the plan are not required. It is assumed that the Commission will agree to a change in policy permitting application of these criteria to power plants.

In order to attain the objective stated above, the reference ALMR design emphasizes accident prevention, long response times between the initiation of an accident and the release of any radiation, and containment and mitigation of accidents if they should occur.

A number of features contribute to the accident prevention capability of the ALMR. Chief among these are the inherent reactivity feedbacks which terminate power ramps resulting from ATWS events, the passive RVACS decay heat removal system, a highly reliable reactor protection system, the low operating pressure and non-corrosive nature of the sodium coolant, and the separation of the safety grade automatic reactor protection system from the non-safety grade plant control system in order to minimize operator error as a contributor to accidents. Supplementing these primary features are a number of secondary features including redundant and diverse reactor shutdown systems, passive gas expansion modules to help terminate loss of flow accidents, electronically positioned mechanical rod stops to limit reactivity insertion during control rod withdrawal accidents, the pool design of the primary system which eliminates any external piping carrying primary coolant during operation, and seismic isolation of essentially all the safety related equipment. As discussed in Section G.4.11.3, these accident prevention features enable the ALMR to meet the NRC safety goals on prompt fatalities and long term cancer fatalities by prevention alone.

A number of features contribute to the long time constants between accident initiation and the release of any radioactivity to the environment. Chief among these are the large heat capacity and fission product scrubbing capability of the primary sodium pool, the large margin between sodium operating temperatures and the sodium boiling temperature, the high heat conductivity of the metal fuel which limits its temperature rise, and the long holdup time of the containment. Supplementing these primary features are a number of secondary features including hermetic seals on all primary coolant boundary openings, design of the core support structure to maintain any core debris within the structure in a subcritical and coolable state, and the increase in RVACS heat removal capability as vessel temperatures rise. As discussed in Section G.4.11.3, these long time constants enable the ALMR to meet the 36 hour requirement with considerable margin.

Sections G.4.1 and G.4.19 discuss ALMR containment and mitigation of severe accidents, respectively. The reactor enclosure is being designed to withstand the consequences of a maximum hypothetical core disruptive accident (HCDA) without breach of the primary system boundary. The reactor lower internal structure is being designed to hold subcritical and keep cool a full core melt without breach of the primary system boundary. In the event that an accident somehow breaches the primary system boundary, the containment is being designed to mitigate any off-site radiological release to levels well below the protective action guidelines.

The staff has stated its belief in Section 3.1.2.4 of the draft SER that emergency planning requirements for advanced reactors should be based upon the characteristics of those designs. The ALMR characteristics of prevention, long time constants, and containment/mitigation combine to produce a reactor that does not exceed the lower level PAGs and gives ample time for ad hoc evacuation based on the staff's 36 hour guideline. Therefore, the ALMR approach to emergency preparedness is to develop an emergency plan, but not include provisions for early notification, detailed evacuation planning, and exercising of the off-site plan. The advantage of not requiring these provisions is considerable, since extensive coordination between the reactor licensee and local government authorities is not required. In addition, such a plan should enhance public acceptance.

The emergency plan will be developed in accordance with applicable requirements set forth in 10CFR50.47 and 10CFR50, Appendix E for emergency planning, and in NUREG 0654, Criteria for Preparation and Evaluation of Radiological Emergency Response. It will identify notification and communication methods for alerting responsible off-site individuals in the event of a severe accident. It will be directed to ensuring that:

- Adequate measures are taken to protect employees and the public
- All individuals having responsibilities during an accident are properly trained
- Procedures exist to provide the capability to cope with a spectrum of accidents ranging from those of little consequence to those associated with a major radioactive release to containment

- o Equipment is available to detect, assess, and mitigate the consequences of such occurrences

- o Emergency action levels and procedures are established to assist in making decisions

G.4.11.3 Rationale for Emergency Preparedness Approach

This section discusses the analyses and evaluations that have been performed to justify the approach to emergency preparedness discussed in Section G.4.11.2. The discussion addresses accident prevention, long response times, and accident containment/mitigation.

<u>**Accident Prevention**</u>

In order to place numerical values on accident prevention probabilities, it is necessary to define a complete list of initiating events. Such a list has been prepared for the reference ALMR by developing a top-down deductive (fault tree) logic, a bottom-up inductive (FEMA) logic, and comparing with checklists from past probabilistic risk assessments (PRAs) and safety analyses. This approach accounts for initiators caused by front line system faults, support system faults, and external events.

Seven generic initiating event groups have been identified using the above approach. These are:

a. Reactivity insertion
b. Core flow reduction and primary coolant boundary leaks
c. Heat removal reduction
d. Local faults
e. Electric power system faults
f. Transients
g. External events

System analyses were then performed on each of the generic initiating event groups to identify internal faults, interfacing system faults, systems interactions, and system dependencies. The resulting initiating events are listed in Table G.4.11-1.

Table G.4.11-1

LIST OF INITIATING EVENTS

Event	Frequency Per Module Year	Main Contributors or Bounded Events
A. Reactivity Insertion		
1. 0 - 6¢ step	$2x10^{-3}$	- Stick and slip - nominal control assembly insertion - 0-0.2g earthquake - Withdrawal of one control assembly
2. Potential for 7 - 18¢ step	10^{-3}	- Stick and slip - maximum control assembly insertion - 0.2 - 0.6 g earthquake - Withdrawal of 2 or 3 control rods
3. Potential for 19 - 36¢ step at 2¢/sec	5x10-4	- Withdrawal of 4 to 6 control rods at nominal speed
4. Potential for >36¢ ramp at 2¢/sec or 19-36¢ step	$5x10^{-5}$	- Withdrawal of 4 to 6 control rods with failure of rod stop - 0.6 - 1.2g earthquake
B. Core Flow Reduction and Primary Coolant Boundary Leaks		
1. Coastdown of one EM pump	0.2	- Loss of power to one EM pump - Discharge line leak

Table G.4.11-1 (Continued)

LIST OF INITIATING EVENTS

Event	Frequency Per Module Year	Main Contributors or Bounded Events
B. <u>Core Flow Reduction and Primary Coolant Boundary Leaks</u> (Cont'd)		
2. Instantaneous loss of flow from one EM pump	0.15	- Failure of EM pump coils - Fire in cable between EM pump and breaker - Breaker short to ground - Discharge line break
3. Coastdown of two EM pumps with IHTS pump trip	$2x10^{-3}$	- Loss of power to bus feeding 2 EM pumps and IHTS pump
4. Coastdown of two EM pumps without IHTS pump trip	$2x10^{-3}$	- Loss of power to bus feeding 2 EM pumps and SG recirculation pump
5. Instantaneous loss of flow from one EM pump with coastdown of another	$2x10^{-3}$	- Instantaneous loss of pumping power from one EM pump (Event B.2) which causes instability in bus common with the other pump causing the breaker to open
6. Coastdown of four EM pumps	$5x10^{-2}$	- Loss of power from preferred power supply
7. IHX leak	10^{-3}	- IHX tube leak
8. Reactor vessel leak	10^{-6}	- Reactor vessel leak
C. <u>Heat Removal Reduction</u>		
1. IHTS pump failure	$5x10^{-2}$	- Pump shaft seizure - Pump leak
2. IHTS piping leak	10^{-2}	- IHTS piping leak
3. Small SG leak	10^{-3}	- SG leak detected by hydrogen detectors - shutdown & repair - no further consequences

Table G.4.11-1 (Continued)

LIST OF INITIATING EVENTS

Event	Frequency Per Module Year	Main Contributors or Bounded Events
C. Heat Removal Reduction (Cont'd)		
4. Medium SG leak	10^{-4}	- Small leak propagating to rupture relief disks connected to equalization line - shutdown and repair - no further consequences
5. Design basis SG break	10^{-6}	- Medium leak propagating to multiple tube leaks - rupture disks break, SG water is dumped and SG is isolated - shutdown and repair - no further consequences
6. Beyond design basis (or unprotected) SG break	10^{-9}	- Design basis break with delayed or inadequate isolation or dumping of SG
7. BOP faults	0.2	- T/G trip - Condensate water system failure - Feedwater system failure
8. RVACS blockages		
<25%	10^{-1}	- Flying objects
25%-75%	10^{-3}-10^{-7}	- Tornado flying objects
75%-90%	10^{-9}	- Hail
>90%	$<10^{-9}$	- Sand/dust storms - Severe seismic event - Double vessel leak
D. Local Faults		
1. Blockage of passage between 3 neighboring fuel pins	10^{-1}	- Foreign material - Loose wire wrap - Excessive pin bowing - Cladding swelling
2. Cladding failure two or more pins	10^{-2}	- Random failure

Table G.4.11-1 (Continued)

LIST OF INITIATING EVENTS

Event	Frequency Per Module Year	Main Contributors or Bounded Events
D. Local Faults (Cont'd)		
3. Melting of one fuel rod	10^{-4}	- Excessive enrichment beyond specs and loading error
4. Partial sub-assembly inlet blockage <20%	10^{-3}	- Manufacturing and QA error
5. Subassembly blockage >85%	10^{-9}	- Pin to pin failure propagation with failure of detection and shutdown - Total inlet blockage because of manufacturing error, errors in QA preoperation testing
E. Electric Power System Faults		
1. Station blackout		
>36 hours	10^{-7}/plant yr	- Loss of preferred and reserve power supplies
<36 hours	10^{-3}/plant yr	
F. Transients		
1. Spurious scrams	0.4	- Spurious scrams - Transients inadequately handled by PCS
2. Normal shutdown	0.7	- Shutdown for refueling or scheduled maintenance
3. Forced Shutdown	0.9	- Shutdown for event not accounted for above but must be repaired before power operation restart

Table G.4.11-1 (Continued)

LIST OF INITIATING EVENTS

Event	Frequency Per Module Year	Main Contributors or Bounded Events
G. External Events		
1. 0.1-0.2g earthquake	10^{-3}	- OBE level earthquake
2. 0.2-0.4g	$3x10^{-4}$	- SSE level earthquake
3. 0.4-0.9g	$1.9x10^{-5}$	- Earthquake >SSE but well within seismic isolation capability to maintain seismic gap
4. 0.9-1.2g	$7.1x10^{-7}$	- Earthquake which may close seismic gap
5. External events used for RVACS evaluation:		
a) aircraft crash		
b) avalanche		
c) hazardous material on-site		
d) coastal edge corrosion		
e) drought		
f) internal fires		
g) external fires		
h) internal floods		
i) external floods		
j) low air temperature, snow and ice storms		
k) tornadoes		
l) hazardous material off-site		
m) land slide		
n) lightning		
o) meteorites		
p) sand/dust storms		
q) seismic events		
r) volcanic ash		
s) T/G missiles		
t) soil shrink/swell		
o) transportation accidents		

There are a few differences between the list of initiating events in Table G.4.11-1 and a similar list presented in the original PRISM PSID (Reference G.4.11-1). The new list adds the following events:

B.2 Failure of the primary coolant EM pump wiring and cable, which renders the synchronous machine of that pump ineffective

B.5 Instantaneous loss of flow in one EM pump, which leads to instability in the electric power supply, resulting in loss of power to the other EM pump connected to the same bus

G.5 Explicit external events which could lead to RVACS blockage

To assess the impact of the above differences on the risk estimates of Reference G.4.11-1, the initiating events of Reference G.4.11-1 were ranked in terms of their contribution to risk. This ranking is presented in Table G.4.11-2. It can be seen that the risk is dominated by the large seismic event (>0.9g) which contributes about 85% of the total risk. It is estimated that Events B.2 and B.5 add about 15% to the overall risk.

In order to compare the risk from these initiating events to the NRC safety goals, the safety goals must be quantified. The safety goals have been defined by the NRC Commissioners as follows (Reference G.4.11-2):

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.

- The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

Table G.4.11-2

INITIATING EVENTS RANKING

Rank	Initiating Event	% Contribution To Individual Risk	% Contribution To Societal Risk
1	Earthquake > 0.9g	84.7	86
2	Loss of One Primary Pump	10.6	9.6
3	Loss of Substantial Primary Coolant Flow	3.4	3.2
4	Loss of Operating-Power Heat Removal	1.0	0.96
5	Earthquake 0.3-0.9g	0.13	0.14
6	Potential for Reactivity Insertion of 18-36¢	$5x10^{-3}$	$5x10^{-3}$
7	RVACS Blockage	$1x10^{-5}$	$5x10^{-3}$
8	Forced Shutdown	$5x10^{-6}$	$2x10^{-3}$
9	Normal Shutdown	$2x10^{-6}$	$1x10^{-3}$
10	Potential for Reactivity Insertion >36¢	$3x10^{-7}$	$3x10^{-7}$
11	All Others	$<1x10^{-10}$	$<1x10^{-10}$

The individual risk goal can be quantified as $5x10^{-7}$ prompt fatalities per year of plant operation, while the societal risk goal can be quantified as $1.9x10^{-6}$ latent fatalities per year of plant operation. The core damage frequency from the list of initiating events in Table G.4.11-1 is estimated to be an order of magnitude or more below these values, as presented in Table G.4.11-3. Therefore, it is concluded that the reference ALMR can meet the safety goals by prevention alone.

Table G.4.11-3

CORE DAMAGE FREQUENCY FROM THE OPERATION OF AN ALMR PLANT

Risk Measure	NRC Goal (Less Than)	ALMR Core Damage Frequency
Individual Risk (Probability of Prompt Fatality Per One Year of Plant Operation, 0-1 mi)	$5x10^{-7}$	$<2x10^{-8}$
Societal Risk (Probability of Latent Cancer Fatality Per One Year of Plant Operation, 0-10 mi)	$1.9x10^{-6}$	$<2x10^{-8}$

Since the proposed Staff criteria for not requiring traditional off-site emergency planning includes the performance of a PRA for events in categories EC-I through EC-IV, the initiating events in Table G.4.11-1 have been assigned to the event categories to indicate how they will be addressed when the required PRA is performed. This allocation is shown in Table G.4.11-4.

Table G.4.11-4

RELATION OF INITIATING EVENTS TO NRC EVENT CATEGORIES

Item	Event Category EC-I	EC-II	EC-III	EC-IV
Frequency Range per Plant Year	$P>10^{-2}$	$10^{-2}>P>10^{-4}$	$10^{-4}>P>10^{-7}$	$10^{-7}>P$
Frequency Range per Module Year*	$P>10^{-3}$	$10^{-3}>P>10^{-5}$	$10^{-5}>P>10^{-8}$	$10^{-8}>P$
ALMR Events	A1,A2,B1,B2 B3,B4,B5,B6 B7,C1,C2,C3 C7,C8 (blockage <25%),D1,D2 D4,F1,F2,F3 G1	A3,A4,C4,C8 (blockage <50%),D3,E1 G2,G3	B8,C5,C8 (blockage <90%),G4	C6, C8 (blockage >90%),D5

*Obtained by dividing values per plant year by 10 as an approximation to account for 9 modules per reference ALMR plant

G.4.11.4 Long Response Time

Amendment 11 to the PSID (Reference G.4.11-3) addressed the question of response time by evaluating the time to reach each of the following five limits, following a spectrum of events:

Fuel Failure
Sodium Boiling
Safety Structural Failure
Lower Level PAG Limits
10CFR100 Limits

It was shown in Table E.9-1 of Amendment 11 that none of the above limits were ever reached for any of the design basis events or beyond design basis ATWS events evaluated in the PSID, and fuel failure limits were reached for only Bounding Event Nos. 1b and 3 added by the Staff to EC-III by engineering judgment, with no other limit being reached. Independent evaluation by the Staff and its consultants, however, identified concerns for not only Bounding Event Nos. 1b and 3, but also Bounding Event No. 4 plus Bounding Event No. 7 added after Amendment 11 was submitted (see Section 6.2.6 of the draft SER). Table G.4.11-5 lists the bounding events, and identifies the four which are of concern to the Staff.

Table G.4.11-5

BOUNDING EVENTS NRC ADDED TO EC-III BY ENGINEERING JUDGMENT

	1.	Inadvertent withdrawal of all control rods without scram for 36 hours
		a. With forced cooling
*		b. With RVACS cooling only
	2.	Station blackout for 36 hours
*	3.	Loss of forced cooling plus loss of ACS/RVACS, with 25 percent of RVACS unblocked after 36 hours
*	4.	Instantaneous loss of flow from one primary pump with failure to scram, and coastdown flow for other pumps
	5.	Steam generator tube rupture without isolation or water dump
	6.	Large sodium leak
*	7.	Blockage of flow to or from one fuel assembly
	8.	External events

* Means Staff concern

The bounding events were added to EC-III largely to test a reactor design which had an unconventional containment, and for which significant mitigation capability was not claimed. Since Amendment 11 was submitted, numerous changes have been made in the ALMR design, as described in Section G.2. In addition, more is known today about the behavior of both the reactor and the metal fuel. The result is that redefinition of Bounding Event No. 3 to stipulate only <u>75%</u> blockage of RVACS for 36 hours is warranted, based on redesign of the reactor to withstand any conceivable HCDA and a full core melt (Section G.4.19), and on the addition of a pressure retaining containment dome over the head access area (Section G.4.1). Also, it is expected that flow blockage can be limited to one fuel assembly, without propagation to adjacent assemblies (Section G.4.6). Reanalyses of the bounding events, (Section G.4.16), using the reference ALMR design and a redefined Bounding Event No. 3, show that none of the limits used in Table E.9-1 of Amendment 11 are reached for Bounding Events 1-6, with Bounding Events 7 and 8 to be evaluated in future work. Therefore, response times for the full spectrum of EC-I, EC-II and EC-III events up through Bounding Event 6 are essentially unlimited, giving ample time for operator recovery actions and/or ad hoc evacuation. It is expected that similar long response times will be substantiated for Bounding Events 7 and 8. Table G.4.11-6 updates Table E.9-1 of Amendment 11 to reflect the current design and analyses.

G.4.11.5 Containment and Mitigation

The preceding paragraphs discussed the low probability of severe events initiating, and the long response times for recovery actions and/or ad hoc evacuation if they do occur. Nevertheless, a third level of protection has been incorporated into the reference ALMR design. This third level serves to contain severe accidents within the primary system boundary, and to mitigate any release of radiation to the outside environment if the primary system boundary were somehow breached.

Section G.4.19 evaluates the two most severe accidents postulated for the ALMR - an energetic HCDA event and a full core melt. In addition, Section G.4.5 evaluates the question of sodium voiding which could lead to an HCDA. These evaluations show that voiding which could initiate an HCDA is of extremely low probability, less than 2×10^{-9} per initiating event severe enough to demand RPS action (i.e., scram). These evaluations also show that it appears feasible that the ALMR reactor vessel and closure can accommodate HCDA loads resulting from energetics on the order of 500 MJ without loss of structural integrity, disengagement of the rotatable plug from the reactor closure, or expulsion of sodium. This level is more than

Table G.4.11-6

RESPONSE TIMES FOR VARIOUS EVENTS

	TIME TO LIMITS				
EC-I, EC-II	FUEL FAILURE (hours)	SODIUM BOIL (hours)	SAFETY STRUCTURAL FAILURE (hours)	LOWER PAG LIMITS (hours)	10CFR100 LIMITS (hours)
FAST RUNBACK	*	*	*	*	*
SCRAM	*	*	*	*	*
LOSS OF NORMAL SHUTDOWN COOLING	*	*	*	*	*
LOCAL FAULTS	*	*	*	*	*
SODIUM SPILLS	*	*	*	*	*
FUEL HANDLING & STORAGE	*	*	*	*	*
COVER GAS RELEASE	*	*	*	*	*
EC-III					
UNPROTECTED LOSS OF FLOW	*	*	*	*	*
UNPROTECTED TRANSIENT OVERPOWER	*	*	*	*	*
UNPROTECTED LOSS OF HEAT SINK	*	*	*	*	*
UNPROTECTED 6-ROD TRANSIENT OVERPOWER	*	*	*	*	*
NRC BOUNDING EVENTS (EC-III)					
UNPROTECTED WITHDRAWAL OF ALL CONTROL RODS FOR 36 HOURS (with forced cooling)	*	*	*	*	*
UNPROTECTED WITHDRAWAL OF ALL CONTROL RODS FOR 36 HOURS (RVACS only)	*	*	*	*	*
STATION BLACKOUT FOR 36 HOURS	*	*	*	*	*
LOSS OF FORCED COOLING, LOSS OF ACS, LOSS OF 75% OF RVACS, WITH SCRAM FOR 36 HOURS	*	*	*	*	*
INSTANTANEOUS LOSS OF FLOW FROM ONE PRIMARY PUMP	*	*	*	*	*
STREAM GENERATOR TUBE RUPTURE WITHOUT ISOLATION OR WATER DUMP	*	*	*	*	*
LARGE SODIUM LEAK	*	*	*	*	*
FLOW BLOCKAGE TO OR FROM ONE FUEL ASSEMBLY	-------REQUIRES FURTHER EVALUATION-------------				
EXTERNAL EVENTS	--------------- TO BE EVALUATED ---------------				

NOTE: * means reactor conditions stabilize, and limits are never reached an order of magnitude greater than the anticipated energetics from a HCDA in an ALMR metal fuel core. Finally, these evaluations show that if a full core melt occurred, perhaps initiated by an HCDA event, it appears feasible that the melt will be contained within the core support structure in a subcritical and coolable geometry. Based on these evaluations, there would be no threat to the public, and hence no requirement for evacuation.

Even if the primary system boundary were somehow breached, the low leakage containment structure, consisting of the containment vessel and containment dome, would provide a holdup and attenuation function to mitigate the consequences of such a remote event. Section G.4.1 evaluates the ALMR containment for the design basis conditions postulated in Table G.4.11-7.

Table G.4.11-7

CONTAINMENT DESIGN BASIS

	Item	Magnitude: Early Phase (0-10 Sec)	Magnitude: Sodium Fire Phase (10 Sec - 6 Hrs
A.	Materials Released to Containment Through Reactor Closure		
	Noble Gases (Xe, Kr)	100%	0%
	Halogens (Br, I)	0.1%	0.8%
	Alkali Metals (Cs, Rb)	0.1%	1.6%
	Te, Ru	0.1%	0.004%
	Sr, Ba	0.01%	0.0016%
	Fuel & Other Fission Products	0.01%	0.0008%
	Na-22, Na-24	None	0.4%
B.	Energy Sources		
	Sodium Fire (Within Reactor)	None	~1700 lbs
	Decay Heat	Yes	Yes
C.	Leak Rate (Containment Dome)	<1%/day @ 25 psig/700°F	

Calculations discussed in Section G.4.1 show that the above releases are not a severe challenge to containment, and that the release of radiation to the environment is well within the lower level PAGs. Section G.4.1 also discusses maintenance and refueling accidents, and shows that the release of radiation to the environment for these events is also within the lower level PAGs.

G.4.11.6 Summary

Work performed to date on the ALMR design shows:

a. there is a high probability of meeting the NRC safety goals and lower level PAGs by prevention alone,

b. there are long response times between the initiation of an accident and any radiation release, and

c. the design incorporates robust capability to contain and mitigate the consequences of severe accidents to levels below the NRC safety goals and the lower level PAGs.

Additional work, including analysis, laboratory testing, and the prototype test, must be performed to evaluate and quantify these attributes in greater detail and to a higher confidence level. However, the work completed to date indicates that the ALMR design does possess these attributes. If the additional future work confirms this, then a basis exists for adopting an off-site emergency plan which does not include early notification, detailed evacuation planning, and provisions for exercise of the plan.

G.4.11.7 References

G.4.11-1 "PRISM Preliminary Safety Information Document", GEFR-00793, November 1986.

G.4.11-2 "Safety Goals for the Operations of Nuclear Power Plants; Policy Statement", Nuclear Regulatory Commission, Federal Register, Vol. 51, No. 149, Monday, August 4, 1986.

G.4.11-3 "Transmittal of Amendment 11 to the Preliminary Safety Information Document (PSID)", GE Letter XL-897-88108, June 3, 1988.

G.4.12 Role of Operator

G.4.12.1 SER Position On The Role Of The Operator

The operator must be protected and provided with appropriate communications. The operator is considered as a backup to the safety systems. (Reference SER Section 13.2.3)

G.4.12.2 Reference Approach for Operator's Role

The Seismic Category II, tornado hardened control building is located within the nuclear island protected area boundary. Included as part of the control building are the control room (CR), technical support center, information management center, and communication room. The control room is designed to protect the operator from environmental hazards and to ensure access to the adjacent remote shutdown facility. The location of the control room also protects the operator from potential personnel intrusion.

The remote shutdown facility (RSF) is a Seismic Category I, tornado hardened structure located in the radwaste building adjacent to the control building. Access between the control room and the remote shutdown facility is provided through a 120-foot long, Seismic Category II, tornado hardened tunnel link. An uninterruptible Class 1E power supply with a 36-hour capacity is provided, as a backup, for the remote shutdown facility electrical and habitability systems. A Class 1E interface to the safety systems is provided in the remote shutdown facility. A detailed description of the remote shutdown facility and operator habitability features is provided in Section G.4.10.2.

For the ALMR, safety actions are performed automatically by the reactor protection system (RPS) (which scrams the control rods whenever safety setpoints are exceeded and then turns off the primary EM pumps), the synchronous machine (which passively provides primary flow coastdown), and the passive RVACS (which removes reactor heat through natural air circulation around the reactor vessel). The ALMR operator can scram the plant manually either from the non-safety related CR console or from a Class 1E interface provided in the RSF. Capability for shutdown and post-accident monitoring (which includes RPS and other Class 1E sensor data) is provided at the non-Class 1E CR console and at the Class 1E console in the RSF. The Class 1E data are buffered by safety related coupling devices.

ALMR operator actions are primarily related to optimal operation of the plant, high plant availability, and protection of the plant equipment. These actions are not safety related and are performed from the CR console. The CR is the center of plant operations. The CR console and its displays provide the operator with well integrated plant state information, diagnostics, and operator aids as needed for highly efficient plant operation. The ALMR is highly automated with manual backup provided for all automatic actions. Recovery actions for service and startup, following a scram or normal outage, are performed by the operator from the CR. Voice and TV communication is provided in the CR and with all on-site and off-site facilities with which the CR operator communicates. Capability is provided for the CR operator to communicate with roving operators using walkie-talkies. Battery backup for eight hours is provided for all CR equipment.

In the event of an accident where the CR is unavailable, the RSF is used to initiate Class 1E scram and perform Class 1E shutdown, post-accident monitoring, and initiation of recovery actions. The operator also maintains communications with both on-site and off-site locations from the RSF. Normally all accident monitoring data is automatically forwarded to the on- and off-site locations. The primary links to all off-site locations are telephone lines, with microwave links provided as a backup. The operator performs a backup communication role using portable radio if the automatic data links are not working.

In addition to the scram and monitoring interface with the RPS, the operator also has an interface with the rod stop system (RSS) and RPS where he provides permissives to allow these systems to make adjustments. The adjustments are really made by these Class 1E systems, and the operator's role is merely that of providing permissives to implement them.

The safety grade man-machine interface electronics in the RSF are backed up by equivalent safety grade electronics in the RPS instrument vaults. A description of the RPS vault facility features is given in G.10.4.2.

G.4.12.3 Rationale For Operator's Role

The ALMR operator provides an additional line of defense in accident situations. In this role, the operator:

a. Monitors and verifies performance of safety systems, and has the capability to initiate reactor shutdown by manual scram or manual activation of ultimate shutdown system,

b. Maintains communication with on-site and off-site personnel,

c. Initiates recovery actions following an event.

Plant facilities have been designed considering the functions the operator must perform during normal plant operations and in response to off-normal and design basis events. The facilities provide requisite operator protection and habitability, and the electrical systems provide the necessary reliability and redundancy to ensure the operator can perform his responsibilities in support of plant safety systems.

The location of the control room (CR) within the nuclear island (NI) protected area boundary protects the operator from potential intrusion. The classification and design of the Seismic Category II, tornado hardened control room ensures the operator is protected from natural phenomena. In the unlikely occurrence of a natural disaster or under any other accident environment which renders the control room uninhabitable, the operator can safely proceed to the adjacent remote shutdown facility through the Seismic Category II, tornado hardened tunnel link with the control room. The remote shutdown facility is a Seismic Category I, tornado hardened structure with Class 1E batteries which supply uninterruptible power, as a backup, for 36 hours for post-accident monitoring, communications, and habitability functions.

Because of the passive and inherent response of the plant to safety challenges, operator actions are not immediately required. However, the operator does have a role in post-accident monitoring, communications with on-site and off-site authorities, and initiation and direction of recovery actions. Communications with the technical support center, operations support center, and, through a data link, with the emergency off-site facility are conducted by the operator from the control room and, alternatively, from the remote shutdown facility.

Diverse means of reactor shutdown are available for operator action. Normal fast runback for reactor shut down is accomplished by plant control system action. The operator can also initiate a non-Class 1E reactor scram from the control room console. A safety grade, Class 1E reactor scram can be initiated from the remote shutdown facility. Passive and inherent reactor responses will terminate ATWS events at low reactor power and acceptable temperatures in the unlikely event that all active systems fail to function. Activation of the ultimate shutdown system by an operator from the remote shutdown facility will insert B_4C balls into a center core assembly. The reactivity worth of the absorber is sufficient to bring the reactor from full power to cold shutdown.

If the RPS is functional, it automatically protects against any event that threatens plant safety, and the operator has no direct safety role. Even if the RPS fails to scram the reactor, the passive features bring the reactor to a stable condition and give the operator abundant time to bring the reactor to a cold shut down condition. Although the operator (who has the capability of providing manual backup for all automatic control actions) can manually scram the reactor if the RPS fails, operator action is not required to safely mitigate ATWS events.

Consequently, ALMR operator interaction with plant safety systems has been specifically designed to provide all the monitoring capability the operator needs, but with very limited control capability such that plant safety cannot be degraded no matter what actions are taken.

G.4.13 Multi-module Control

G.4.13.1 SER Position on Multi-Module Control

Operation with multi-module control needs to be adequately demonstrated (Reference SER Section 13.2.4).

G.4.13.2 Reference Multi-Module Control Design Features and Approach

The ALMR power block consists of three reactors, each having one intermediate loop and one steam generator, with the three steam generators headered together to provide steam to a single turbine. From the control point of view, this configuration is similar to many three loop monolithic plants (LMFBRs and PWRs) except the ALMR has more operational flexibility because of the multiple reactors. The reactors are small and simple (only six control rods, no valves) and since the loops are not coupled to the same reactor, they do not need to be matched and balanced. For these reasons, control of the ALMR power block is judged to be simpler than control of existing monolithic plants.

A four stage plan has been established for development of the ALMR control system. This plan shows that the ALMR multi-module control will be adequately tested and demonstrated as the design progresses. In the first stage, control models are developed and tested using simulation codes. The design is presently in this stage, with the control models being developed and tested at GE and ORNL. Man-machine interface requirements will also be developed in this stage using task analysis for normal and faulted operating conditions. In the second stage, the control models will be loaded as software into prototype controllers, and a hybrid test will be performed with these prototype controllers connected to a real-time plant simulator. During this stage, a man-machine interface prototype will be developed and connected to the plant simulator for operator response evaluation and task analysis verification. Key features of the ALMR control system design will also be tested at EBR-II. In the third stage, integrated testing with the prototype controllers and man-machine interface connected to the plant simulator will be performed. This test will ensure that the controllers and operator's console for the ALMR prototype test will meet performance requirements. Finally, in the fourth stage, the ALMR prototype test will be conducted. In this test, the prototype module will send real plant data to the controllers and operator's console, and the other modules will be simulated. Results from this test will be used to finalize the controller and man-machine interface designs.

The two key aspects of ensuring that ALMR multi-module control system meets requirements are: (1) demonstrating that a single operator can safely and efficiently monitor and direct the operation of a power block, and (2) through simulation and actual operation show that the multi-module controller is stable and properly terminates plant upsets. The reference design and verification plan in these two areas is described below.

a. The PCS contains a sufficient level of automation and operator aids to enable one control room operator to direct the operation of each power block. To simplify plant operation, the ALMR features constant flow sodium pumps and constant speed recirculation and feedwater pumps. Use of a passive safety grade cooling system (RVACS) in the ALMR also greatly reduces the operators' interface with safety systems. The power block operator directs operations from a single control room console, which is provided with multiple touch screens and touch panels, as well as a large overview screen. The shift supervisor and roving local operators are available to assist power block operation when requested. The acceptability of the operator-controller interfaces and operator workloads are determined by a series of studies and tests, beginning with an initial allocation of operator and controller functions, followed by real time simulator and control room and console mockup studies, and then actual prototype reactor operation and testing.

b. The ALMR control system uses model-based optimal controllers for improved plant operation. These controllers are more robust and provide improved capability of responding to and terminating upset events. Conventional proportional-integral controller models have been used previously in ALMR simulations studies and have demonstrated the feasibility of multi-module control for various events and power levels. The new optimal controllers now being developed have shown improved performance under simulated testing. These controllers are directed by improved block and plant level supervisory controllers which utilize fault diagnostics and a knowledge of the current and desired final operating conditions to select the proper plant operating strategy. Real time testing of the controllers using distributed hardware will follow. These controllers will be then used to operate the prototype ALMR module and simulate two additional reactor modules for final multi-module control verification.

G.4.13.3 Rationale for Multi-Module Control

Automation and operator aids are selected to ensure the vast majority of the operator's time is available for monitoring power block operation. This enables the operator to have a "minds-on" instead of a "hands-on" approach to plant operation, which allows the operator to have a better overview of power block operations without becoming extensively involved in manual manipulations. Although capability to manually control all major actuators is available, most operator manual control actions under normal conditions consist of directing changes in the modes of operation and granting permissives to continue fixed, automatically controlled sequences of operation. The large negative reactivity feedbacks and the constant speed pumps used during power operation greatly simplify the operator's tasks in directing operation of the ALMR power block. Since operator actions are not required for immediate response to insure reactor safety (as described in G.4.12.2), the operator can concentrate on efficient, economical power block operation. Availability of multiple displays and all operator interfaces at a single control console in the control room, minimizes the physical travel required by the operator, and provides for a higher rate of information transfer between the operator and the power block. The shift supervisor, roving local operators, technical support staff, and maintainers are all provided with continuous displays of the detailed power block parameters to assist in normal operations as well as evaluation of potential component failures, resolution of anomalies during operation, and maintenance or refueling operations. Preliminary operational studies support the design goal of one senior operator being able to operate one power block safely and economically. More extensive studies will follow as the project progresses, with further task analyses and further refinements in the allocation of workload between the operator and controller. Adequacy of the man-machine interface, including operator aids and displays for both normal and a wide range of upset plant conditions, will be demonstrated using a control room mockup connected to a real time multi-module simulator. Final design of the man-machine interface will be based on actual experience obtained during the ALMR prototype test with one reactor module and the real time multi-module simulation tests. This design process will produce a man-machine interface design in accordance with applicable NRC guidelines.

The design of power block and local controllers will be thoroughly evaluated by a planned series of design steps and simulation studies prior to plant operation. Power block controller stability has already been demonstrated in simulations using proportional-integral controllers for: a) single and multiple module scrams and fast runbacks, b) load following over the 25% to 100% power range, and c) a number of limiting single module events such as IHTS sodium pump seizure. The simulations have been used to verify and refine operating strategies, which are being incorporated in the supervisory controller. Modern optimal controllers are being developed at GE and ORNL for improving local and supervisory control. The complexity of these controllers is greatly reduced by the simple plant control configuration including use of constant sodium flow and constant speed feedwater pumps. The effectiveness of these controllers for both the control and diagnostic functions will be evaluated starting from local subsystem engineering simulations and progressing to the use of a full scale dedicated control room simulation facility. Actual controllers will be interfaced with the plant simulation to verify controller performance. This will also permit real-time preoperational testing of the controller-to-plant interfaces. Plans also call for demonstrating key features of these controllers in operating test reactors. These pre-tested plant controllers will be used to operate the prototype ALMR module. The prototype module will be tested under various conditions involving a wide range of parameters to ensure adequate verification of the controller performance. These tests will include module startup, normal power operation, shutdown, and several module transients. Controller designs will be refined based on the prototype test results. The major non-prototypic feature of the prototype tests is that it only has one module instead of the three modules in the ALMR power block. However extensive real-time simulation of the multi-module configuration, using actual prototype module performance data, will provide sufficient verification of multi-module control.

The satisfactory controllability and operability of existing nuclear plants with multiple steam generators, and fossil-fired plants with multiple boilers and multiple steam generators, demonstrate the feasibility of multi-module control. Nuclear plants of this type include PWRs such as Muelheim-Kaerlich, gas cooled plants such as WYLFA A & B in Great Britain, and LMFBRs such as SUPER PHENIX at Creys-Malville, France. Fossil-fired plants of this type include the Lyondell cogeneration facility in Houston, Texas with five boilers (which recover heat from five gas turbines) and five parallel steam generators providing steam to one steam turbine-generator.

G.4.14 Security

G.4.14.1 SER Position on Security

Ten open security issues are identified in Section 13.3.3 of the SER. The most significant issues are Item 9, location of the control room and Item 10, location of the sodium-water reaction pressure relief subsystem (SWRPRS). The present location of the control room and the SWRPRS outside the protected area may increase the vulnerability to sabotage. Protection of the operators is important since they represent an important source of knowledge concerning the plant status, design, and behavior which could prove extremely valuable in understanding, responding to, and recovering from an accident situation. The SWRPRS is important in maintaining primary system and containment integrity. Additional assessment is needed to support its present location outside the protected area or, alternatively, locate key features of the SWRPRS inside the protected area.

The following items summarize the ten open security issues cited in the SER. The item numbers below correspond to the numbers in Section 13.3.3 of the SER.

1. A design change (such as including a suitable Curie point magnet backup scram system) should be considered that would reduce reliance on security systems for prevention of a sabotage-induced loss of flow ATWS event.

2. Exceptions to the isolation zone requirements at the NI guard house and at the warehouse are conditionally acceptable. Coverage of roofs, walls, and interior structures of these buildings by intrusion detection and assessment systems will be needed at the interface between the balance of plant and protected areas. Alarms on vital area access points will also be needed to comply with 10 CFR 73.55(D) and (8).

3. A site plan that placed the protected area perimeter farther from the vital areas 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 of response.

4. A combination of barriers and intrusion detection alarms are needed to ensure that the plant response force is alerted to attempts to penetrate RVACS vents and inspection ports in time to intervene.

5. Door alarm mechanisms will have to be selected for vital area doors which provide adequate delay while ensuring provision for timely access and rapid exit for emergency situations.

6. On-site secondary power supplies for security equipment is required to be protected as vital by 10 CFR 73.44(e).

7. Response time for local law enforcement authorities to arrive in force must be considered before recommended reductions from the nominal force of 10 specified in 10 CFR 73.55 would be considered acceptable.

8. Plant design for protection against insider sabotage threat will be considered in subsequent revisions of the Safeguards and Security report and the risk of tampering and vandalism will be reported in a probabilistic risk assessment (PRA) update.

9. The operations center should be located within the NI protected area, with bullet restraint alarm stations and equipment, and personnel in one location to ensure that a single adversary action could not negate the security force's effectiveness.

10. Additional justification is needed to support the location of the SWRPRS outside the protected area.

G.4.14.2 Reference Design Features and Approach For Plant Security

A conceptual physical security system has been developed, based on the unique aspects of the ALMR plant, which meets regulatory requirements, addresses the design basis threat, and minimizes interference with reactor safety, operations, and maintenance. Underlying this security concept are four layers of plant protection:

- physical security
- design features
- damage control elements
- mitigation

Plant design features enhancing safety and physical security include:

- Inherent reactor shutdown through negative reactivity feedbacks
- Safety grade reactor shutdown and passive decay heat removal
- Embedded reactor modules protected by the reactor silo and head access area structures
- Self-protecting fuel and blanket assemblies
- Safety grade remote shutdown facility
- Safety grade plant facilities and systems located within the NI protected area (except for the SWRPRS rupture discs)
- Event Category III events do not result in off-site radiological consequences
- The reactor primary coolant system boundary is not challenged in any design basis or Event Category III event

The plot plan for the reference plant design, shown in Figure G.4.14-1, has been modified, changing the arrangement and location of several key buildings. Significant changes which impact the plant security capabilities are summarized in Table G.4.14-1.

Table G.4.14-1

PLANT BUILDING LOCATION CHANGES

Building, Facility	Location in 1986-1987 Design	Location in Reference Design
Control Building	Outside NI Protected Area	Inside NI protected Area
Remote Shutdown Facility (and Post Accident Monitoring Facility)	Non-safety Grade Facility in Reactor Service Building	Safety Grade, Seismic Category I, Tornado Hardened Structure in Radioactive Waste Building
NI Guardhouse and Personnel Service Building	Shared Common Wall With Administration Building	NI Guardhouse and Personnel Service Building Moved Within NI Protected Area
Location of SWRPRS	In Steam Generator Building	In Steam Generator Building With Additional Access Controls
Warehouse	In BOP Area With Access From NI	Separate Warehouses for NI and BOP

The four tier plant physical security scheme consists of:

- o Owner controlled area with a conventional security boundary at the site boundary
- o Balance of plant (BOP) area which includes non-safety related facilities and systems with manned access and personnel controls
- o Nuclear island (NI) protected area containing all safety related facilities and systems
- o An optional fuel cycle facility area located within the NI protected area and provided with an additional manned access control station and an additional physical barrier.

The protected area boundary located around the NI perimeter consists of a vehicle barrier, two chain link fences, inner and outer isolation zones, intrusion detection and assessment sensors, exterior lighting, and continuously manned access control stations for personnel entry. An uninterruptible power supply, protected as vital, provides power for security system electronics and exterior lighting for the NI protected area boundary.

The control building has been relocated within the high security boundary in reference plant design.

The location of the SWRPRS in the steam generator building (SGB) presents a special case for plant security. The ability of the SWRPRS to operate correctly is essential to protect the IHX and reactor pressure boundary from the effects of a sodium water reaction. Although the SGB is located outside the NI protected boundary, additional features have been added to the SGB to protect the SWRPRS from sabotage and terrorist attacks. Limited access to the SGB equipment area is achieved through tamper resistant access control to the building with a second controlled access to the steam generator silo where the SWRPRS equipment and piping are located. Access to the electrical equipment vaults and control and monitoring room is unrestricted but separate from the access to the SGB equipment area. Evaluation of the SWRPS is continuing to ensure no credible sabotage action in the SGB can cause a significant radiological accident or release in the reactor.

G.4.14.3 Rationale Supporting Reference Plant Security Design

Responses to the ten issues cited in the SER are provided in this section. The numbers of each response correspond with the issue numbers of the SER. Since these issues deal with security and safeguards information, as defined by 10 CFR 73, which is subject to limited and controlled distribution, only general plant security features are discussed which are responsive to the SER issues. Additional details of the ALMR plant security systems and threat assessment are provided in the ALMR Safeguards and Security Assessment report (Reference G.4.14-1).

Issue No. 1 - Passive Response to Loss of Flow ATWS Event:

Modifications to the reactivity control and shutdown system include the addition of three gas expansion modules (GEM), electronically positioned mechanical rod stops, and an ultimate (cold) shutdown system (USS) occupying the center core position. The GEMs are passively activated through pressure changes in response to loss of flow events and require no operator action or active systems. They respond to an initiating event by replacing sodium within each GEM assembly with gas, thereby adding negative reactivity to the core. The electronically controlled mechanical rod stops can be repositioned only with operator permissives after the rod stop controller has determined that a position adjustment is necessary for continued full power operation. Rod stop position adjustment is limited to an equivalent reactivity addition of not more than 0.40$. Activation of the USS will bring the reactor to a cold shutdown condition from the stable power levels attained by inherent negative reactivity responses following ATWS events. These additions ensure a safe, acceptable response to unprotected loss of flow events and unprotected rod withdrawal events, and ensure the reactor can be brought to a cold shutdown state from 135 % power levels achieved through inherency and reactivity feedback effects following ATWS events. Additional details on these reactivity control and shutdown systems can be found in Sections G.2, Design Description, and G.4.16, Safety Analyses.

Issue No. 2 - Isolation Zone Requirements:

The NI guardhouse is located within the NI away from the BOP area as shown on the plot plan, Figure G.4.14-1. Only one outside wall, at the entrance to the NI, is part of the NI perimeter. This hardened, unscalable wall is constructed to a height of no less than 18 feet above ground level. Wall penetrations are minimized. Penetration detection and assessment sensors, intrusion detection sensors, and physical barriers are provided at personnel and vehicle access points to the NI. The NI personnel service building is relocated within the NI protected area and provided with a minimum 20-foot clear zone from the security fences. A separate warehouse located within the NI protected area is provided for the NI. There is a minimum 20-foot clear zone maintained between the BOP facilities and the NI protected area outer fence. Details of the safeguards and security system for the reference plant are provided in Reference G.4.14-1.

Issue No. 3 - Protected Area Perimeter Fence Location:

In the revised plot plan, the protected area perimeter fence encompasses a larger area due primarily to the larger footprint of the reactor building (to accommodate the containment structure within the head access area). The location of the double perimeter fence maintains the prescribed outer and inner clear zones (20 feet) between adjacent structures and the required distance (25 feet) between the two perimeter fences. Vulnerability analysis of the design basis threat has shown that the location of the armed response force provides for a sufficient response time to defeat the threat.

Issue No. 4 - RVACS Intrusion Sensors:

Intrusion detection sensors and alarms are provided for the RVACS ventilation stacks. See Reference G.4.14-1 for additional details of the RVACS security system.

Issue No. 5 - Vital Area Door Alarms:

Appropriate requirements will be established for alarm mechanisms and door hardware which ensure access and exit functions are satisfied for emergency conditions. Specific alarm mechanisms and door hardware will be selected during the detail design phase.

Issue No. 6 - Power Supplies for Security Equipment:

An uninterruptible power supply, protected as vital, provides power for security system electronics and exterior lighting for the NI protected area boundary for a minimum period of eight hours.

Issue No. 7 - On-Site Response Force:

The size of the on-site response force was determined from a vulnerability analysis in response to a design basis threat (see Reference G.4.14-1). Support from local law enforcement authorities was not included in the determination of the ALMR response force size. Of course, local law enforcement authorities would be informed of any challenge to the site, including beyond design basis threats.

Issue No. 8 - Sabotage Threat From Insider Actions:

An assessment of insider actions (see Reference G.4.14-1) concludes that fuel damage or theft, even with insider assistance, is not credible. Insider assistance could facilitate adversarial actions against vital systems, but would not be sufficient to overcome the design features and security provisions to make the threat credible. The design of the plant and control systems prevents individual control over reactor operations. The operator's role is one of providing permission for the automatic controls to take action; the operator cannot dictate the action to be taken to control the plant which could lead to an off normal condition. Plant control logic is provided as firmware or hardware in the plant control system (PCS) and the reactor protection system (RPS) and is inaccessible to an individual.

Issue No. 9 - Control Building Location:

Relocation of the control building within the NI protected area responds to Item 9 of SER Section 13.3.3. The control building is a Seismic Category II, tornado hardened structure as is the underground access to the adjacent remote shutdown facility (RSF). The RSF is a Seismic Category I, tornado hardened structure. The control building contains no nuclear safety related equipment or functions. However, its location within the NI protected area provides operators with protection against terrorist activity and provides improved access to the RSF and reactor facilities. Additional control building details are provided in Section G.4.10.

Issue No. 10 - Protection of SWRPRS:

The sodium water reaction pressure relief subsystem (SWRPRS) consists of two rupture discs connected in series between the shell side of the steam generator and the reaction products separation tank (RPST). These rupture discs, and the line leading from the steam generator to the reaction products separation tank, protect the intermediate heat exchanger (IHX) tubes from potential over pressurization in the IHTS piping which would result from a sodium water reaction due to a steam generator tube leak. Because of their function, the SWRPRS discs are classified as safety related and therefore physical security protection is required to protect them against sabotage. In addition, the RPST vent line, the 30-inch diameter relief line, sodium drain lines leading to the two drain tanks, and the sodium drain tank vent lines are required for the pressure relieving performance of the SWRPRS. Therefore, physical protection of this equipment is also needed to protect against sabotage. Restriction or blockage in any of these lines could defeat the pressure relieving capability of the SWRPRS, although damage affecting only the shape or integrity of the non-safety related components will not affect their function.

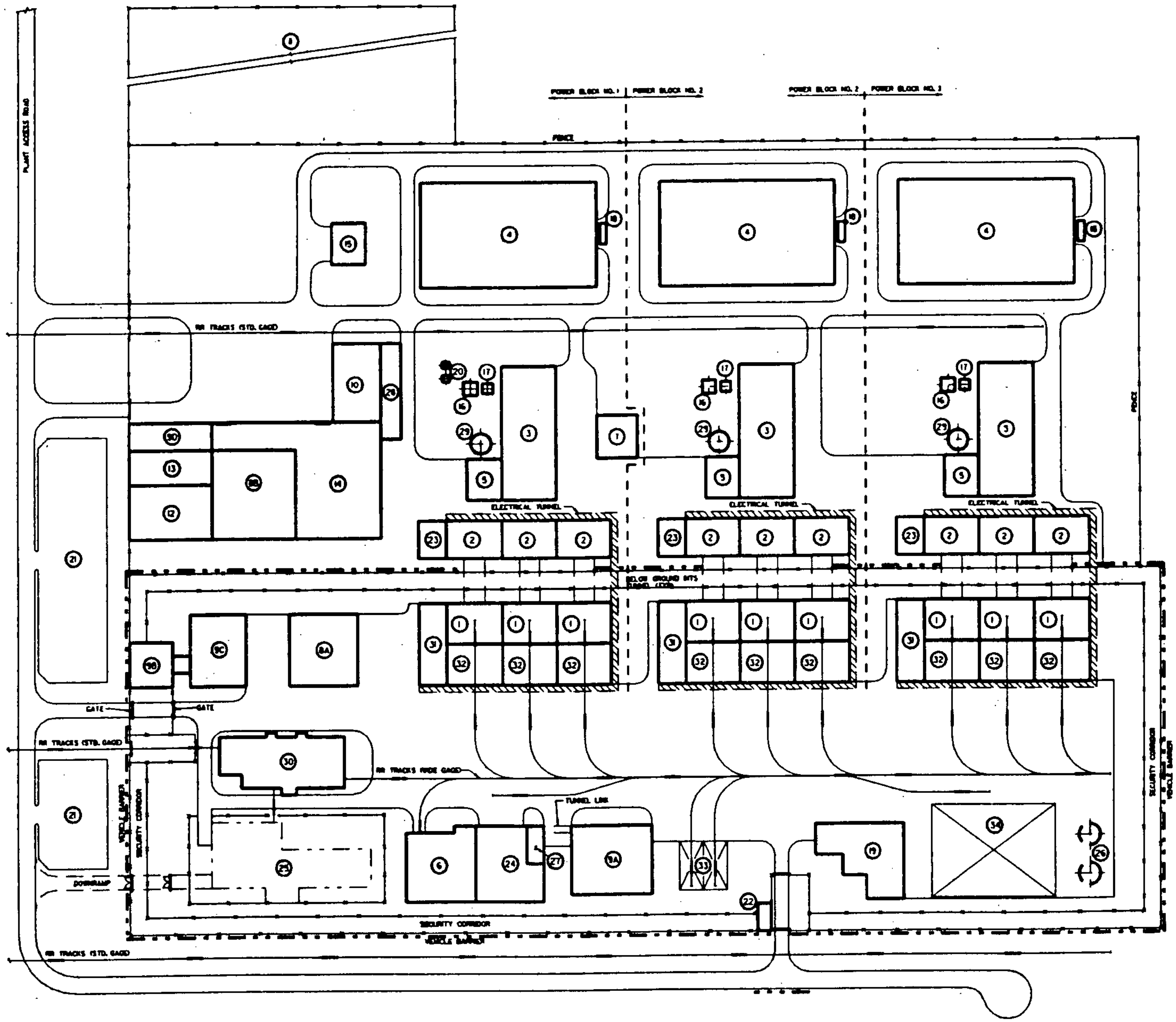
The SWRPRS is located within the steam generator building (SGB), approximately 45 feet below grade. Two rupture discs in series are located in the 30-inch diameter SWRPRS relief line connecting the steam generator to the reaction products separation tank. The RPST vent is a 30-inch diameter line leading to the SGB roof. Two 30-inch diameter lines drain sodium from the RPS tank to the two sodium drain tanks, each also equipped with a 30-inch diameter vent line to the SGB roof.

Although located outside the site high security boundary, a protective area with special features is established within the SGB to protect the SWRPRS against sabotage and terrorist attacks. Limited access to the SGB equipment area is achieved through tamper resistant access control to the building with a second controlled access to the silo where the SWRPRS equipment and piping are located. A buddy system will be adopted which never allows less than two people access at one time. Access to the electrical equipment vaults and control and monitoring room is unrestricted but separate from the access to the SGB equipment area. The limited access will have only minimal impact on maintenance functions, but will have no impact on operation and safety functions. Additional details of the SWRPRS safeguards and security protection features for the SWRPRS are furnished in References G.4.14-1 and G.4.14-2. Modifications to the SGB to protect the SWRPRS will substantially reduce the vulnerability of this system to sabotage.

G.4.14.4 References

G.4.14-1 BNI-8902, ALMR Safeguards and Security Assessment, Bechtel National, Inc., November, 1989

G.4.14-2 BNI-134, SWRPRS Disc Physical Protection, letter from CR Snyder to CE Boardman, February 12, 1990



LEGEND

1. Reactor Facility
2. Steam Generator Facility
3. Turbine Generator Facility
4. Cooling Tower
5. Circulating Water Pump House
6. Reactor Maintenance Facility
7. Gas Turbine Facility

8A Warehouse (NI)
8B. Warehouse (BOP)
9A. Control Building
9B. Nuclear Island Guardhouse
9C. Nuclear Island Health Physics /Personnel Service Building
9D. Balance of Plant Guardhouse /Personnel Service Building

10. Balance of Plant Services Facility
11. Switchyard
12. Administration Building
13. Training Center
14. Maintenance Building
15. Sanitary Waste Treatment
16. Gen. Step Up Transformer
17. Auxiliary Transformer
18. Transformer
19. Assembly & Storage Facility
20. Station Service Transformer
21. Parking Lot
22. Construction & Outage Guardhouse
23. Inert Gas Storage
24. Radwaste Facility
25. Fuel Cycle Facility (Optional)
26. Spent Components Temporary Storage
27. Remote Shutdown Facility
28. Storage Tanks
29. Condensate Storage Tanks
30. Fuel Service Facility
31. Sodium Purification Vault
32. Electrical Equipment Vault
33. Refueling Enclosure & Cask Transporter Parking
34. Transilift Operating Area

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Figure G.4.14–1 ALMR PLANT PLOT PLAN

G.4.15 Prototype Test

G.4.15.1 SER Position on Prototype Test

Sections 3.1.3, 14.3.2 and 14.4.4 of the draft SER (NUREG 1368) address the issue of prototype testing. In Section 3.1.3, the Staff states:

> "PRISM has as its stated objective the development of a standardized plant design that would be submitted to NRC for design certification. It is expected that formal application for a standard plant review of PRISM will be in accord with the rulemaking, as finalized, on standard design certification (10 CFR Part 52). It is the intent of 10 CFR Part 52 to address the standardization criteria associated with advanced designs, including PRISM, by addressing the following standardization issues:
>
> (1) scope and level of detail of design to be standardized
> (2) plant options (number of reactor modules) to be standardized
> (3) prototype testing
>
> These criteria are intended to ensure that before a design certification is granted for the design of any plant that is significantly different from one that has been built and operated, high confidence in the performance of the safety features of that design (must be) demonstrated. The staff considers the approach taken so far with PRISM to be consistent with 10 CFR 52. The scope of the PRISM design to be certified remains an open item (GE proposes to certify only the PRISM nuclear island) and should be resolved consistent with the provisions of 10 CFR 52.47".

In Section 14.3.2, the Staff states:

> "The most important factor in the safety tests is that the reactor module and key supporting systems are prototypical. The current plan calls for a true prototypical unit. However, certain special instrumentation may be required to be installed to obtain data sufficient to validate analytical tools".

In Section 14.4.4, the Staff states:

> "The need to have a prototypic steam generation system on the test unit is dependent upon the scope of the design to be certified. The use of air dump heat exchangers may be an acceptable alternative if the scope of the design to be certified does not include the power conversion system and if the prototype testing confirms that the non-certified portion of the plant cannot significantly affect the safe operation of the plant. Similar considerations also apply to the multi-module control system. If it is to be certified it must be demonstrated. If not, the test program must verify that it cannot significantly affect the safe operation of the plant".

G.4.15.2 Prototype Test Approach for Reference ALMR

G.4.15.2.1 Introduction

One of the major challenges to the deployment of nuclear power has been the time, cost, and effort required to obtain regulatory licensing. The need to improve the situation has been well recognized by both government and industry, and a number of remedies have been proposed. One of the most important remedies is the regulatory certification of standard designs. The NRC has issued both a nuclear power plant standardization policy statement (Reference G.4.15-1) and a rule identified as 10 CFR 52 - "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants" (Reference G.4.15-2) to this end. It is expected that the implementation of this approach will be of major benefit, especially when applied to new designs. Also, regarding new designs, the NRC has issued a policy statement on advanced reactors which sets forth desired attributes of improved safety characteristics for advanced designs, and encourages early interaction of the designer with the NRC to ease the licensing process (Reference G.4.15-3).

An assessment of the overall situation spotlights the central problem in licensing and commercializing a new reactor type such as the ALMR. The NRC, in rule 10 CFR 52, indicates that prototype reactor testing and operation may be required before granting a Standard Design Certification (SDC) for a new reactor type. However, electric utilities indicate that they are unwilling to invest in commercial units of a new reactor type until they are assured that such a reactor will receive a license. A low cost prototype test is one alternative which can resolve the licensing issues before the utilities are required to commit substantial financial resources to commercial plants.

The ALMR, with its modular design and separate nuclear safety related island, permits the option of an affordable prototype test. One reactor module, costing a fraction of a complete plant, can be built and subjected to a series of tests to demonstrate its inherent and passive safety characteristics, in order to resolve the licensing issues. A non-safety related turbine island can then be added, permitting operation as a power producer to demonstrate availability, operating, maintenance, reliability and inspection characteristics, and to recover a majority of the investment. Based on the results of these tests, a SDC can be obtained, assuring licensability of the commercial ALMR prior to utility commitment of resources for the first full size commercial plant.

An ALMR Licensing Plan has been developed to meet the requirements of 10CFR52 and to address the concerns raised by the Staff in the draft SER concerning the prototype test. The Plan identifies the following three objectives:

- Establish the <u>licensability</u> of ALMR nuclear power plants prior to commitments by the utilities to buy.

- Reduce uncertainties in the <u>schedule</u> of licensing ALMR nuclear power plants.

- Reduce the <u>costs</u> of meeting licensing requirements.

These objectives will be achieved by obtaining a standard design certification for the ALMR, based on the testing and operation of a prototype module.

G.4.15.2.2 Approach

In order to meet the requirements of 10 CFR 52, address the Staff's concerns, and achieve the three objectives listed above, a two-pronged approach has been developed. This approach can be summarized as follows:

- o Design a standard ALMR plant whose safety portion is licensable by certification and amenable to an affordable prototype test.

- o Perform safety tests and subsequent power operation on a prototype reactor module in order to establish the basis for standard design certification.

A number of innovative features used in the ALMR design contribute to the viability of this approach. The first two key features are a high degree of inherent core reactivity control and passive core decay heat removal. These two features reduce the challenges to engineered safety systems, permitting the running of safety tests for a number of severe events - such as loss of heat sink without scram, loss of coolant flow without scram, transient overpower without scram - while preserving the reactor for additional tests and power production. These features also reduce the need for active safety systems, and for operator action to shutdown the reactor in the event of off-normal conditions. Thus, from the point of view of reactor safety, the reactor system can be decoupled from the remainder of the plant, provided the non-nuclear safety related portion of the plant is designed so that no failure of it can jeopardize the reactor. Only the reactor module, reactor protection system, and reactor service systems, are required to be nuclear safety related. The operator safety functions following an accident can be limited to securing the plant to cold shutdown status, monitoring post accident conditions, providing mitigating actions, communicating plant conditions to outside personnel, and initiating recovery actions.

The third feature is modularity. A typical ALMR power plant is comprised of three identical power blocks, with each power block comprised of three identical reactor modules providing thermal power for one turbine generator. Thus, a safety test on one reactor module and associated systems can demonstrate the safety characteristics for a complete plant nine times its size and cost.

The fourth feature is factory fabrication. Essentially all of the plant can be fabricated in modules in a factory, shipped to the site by rail or barge, and assembled into exact replicas of the reactor which has been tested and certified. This permits a high degree of reliability, quality control, cost control, and replicability.

Incorporation of the above detailed features, plus use of the design principles of defense-in-depth, redundancy, and diversity, yield a standard plant design which fully meets the NRC Safety Goals (Reference G.4.15-4), and the NRC policy statements on the Regulation of Advanced Nuclear Power Plants (Reference G.4.15-3) and on Severe Reactor Accidents Regarding Future Designs and Existing Plants (G.4.15-5).

The licensing program for the ALMR is comprised of two major elements. The first element is the design, construction, and operation of the standard plant, coupled with NRC review and approval, leading to Final Design Approval (FDA). The second element is the design, construction, safety testing, and operation of a prototype reactor module, leading to Standard Design Certification (SDC) by the NRC.

G.4.15.2.3 Standard Plant Design and Certification

The first element of the overall ALMR licensing program is the standard ALMR power plant design and certification.

<u>Design</u>

The standard plant design will be developed in three phases: the Advanced Conceptual Design phase, the Preliminary Design phase, and the Final Design phase. Supporting this design effort are extensive component and fuel R&D programs (Section G.3.2). A series of documents will be issued by GE summarizing the status for each phase. These documents will describe a complete power plant, even though certification will be requested for the power block and key support systems only.

The primary licensing documents to be issued will be a Preliminary Safety Information Document (PSID) for the Advanced Conceptual Design phase, a Preliminary Safety Analysis Report (PSAR) for the Preliminary Design phase, and a Final Safety Analysis Report (FSAR) for the Final Design phase. Supporting each of these documents will be Probabilistic Risk Assessments (PRAs), and Safeguards and Security Assessments.

In addition to the above documents, Quality Assurance (Q/A) Plans, Construction Plans, and Operation & Maintenance (O&M) Plans will be developed. The ultimate purpose of these plans will be to ensure that the follow-on commercial plants are designed, fabricated, constructed, tested, operated, and maintained in a manner essentially identical to the prototype test module in its final certified configuration.

All of the above documents will be submitted to the NRC Staff. It is expected that the NRC Staff will review and evaluate these reports, and issue a Safety Evaluation Report (SER) on each one. It is also expected that the Advisory Committee on Reactor Safeguards (ACRS) will review these SERs, and issue letters stating their conclusions and recommendations.

The ALMR design team will then use the conclusions and recommendations contained in the SERs and ACRS letters as iterative feedback. This feedback can be expected to guide the development of, and changes in, the design, planning, construction, O&M procedures, and analyses to support the standard design. This feedback will also be used to help select the safety and operational tests to be run on the prototype to demonstrate performance.

Certification

In addition to the design reports discussed above, four additional documents will be issued by GE: a Certification Basis Agreement, a Prototype Safety Test Plan, a Prototype Safety Test Report, and an Application for Standard Plant Final Design Approval and Certification.

The Certification Basis Agreement will be prepared in cooperation with the NRC Staff. It will clarify and summarize the information required by the Staff to support an application for standard design certification. It will define the standards and criteria required for design certification by test, where these standards and criteria are not now addressed in current Standard Review Plans. It will also establish the procedural steps, schedule, and actions required of both GE and the NRC for what will be the first liquid metal cooled reactor proceeding through the certification process by test and rulemaking.

The Prototype Safety Test Plan will be prepared in cooperation with the NRC Staff, and will detail the safety tests to be performed on the prototype test module. The tests will cover events in categories EC-I, EC-II, and EC-III for the initial core. (The Prototype Power Operation Test Plan, to be prepared later, will address additional tests to be performed as the initial core transitions to its equilibrium state). The tests will be selected to demonstrate, in conjunction with accompanying analyses and laboratory tests, that the ALMR standard plant can be operated in a manner that meets all the applicable safety criteria. This Test Plan will be submitted to the NRC Staff for review and an SER.

The next document to be issued will be the Prototype Safety Test Report, which will document results of the safety tests performed on the prototype test module. It will be reviewed and evaluated by the NRC Staff and ACRS, who will be requested to issue their customary SER and letter.

The Application for Standard Plant Final Design Approval and Certification will contain the design, analysis, and test plan information required by 10 CFR Part 20, Part 50 with appendices, Part 52, Part 73, and Part 100. The Prototype Safety Test Report will be included. Design information will also be included for the complete power plant. Site specific information will not be included.

The NRC Staff will be requested to review and evaluate the information contained in the Application for Standard Plant Final Design Approval and Certification. The ACRS is expected to also review and evaluate the

Application for Standard Plant Final Design Approval and Certification, and issue a letter summarizing its conclusions and recommendations.

If all of the above described information is satisfactory, the Staff is expected to issue a Standard Plant FDA, signifying that the Staff finds the design satisfactory for standard plant design certification.

Following completion of these steps, the NRC Commissioners are expected to initiate rulemaking proceedings for Standard Plant Design Certification, according to the guidelines in 10 CFR 52. Rulemaking gives the public a chance to provide input into the design certification process. The intent is to limit public participation in the standard design certification process to this one rulemaking hearing, with any later public participation limited to site specific issues.

The final step will be issuance of a Standard Plant Design Certification by the NRC.

G.4.15.2.4 Prototype Safety Testing and Power Operation

The second element of the overall ALMR licensing program is the Prototype Project made up of safety testing of a prototype reactor module, followed by electrical power generating operation of a one reactor module power plant, or alternatively, a three reactor module power block. As mentioned above, the modular design of the ALMR, coupled with a nuclear safety related island separate from the non-safety related portion of the plant, permit the building and testing of just one reactor module to demonstrate the safety characteristics for a complete plant.

<u>Prototype Safety Testing</u>

Safety testing of the ALMR prototype will be performed on one reactor module and intermediate heat transfer system (IHTS) with all the required safety related protection and auxiliary systems. Only those portions of the balance-of-plant (BOP) will be included which are necessary to perform the safety testing. Other key support systems will be demonstrated separately. The prototype test module will include the reactor module, the IHTS with its steam generator building, but will omit the steam generator and all downstream systems. The secondary sodium from the reactor will dump its heat directly to air by means of a sodium-to-air heat exchanger system. Any other necessary interfaces with the BOP will be simulated.

Alternatives to the above configuration will be considered, including the incorporation of a steam generator and steam-to-air heat exchanger in place of the sodium-to-air heat exchangers.

In order for certification by prototype test to be successful, it must be shown that the reactor module and IHTS of the follow-on standard plants have been designed and constructed in a manner essentially identical to the prototype test module in its final certified configuration, and that the non-nuclear safety related portions fall within the interface simulation limits used in the prototype test. This means that the design and construction of the prototype test module must be well documented. A separate Quality Assurance Plan will therefore be written to specify how the prototype test module design, procurement practices, fabrication procedures, construction techniques, and inspections are to be determined, verified, and documented so that replication can be assured. A separate Construction Plan will also be written to document the factory fabrication, site assembly, and system checkout techniques to be used. The techniques to be used must be understood by the nuclear industry, the regulators, and the utilities in order to gain support for their use.

The detailed design of the reactor module and IHTS for the prototype test will be essentially identical to that of the standard plant. The two designs will therefore be completed concurrently. In suppport of fulfilling applicable requirements for a CP/OL, a FSAR and PRA, documenting design of the prototype reactor module and IHTS, will be prepared and submitted to the NRC Staff for review. Concurrently, the Q/A and Construction Plans will be submitted for review. It is expected that the Staff will issue SERs on these documents, and that the ACRS will review and issue letters summarizing its conclusions and recommendations.

In addition, it is expected NRC inspections, tests, analyses, and documentation will be performed to provide the data base for licensing actions.

Early in the design and construction of the prototype test module will be completion of the safety test planning. Safety test planning will be documented in three levels, a Safety Test Plan (mentioned above), Safety Test Specifications, and Safety Test Procedures. It is currently planned that only the Safety Test Plan will be submitted to the NRC Staff and ACRS for their review, SER and Letter. This Plan will cover start-up tests, pre-operational tests, baseline tests, and the certification safety tests. The safety tests will be selected in cooperation with the NRC Staff to ensure they demonstrate the characteristics required by the Staff for eventual certification of the standard plant design. Events in categories EC-I and EC-II will be included, plus a selection of events in EC-III. Some EC-III events can be included, either at rated or less than rated conditions, because of the benign response of the ALMR to these events, attributable to the inherent and passive safety characteristics built into the design. Table G.4.15-1 summarizes a preliminary list of tests to be considered.

It is recognized that not all EC-II and EC-III events can be fully tested, either because they are too expensive, or because they could damage the prototype test module. These events will either be run under less than rated conditions to validate analytical models, or they will be addressed directly by analysis and laboratory testing, using supportive probabilistic risk assessment (PRA) techniques. Table G.4.15-2 summarizes a preliminary list of such events.

The Safety Test Plan will identify criteria for acceptance of test adequacy, and the alternatives for new or expanded tests if the criteria are not satisfied. Following completion of each phase of testing, a Safety Test Report will be issued. These reports will document the results on the prototype test module. It is expected that the NRC Staff and ACRS will review these reports, and issue SERs and letters summarizing their conclusions and recommendations. If the reviews indicate that additional testing must be performed, or that modifications must be made to the prototype test module, such additional tests and modifications will be addressed.

Table G.4.15-1

PRELIMINARY LIST OF PROTOTYPE SAFETY TESTS

CONVENTIONAL TESTING

- o Pre-operational Testing
- o Baseline In-service Inspections
- o Hot Functional Testing
- o Fuel Loading
- o Start-up Testing
  - Pre-criticality Testing
  - Low Power Ascension Testing
- o Duty Cycle Testing

SAFETY BENCHMARK TESTING

- o Inherent Response Characterization Testing
  - Reactivity Feedbacks
  - Structural Responses
- o Inherent Response Verification Testing
  - RVACS Heat Transfer
  - Seismic Response

SAFETY TESTING

- o EC-I and EC-II Events (with scram)
  - Normal Shutdown with Primary Flow Coastdown
  - Reactivity Addition with Primary Flow Coastdown
  - Loss of IHTS with Full Primary Flow
- o EC-III Events (with delayed scram)
  - Reactivity Addition with Full Primary Flow
  - Reactivity Addition with Primary Flow Coastdown
  - Reactivity Addition with Loss of Power
  - Loss of Primary Flow
  - Loss of IHTS with Full Primary Flow
  - Loss of IHTS with Primary Flow Coastdown
- o EC-III Events (with scram)
  - Partially Blocked RVACS Performance with Loss of IHTS but with Primary Flow Coastdown

SEISMIC TESTING

- o Free Vibration Testing
- o Forced Vibration Testing

SURVEILLANCE ACTIVITIES

- o Operability and Reliability Monitoring
- o On-Line Maintenance Demonstration
- o On-Line In-service Inspection
- o Post Safety Test Inspection (prior to Power Operation Phase)

Table G.4.15-2

EVENTS TO BE EVALUATED BY ANALYSIS AND LABORATORY TESTING

Potential Severe Accident Initiator	Event Description	Variations to Establish Margins
Steam Generator Failure	Design basis leak followed by failure of water/steam dump system	Increase number of tubes failed
Large Sodium Leaks	Sodium pipe leak	Increased leak size up to double ended guillotine break
	Primary piping leak without scram	Additional pipe failures
	Reactor Vessel leak	Vary leak rate and time for worst scenario
External Events	Seismic up to and beyond Safe Shutdown Earthquake	Key component seismic fragilities
Station Blackout	36-hour station blackout without scram	Station blackout without scram for extended times

Prototype Power Operation

Following completion of the safety tests, the reference plan is to add a steam generator and turbine-generator to the prototype test module to convert it into a one reactor module power plant. Since this power plant will have only one reactor, it will not be fully prototypic of the three reactor power block. However, it will permit the accumulation of availability, operating, maintenance, reliability, and inspection experience, it will permit repeating selected safety tests during the transition from the initial core to the equilibrium core to verify the effects of core burnup, and it will permit the sale of electricity to pay back a majority of the investment.

Alternatives to the above configuration will be considered, including the incorporation of two additional reactor modules and a full size turbine-generator, in order to achieve a fully prototypic three reactor module power block.

To accomplish this phase of the test program, the prototype BOP design will have to be completed and documented in a supplement to the prototype FSAR and PRA. The NRC Staff and the ACRS are expected to review these supplements, and issue SERs and letters summarizing their conclusions and recommendations. The Staff is then expected to proceed with licensing activities in order to be able to issue a CP/OL for the one reactor power plant, or the three reactor module power block.

Following issuance of the CP/OL, the BOP portion, (and additional reactor modules if required), will be fabricated, constructed, and checked out. Although the BOP is not nuclear safety related, Q/A and construction plans will be prepared and submitted to the Staff and ACRS for information.

Concurrent with construction of the BOP, a Prototype Power Operation Test Plan will be prepared, along with accompanying Specifications and Procedures. The Power Operation Test Plan will list proposed pre-operational and baseline tests, power operation tests required to evaluate core burnup and transition effects from the initial core to the equilibrium core, and tests to confirm that the integration of the BOP with the reactor module and IHTS falls within the simulation interface envelope used in the safety tests.

A series of Power Operation Test Reports will be issued, documenting trends in plant performance for varying degrees of core burnup out to equilibrium conditions. Since prototype power plant operation will lead any follow-on commercial plant by several years, operating trends will be known and evaluated well in advance of commercial plant needs.

G.4.15.3 Factors Affecting Prototype Test Approach

G.4.15.3.1 Requirements

The NRC has recently issued 10 CFR 52 - "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants" (Reference G.4.15-2). Paragraph 52.47 (a) of this rule reads, in part, as follows:

> (1) An application for design certification must contain:
>
> (i) The technical information which is required of applicants for construction permits and operating licenses ... ;
>
> (ii) Demonstration of compliance with any technically relevant portions of the Three Mile Island requirements ... ;
>
> (iii) The site parameters postulated for the design, and an analysis and evaluation of the design in terms of such parameters;
>
> (iv) Proposed technical resolutions of those Unresolved Safety Issues and medium- and high-priority Generic Safety Issues ... which are technically relevant to the design;
>
> (v) A design-specific probabilistic risk assessment;
>
> (vi) Proposed tests, inspections, analyses, and acceptance criteria ... ;
>
> (vii) The interface requirements to be met by those portions of the plant for which the application does not seek certification ... ;
>
> (viii) Justification that compliance with the interface requirements ... is verifiable through inspection, testing ..., or analysis ... ;

(ix) A representative conceptual design for those portions of the plant for which the application does not seek certification ...

(2) The application must contain a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design ...

Paragraph 52.47 (b) of this rule reads, in part, as follows:

(2)(i) Certification of a standard design which differs significantly from the light water reactor designs ... or utilizes simplified, inherent, passive, or other innovative means to accomplish its safety functions will be granted only if

(A)(1) The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof;

(2) Interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof;

(3) Sufficient data exist on safety features of the design to assess the analytical tools used for safety analyses ... ; and
(4) The scope of the design is complete except for site-specific elements ... ; or

(B) There has been acceptable testing of an appropriately sited, full-size, prototype of the design over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. If the criterion in paragraph (b)(2)(i)(A)(4) of this section is not met, the testing of the prototype must demonstrate that the non-certified portion of the plant cannot significantly affect the safe operation of the plant.

(ii) The application for final design approval of a standard design ... must propose the specific testing necessary to support certification of the design, whether the testing be prototype testing or the testing required in the alternative by paragraph (b)(2)(i)(A) of this section.

G.4.15.3.2 Prototype Configuration

The ALMR Prototype Project addresses the above requirements by a combination of prototype testing, laboratory testing, and analyses. A full-size prototype reactor module will be built to demonstrate the performance of the safety related nuclear island over a wide range of normal and abnormal conditions. Safety analyses and laboratory testing will be used to provide supplementary data for conditions not amenable to prototype testing, and for interaction effects between the nuclear island and those portions of the balance of plant not included in the prototype. All of the equipment will be prototypic with the following three exceptions:

- o A sodium-to-air heat exchanger system replaces the steam generator system
- o The control system will be for one reactor module only
- o Diagnostic instrumentation will be added for the purpose of collecting test data not normally required for power operation

Although the current reference plan is to use a sodium-to-air heat exchanger system in place of a steam generator and steam-to-air heat exchanger system, this choice is open to change if further investigation shows that safety interactions cannot be adequately simulated. In any event, a fully prototypic steam generator system will be added later for the power operation phase of the project.

The use of a single reactor, instead of a prototypical power block (three reactors) control system, poses no safety issue for the safety test itself. However, the issue of control system interaction will arise when certification is requested for a commercial power plant with its control system involving three reactors per power block. To address this issue, an extensive real-time simulation program will be performed to simulate the interaction between the three reactors in a power block and between the three power blocks in a plant. The simulation models will be fine-tuned by using actual data from the prototype test and will include the effects of reactor noise and instrument inaccuracies. A mockup of the operator's console will be used to evaluate operator response to multi-module transients, and to determine the adequacy of the man/machine interface. Confirmation of this multi-module control simulation will be obtained, either in the power operation phase of the Prototype Project if a full power block is added, or with the first commercial plants.

G.4.15.3.3 Test Selection

The choice of tests to be performed will be based on extensive analyses of reactor performance, backed up by an extensive fuel and component test and development program. The tests will be performed in three phases - conventional testing, safety benchmark testing, and safety tests. In addition, surveillance activities will be performed, not only to demonstrate feasibility, but also to verify that the reactor is in a condition to continue with the test program and, ultimately, power operation. The test program will be developed in close cooperation with the NRC Staff, and will be based on startup test programs for commercial power reactors, testing proposed for the Clinch River Breeder Reactor Plant (CRBRP), testing performed at the Fast Flux Test Facility (FFTF) during both its initial core and follow-on safety test program, tests performed at the Experimental Breeder Reactor No. II (EBR-II), and tests performed on other LMRs.

The events of primary interest for defining the safety phase transients will be those postulated sequences which challenge the design, and provide a basis for methods and prediction verification. The test program will be based on the concept of "enveloping" as a means to reduce the number of safety tests. Tests to be included encompass events in categories EC-I, EC-II, and EC-III. Less severe scenarios will be run first before more severe and low probability events are run, in order to minimize risk of damage to the test facility. Extremely unlikely events will be conducted at less than rated conditions to prevent damage. Some events, such as flow blockage, will have to be addressed by analysis and laboratory testing, since they are impractical due to potential for damage to the plant.

G.4.15.3.4 Initial and Equilibrium Cores

The reference plan is to fabricate fuel and blanket assemblies for each initial ALMR core and two reload batches from LWR spent fuel. LWR spent fuel provides both the required source of plutonium, as well as the opportunity to recycle actinides in order to alleviate the long term radiological waste problems associated with LWR fuel. After the second reload batch, each ALMR will begin recycling its own spent and breeder fuel, and will no longer use LWR spent fuel as feedstock.

There are differences in isotopic distribution between ALMR fuel fabricated from LWR spent fuel, and fuel fabricated from ALMR spent and breeder fuel. These differences manifest themselves in different values for:

- o reactivity feedbacks
- o reactivity swing over each operating cycle
- o decay heat
- o power distributions

The initial core of the prototype reactor will be fabricated from LWR spent fuel in order to be prototypic. This poses a problem, however, since most of the safety tests will be performed on this initial core. The results of these tests will be used to certify a standard plant design which encompasses not only the initial core but the transition to an equilibrium core as well as the equilibrium core itself. In order to address this problem, a series of analyses will be performed to predict core and reactor performance from the initial core through the transition to the equilibrium core. Evaluations will be performed to determine how the results of tests on the initial core can be used to predict behavior during the transitional and equilibrium cores. Periodic tests will be performed on in both the safety test and power operation phases to verify that the reactor actually performs as predicted during the transition.

This approach is justified since the prototype reactor will lead any follow-on commercial reactor by several years of operation, giving advance warning of any anomalies.

G.4.15.3.5 Changes to Prototype Design

It is expected that changes will be made to the prototype reactor in response to results from the safety test program. Certification will be requested on the final configuration of the prototype following completion of the safety test program. These changes will be documented so that replication of the final, as-certified design, can be ensured.

G.4.15.3.6 Duration of Power Operation Phase

A key feature of the Prototype Project is the follow-on power operation phase. The purposes of this phase include confirmation of interaction effects for equipment not included in the safety test phase, confirmation of transition effects from the initial core to the equilibrium core, demonstration of availability, operability, maintainability, reliability and inspectability, demonstration of the metal fuel pyro reprocessing cycle, and recovery of the majority of the investment.

The duration of this phase will be based on the expected lifetimes of key components in the prototype reactor. As discussed in Section G.4.3, the design life of the plant is 60 years. However, it is expected that up to half of this design life will be consumed in performance of the safety tests, many of which will be quite severe. Therefore, the reference plan is to operate the prototype for 30 years in the power producing mode.

G.4.15.4 Summary

A Prototype Project will be performed which meets the requirements of 10 CFR 52, and which addresses the concerns raised by the Staff in the draft SER. Close cooperation between GE and the Staff will be maintained during all phases of the planning and execution of the Prototype Project to insure that the information generated will be sufficient to support standard design certification by the NRC.

G.4.15.5 References

G.4.15-1 "Nuclear Power Plant Standardization", Policy Statement, Federal Register, Vol. 52, No. 178, September 15, 1987.

G.4.15-2 "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Reactors", Federal Register, Vol. 54, No. 73, April 18, 1989.

G.4.15-3 "Regulation of Advanced Nuclear Power Plants; Statement of Policy", Federal Register, Vol. 51, No. 130, July 8, 1986.

G.4.15-4 "Safety Goals for the Operation of Nuclear Power Plants; Policy Statement; Correction and Republication", Federal Register, Vol. 51, No. 162, August 21, 1986.

G.4.15-5 "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants", Federal Register, Vol. 50, No. 153, August 8, 1985.

G.4.16 Safety Analysis

G.4.16.1 SER Position on Transient Safety Performance

The draft SER (Section 3.1.2.3) defines the following acceptance criteria: If the ALMR design is to be accepted for NRC certification of a design without a containment building, specific measures must be taken to ensure that no core melt accidents, no accidents with significant positive reactivity feedback, or other accidents with potential for a large radiological release are in the EC-I, EC-II, or EC-III spectrum.

As stated in SER Section 3.1.2.1, "... a key test in accepting the proposed PRISM design is the confidence one can place in the ability of PRISM to prevent accidents which could lead to significant core damage or off-site release of radioactive material... the staff has included in Event Category III a set of bounding events for PRISM whose purpose is to account for uncertainties in design and reliability and acknowledge the difficulty in being able to identify, particularly at this stage of the design, all failure modes of a system or component...Accordingly, the set of bounding events selected for consideration at the conceptual design stage was intended to provide for a sufficient test of the conceptual design such that accurate knowledge of the failure modes and failure probabilities of the safety features of the design would not be critical to assessing or understanding its safety... These bounding events should be reviewed in the future to determine if design changes, additional design detail or R&D program results indicate a change should be made."

Section 15.3.5 of the draft SER states: "The Staff has identified those events which it believes should be considered in the PRISM design, with emphasis at this stage of the review on the bounding events in EC-III..."

Four of these bounding events were judged by the Staff to not meet the EC-III criteria. Based on the transient analyses submitted by GE, Bounding Event Nos. 1B, all-rods UTOP with RVACS cooling only, and 3, complete loss of decay heat removal capability for 36 hrs, have potential for fuel motion and a resulting positive reactivity insertion (SER Section 15.10.5). Bounding Event No. 4, loss of flow without scram, with seizure of one primary pump, was judged to be unacceptable because the analysis done by Brookhaven National Laboratory (SER Section 15.10.5) indicated the event could lead to sodium boiling and possible energetics, although the GE analysis showed large margins to boiling. Event 7, flow blockage of a single fuel assembly, was not addressed by GE in the PSID but was judged by the Staff to have the potential for sodium boiling and possible energetics (SER Section 15.10.5).

The Staff summarizes that until these bounding events are resolved, it "... cannot conclude that PRISM has the potential to achieve a level of safety at least equivalent to current generation LWRs." (SER Section 15.10.6)

G.4.16.2 Summary of Core Passive Safety Performance

A number of core and reactor design changes have been made since the 1986-1987 PRISM design which modify and, in general, improve safety performance during the bounding events; these features are presented in Section G.2.2. Key changes are an increase in core power from 425 MWt to 471 MWt, an increase in core inlet/outlet temperatures from 610/875°F to 640/905°F, an increase in the number of fuel pins in an assembly and corresponding reductions in pin diameter and linear pin power, the addition of the ultimate shutdown system, the addition of control rod withdrawal limiters to limit rod withdrawal during unprotected transient overpower events, and the addition of three gas expansion modules (GEMs) to add negative reactivity upon the loss of flow and thus limit core temperatures during unprotected loss of flow events.

The core-related bounding events in the reference ALMR have been run on the GE ARIES plant system transient code on a nominal basis as appropriate for Category III events. A summary of peak temperatures and margins to limits derived from the EC-III criteria is given in Table G.4.16-1. All bounding events now meet the EC-III criteria, assuming a redefinition of Bounding Event No. 3 to require accommodation of 75% RVACS blockage, or alternatively, 100% blockage for 12 hours followed by 25% unblockage.

Table G.4.16-1

SUMMARY OF PEAK TEMPERATURES REACHED DURING BOUNDING EVENTS

	Peak Clad Temp (°F)	Peak Coolant Temp (°F)	Mixed Mean Core Outlet Temp (°F)	Cladding Loss By Liq. Phase Formation (Mils)	Margin To Na Boiling (°F)
1A All-Rods UTOP	1303	1252	1097	<0.005	708
1B All-Rods UTOP, RVACS Only	1495	1479	1344	0.22	281
2 ULOF/LOHS	1312	1291	1191	<0.001	469
3 Loss Of Decay Heat Removal					
3A 75% RVACS Blockage	1215	1215	1215	NONE	580
3B 100% Blockage, 12 Hrs	1290	1290	1290	NONE	500
4 ULOF/LOHS, One Pump Seized On Coastdown	1355	1335	1193	<0.001	425
5 Rupture Of Steam Generator Tubes With Failure To Isolate Or Dump Water	See Section G.4.8.3				
6 Large Sodium Leaks	See PSID Amendment 11				
7 Assembly Flow Blockage	See Section G.4.6				
8 External Events	Awaiting Definition By NRC Staff				

It is worth noting that, since the current ALMR reactor design provides (1) mitigation of hypothetical core disruptive accidents (HCDA) and core melt events within the primary system boundary (Section G.4.19), and (2) separate low-leakage, pressure retaining containment (Section G.4.1), the draft SER EC-III criteria and bounding events may not now be applicable. Credit for mitigation and containment capability is taken by redefining Bounding Event No. 3 to require only 75% RVACS blockage, or alternatively, 100% blockage for 12 hours followed by 25% unblockage. With this redefinition, all the bounding events can, or will, be shown to meet the acceptance criteria.

A discussion of Bounding Events Nos. 1A, 1B, 2, 3A, 3B, and 4 follows in this section. Additionally, the operation, test experience and associated potential safety issues with the GEMs and control rod stops are discussed in some detail in this section.

Four bounding events are not presented in this section. Bounding Event No. 5, rupture of steam generator tubes with failure to isolate or dump water from the steam generator, is discussed in Section G.4.8.3. The analysis of Bounding Event No. 6, large sodium leak, included in PSID Amendment 11 is still current. Because separate questions were raised in the draft SER on Bounding Event No. 7, flow blockage of a single assembly, the analysis of this event is not presented here but is treated separately in Section G.4.6. Analysis of Bounding Event No. 8, external events consistent with those imposed on LWRs, is waiting Staff definition of the events.

G.4.16.3 Analysis of Core-Related Bounding Events

G.4.16.3.1 Analytical Approach

The core-related NRC bounding events have been analyzed using the GE ARIES plant transient analysis code. In each case, a nominal analysis has been performed. The ARIES code is very similar to, and has been shown to give results in excellent agreement with, the national LMR transient safety analysis code, SASSYS, which has been validated by comparison to EBR-II and FFTF integral test data. ARIES has also been shown to be in excellent agreement with the SSC-PRISM code developed by Brookhaven National Laboratory in support of the Staff (Reference G.4.16-1).

G.4.16.3.2 Damage/Failure Limits

The relevant damage/failure limits to insure that the EC-III criteria are met are the following:

a. Cladding Failure - High temperature cladding creep rupture is the principal fuel pin failure phenomenon during transients. The ferritic alloy HT9 has significant degradation in creep strength at elevated temperatures. A typical end-of-life fuel pin at a 1400°F peak cladding midwall temperature will fail by creep rupture in about 45 minutes, including the effects of cladding internal wastage caused by formation of a low melting temperature alloy of the metal fuel and cladding. Below the alloy melting temperature of 1300°F, the alloy formation is limited to a solid diffusion process, and cladding degradation is extremely slow. Once the alloy has melted, the wastage rate increases rapidly. Figure G.4.16-1 relates the cladding wastage rate, as determined experimentally by Argonne National Laboratory, to the temperature at the fuel-clad interface. As a preliminary design limit, the cladding attack has been limited to less than 10% of the wall thickness, or 2 mils.

b. Local Sodium Boiling - To avoid local sodium boiling within the core, the peak coolant temperature in the core is limited to 1700°F. Conservative saturation temperatures are 1760°F for conditions in the core with the primary pumps not operating, and 1960°F with the primary pumps operating at full flow.

c. Structural Integrity - The reactor vessel, internal structures and reactor components are protected from thermal creep damage by limiting the core average outlet temperature to the following limits to ensure ASME Code Level D time-at-temperature criteria are met:

Time at Temperature	Temperature Limit
< 1 hr	1400°F
> 1 hr	1300°F

d. Fuel Melting - Fuel melting, per se, is not a cause of pin failure. TREAT tests, especially M5 and M6, have demonstrated that extensive fuel melting (exceeding 80% of a given cross-section) does not affect the basic pin failure mechanism (Reference G.4.16-2). Failure by cladding creep rupture, with clad thinning by fuel-clad liquid phase formation, is the appropriate mechanistic cladding breach criterion even for pins with molten fuel in contact with the cladding.

G.4.16.3.3 Analysis of Individual Bounding Events

The analyses of the core-related bounding events are performed with the events being initiated while the reactor is at nominal, full power (100%) conditions, with a core inlet temperature of 640°F and a mixed mean outlet of 905°F. The analyses are performed at beginning of equilibrium cycle conditions, when the power in the driver assemblies is the greatest. The peak assembly is representative of fresh fuel in the reactor; however, for conservatism, the fuel conductivity is based on irradiated fuel since the conductivity of fresh fuel drops rapidly during the first 1.5-2 atom % burnup.

BE-1A: All-Rods Withdrawal Without Scram, With Normal Cooling

This event postulates that a malfunction in the reactivity controller causes the shim motor to continue to withdraw the control rods until the driveline reaches the rod stop, and the RPS function of scramming the reactor is absent. Analysis of the withdrawal accident conservatively assumes a 0.40$ insertion limit. The rod stops are positioned to limit the reactivity insertion to approximately 0.30$, less than the 0.40$ limit even with appropriate margin. The reactivity insertion rate is 0.02$ per second, which corresponds to the maximum speed of the shim motor as it sequentially withdraws one rod at a time. All six absorber bundles are assumed to fail to unlatch, or alternately to fail to be driven in, during this event. The heat removal systems continue to function at full capacity.

In this event, the rods are fully withdrawn to the rod stops in 20 seconds. As shown in Figure G.4.16-2, the power rises rapidly as the rods are withdrawn, and it reaches a maximum of 172% of full power in 30 seconds. At this time, the negative reactivity feedback (shown in Figure G.4.16-3), mostly Doppler and thermal expansion, has turned the power rise around. The power then drops over the next 100 seconds, and stabilizes at about 120% of full power. The fuel pin temperatures follow the power changes, and the peak fuel, cladding, and bulk coolant temperatures reach maximums of 1865°F, 1303°F, and 1252°F, respectively, at 31 seconds, all below the established limits. The cladding attack due to eutectic formation during this event is less than 0.1 mils, well below the 2 mil limit. The reactor mixed mean outlet temperature also peaks at 1097°F in 31 seconds, and then levels off at about 1000°F. The temperatures during this event are shown in Figure G.4.16-4. For conservatism during this event, it is assumed that the axial expansion of the fuel is based on the cladding temperature rather than the fuel temperature.

It has been assumed that the power block behaves normally during this event. Since a module undergoing an unprotected transient overpower (UTOP) event reaches an equilibrium power of 120%, the power block would also see a power increase. The turbine-leading supervisory control would try to maintain the power block at 100% power. Since the module undergoing a UTOP is not responding to control signals, the other modules in the power block would reduce power correspondingly to keep the power block at 100%. If the power block remains at 100%, the module undergoing the UTOP receives adequate feedwater flow to continue steady operation at 120% power. If the supervisory control fails to reduce power to the other modules in the power block or if the other modules are also undergoing a UTOP event, there would be flow starvation in the steam generator and an eventual loss of heat sink. If there are UTOPs simultaneously in all of the modules, each steam generator would lose feedwater after about 600 seconds. The peak pin temperatures would remain the same, but the mixed mean coolant temperature would be higher due to the loss of the heat sink, but still less than the 1300°F limit.

BE-1B: All-Rods Withdrawal Without Scram, With RVACS Cooling Only

As noted in Amendment 11, this event is of such low probability that it belongs in EC-IV. However, it is presented here against EC-III acceptance criteria as requested by the Staff.

This event is initiated in the same manner as the UTOP event discussed above, where there is a 0.40$ reactivity insertion at 0.02$ per second.

However, in this event, cooling by the intermediate heat transport system is lost so that heat removal is only from the reactor vessel through the RVACS. It is assumed that the intermediate pump is seized at the start of the transient and that there is no heat removal through the intermediate heat exchanger and balance of plant. Since the GEMs would rapidly provide a large negative reactivity feedback if the primary pumps were stopped, it is conservatively assumed that the primary pumps continue to operate during this event until the high pump inlet temperature (1000°F) trips the pumps.

The reactor power, shown in Figure G.4.16-5, rises rapidly as in the normal UTOP (BE-1A), but peaks at 172% a little sooner (at 21 seconds) because of the additional feedback associated with the loss of the heat sink. The peak fuel temperature, shown in Figure G.4.16-6, is slightly higher at 1885°F, and the cladding and coolant temperatures are significantly higher. The cladding, coolant, and mixed mean outlet temperatures peak at 80 seconds at 1495°F, 1479°F, and 1334°F, respectively, as the primary pumps are tripped. However, this peaking is short since the GEMs provide additional negative feedback (as shown in Figure G.4.16-7). The cladding attack during this event is about 0.2 mils. The mixed mean outlet temperature starts to increase again at about 1400 seconds as the vessel heats up and moves the core away from the control rods. The mixed mean outlet temperature continues to increase until it reaches about 1280°F at 9000 seconds, as can be seen in Figure G.4.16-8.

If the primary pumps are tripped at the start of the event, the flow loss quickly activates the GEMs and the large negative feedback of the GEMs limit the power rise to 103%. This fast shutdown of the power rise results in much lower peak temperatures. The peak fuel temperature is 1586°F, and the peak cladding temperature is 1351°F. These peaks occur within the first four seconds.

BE-2: Unprotected Loss of Flow, Loss of Heat Sink, for 36 Hours

Bounding Event No. 2 is defined by the Staff as a station blackout for 36 hours. As stated in the draft SER, Section 15.10.1, "Assume a station blackout event, which leaves the PRISM module without power for 36 hours.

Assume scram occurs and that natural circulation cooling is the only mode of cooling available." With scram this event is totally benign.

In order to assess the inherent safety capabilities of the ALMR, the event has been conservatively analyzed without scram. In addition, it has been conservatively assumed that no heat removal occurs through the IHX and BOP. The transient is thus analyzed as an unprotected loss of flow and heat sink (ULOF/LOHS) event. Adding further conservatism, the axial fuel expansion is based on fuel temperature rather than on cladding temperature.

In this event, the power and flow drop rapidly at the start of the transient, as shown in Figure G.4.16-9, since the loss of flow activates the GEMs. As shown in Figure G.4.16-10, there is some initial undercooling of the fuel pins before the negative reactivity of the GEMs takes effect. The fuel, cladding and coolant temperatures peak at 1547°F, 1312°F, and 1291°F, respectively, at three seconds into the transient, and then the core starts to cool. As seen in Figure G.4.16-11, there is little negative feedback other than GEMs during the early part of this transient because the GEMs rapidly reduce the power. However, as the primary pump coastdown ends, the coolant starts to heat up again. Since there is no heat sink other than RVACS, the vessel continues to heat up for a while. The heatup of the vessel causes the core to move away from the control rods, and the net effect is positive reactivity feedback due to thermal expansion. At about 1400 seconds into the transient, the effects of the control rod expansion along with the other positive feedback effects (shown in Figure G.4.16-12) overcome the negative feedback of the GEMs and the power starts to rise. The core heats up during this slow power rise, and other feedback mechanisms (Doppler and core radial expansions) become negative and turn the power excursion around. The mixed mean outlet temperature slowly increases as RVACS heat removal comes in balance with the decay power, and it peaks at about 1191°F at 41000 seconds, as seen in Figure G.4.16-13. The cladding attack during this event is less than 0.1 mils.

BE-3: Loss of Decay Heat Removal Capability

The acceptance criteria of SER Section 3.1.2.3, and the bounding events, were imposed by the Staff largely because the original PRISM design emphasized accident prevention over mitigation, and did not have a containment building. Now that the reference ALMR design emphasizes mitigation as well as prevention, and also has a containment building, relaxation of the acceptance criteria and bounding events is warranted. Based on this rationale, a redefinition of Bounding Event No. 3 is considered warranted.

This event was originally defined by the Staff as "Loss of forced cooling plus loss of ACS/RVACS with 25% unblocked RVACS after 36 hours". Scram is assumed to occur. Due to the extremely low probability of total blockage of the RVACS, which requires complete blockage of all four RVACS inlets and all four outlets, and the additional low probability of not being able to unblock at least one of the inlet/outlet stacks within 8-12 hours, the following redefinition of this event was proposed to the Staff at a GE/NRC meeting on February 27, 1990:

- Loss of forced cooling plus 75% RVACS blockage for 36 hrs.

An alternative redefinition was proposed for capability assessment in recovery from 100% blockage:

- Loss of forced cooling plus 100% RVACS blockage for 12 hours, with 25% unblockage after 12 hrs.

These two transients have been analyzed by means of a thermal nodal network model which accounts for:

- Radiation from the reactor vessel to the containment vessel
- Radiation from the containment vessel to the collector cylinder
- Radiation from the collector cylinder to the silo wall
- Natural circulation of air through the RVACS air passages, assuming appropriate amounts of blockage
- Conduction outward through the silo wall and surrounding earth

Although of minor importance, the heat rejection from the bottom of the reactor vessel has also been included. Heat losses through the top closure and from the IHTS piping are neglected.

BE-3A: 75% Blockage of Decay Heat Removal Capability for 36 Hours

Results for the more severe case of extended 75% blockage (i.e., without unblockage at any time) are shown in Figure G.4.16-14. The maximum mixed mean core outlet temperature of 1215°F is reached at about 40 hours. With the decay heat reduced to 0.5% of full power and with a correspondingly low natural circulation flow through the core, the radial temperature peaking is minimal. The peak local sodium temperature is about 540°F below boiling. No cladding failures are predicted, and the EC-III criteria are satisfied with margin.

BE-3B: Complete Loss of Decay Heat Removal Capability for 12 Hours, Followed by 25% Unblockage of RVACS

This transient is analyzed by the same means as for Event 3A. The key temperatures are summarized in Figure G.4.16-15. The effect of the 25% RVACS unblockage at 12 hours is clearly seen in the response of the containment vessel (and other surfaces in contact with the RVACS air flow). The core and reactor vessel temperatures continue to increase for a short time after the partial unblockage, reaching a peak mixed mean core outlet temperature of 1290°F at about 25 hours. No cladding failures are predicted, and the EC-III criteria are satisfied with margin.

BE-4: Unprotected Loss of Flow, Loss of Heat Sink, With Seizure of One Primary Pump

Bounding Event No. 4 is defined in the draft SER, Section 15.10.1, as follows: "Instantaneous loss of flow from one pump (e.g., power is cut to an EM pump with no flow coastdown) and the other three pumps trip and coast down. Consider event without scram." If loss of heat sink is added to this scenario, it is similar to Bounding Event No. 2 as analyzed and discussed above, but with one primary pump failing to coast down. Bounding

Event No. 4 is therefore conservatively analyzed as an unprotected loss of flow and heat sink (ULOF/LOHS) event, with one primary pump failing to coast down. Axial fuel expansion is conservatively based on fuel temperature rather than cladding temperature.

Since the other three pumps continue to coast down normally, there is a coastdown of flow through the core but at a reduced rate. Some of the coolant in the core inlet plenum flows back to the cold pool through the failed pump rather than through the core. Although there are fewer pumps coasting down and the bypass through the failed pump reduces the core flow somewhat, the flow coastdown is not reduced appreciably, as can be seen by comparing Figure G.4.16-16 with Figure G.4.16-9. In fact, the lower core flow early in the transient reduces the pressure drop around the primary circuit such that less kinetic energy in the synchronous machines is expended. This extends the coastdown of the unfailed pumps.

Figure G.4.16-17 shows the temperatures in the early part of the transient. The peak temperature in the fuel is 1562°F, the peak cladding temperature is 1355°F, and the peak coolant temperature is 1335°F. These peaks are all reached in three seconds. The longer term behavior, shown in Figure G.4.16-18, is similar to that of Bounding Event No. 2 in that the expanding vessel pulls the core away from the control rods and, at about 1700 seconds, the reactor undergoes a small power increase, raising core temperatures. As seen in Figure G.4.16-19, the GEMs provide most of the negative feedback early in the event, and the combined driveline-vessel expansion provides an increasing amount of positive feedback. As the core heats up again due to the power increase, the Doppler and core thermal expansion provide additional negative feedback to turn the power increase around, as shown in Figure G.4.16-19. The mixed mean outlet then increases gradually until it peaks at 1193°F at about 41000 seconds as the decay heat drops to the RVACS heat removal capability. The cladding attack during this event is less than 0.1 mils.

BE-5: Rupture of Steam Generator Tubes with Failure to Isolate or Dump Water from Steam Generator

This bounding event is treated in Section G.4.8.3.

BE-6: Large Sodium Leaks (Single Module)

The analysis presented in PSID Amendment 11 is still valid.

BE-7: Flow Blockage of a Single Fuel Assembly

This bounding event is treated separately in Section G.4.6.

G.4.16.4 Gas Expansion Modules

Gas expansion modules (GEMs) are devices designed to passively provide negative reactivity feedback during loss of primary flow (LOF) events. Their principle of operation is to control neutron radial leakage from the core with a gas and sodium filled cavity at the driver core perimeter that is connected hydraulically to the high pressure plenum. When pumps are at full flow, the plenum pressure compresses the gas in the GEM cavity to a level above the core, producing neutron back scattering into the core by the sodium in the cavity. When the flow decreases, the trapped gas expands, displacing the sodium in the core elevation of the cavity. The gas scatters fewer neutrons back into the core and thus produces a negative reactivity feedback. The reactivity worth of GEMs in small diameter cores, such as the ALMR, is sufficient to provide a shutdown system with only a few devices.

G.4.16.4.1 Design and Operation

The GEM is essentially an empty assembly duct, sealed at the top, open at the bottom and connected to the core high pressure inlet coolant plenum. Figure G.4.16-20 shows the ALMR GEM design. The upper section consists of a handling socket and a sealing plug at the top of a duct. A hexagonal cross section duct, with a wall thickness slightly greater than the standard fuel and blanket duct, forms the body. A lower shield block and a nosepiece complete the bottom end.

An Inconel lower shield block maximizes the gas volume available below the core. Maximum gas volume is essential to maximize the "stroke" of the GEM sodium level. Holes occupying about 40% of the cross-sectional area of the lower shield have been provided to permit the GEM gas cavity to communicate freely with the high pressure inlet plenum. The upper shielding is HT9 to reduce cost. The length of the assembly above the sealing plug is ample for shielding with the less efficient HT9 neutron reflection. The adequacy of the lower shielding needs verification in additional core shielding studies.

At completion of GEM insertion into the core by the in-vessel transfer machine, the trapped helium cover gas bubble is compressed into the cavity by the static head of the sodium and by ~5% primary flow pump head, such that the sodium level rises through the nosepiece to a level near the bottom of the active core. When the pump pressure increases to full flow, the sodium level in the cavity rises until the gas pressure balances the coolant plenum pressure. The elevation of the top of the gas cavity is set by the upper shield plug such that the sodium-gas interface is then above the active core.

The GEMs provide negative reactivity feedback upon loss of primary flow and pressure. The loss of pressure causes the gas bubble to expand, driving sodium from the assembly and restoring gas in the active core region. This displacement of sodium from the core perimeter increases radial, and some axial, neutron leakage, producing a negative reactivity feedback.

Figure G.4.16-21 illustrates the elevation of the sodium-gas interface for various reactor states. Initial rapid depressurization of a GEM results in almost adiabatic expansion of the gas. Thus the initial gas expansion is much smaller than the final expansion state. The figure assumes steady state conditions have been established for the final states. Transient studies indicate that the GEM gas volume has little heat capacity and that heat transfer from the duct walls to the gas occurs very rapidly. As a result, the steady state sodium elevations indicated in the figure are rapidly established.

This heatup phenomenon is illustrated in plots of GEM performance predicted by transient models in the ARIES code during a flow coastdown event. Figure G.4.16-22 plots sodium elevation in an ARIES-modeled GEM as a function of time in the event. Core average flow rate is also plotted in the figure. Initially in a loss of flow event, primary flow drops to 60% of full flow in a few seconds. Then the flow coastdown power controller decreases flow slowly to natural circulation conditions over about 200 seconds, due to coastdown of the synchronous machines. Over the 500 seconds of transient plotted, temperatures in the reactor are almost constant, so the gas expansion will exhibit characteristics of initial adiabatic expansion followed by heatup to the original temperature. As shown, the GEM gas expansion is largely complete by 100 seconds, while the flow is still decreasing slightly. Coolant pressure changes are largely complete by 100 seconds, thus little further expansion is to be expected if gas reheat is rapid compared to the coastdown rate.

The initial adiabatic expansion and rapid follow-on heatup are best shown by cross-plotting the sodium elevation as a function of fractional core flow rate during the event, as in Figure G.4.16-23. As shown in the figure, the initial pressure reduction corresponding to 60% flow causes an almost adiabatic gas expansion with little reduction in the sodium elevation within the GEM. The gas expansion during the remaining flow coastdown evidences a behavior that indicates gas heatup toward the original temperature that is faster than the flow coastdown pressure drop. By the time the flow coastdown is completed, the gas has reheated to the steady state temperature of the coolant around the GEM and the expansion has been completed.

G.4.16.4.2 Test Experience

Nine GEMs were loaded into the FFTF core in July, 1986. Tests were performed with the reactor near critical to determine their feedback worth, sensitivity to system temperature and worth versus time during a flow coastdown. In July, 1986, the nine GEMs were reloaded into the core and a series of ULOF tests were performed. The test program was a bootstrap series of increasing severity loss of flow transients, culminating in a loss of flow from 100% flow and 50% power with the normal scram trips replaced by special trips based on in-core temperature sensors (References G.4.16-3 and 4). The reactivity worth of the GEMs matched analytic predictions closely and the transient performance of the GEMs was as predicted, providing confidence in their use to terminate loss-of-flow events.

The analysis predicting FFTF transient response with GEMs was done with the national standard LMR safety code, SASSYS, and was validated against the reactor integral tests (Reference G.4.16-5). As stated in Section G.4.16.3.1, ARIES has been shown to give results in excellent agreement with SASSYS.

G.4.16.4.3 Potential Safety Issues

The potential safety issues associated with GEMs that have been identified to date are:

a. Accuracy or confidence concerning in situ GEM performance verification tests.

b. Ensuring that the GEMs still contain gas to operate when called upon to do so. GEM reliability, detection of leakage, and worth degradation from leakage are key issues.

c. Prevention or accommodation of an inadvertent reactivity insertion caused by restarting pumps with the rods partially out.

Life and reliability testing will be necessary to qualify the GEMs for long life reactor service.

<u>In Situ Testability</u>

The tests used to determine FFTF GEM worth demonstrated that the reactivity feedback from GEMs could be measured reliably with the reactor subcritical using pump speed changes, coolant temperature changes and the neutron monitoring system. In situ testing may thus be assumed to be available for the ALMR GEMs.

The analyses performed in support of the FFTF LOF test program demonstrated that the recently developed 3-D neutronics analysis methods are able to predict GEM feedback worth with sufficient accuracy for core design. It is thus expected that module prototype tests will confirm the predicted worth and feedback behavior.

GEM Functional Reliability

Methods to monitor the leak tightness of a GEM are being evaluated. At this stage of design, it is assumed that the GEMs will be replaced after five cycles, a lifetime comparable to that of a blanket assembly. It is also assumed that control system reactivity tests at the start of each refueling outage will be used to identify any GEM that has failed and leaked sufficient gas to measurably affect its worth. Similar tests after refueling will be part of the startup process. These tests will verify that the core is starting the cycle with GEMs satisfying worth requirements. A GEM development program will be needed to generate the quantitative hardware reliability data necessary to predict the GEM system reliability throughout the cycle. Alternatively, a tag gas in the GEMs with the capability to trigger one of the cover gas or sodium monitoring devices could be employed.

Future design trade studies must address functional reliability. However, adequate reliability should be readily achievable.

Inadvertent GEM Reactivity Insertion

The ALMR reference core design addresses GEM reactivity insertion principally by accommodation. Since the core needs only a small feedback effect from the GEMs to compensate for control rod withdrawal by vessel thermal expansion, only three GEMs are used. The expected total negative reactivity insertion from the three is about 0.70$. A rapid insertion of this magnitude is not sufficient to cause a prompt critical event. However, in the limit, the reactor and containment are being designed to accommodate a core melt event and commensurate energetics.

A potential scenario leading to such a hypothetical event would be a cold critical test with pumps off. If the pumps were then inadvertently started, the GEM reactivity insertion of about 0.70$ would occur. This insertion is comparable to the cold critical-to-full power temperature defect. Thus, the expected core response would be a rapid power rise overshooting full power as limited by Doppler and fuel expansion feedbacks. The core power would then establish an equilibrium near full power as the slower thermal feedbacks become established.

As currently designed, the RPS prevents an event of this type. The control rods cannot be pulled more than one at a time, for rod drop testing, with pumps not at full flow. For more than one control rod to be lifted, primary pumps must be fully on.

Accident analyses of GEM failures have not yet been performed. Additional event hazards associated with GEMs have not yet been identified. It is believed that overall public risk is improved by the addition of GEMs. The presence of GEMS may increase the probability of a containment challenge as a result of an event assuming unusual conditions (rods critically banked and no flow, then inadvertent pump start and no scram). On the other hand, the GEMs reduce the challenge from ULOF events, which are of higher probability since they involve fewer sequential failures or errors.

G.4.16.5 Control Rod Stops

Electronically positioned mechanical rod stops provide an upper bound to the amount of reactivity that can be added to a core as a result of an uncontrolled rod withdrawal event. The rod stop system (RSS) selected for the ALMR is characterized by: (1) a redundant electronic controller for rod stop position adjustment, and (2) mechanical out-motion blocks in each control rod drive mechanism. Functional requirements and a design description are provided in Section G.4.2, Shutdown Systems.

G.4.16.5.1 Operation of Rod Stop System

In general, the operation of the RSS provides a passive out-motion blocking function and an active stop position adjustment function. During most of the core operating cycle the RSS controller is powered and performs a monitoring function, while the stop adjustment motor power supply, stepper controller, and drive selector are unpowered. During this time, the rod stops are in fixed positions and passively limit out-motion by physical interference with CRDM carriage motion.

During the active stop adjustment process, the controller determines the appropriate stop settings. Operator permission is required to proceed with stop movements. The operator is unable to alter the controller-predicted settings, but can refuse movement permission (e.g., if there is a safety question or an administrative limit).

The PCS monitors the rod and stop positions reported by the RPS and RSS. Logic is included in the PCS to terminate rod out-motion prior to impact with the stop. This feature reduces wear on the drive and stop mechanisms if the operator or automatic control logic of the PCS overlook the rod stop reset request or the stop positions.

G.4.16.5.2 Reactor Usage Precedents

Existing Boiling Water Reactors (BWRs) employ rod blocks to limit the consequences of a rod withdrawal error during normal plant operation. An abnormal operation that might result in local fuel damage is prevented by the rod block enforcement functions of the rod control and information system (RC&IS). The RC&IS is the BWR analog of the subsystem of the ALMR plant control system (PCS) that controls rod motion.

Rod block signals can be generated by the RC&IS and by several related monitoring subsystems. The rod block function is logic based, not physical. A rod motion inhibiting signal is generated and transmitted to the rod server modules of the RC&IS. The signal prevents the servers from actuating the control rod drives. The RC&IS and the rod blocks have no effect on the scram function of the reactor protection system. Only plant control system operations are affected.

The expected frequency of an inadvertent movement of more than one rod, due to failure, is less than or equal to once in 100 reactor operating years. The RC&IS design assures that no credible single failure or single operator error can cause or require a scram or require a plant shutdown.

G.4.16.5.3 Potential Safety Issues

Potential safety issues associated with rod stops that have been identified are: (1) the possibility of misadjustment, and (2) the effects of uncertainties. The rod stop system design includes features to reduce the chances of stop misadjustment. Uncertainties are accommodated by design margins.

Rod Stop Misadjustment

The redundancy of the rod stop controller will make misadjustment by controller malfunction a low probability event. In addition, only RPS verified data are used to predict a new stop setting, rendering incorrect data a low probability cause of misadjustment.

To ensure the accuracy of the banked full power position prediction, the prediction is only made when core power is at or near full power. This limits the uncertainty in the extrapolation the controller must do in estimating the current full power rod bank position and reduces the probability of a setting error.

An administrative control is also employed to reduce the probability of an error in rod stop setting. The controller provides the predicted rod stop position to the operator and requests permission to reset the stops to that position. The controller repositions the rod stops only with operator permission. If the operator disagrees with the setting prediction of the controller, then no permissive is given and no readjustment occurs.

The operator cannot instruct the controller to use a setting different from the controller-predicted position, thus preventing operator errors or sabotage from incapacitating the passive blocking function.

Since the ALMR cores have a reactivity loss during the cycle, the failure to reset a rod stop has a conservative consequence. The potential rod runout reactivity insertion decreases with time from the last reset until the stops interfere with the normal banked rod positions. Then core power decreases as a result of the inability to withdraw the rods to continue full power operation.

Stop Misadjustment During Operation

The potential, during rod stop adjustment, for a UTOP event more severe than the limiting 0.40$ insertion is very low. The features that contribute to risk reduction from rod stop misadjustment apply equally whether the core is in operation or shutdown. The adjustment process does not interfere with normal operation, nor does operation interfere with the stop adjustment process. Normal control rod movements are curtailed during rod stop adjustments, but rod fast runback and scram functions are unaffected during stop adjustment. No increase in the probability of an excessive magnitude UTOP is expected as a result of adjusting the stops during operation. The same probability would be expected if the stops were only moved during a core shutdown.

Design Margin And Uncertainties

Core performance analyses indicate the ALMR core can accommodate up to 0.40$ of reactivity insertion within the EC-III limits. Based on engineering judgment, the nominal rod stop setting limit has been set to 0.30$ to maintain a margin for rod worth and position uncertainties, rod stop setting uncertainties, and core performance uncertainties. During detailed design, the appropriate margin between the core capability and the stop setting for an acceptable probability of not exceeding the EC-III limits will be determined by a combination of deterministic and probabilistic analysis. The margins desired may also be varied based on actual reactor power and time into the cycle to permit normal power maneuvers to be safely performed.

G.4.16.6 References

G.4.16-1 Slovik, G. C., et al., "Evaluating Advanced LMR Reactivity Feedbacks Using SSC," ANS International Topical Meeting on Safety of Next Generation Power Reactors, Seattle, WA, May 1-5, 1988

G.4.16-2 Bauer, T. H., et al., "Behavior of Metallic Fuel in TREAT Transient Overpower Tests," ANS International Topical Meeting on Safety of Next Generation Power Reactors, Seattle, WA, May 1-5, 1988

G.4.16-3 Waldo, J. B., et al., "Application of the GEM Shutdown Device to the FFTF Reactor," ANS Transactions, 53:312 (1986)

G.4.16-4 Campbell, L. R., et al., "Reactivity Worth of Gas Expansion Modules (GEMs) in the Fast Flux Test Facility," ANS Transactions, 53:457 (1986)

G.4.16-5 Padilla, A., and S. W. Claybrook, "Posttest Analysis of the FFTF Inherent Safety Tests," ANS Transactions, 54:250 (1987)

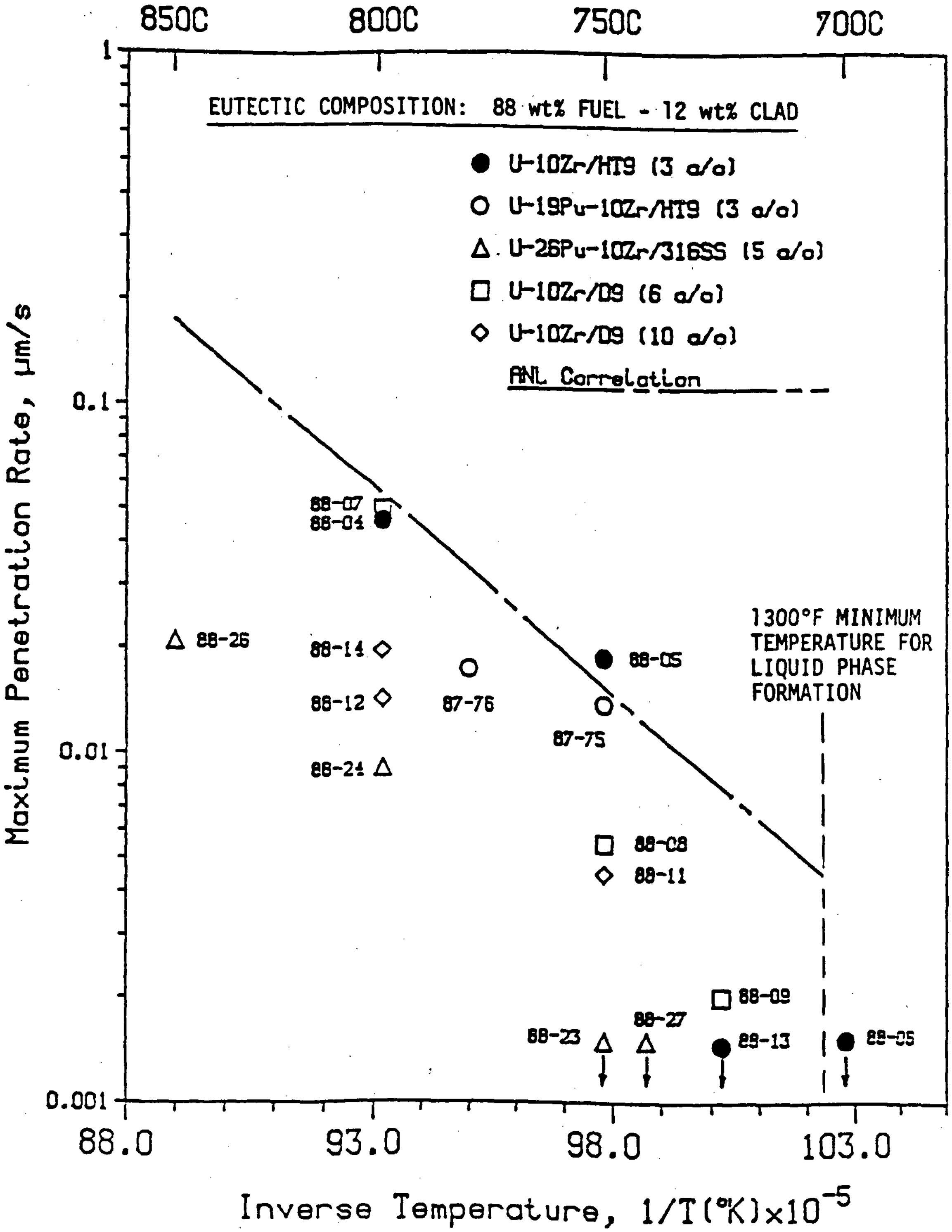


Fig. G.4.16-1 RATE OF CLADDING ATTACK BY FUEL-CLAD LIQUID PHASE

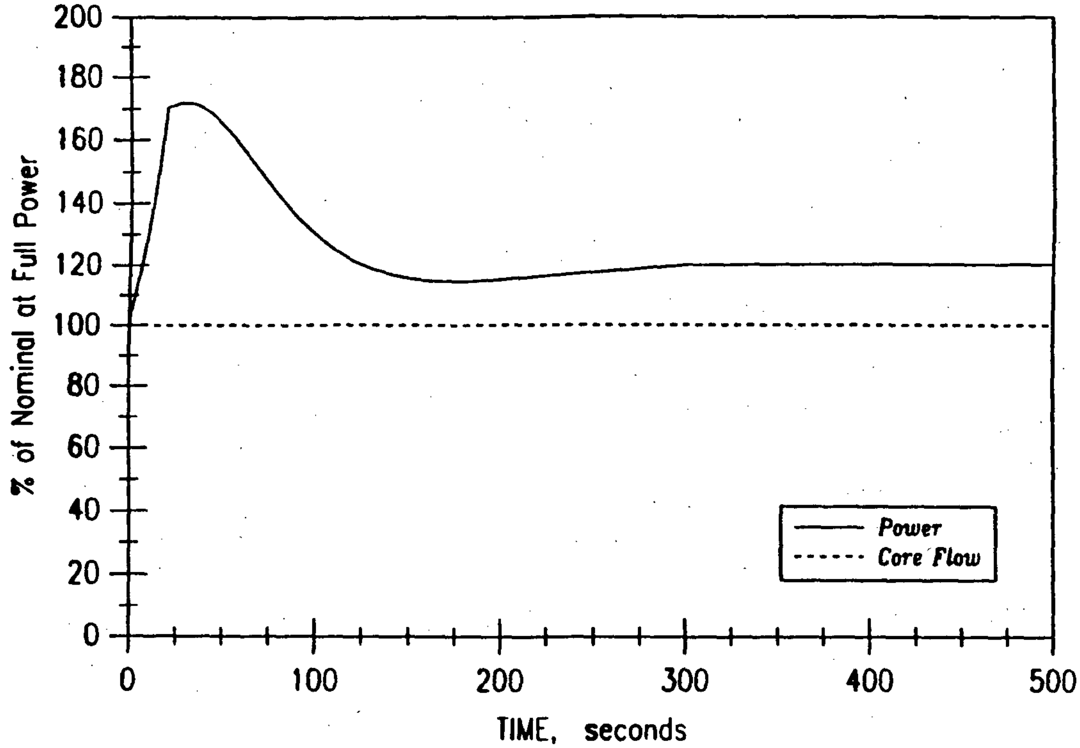


Fig. G.4.16-2 EVENT 1A, ALL-RODS WITHDRAWAL WITHOUT SCRAM, WITH FORCED COOLING: POWER AND FLOW

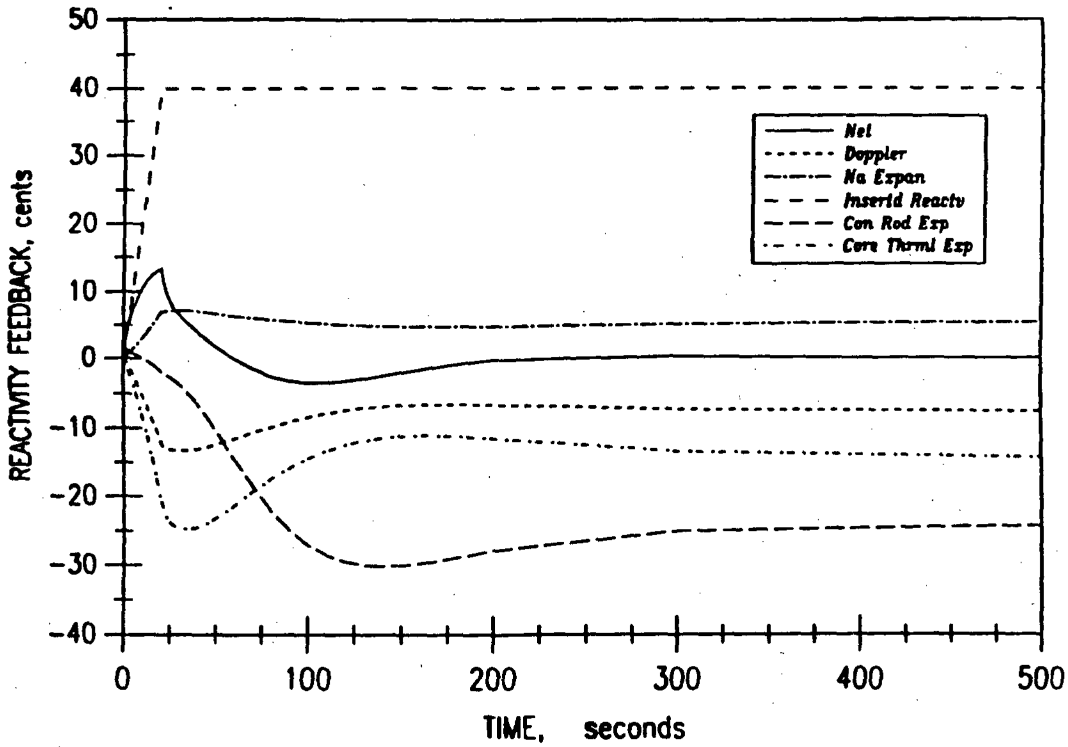


Fig. G.4.16-3 EVENT 1A, ALL-RODS WITHDRAWAL WITHOUT SCRAM, WITH FORCED COOLING: REACTIVITY FEEDBACKS

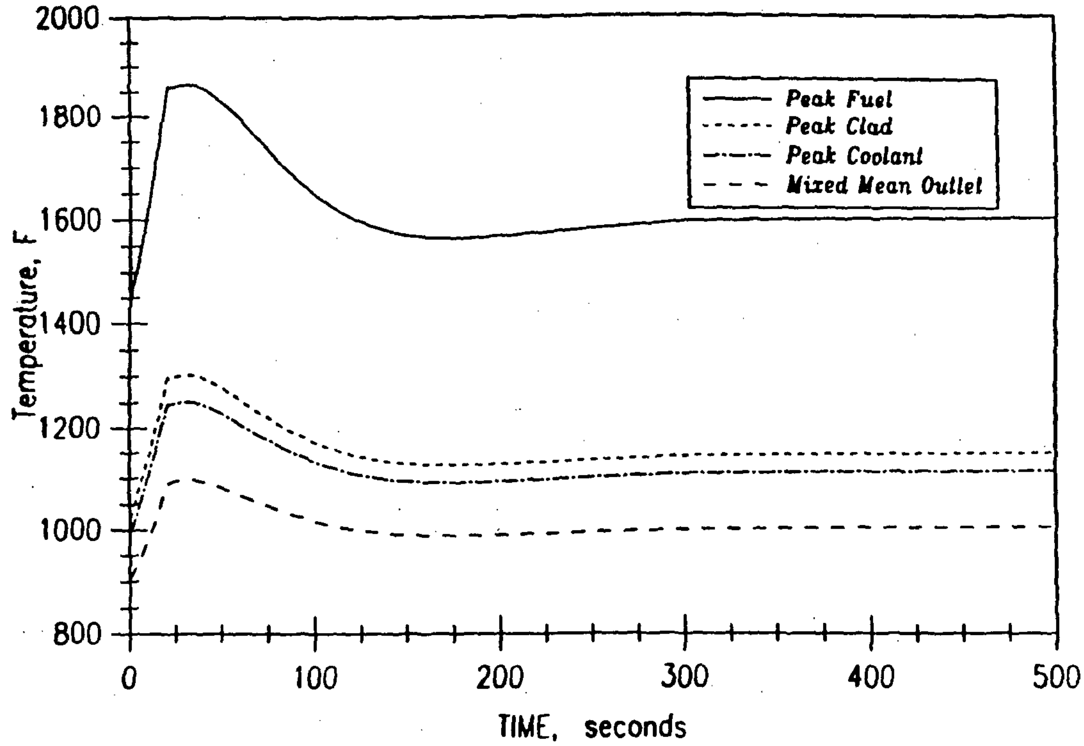


Fig. G.4.16-4 EVENT 1A, ALL-RODS WITHDRAWAL WITHOUT SCRAM, WITH FORCED COOLING: CORE TEMPERATURES

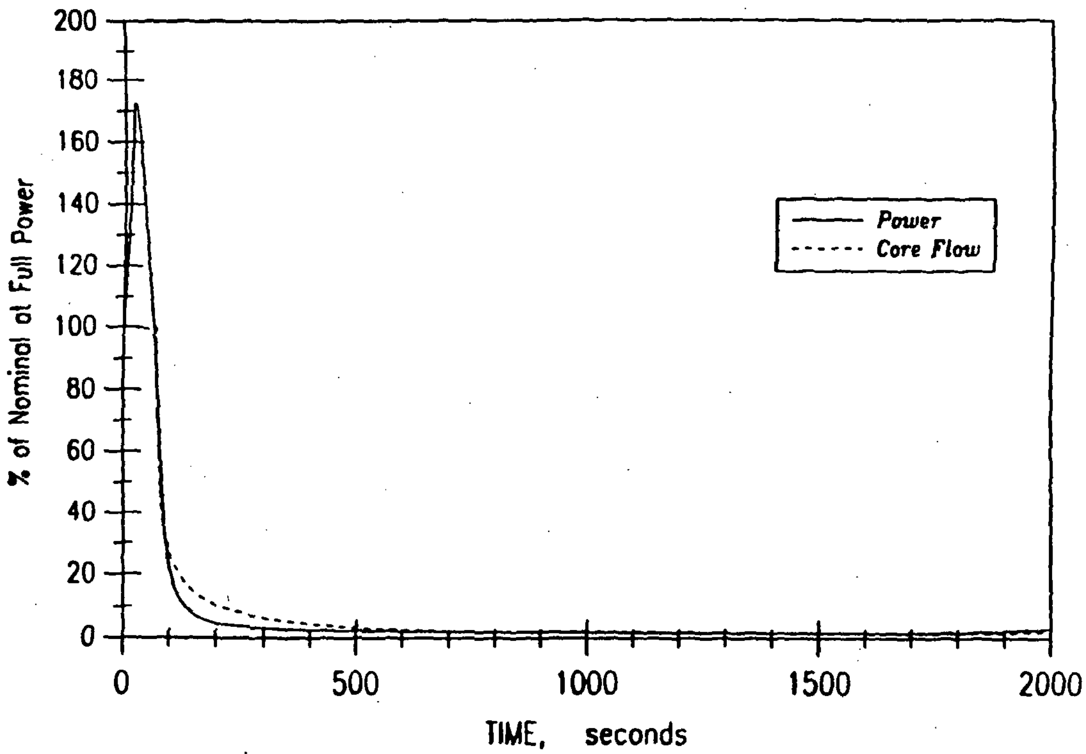


Fig. G.4.16-5 EVENT 1B, ALL-RODS WITHDRAWAL WITHOUT SCRAM, WITH RVACS COOLING: POWER AND FLOW

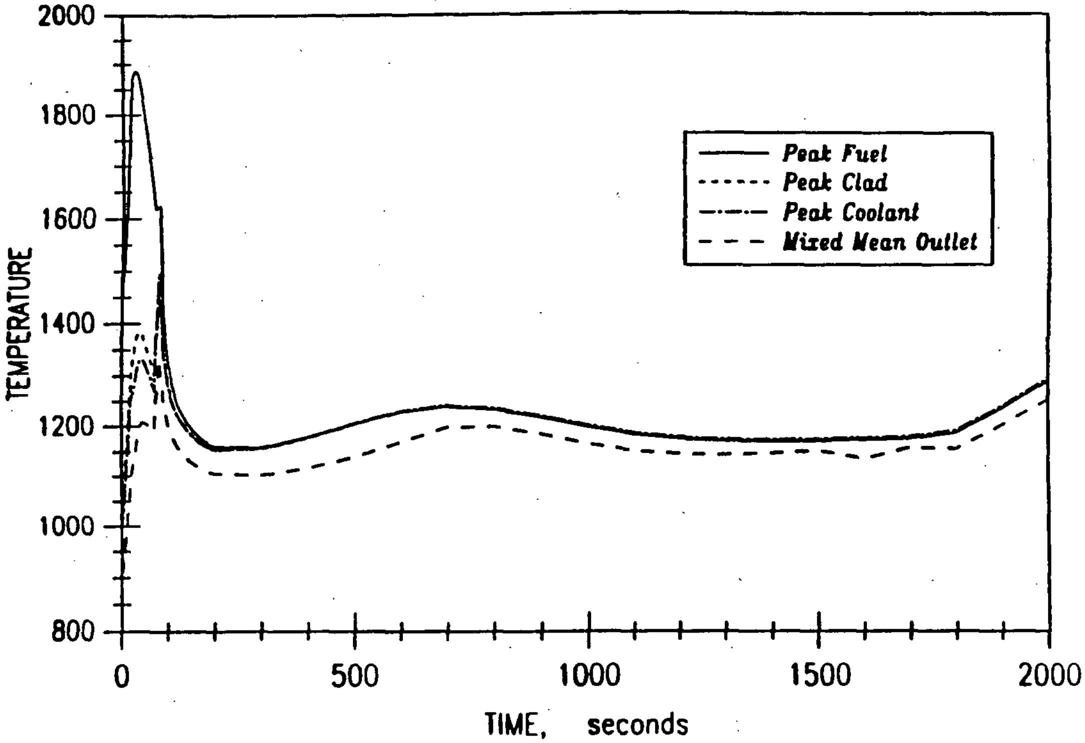


Fig. G.4.16-6 EVENT 1B, ALL-RODS WITHDRAWAL WITHOUT SCRAM, WITH RVACS COOLING: CORE TEMPERATURES

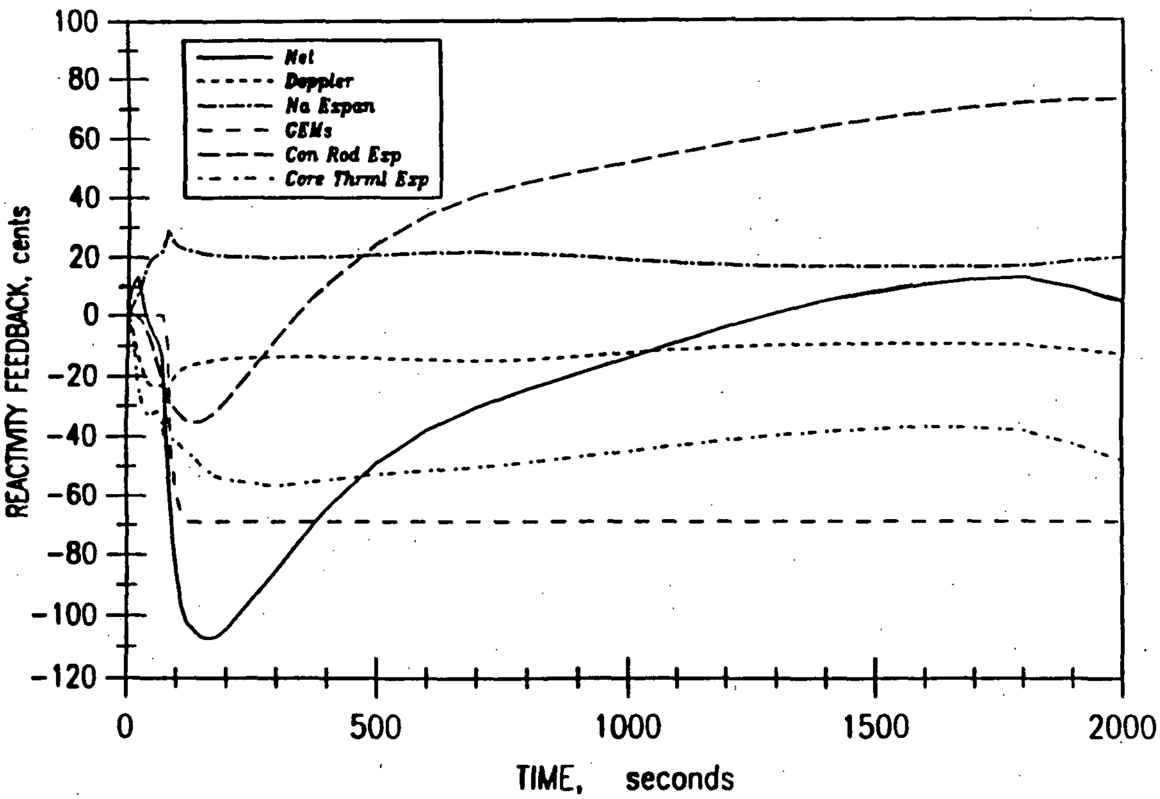


Fig. G.4.16-7 EVENT 1B, ALL-RODS WITHDRAWAL WITHOUT SCRAM, WITH RVACS COOLING: REACTIVITY FEEDBACKS

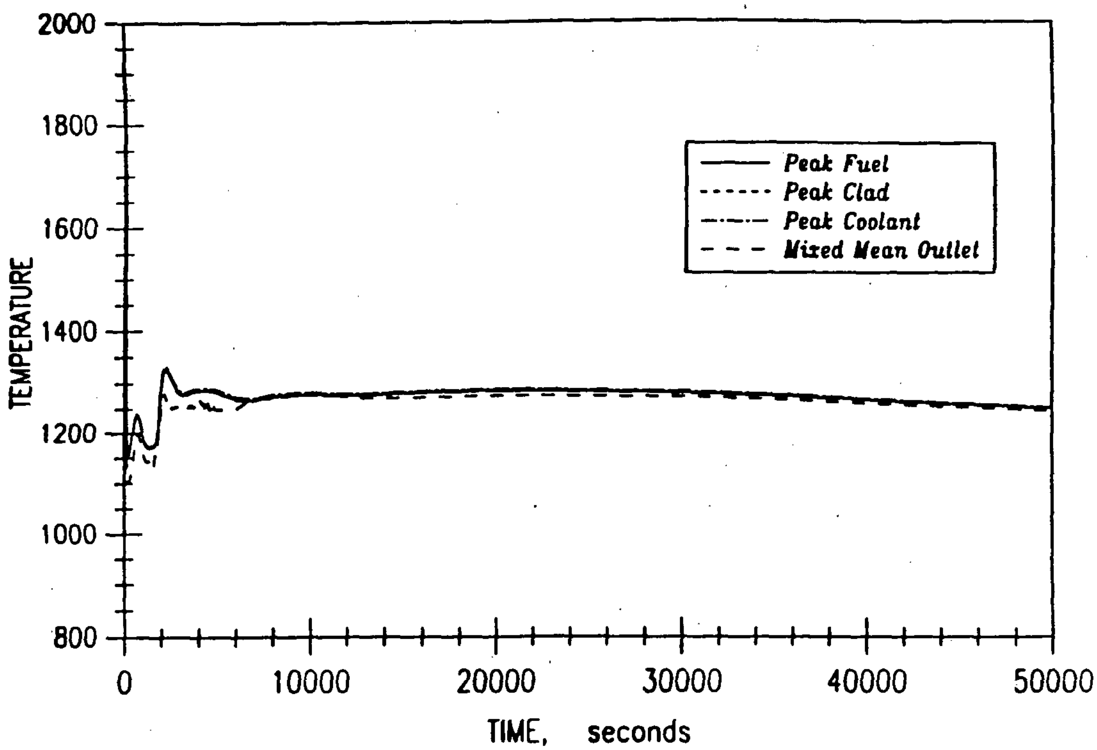


Fig. G.4.16-8 EVENT 1B, ALL-RODS WITHDRAWAL WITHOUT SCRAM, WITH RVACS COOLING: LONG-TERM CORE TEMPERATURES

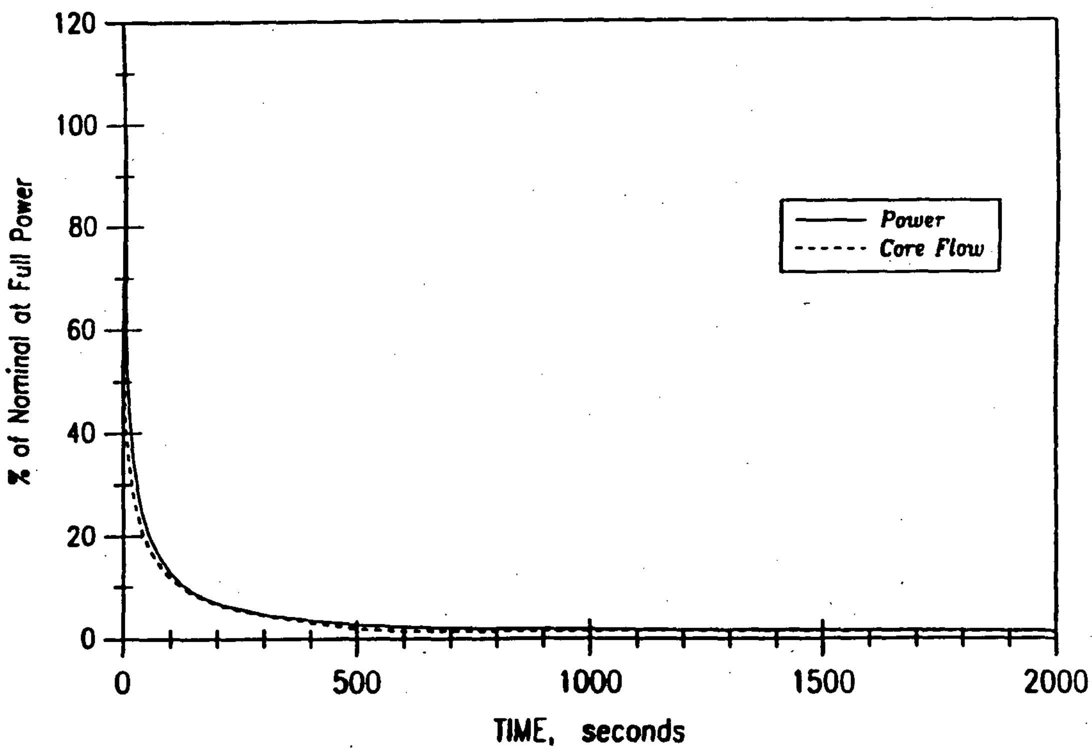


Fig. G.4.16-9 EVENT 2, UNPROTECTED LOSS OF FLOW AND IHTS HEAT SINK: POWER AND FLOW

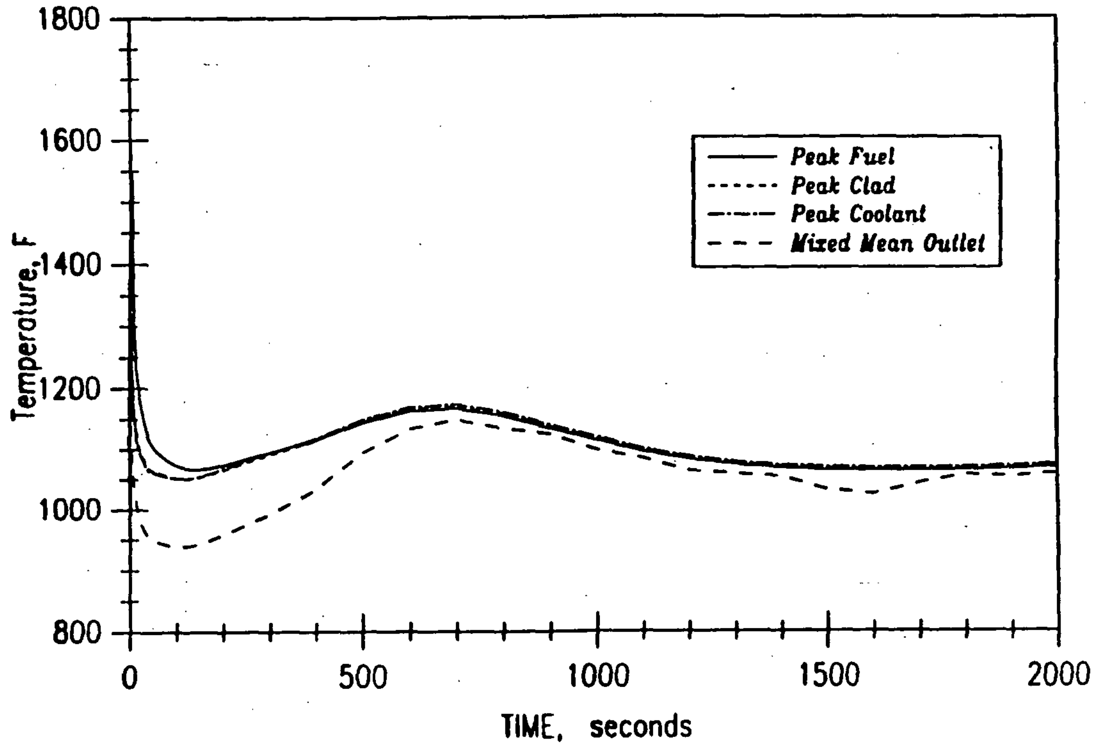


Fig. G.4.16-10 EVENT 2, UNPROTECTED LOSS OF FLOW AND IHTS HEAT SINK: CORE TEMPERATURES

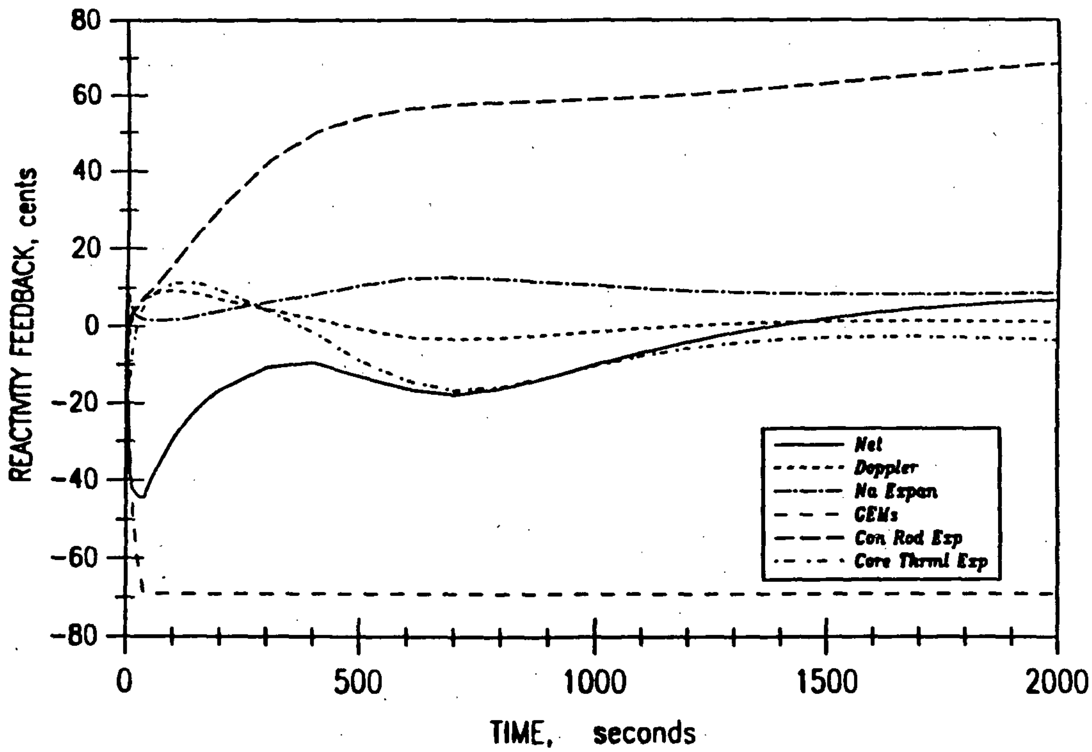


Fig. G.4.16-11 EVENT 2, UNPROTECTED LOSS OF FLOW AND IHTS HEAT SINK: REACTIVITY FEEDBACKS

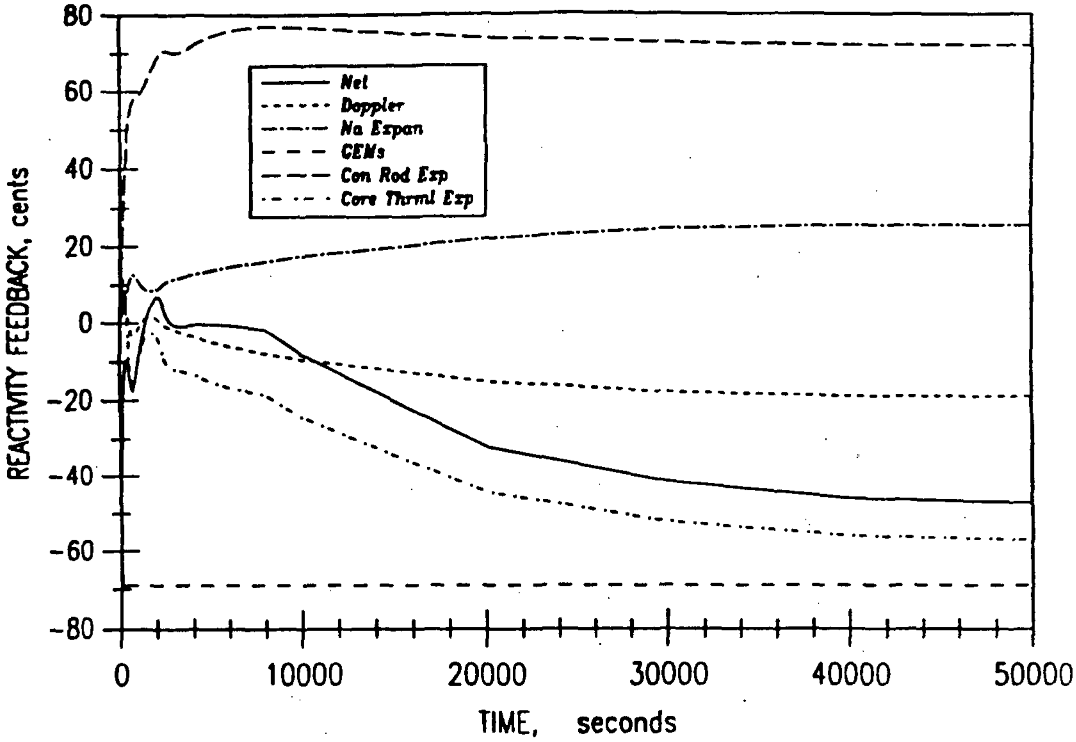


Fig. G.4.16-12 EVENT 2, UNPROTECTED LOSS OF FLOW AND IHTS HEAT SINK: LONG-TERM REACTIVITY FEEDBACKS

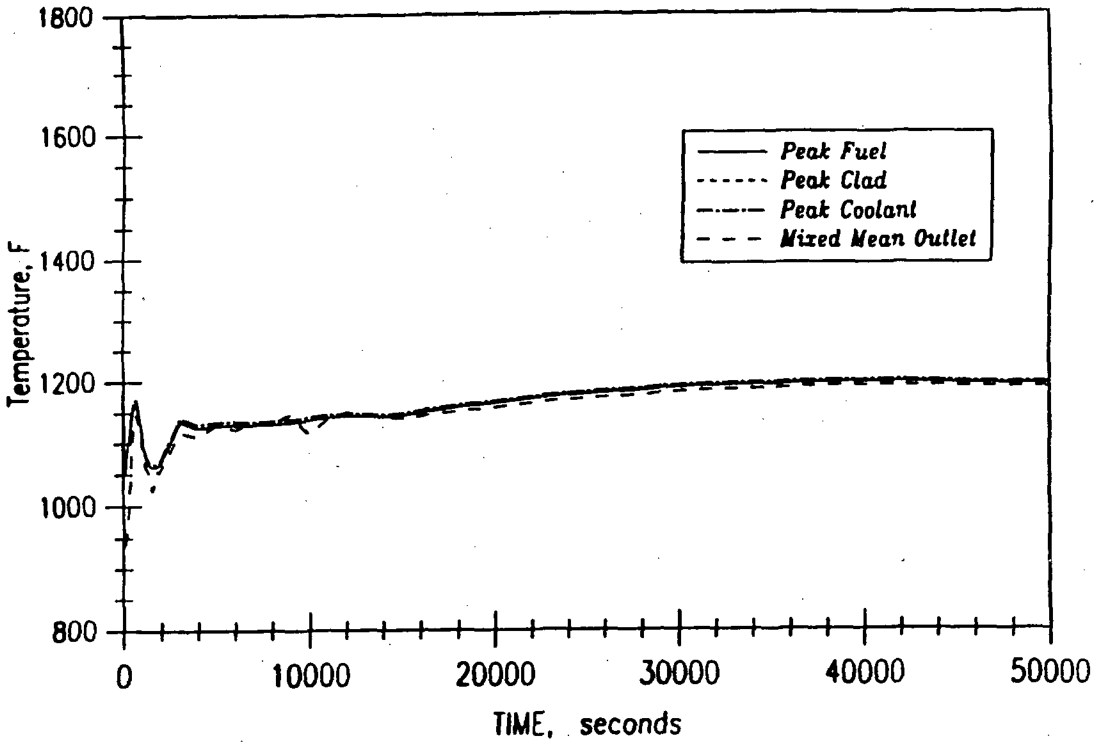


Fig. G.4.16-13 EVENT 2, UNPROTECTED LOSS OF FLOW AND IHTS HEAT SINK: LONG-TERM CORE TEMPERATURES

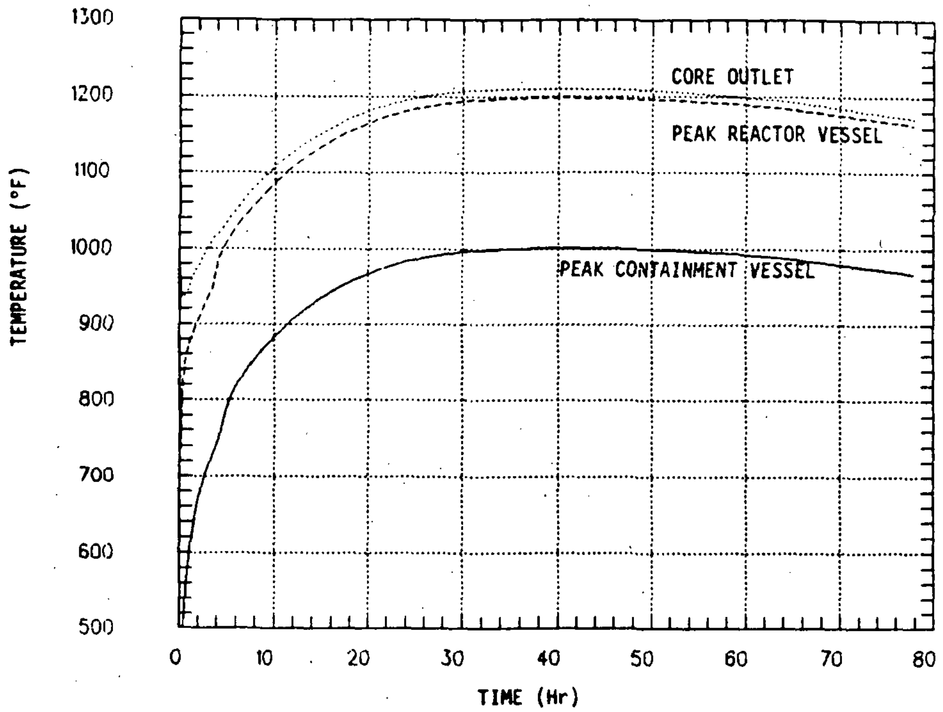


Fig. G.4.16-14 EVENT 3A, LOSS OF IHTS HEAT SINK AND 75% BLOCKAGE OF RVACS: LONG-TERM TEMPERATURES

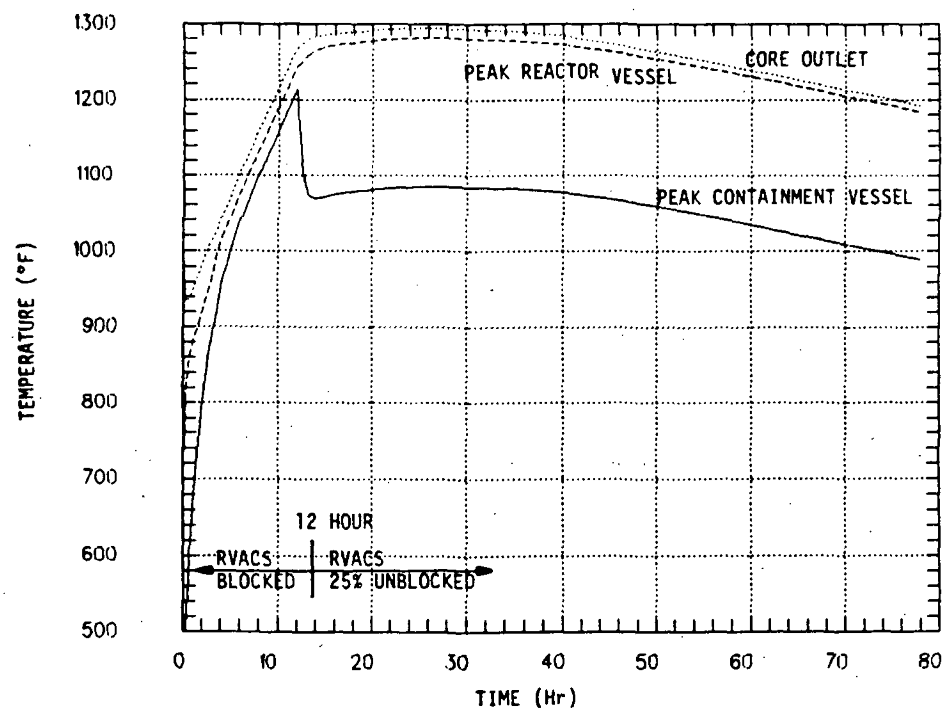


Fig. G.4.16-15 EVENT 3B, LOSS OF IHTS HEAT SINK , 100% BLOCKAGE OF RVACS, 25% UNBLOCKAGE AT 12 HOURS: LONG-TERM TEMPERATURES

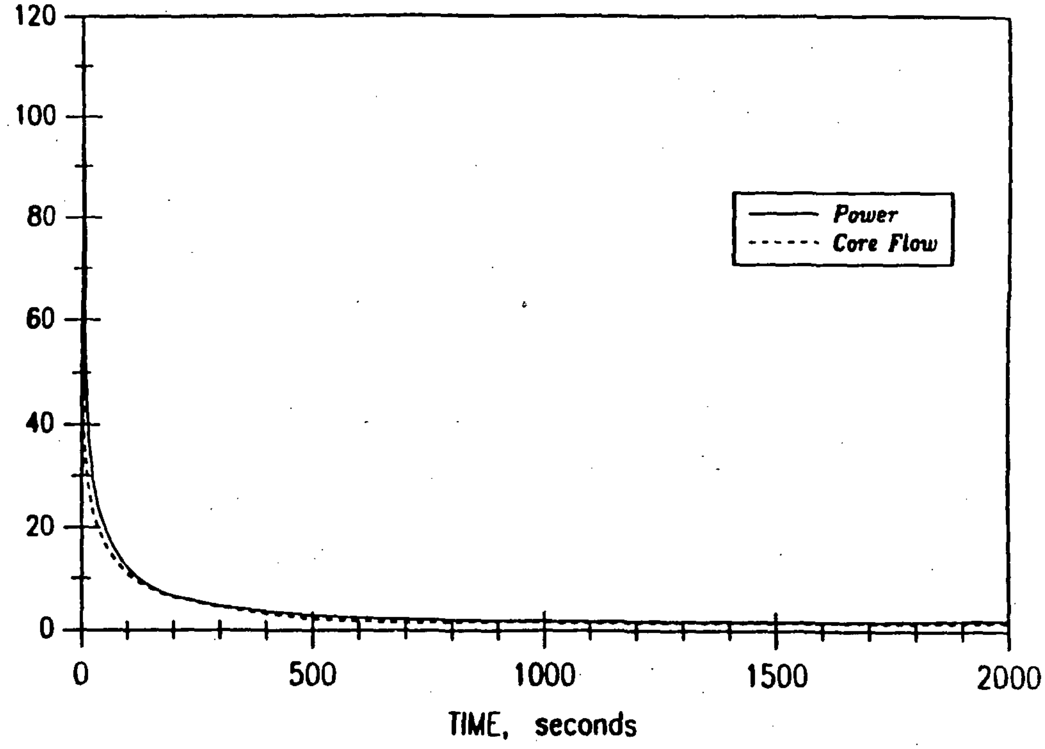


Fig. G.4.16-16 EVENT 4, UNPROTECTED LOSS OF FLOW AND IHTS HEAT SINK, COASTDOWN ON THREE PUMPS: POWER AND FLOW

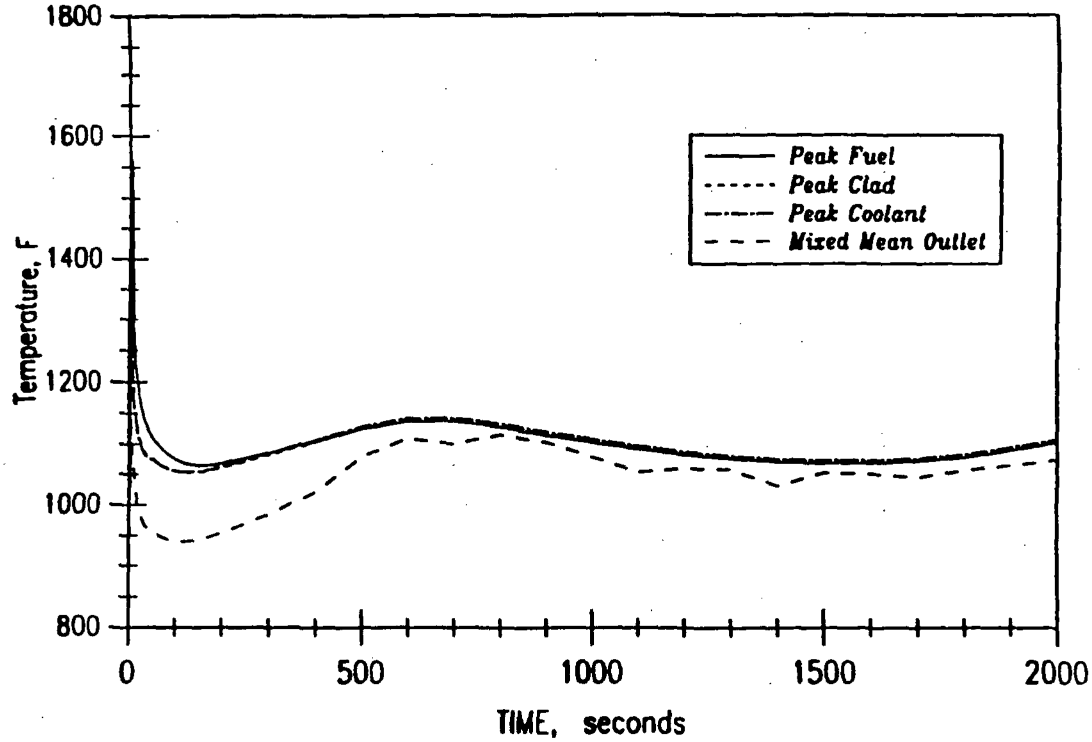


Fig. G.4.16-17 EVENT 4, UNPROTECTED LOSS OF FLOW AND IHTS HEAT SINK, COASTDOWN ON THREE PUMPS: CORE TEMPERATURES

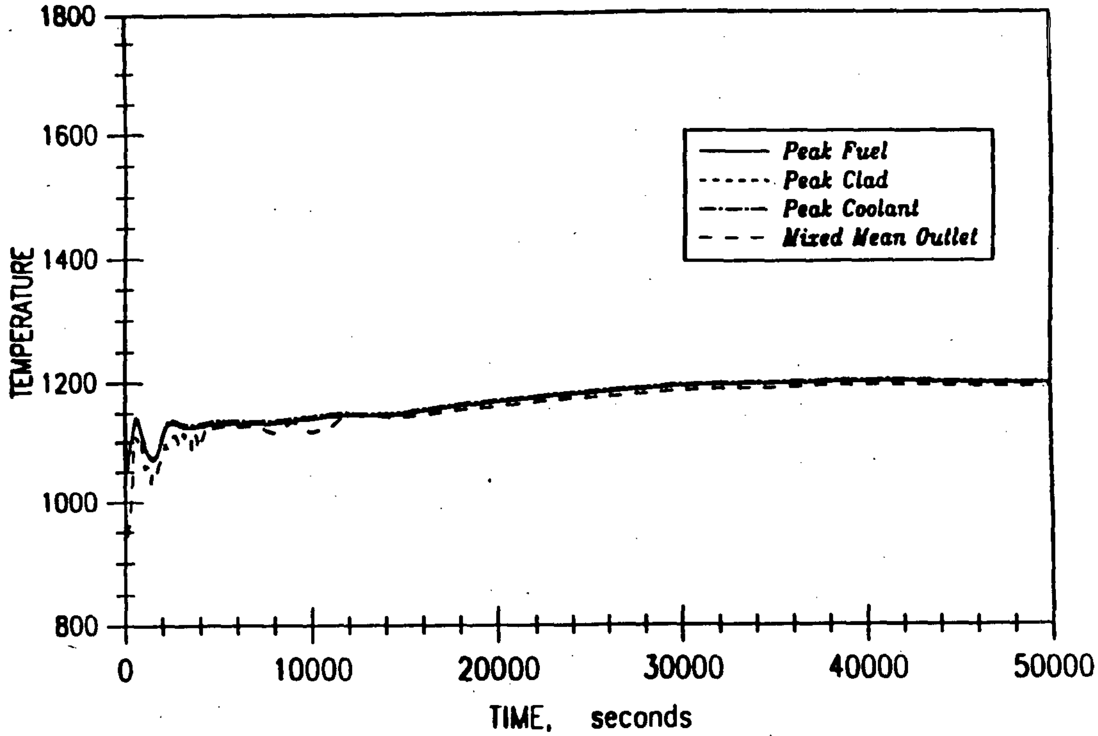


Fig. G.4.16-18 EVENT 4, UNPROTECTED LOSS OF FLOW AND IHTS HEAT SINK, COASTDOWN ON THREE PUMPS: LONG-TERM CORE TEMPERATURES

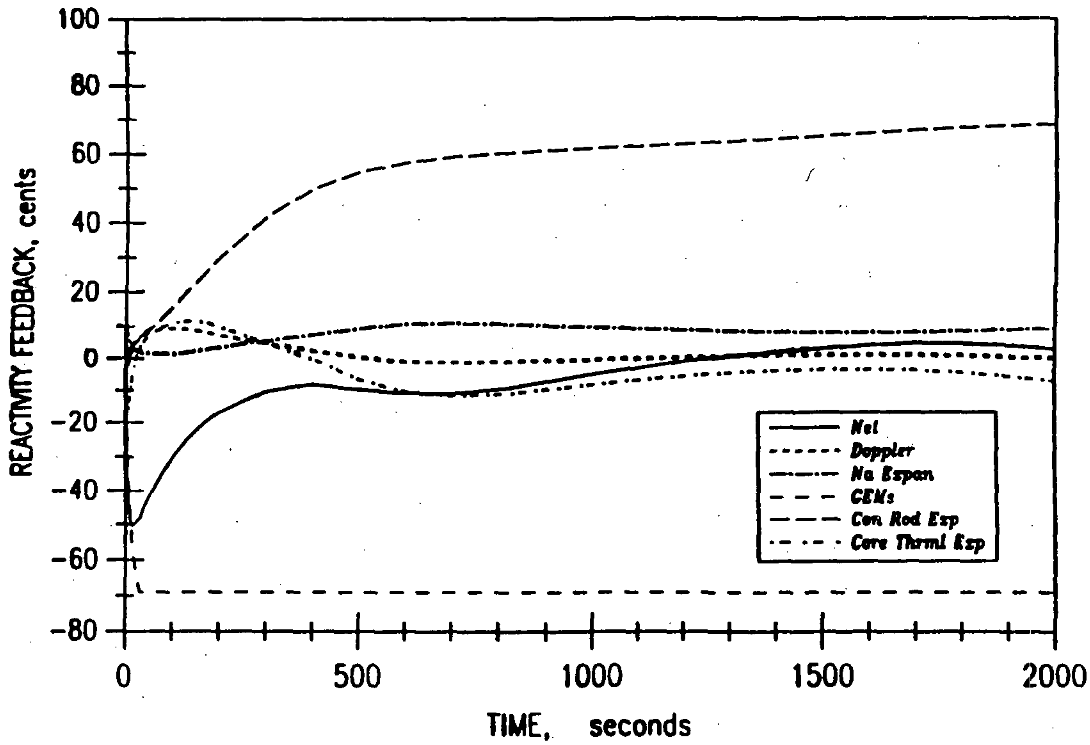


Fig. G.4.16-19 EVENT 4, UNPROTECTED LOSS OF FLOW AND IHTS HEAT SINK, COASTDOWN ON THREE PUMPS: REACTIVITY FEEDBACKS

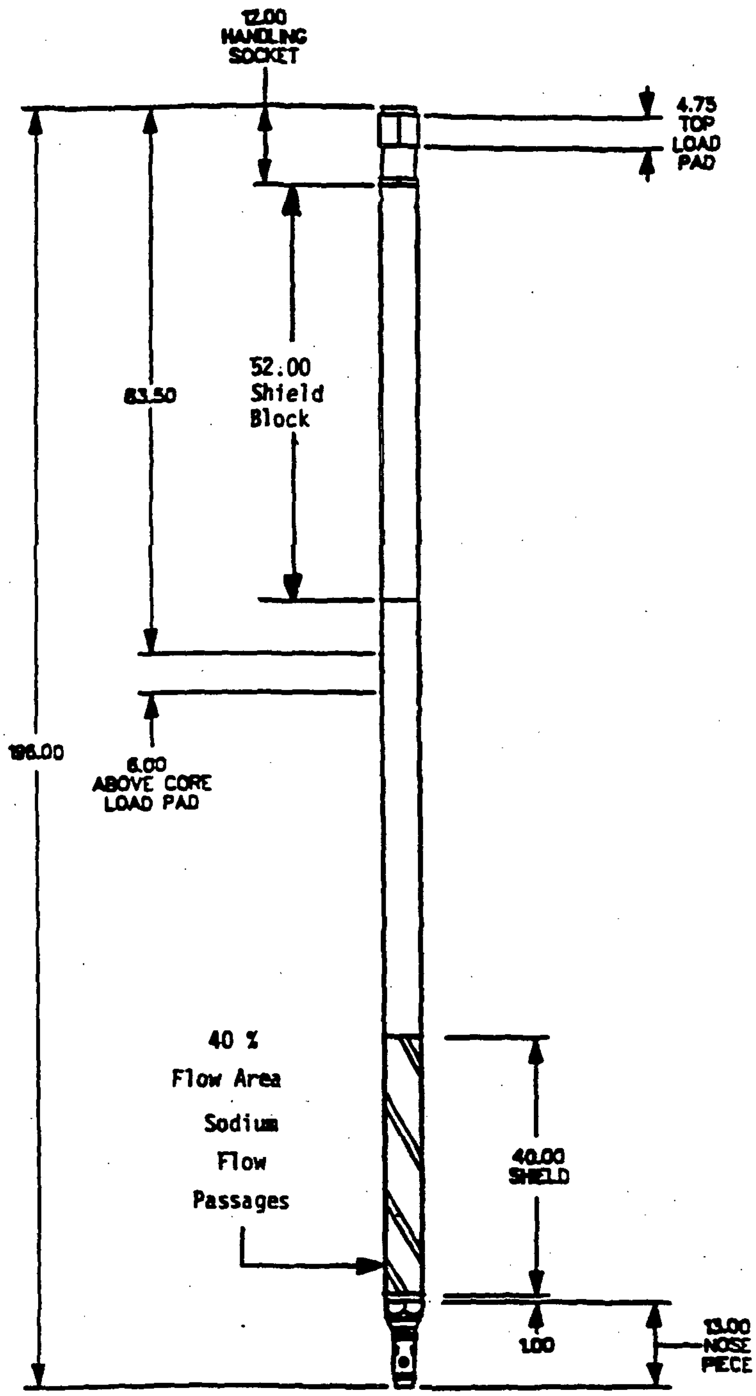


Fig. G.4.16-20 **GAS EXPANSION MODULE**

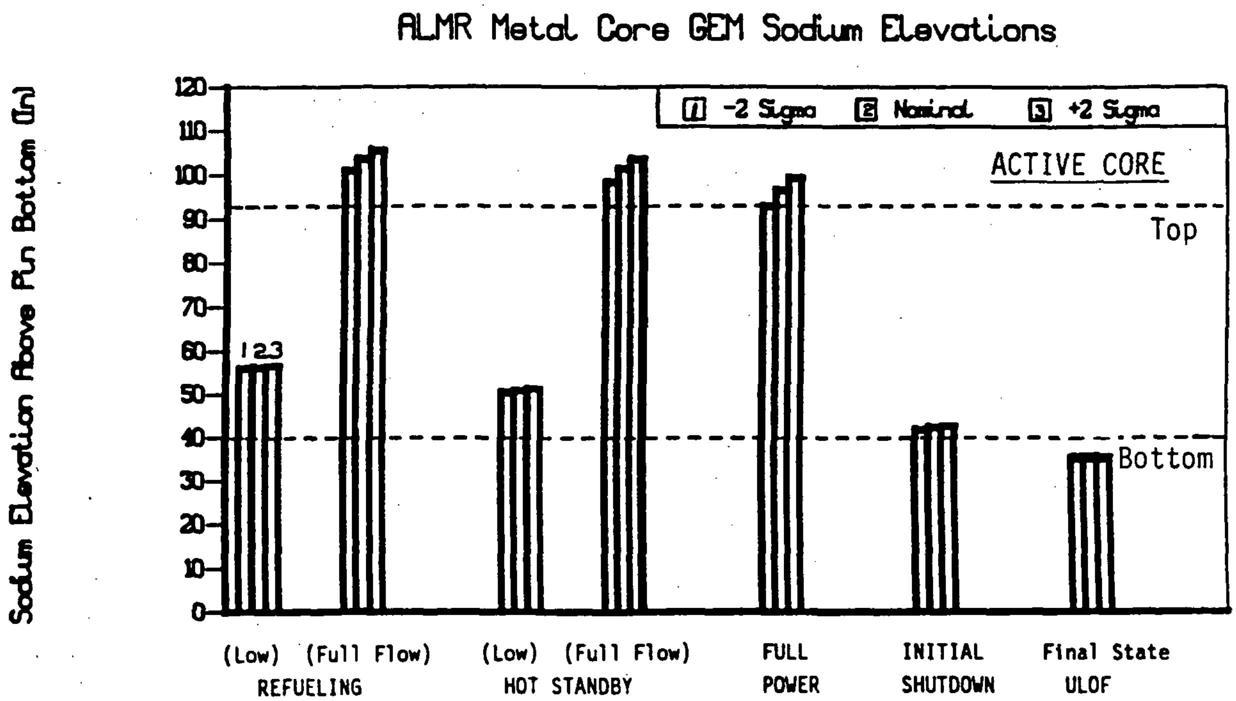
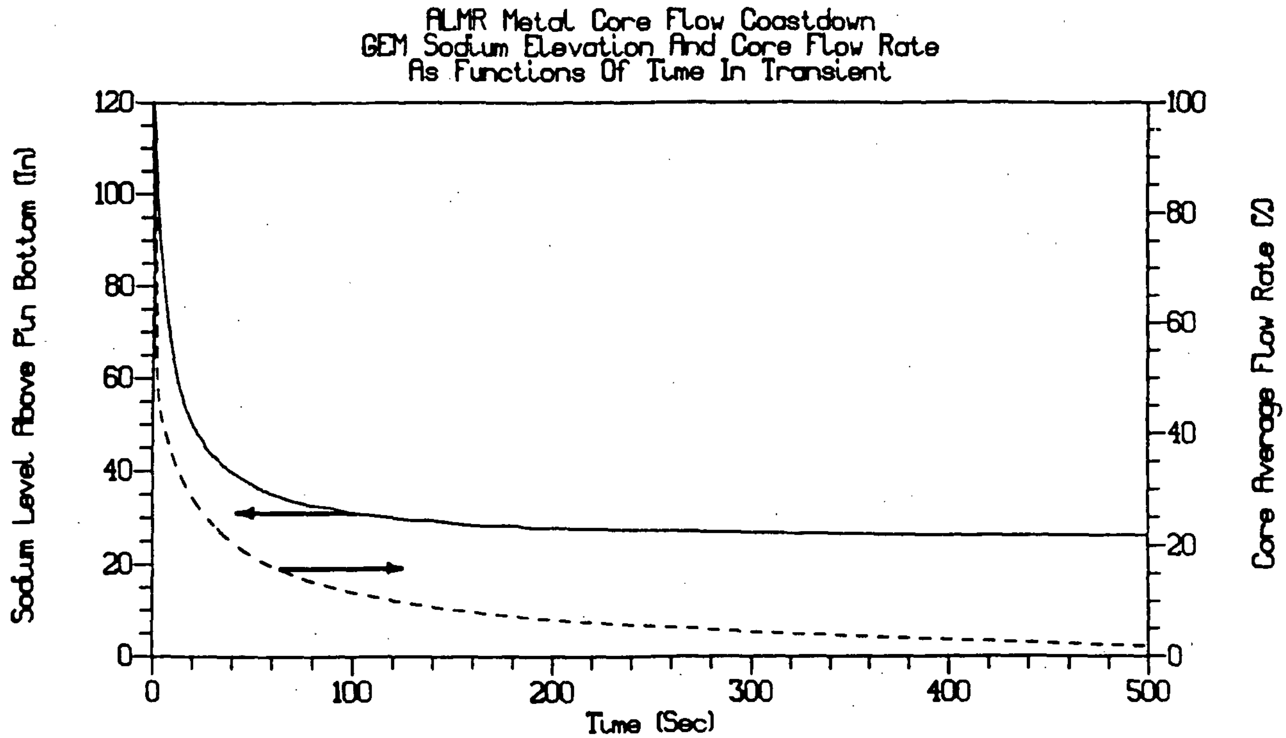
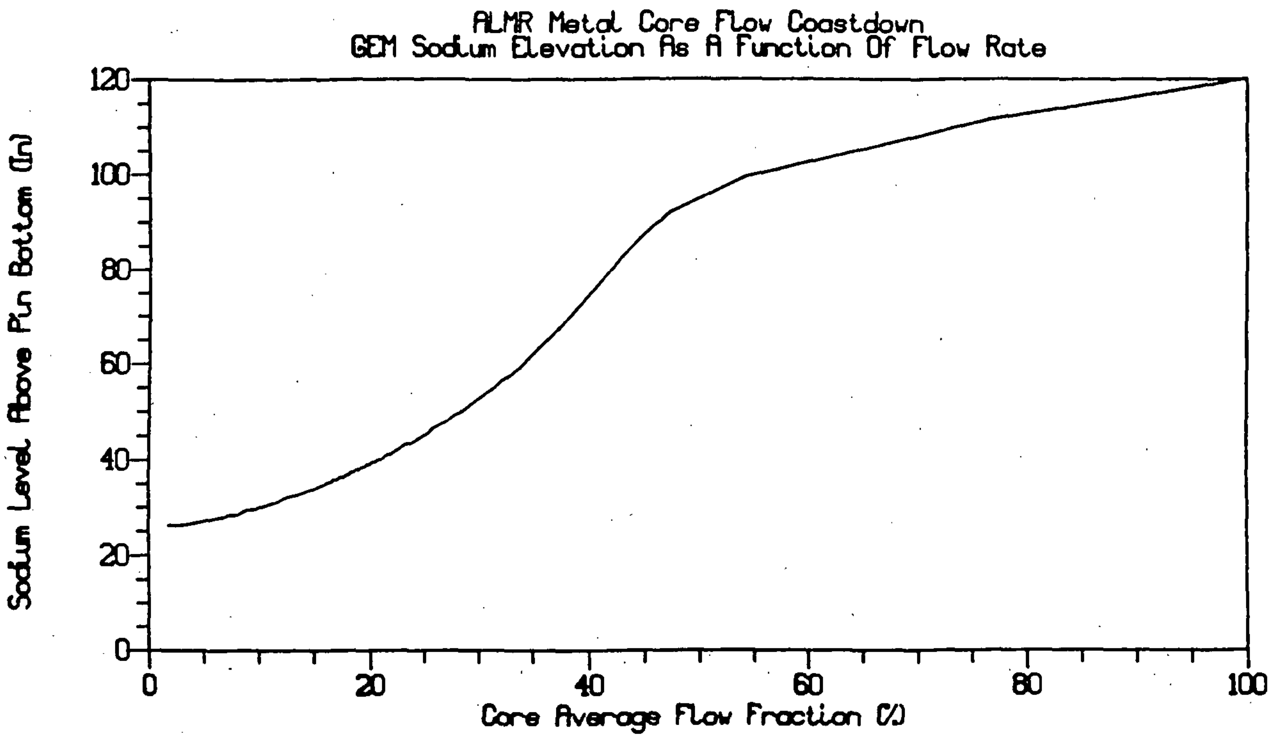


Fig. G.4.16-21 GEM SODIUM ELEVATIONS



Notes: 1) ARIES GEM tank top at about 160 inches
2) ARIES GEM does not include lower shielding

Fig. G.4.16-22 GEM SODIUM ELEVATION AND CORE FLOW RATE AS FUNCTIONS OF TIME DURING PRIMARY FLOW COASTDOWN TRANSIENT



Notes: 1) ARIES GEM tank top at about 160 inches
2) ARIES GEM does not include lower shielding

Fig. G.4.16-23 GEM SODIUM ELEVATION AS FUNCTION OF FLOW RATE DURING PRIMARY FLOW COASTDOWN TRANSIENT

G.4.17 Station Blackout

G.4.17.1 SER Position on Station Blackout

Lack of Class 1E emergency diesel power may make station blackout frequency much greater than for LWRs (Reference SER Section A.3.2, Item 2).

The frequency of station blackout for PRISM is estimated to be 3×10^{-5} per year. This frequency is comparable with current LWRs which have safety grade emergency diesels. PRISM's lack of a Class 1E emergency generator to pick up the house load during loss of off-site power may increase the frequency of station blackouts much higher than reported in the PRISM PRA. The consequential higher frequency of RVACS operation may lead to permanent damage of the reactor vessel. The plant design capabilities to withstand such an event and the ability to inspect the reactor components following the event will determine whether PRISM will be allowed to have a station blackout frequency higher than current LWRs (Reference SER Section A.3.2, Item 2).

G.4.17.2 Reference Design Features and Approach For Station Blackout

Station blackout is defined as the loss of all off-site AC power to the essential and nonessential electrical buses with concurrent turbine trip and the unavailability of the redundant on-site emergency AC power systems (Reference G.4.17-5). For the ALMR, station blackout (Bounding Event No. 2) is defined as loss of all AC power for 36 hours. Station blackout frequency estimates are based on PRA evaluation (Reference G.4.17-1). For such an event to occur, the following conditions must both be postulated:

a. Off-site power lost and not recovered
b. Runback capability lost (i.e., loss of ability to reduce power and provide house load AC power from all three of the 465 MWe plant turbine generators)

The ALMR includes features which passively mitigate both the safety and investment impacts of a station blackout on the plant. Reactor scram is provided by the battery backed Class 1E RPS. Alternative safety grade means to shut down all reactors is provided on loss of normal AC power by de-energizing the electromagnetic latches, allowing the control rods to enter the core by gravity, and, as a diverse backup, by separate drive-in motors powered by battery backup, which rapidly run in each control rod driveline. Post shutdown monitoring is provided by the remote shutdown facility, which is also battery backed with 36 hour capacity. Decay heat removal is passively accomplished by the reactor vessel auxiliary cooling system (RVACS) and an auxiliary cooling system (ACS) as part of the intermediate heat transport system and the steam generator system.

The RVACS is a passive safety grade shutdown heat removal system. Reactor decay removal by RVACS alone is sufficient for maintaining the primary coolant temperature below 1200°F, following reactor scram, without any additional heat removal system. ACS operation in a natural circulation mode, in combination with RVACS, further reduces the peak primary coolant temperature below 1000°F. Because of its simplicity, passive operation, resistance to operational failure, and ability to maintain reactor temperatures at acceptable levels, the RVACS is the only shutdown heat removal system required to ensure reactor safety. The ACS is non-safety grade and is provided to increase plant availability by decreasing post-shutdown cooldown periods.

Upon loss of off-site power, the local turbine controller will quickly close the main steam throttle valves to maintain approximately 8% of rated steam flow to the turbine, matching house load requirements. Simultaneously, the turbine bypass valves will open diverting approximately 60% of the steam flow to the condenser. The remaining 32% of the steam flow will be initially vented to maintain steam pressure with safe levels. The reactor plant control system (PCS) will run back the control rods, initially reducing reactor power to the reactors to about 68% of rated which terminates the steam venting. Power requirements to maintain operation of the auxiliary equipment is approximately 120 MWe for a three block plant. If the station blackout continues for an extended period, subsequent reactor power maneuvers will be performed to optimally balance house load demands with turbine output and eliminate the bypass. Operation of one reactor produces adequate power to sustain full circulation of the primary and intermediate sodium and feedwater systems in the other eight reactors. In this case, the consequences of a loss of off-site AC power are benign as the normal shutdown heat removal paths are maintained.

The arrangement of the ALMR plant with three 465 MWe (net) turbines (each with 60% bypass capability) is well suited for picking up the house load of 40 MWe per block even in the event of complete grid load rejection. Plant controls are designed to provide a runback capability to reduce reactor power and turbine output to house load levels without sustaining a turbine trip. The control scheme is patterned after similar controls successfully used in several operating PWR plants. If the control system fails for one or two turbines and these turbines do not stay on line, the remaining turbine is capable of supplying the house load. Only if all three turbines are lost and reserve power is not available to supply house load, will the reactors scram and the passive heat removal systems be required to remove decay heat.

During station blackout, the ACS will continue to operate in a natural circulation mode. The additional decay heat removal capability of the ACS reduces the expected peak sodium and vessel temperatures to approximately 920°F. The corresponding peak temperature for the 316 stainless steel reactor vessel is approximately 815°F and about 450°F for the 2-1/4Cr-1Mo containment vessel. These temperatures are significantly below the respective design temperatures for the vessel materials.

G.4.17.3 Rationale Supporting Acceptability of Station Blackout for ALMR

Factors considered in the risk evaluation for station blackout include: (1) the likelihood and duration of the loss of off-site power, (2) the reliability of the on-site AC power system, and (3) the potential for severe accident sequences after a loss of all AC power, including the capability to remove core decay heat without AC power for a limited time period (Reference G.4.17-5).

Bounding Event No. 2 (BE-2) is defined as a station blackout which leaves the ALMR plant without off-site AC power for 36 hours.

The frequency of station blackout lasting longer than 36 hours for the ALMR is estimated to be 1×10^{-7} per year, and lasting less than 36 hours is estimated to be 1×10^{-3} per year. The combination of the following three event frequencies are used to obtain these estimates:

1) Loss of off-site power initiating event: probability of 4×10^{-2} per year;
2) Failure of all three turbine generators to pick up house load following loss of off-site power: probability of 3×10^{-2} per demand;
3) Conditional probability of not recovering off-site power within 36 hours: probability of 1×10^{-4} per demand.

The normal response to station blackout is described in Section G.4.17-2. Should the plant runback system fail to perform as designed, and the reactors scram producing no power for plant load demands, only the passive RVACS and ACS decay heat removal systems would remain operational. For the safety evaluation of station blackout, it is assumed that the reactor scrams, and natural circulation cooling is the only mode of cooling available. Assuming loss of active cooling, the ACS will continue to operate in a natural circulation mode. Natural circulation of the primary sodium will be established and the reactor decay heat will be removed by the ACS and the RVACS. The use of non-safety related, as well as safety related, equipment and systems to cope with station blackout is consistent with Reg. Guide 1.155 (Reference G.4.17-4).

Two factors affect the ability of the ALMR to passively and safely withstand the consequences of a station blackout: the resulting material conditions (temperature and pressures), and the material degradation limits at these conditions.

Estimated core outlet temperature histories following reactor scram are shown in Figure G.4.17-1 following the onset of decay heat removal by the RVACS and with natural circulation through the ACS. Core outlet temperature will rise about 15°F, peaking at about 920°F (nominal) after about five hours for a startup core (LWR recycle Pu plus minor actinides) before decreasing as the sensible heat is depleted. The design of the reactor internals promotes natural circulation within the reactor vessel and results in fairly uniform temperatures.

The higher temperatures following a blackout would degrade the structural performance by imposing thermal stress cycles on the reactor components and by decreasing material load carrying capacity. The ASME Code guards against failure by these loads by: 1) limiting the number of load cycles through a temperature-dependent fatigue limit, 2) limiting the duration and magnitude of loading through a time-temperature-dependent stress limit, and 3) prescribing an enveloping limit on the combined creep-fatigue damage for the entire loading history. The Code specifies different design limits for: 1) normal operation (Level A) including anticipated transients (Level B) for which the stresses, deformations, and damage are limited to permit plant operation without any remedial measures, 2) emergency conditions (Level C) for which larger deformations are permitted requiring inspection and repair to ensure adequate performance before resuming normal operation, and 3) faulted conditions (Level D) which maintain the pressure boundary and coolant path integrity but permit large deformations which may make further operation very difficult.

Conservative analyses show that the loads imposed by station blackouts remain within the ASME Code Level B design limits. The principal stresses imposed by a blackout are the stresses associated with reactor scram when the cold sodium imposes a thermal transient over a short time, and sodium stratification imposes axial thermal gradients over a somewhat longer time period. The thermal gradients from the subsequent decay heat loads during the blackout are mild compared to the scram-associated temperature gradients. However, for conservatism, the strain cycles associated with the blackouts were assumed to be the same as those for the normal scram cycle. Additional conservatism was added to the evaluation by assuming the material design limits for the blackout loads are the ASME Code limits at 1300°F. This temperature is considerably higher than the peak coolant temperatures shown in Figure G.4.17-1. Finally, 20 station blackouts per 60-year plant life were assumed to ensure a conservative event frequency. Table G.4.17-1 shows the number of duty cycle events, associated peak temperatures, and fatigue damage together with the corresponding values for the station blackout. The 2000 non-blackout load cycles in the table conservatively envelop the ALMR duty cycle. The 20 cycles assumed for station blackout conservatively envelop the estimated six blackout events over 60 years based on a frequency of one blackout per 10 site-years cited in Reference G.4.17-5 and the current ALMR estimate of 6×10^{-6} per 60 plant years.

TABLE G.4.17-1 FATIGUE DAMAGE IN REACTOR STRUCTURES

Component	Thermal Stress Cycle, psi	Maximum Temperature, °F	ASME Code Fatigue Limit, No. of cycles N	Number of Load Cycles n	Fatigue Damage n/N
Reactor Vessel:					
Design Duty Cycle	34000	850	> 1000000	2000	0.002
Station Blackout		1300	400	20	0.050
Fatigue Damage					0.052
Reactor Liner, UIS:					
Design Duty Cycle	21750	950	100000	2000	0.020
Station Blackout		1300	10000	20	0.002
Fatigue Damage					0.022
Containment Vessel:					
Design Duty Cycle	5000	500	> 1000000	2000	0.002
Station Blackout		800	> 1000000	20	0.000
Fatigue Damage					0.002

Notes: 1. Blackout loading enveloped by assuming a 1300°F peak core exit coolant temperature instead of <1000°F expected temperature.
2. 20 site blackouts assumed.

The ASME Code limits elevated temperature creep damage by specifying limits on the duration of loading at different temperatures. As shown in Figure G.4.17-1, the coolant and therefore component temperatures would remain below 920°F during station blackout with heat removal by natural circulation through the ACS and RVACS. These temperatures would produce insignificant creep damage at the pressure and gravity stresses operating after scram. If ACS, being a non-safety grade system, is arbitrarily assumed to fail, the coolant temperature would increase to a peak value of

1100-1200°F depending on the condition of the RVACS surfaces, and then decrease. The station blackout will be corrected in less than 36 hours (a short time compared to the durations shown in Figure G.4.17-1) with a corresponding reduction in the elevated temperature exposure. The effect of these various possible coolant temperature histories were enveloped by assuming a peak coolant temperature of 1300°F maintained for 100 hours for each of the assumed 20 blackouts during a reactor design life giving a total exposure of 2000 hours and corresponding creep damage levels shown in Table G.4.17-2. The component temperatures in the table reflect the effects of natural circulation which keeps the core in a near isothermal condition, thereby minimizing thermal gradients, while maintaining the core support structure more than 100°F cooler than the core exit coolant. The temperature drop across the reactor-containment vessel annulus keeps the containment vessel more than 400°F cooler than the core exit coolant.

TABLE G.4.17-2 CREEP DAMAGE IN REACTOR STRUCTURES

Component	Pressure Gravity Stress, psi	Assumed Temper-ature, °F	ASME Code Life Limit, hours T	Assumed Load Duration t	Creep Damage t/T
Reactor Vessel	1640	1300	> 300000	2000	0.007
Core Support	4220	1200	> 300000	2000	0.007
Containment Vessel	2700	900	> 300000	2000	0.007

Notes:
1. Creep damage during non-blackout duty cycle is negligible.
2. Blackout loading enveloped by assuming a constant 1300°F core exit coolant temperature for 100 hours instead of the expected peak temperature of <1000°F for the blackout duration.
3. 20 site blackouts assumed.

The stresses associated with a blackout are similar to the stress levels for normal scram and are enveloped in the normal design margins. The only additional effect of station blackout is to increase the creep-fatigue damage by increasing the duration of elevated temperature exposure and by increasing the number of load cycles. As indicated by the conservative damage estimates in Tables G.4.17-1 and G.4.17-2, the maximum combined creep-fatigue damage including the effect of blackout is 0.059 (0.052 fatigue damage + 0.007 creep damage) estimated for the reactor vessel which is insignificant compared to the ASME Code creep-fatigue damage limit of 1.0.

Thus, station blackouts will not load the reactor system beyond the normal operation loads. With the structural damage from the blackout determined by the temperatures reached by the core exit coolant, the normal temperature monitoring will be sufficient to ensure that the reactor system is not challenged beyond the conservative temperatures and loads assumed in the evaluation.

G.4.17.4 Summary

Station blackouts will not load the reactor system beyond the normal design loads and need not be considered as a separate challenge to the reactor design. With the structural damage from the blackout determined by the temperatures reached by the core exit coolant, the normal monitoring of the coolant temperatures will be sufficient to ensure that the reactor system is not challenged beyond the normal operation loads.

The probability of station blackout for the ALMR is low, principally due to the ability of ALMR's multi-turbine arrangement of small turbines with large bypass and the turbine controller to pick up the house load. However, even with the low probability of station blackout, the passive decay heat removal systems are highly reliable and effective. The maximum temperatures achieved during a station blackout with RVACS and ACS natural circulation decay heat removal is approximately 920°F for a start up core. The temperature peak is even less for an equilibrium core. At these temperatures, the stress levels for the reactor and containment vessels are similar to normal scram and are enveloped in the normal design margins. The short exposure to elevated temperatures result in minimal increase in the creep-fatigue damage, which remains substantially below ASME damage limits. Thus material degradation conditions are not experienced. Consequently, station blackout events will not lead to degraded core conditions, or initiate severe accident sequences, and are not a concern to the integrity of the primary coolant or the containment boundaries.

G.4.17.5 References

G.4.17-1 GEFR-00873, Probabilistic Risk Assessment of the Advanced Liquid Metal Reactor, November 1989

G.4.17-2 GEFR-00832, PRISM Reactor Structural Evaluation, September 1988

G.4.17-3 GEFR-00833, PRISM Thermal Hydraulic Analysis, October 1988

G.4.17-4 USNRC Regulatory Guide 1.155, Station Blackout, August 1988

G.4.17-5 53 FR 23203, 10 CFR 50, Final Rule, Station Blackout, June 21, 1988

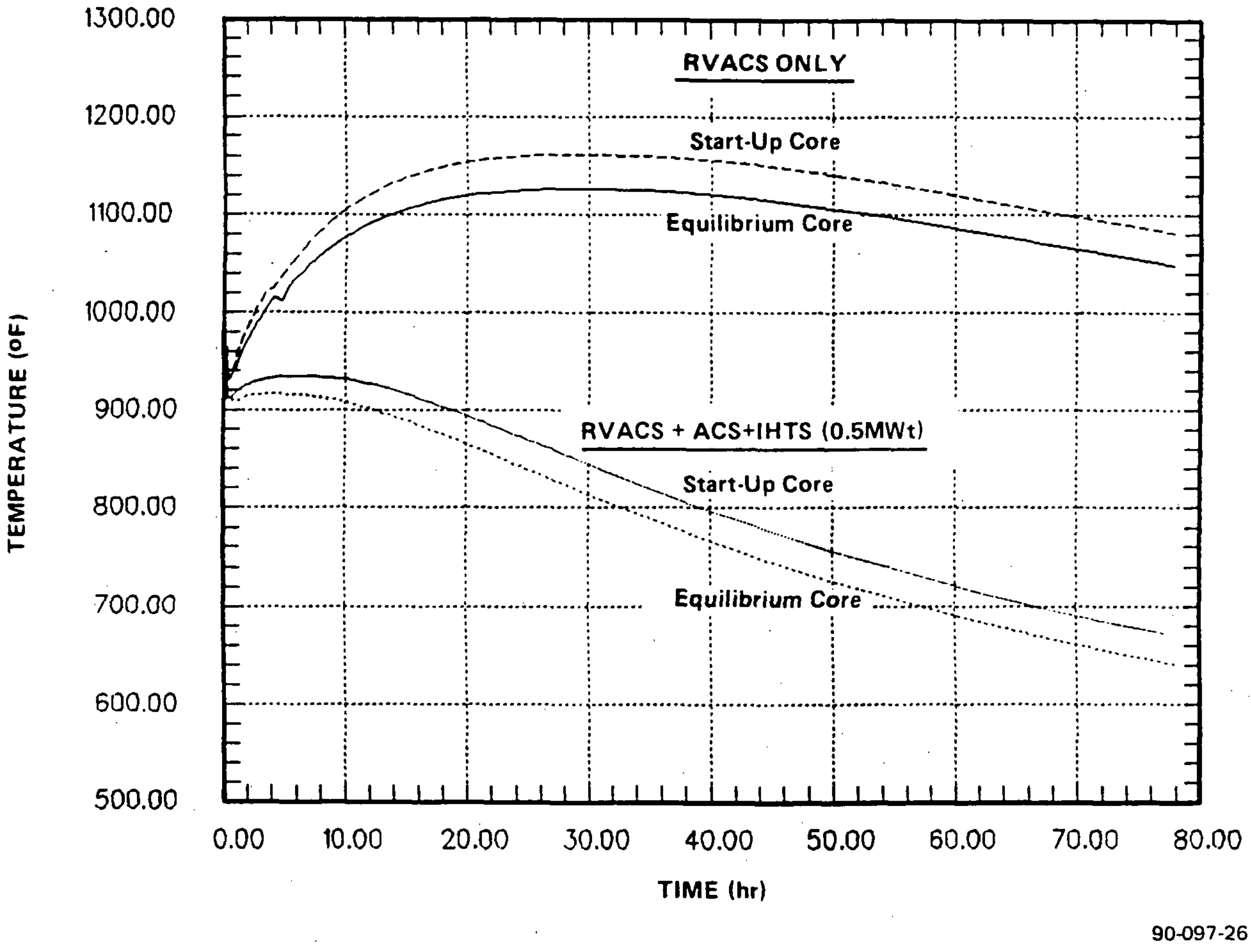


Figure G.4.17–1 AVERAGE CORE OUTLET TEMPERATURES AS FUNCTIONS OF TIME FOR RVACS ONLY AND RVACS PLUS ACS CASES

G.4.18 Risk Assessment

G.4.18.1 SER Position on PSID Risk Assessment

Section A.7 of the draft SER notes that "Redundancy, diversity, and passive safety features designed into PRISM resulted in very low PRA risk estimates." However, because of "...large uncertainties in the front end of the PRA..." and "...large uncertainties in phenomenological treatment of the core response and consequence analysis...", the SER states: "...it is the staff's judgment that only limited uses can be made of the PRA." The SER identified nine caveats which should be taken into consideration when the PRA estimates are used as a means of judging the PRISM's safety capacity. These are:

a. "The PRA lacks the detail and data required to substantiate system reliability estimates. Major weaknesses include essentially unmodeled common-cause failures, human factors, and support system failures. It is also believed that some of the basic event probabilities have been underestimated." The SER (Pages A-3, A-4 and A-14) has identified the following specific events as having been assigned optimistic probabilities:

   (1) Catastrophic reactor vessel failure
   (2) Station blackout
   (3) Steam generator tube rupture
   (4) Inadvertent control rod withdrawal
   (5) Reactor protection system failure
   (6) Primary pump coastdown system failure
   (7) Shutdown heat removal system failure
   (8) Seismic isolator system failure
   (9) Loss of the inherent feedback capability

b. "External events other than seismic have not been quantified. Seismic analysis is limited to the hazard curve assumed for the GESSAR II site. Fragilities are simply based on engineering judgment.

c. "An assessment of system interactions among safety systems, support systems, and other modules have not been performed.

d. "Source term estimates may be low for some scenarios as a result of extrapolating from oxide fuel to metal fuel." The SER singles out the release fractions of strontium (Sr) and barium (Ba) as possibly being underestimated.

e. "Retention of fission products in the head access area appears optimistic and needs to be substantiated by additional analysis.

f. "A mechanistic analysis of the accident sequences has not been performed." Moreover, the SER expresses the concern that there is very limited experience with modeling of metal fuel performance under an unprotected loss of flow accident.

g. "Uncertainties have not been quantified, nor are they well understood at this conceptual design stage.

h. "The role of the operator is not apparent from the PRA. Credit in the form of operator recovery has been taken, although it has not been established what actions will be taken or if operators will even be available to perform such actions.

i. "...a greater effort will be needed to achieve reasonable completeness at the lower end of the probability frequency spectrum..." in order to substantiate the very low risk estimates reported in the PRISM PRA.

The draft SER recommends that the above items "...should be addressed at a later design stage." The following sections provide an early discussion of these items in view of the design changes and analyses which have occurred since submittal of the PRA in the 1986-1987 PRISM PSID. Section G.4.18-2 provides an overview of design and analytical developments, and their relation to the above items. Section G.4.18-3 discusses current evaluations and future plans relevant to each item. It is the intent of the ALMR program to work closely with the Staff and its consultants to address each of the issues as the ALMR PRA evolves from the conceptual design phase to the preliminary design and final design phases.

G.4.18.2 Design Changes and Recent PRA Evaluations Relevant to the SER PRA Concerns

Amendment 12 to the PSID and some sections in this amendment describe the changes in the ALMR design and design requirements which took place since the 1986-1987 PSID was issued. Tables G.2.2-1 and G.2.2-2 list these changes.

Many of the changes shown in Tables G.2.2-1 and G.2.2-2 increase the design margins for the prevention and mitigation of accidents, and consequently are expected to result in lower risk estimates or higher confidence in these estimates. This is clearly demonstrated by the following relations between some of these design changes and various risk attributes.

a. <u>Changes directed at decreasing the risk from external events</u>. These include:

   (1) Seismic isolation of the EM pump synchronous machine and the RPS.
   (2) Tornado hardening of the portable refueling enclosure, steam generator building, and control building.
   (3) Upgrading the portable refueling enclosure to Seismic Category I, and the steam generator building and control building to Seismic Category II.

b. <u>Changes directed at reducing the frequency of core damage</u>. These include:

   (1) Use of Gas Expansion Modules (GEMs) to provide extra inherent negative reactivity feedback for loss of flow accidents
   (2) Use of a diverse reactor shutdown system
   (3) Use of control rod withdrawal limiters (rod stops)

c. Changes directed at preventing the release of radioactive material from the reactor vessel. These include:

(1) Design of the reactor vessel head with the goal of accommodating the dynamic and static loading of a hypothetical core disruptive accident (HCDA)

(2) Design of the reactor internals with the goal of accommodating a whole core meltdown and retaining it inside the reactor in a coolable and stable configuration.

d. Changes directed at preventing the release of radioactive material to the environment. These include:

(1) Use of a low leakage pressure retaining containment dome designed to retain its integrity under an HCDA, followed by a sodium fire which consumes all of the containment oxygen, while maintaining the site dose below the Protective Action Guideline (PAG) and 10CFR100 limits.

The impact of the above changes on the PRA analytical areas is shown in Table G.4.18-1. The table also shows the relation between these changes and the nine SER PRA concerns stated in the previous section.

Since the 1986-1987 PSID, a significant effort has been spent to identify and reduce the data and modeling uncertainties in the risk estimates. The effort focused initially on the front end of the PRA and resulted in the following developments:

a. A generic and LMR-specific component reliability data base. The data base was derived mainly from the NRC Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR) and the DOE Centralized Reliability Data Organization (CREDO) as of February 1990. These sources were supplemented by data from the CRBRP PRA, Seabrook PRA, Wash-1400, and IEEE Std-500.

Table G.4.18-1 - RELATION BETWEEN PRISM/ALMR CHANGES SINCE 1986-1987 PSID AND SER PRA ISSUES

Area of Design Change	PRA Item Impacted	SER PRA Issue Impacted
Reactor Power	Core Response Event Tree, Source Term	
Primary Sodium Hot/Cold Leg Temperature	Minor Effect	
No. Fuel Pins/Assembly	Minor Effect	
Cold Shutdown After Stabilization by Inherency for ATWS Events	Core Response Event Trees	1
Accommodation of ULOF/LOHS Accidents	System Event Trees	1
Accommodation of UTOP	Reactivity Insertion Initiating Events	1
Accommodation of Core Melt	Containment Response Event Trees, Source Term	4, 6
Accommodation of HCDA	Containment Response Event Trees, Source Term	4, 6
Ex-vessel Storage for Core Unloading	Recovery from Primary Coolant Leak and Minor Core Accidents	8
Seismic Isolation of EM Pump Synchronous Machine	System Event Trees	1, 2, 3
Seismic Isolation of RPS Electronics	System Event Trees	1, 2, 3
HAA & Refueling Enclosure Containment Capability	Containment Event Trees, Source Term	4, 5
Portable Refueling Enclosure Seismic and Tornado Qualification	Refueling Accidents	
Steam Generator Building Seismic and Tornado Qualification	Initiating Events	1
Steam Generator Type	Initiating Events	1
Steam Generator SWRPRS Rupture Disc Qualification	Initiating Events	1
IHTS Auxiliary Cooling System	System Event Trees	1, 3
Control Building Location	Initiating Events, Recovery Events	1, 8
Control Building Seismic and Tornado Qualification	Initiating Events, Recovery Events	1, 2, 8
Remote Shutdown & Post-Accident Monitoring Facility	Recovery from Accidents	8
Fuel Cycle Facility Reference Location	NA	NA

b. Failure Mode and Effects Analysis (FMEA) of the reactor protection system, the EM pump synchronous machine, and the plant control system.

c. A comprehensive set of initiating events derived from past PRAs, safety analysis reports and a master logic diagram which was developed to ensure completeness of the initiating events, and to identify underlying dependences and uncertainties.

The above work was documented in November 1989 in Reference G.4.18-1. Work during the current year has been directed at identifying and evaluating data and modeling uncertainties in the remaining areas of the PRA, especially those uncertainties identified in the draft SER. Initial activities evaluated the importance of uncertainties in the initiating events and core damage frequencies, and developed an accident progression diagram which identifies accident progression paths, key phenomena and uncertainties. The diagram is being used as a road map for discussion with experts to establish R&D needs and priorities, and to quantify current uncertainties.

In order to address uncertainty concerns raised in the draft SER, three complementary approaches are being used. These are:

a. Importance analysis - This includes sensitivity and parametric evaluations designed to assess the relative significance of risk contributors and uncertainties.

b. Comparative evaluations - These include comparisons with LWR and other LMR risk attributes.

c. Specific investigations to identify and quantify key uncertainties.

Results of the work completed so far lead to the following preliminary conclusions:

a. ALMR risk estimates are insensitive to very large increases (several orders of magnitude) in the frequency of the initiating events of catastrophic reactor vessel failure, station black out, steam generator tube rupture, and inadvertent control rod withdrawal. These events are dominated by more frequent events which lead to similar or larger consequences.

b. ALMR risk estimates are insensitive to very large increase in the failure probability of the synchronous machines given failure to scram. Conservative assignment of more frequent sequences to the same accident type has significantly reduced the importance of uncertainty in the synchronous machine failure probability.

c. The beta factors used to estimate common mode failure probabilities in the ALMR reactor protection system, reactivity control and shutdown system, and the primary pump coastdown system are either equal to or more conservative than the beta factors recommended in the Advanced Light Water Reactor (ALWR) guidelines (Reference G.4.18-2). The ALWR beta factors present reasonably conservative state-of-the-art values.

d. Although detailed analysis of dependent failures which might result from human error, systems interactions and support system failures is yet to be completed and continuously updated as the design evolves, qualitative engineering evaluations indicate that the ALMR should be less vulnerable to such failures than conventional reactors because of its less reliance on support systems and its reduced man machine interface.

e. The 21 external events compiled by the Advanced Reactor Severe Accident Program (Reference G.4.18-3) were used for RVACS evaluation. The RVACS elevations, dimensions and geometry were found to limit the frequency of RVACS blockages greater than 75% to less than 10^{-7} per year. The results indicate that the frequency of RVACS blockage used in the PRISM PRA might be conservative by a factor of 10.

f. Seismic fragility analysis of seismically isolated ALMR structures and major components have been performed. Early results indicate that the ALMR has significant seismic margins which ensure structural integrity and component function up to at least 1.5g peak ground acceleration.

g. An importance analysis was performed to estimate the relative risks of the ALMR fission products of noble gases, halogens (I, Br), alkaline metals (Cs, Rb), Te, alkaline earths (Sr, Ba), noble metals (Ru), and the sodium coolant. The analysis used the SMART consequence analysis computer code (Reference G.4.18-4) which calculates early doses and early health effects at different distances and for different weather conditions. The relative importance was measured by the risk from equal release fractions released at the same time and rate. The fission products were found to rank as follows in increasing levels of risk: noble gases, alkaline metals, radioactive sodium, alkaline earths, halogens, Te, and Ru. It is interesting to note that Sr and Ba (alkaline earths) produce less risk of early dose and early health effects than do Te and Ru. The latter radionuclides are more volatile in their oxide form than their elemental form.

h. The draft SER expressed a concern over the assumptions used in the PRISM PRA for the retention capability of the head access area (HAA). The ALMR design changes for accident prevention (GEMS and ultimate shutdown system) and mitigation (HCDA accommodation, core meltdown accommodation and containment) provide retention capabilities which exceed those assumed in the 1986-1987 PRA. On-going evaluations of these changes and the IFR metal fuel program are expected to confirm this conclusion and to ensure that the risk level remains low.

i. To umbrella the uncertainties resulting from lack of metal fuel data and detailed mechanistic analysis, the WASH-1400 release category PWR1 was used in SMART code calculations assuming end-of-equilibrium cycle inventory for all in-vessel radioactive material in the ALMR. The category includes maximum release fractions of radioisotopes which are volatile under the accident conditions assumed in WASH-1400. In particular, the category includes a release fraction of 0.4 for Ru which is highly volatile under molten oxide fuel/water coolant interaction. Such a release is practically impossible in the metal fuel/sodium coolant system. The results indicate that the public at risk is reduced significantly as a result of the smaller radioactivity inventory. Using more realistic release assumptions, which account for the sodium coolant thermal capacity, has reduced the risk further even without any credit taken for containment attenuation or holdup.

j. The draft SER has raised the concern that uncertainties have not been quantified or understood. Systematic procedures such as the master logic diagram of Reference G.4.18-1 and the accident progression diagram currently under development and evaluation will be consistently applied in the ALMR Program to organize and evaluate uncertainties and to track down R&D development needs and priorities to reduce these uncertainties. The IFR metal fuel program and the ALMR safety test are expected to furnish the needed information to confirm the ALMR safety.

k. The draft SER raised the concern of the availability of the operator for recovery actions. Although detailed analysis of this area of the PRA is planned for the long term, importance analysis will be performed in support of the project's needs and plans to specify allowed outages for recovery. It should be noted that the control room has been relocated within the safety fence and tornado hardened and upgraded to Seismic Category II.

l. A greater effort has been and will continue to be spent to ensure completeness of the PRA. It should be recognized, however, that with the design changes enhancing the mitigation capability of the ALMR, low probability sequences which do not bypass these mitigating provisions will not impact the risk. It is expected that external events, specially seismic will continue to dominate the risk as a result of the uncertainties of systems performance and interactions under such events.

G.4.18.3 Data and Analysis To Support Response to SER PRA Issues

The SER concerns quoted in Section G.4.18-1 can be grouped under three general areas:

a. Issues related to core damage frequency. These are Issues a, b, and c in Section G.4.18.1.

b. Issues related to core meltdown phenomena and consequences. These are Issues d, e, and f in Section G.4.18.1.

c. General PRA methodology issues. These are Issues g, h, and i in Section G.4.18.1.

This section is organized according to the above general areas.

G.4.18.3.1 Issues Related to Core Damage Frequency

These issues involve the uncertainties in the frequency of the initiating events and the data and modeling uncertainties in the system event trees and systems analysis. For convenience, these issues are discussed under three separate items: Internal Initiating Events, External Events, and Systems Analysis.

<u>Internal Initiating Events</u>

The draft SER expressed concern over the low frequency values of four initiating events: catastrophic reactor vessel failure, station blackout, steam generator tube rupture, and inadvertent control rod withdrawal. As stated earlier, a significant effort was spent in 1989 to assess the importance of uncertainties in the frequency of the initiating events, to establish a comprehensive set of properly defined initiating events, and to develop an ALMR-specific component reliability data base. Relevant summary of this work is presented below.

Importance of the uncertainty in an initiating event frequency was evaluated by calculating the factor by which the frequency must increase before the risk is doubled. The larger the risk doubling factor for a given initiating event, the more forgiving is the risk to uncertainties in this initiating event. Table G.4.18-2 shows the calculated factors. As seen in the table, the risk is most sensitive to the frequency of the large earthquake event. An increase in the frequency of this event by a factor of 2.35 doubles either the public or individual risk. In contrast, the frequency of initiating Event IE2 (potential for reactivity insertion between 18¢ and 36¢) must be increased by a factor of 4×10^4 for the risk to be doubled. It is interesting to note that all of the initiating events of concern in the draft SER belong to the group of initiating events labeled "All Others" in Table G.4.18-2. This means that the frequency of any of these events will have to increase by at least 10 orders of magnitude before the risk is doubled.

Table G.4.18-2

RISK SENSITIVITY TO UNCERTAINTY IN THE FREQUENCY OF INITIATING EVENTS

Rank	Initiating Event	Estimated Frequency (per module yr)	Allowed Frequency Increase Factor Before Risk Is Doubled
1	IE6 (Earthquake > 0.875g)	7×10^{-7}	2.35
2	IE10 (Loss of One Primary Pump)	0.16	20
3	IE11 (Loss of Two or More Primary Pumps)	0.05	60
4	IE12 (Loss of Operating-Power Power Heat Removal)	0.08	200
5	IE5 (Earthquake 0.3-0.875g)	1.9×10^{-5}	1600
6	IE2 (Potential for Reactivity Insertion 18-36¢)	10^{-4}	4×10^4
7	IE21 (RVACS Blockage)	10^{-8}	2×10^7
8	IE19 (Forced Shutdown)	5.5	4×10^7
9	IE19 (Normal Shutdown)	0.6	10^8
10	IE3 (Potential for Reactivity Insertion > 36¢)	10^{-6}	3×10^8
11	All Others		$>10^{10}$

The above conclusion indicates that the risk is insensitive to very large increases in the frequency of the initiating events of concern in the draft SER. Despite this conclusion, an investigation was made of each concern.

The investigation led to the following conclusions:

a. The PRISM PRA initiating event of catastrophic reactor vessel failure refers to a complete circumferential rupture of the vessel as a result of fatigue. The probability of this event was estimated based on fracture mechanics analysis which led to the extremely low probability of 10^{-13} per vessel year. The event was identified in a design review as a potential cause of cascaded failure of the guard vessel, which subsequently leads to the loss of coolant and loss of RVACS. Subsequent analysis indicated that extreme seismic events and leaks of both the reactor vessel and guard vessel dominate the probability of double vessel failure. Consequently the updated list of initiating events reported in Reference G.4.18-1 excludes the catastrophic vessel failure on the basis of being an insignificant risk contributor.

b. An independent assessment of the frequency of station blackout for the ALMR was conducted by EG&G using the newly developed data base for the ALMR (Reference G.4.18-1). The result is almost identical to that of the 1986-1987 PSID PRA despite the difference in modeling and data base. It should be noted that the ALMR safety systems do not depend on the availability of electric power. Consequently, significant error in the station blackout frequency estimate will result in negligible impact on the risk estimates. This is confirmed by Table G.4.18-2.

c. The 1986-1987 PRISM PRA initiating event of steam generator tube rupture refers to a beyond-design-basis composite event which involves multiple steam generator tube ruptures, and failure of the multiple protective systems designed to terminate the resulting sodium water reaction. Since a different steam generator design is now used in the ALMR, this event has been reevaluated in the updated list of initiating events in Reference G.4.18-1.

d. Reactivity insertion events, including the inadvertent withdrawal of a control rod, were reevaluated in the updated list of initiating events of Reference G.4.18-1 using the newly developed ALMR data base. Current estimates of the frequency of these events are about an order of magnitude higher than those reported in the PRISM PRA. However, the estimates are still lower than those of a typical LWR as a result of the much smaller number of control rods (six in the ALMR vs about 50 for a typical PWR). The basis for these estimates can be found in Reference G.4.18-1.

External Events

Two studies related to the risk from external events were conducted recently. One of the studies was performed as a part of the effort to develop a comprehensive list of initiating events. This study focused on the vulnerability of RVACS to external events. The other study involved seismic fragility analysis for seismically isolated ALMR structures and major components.

The RVACS is a passive shutdown heat removal system with quadruply redundant inlets and outlets. The system is continuously operating and monitored. The lack of moving parts and the use of simple geometry makes it impossible for any internal failure to degrade the heat removal capability or block the coolant passages. To assess the system vulnerability to external events, the list of external events compiled by the Advanced Reactor Severe Accident Program (Reference G.4.18-3) was used. The list is shown in Table G.4.18-3. The list was screened using a procedure similar to that suggested by the Probabilistic Safety Analysis Procedures Guide (NUREG 2815) to identify those events which are likely to be removed by siting or because of their irrelevance to RVACS failure.

Table G.4.18-3

EXTERNAL EVENTS USED FOR RVACS EVALUATION

a) Aircraft Crash
b) Avalanche
c) Hazardous Material On-Site
d) Coastal Edge Corrosion
e) Drought
f) Internal Fires
g) External Fires
h) Internal Floods
i) External Floods
j) Low Air Temperature, Snow and Ice Storms
k) Tornadoes
l) Hazardous Material Off-Site
m) Land Slide
n) Lightning
o) Meteorites
p) Sand/Dust Storms
q) Seismic Events
r) Volcanic Ash
s) T/G Missiles
t) Soil Shrink/Swell
o) Transportation Accidents

Fourteen failure modes of RVACS were defined to cover the range of blockages up to 100%. The failure modes included:

a. Partial inlet and outlet blockages
b. Combinations of partial inlet and outlet blockages
c. Blockage of inlet or outlet plenum
d. Blockage of the bottom of the reactor silo
e. Blockage of all inlets and outlets
f. Replacement of silo air by another material due to an external event

g. Replacement of silo air by sodium leaking due to a double vessel leak (the reactor vessel and guard vessel)

Each external event was evaluated against the possibility of causing any of the above failure modes. Frequencies of the external events were assigned based on the generic data in Reference G.4.18-3. The fragility of RVACS to external events, and the mean time to recover RVACS, were assigned by judgment based on the perceived severity of the event.

Table G.4.18-4 shows the frequency, mean time to recover, and cause of various blockage sizes. As seen in the table, blockages greater than 90% of RVACS are estimated to occur at a frequency of less than 10^{-9} per year. This is to be compared to the frequency of 10^{-8} which was used in the PRISM PSID PRA. Table G.4.18-2 shows that the frequency of RVACS total blockage has to increase by a factor of 2×10^{7} for the risk to be doubled. This leaves a significant margin for possible dependent failures of other heat removal capabilities involving the turbine condenser and the ACS.

A Level 1 seismic risk assessment was initiated to assess the impact of uncertainties on the probability of core damage initiated by seismic events. Fragility analysis for major ALMR components which impact the core damage frequency was also initiated. Early results of this analysis indicate that the ALMR has considerable margin for seismic events which exceed the SSE level of 0.3g. Horizontal seismic isolation of the critical safety related components, including the primary and containment systems, the reactor protection system electronics, and the synchronous coastdown machines, contribute to the overall seismic capability. These early results show that the safety related components will remain isolated up to 0.9g peak horizontal ground acceleration, at which point the isolation gap is closed. Fragility analysis indicates the components have sufficient seismic capability to withstand at least an additional 0.6g, which provides a total capability of at least 1.5g peak horizontal ground acceleration. This assessment is preliminary however, and work is continuing to address the extremely low probability region above 0.9g.

Table G.4.18-4

RVACS BLOCKAGES CAUSED BY EXTERNAL EVENTS

Range	Frequency/ Module Yr	Recovery Time	Main Cause
0 - 25%	10^{-2} - 1	<8 hours	Flying Objects
25% - 50% (Includes Some Cylinder Wall Fouling)	10^{-3} - 10^{-5}	<8 hours	Tornado Flying Objects
50% - 75% (Includes Blockage of Bottom of Collector Cylinder)	10^{-5} - 10^{-7}	<24 hours	Tornado Flying Objects
75% - 90%	$\sim 10^{-9}$	>24 hours	Hail
>90%	$<10^{-9}$	>24 hours	Sand/Dust Storm Severe Seismic Event Double Vessel Leak

Systems Analysis

The SER has questioned the adequacy of data and modeling dealing with dependent failures. The SER has also expressed concern over the uncertainty in the availability estimates of the safety systems. This section discusses these concerns.

The 1986-1987 PRISM PSID PRA incorporated three types of dependences in the system event trees (Reference G.4.18-6, Page A4-17). These are:

a. Dependence on the initiating event
b. Dependence between system responses
c. Dependencies between subsystems of a system.

During 1989, analysis of the question of dependent failures focused on developing a master logic diagram to identify a comprehensive list of initiating events which properly accounts for dependence on external events and interfacing systems. This analysis is discussed in detail in Reference G.4.18-1. Plans are in place for upgrading the remainder of the PRA areas to ensure completeness and proper account of dependences. This effort started in 1990 by performing importance analysis of accident sequences and evaluation of the common mode failure estimates used in the 1986-1987 PRISM PSID PRA. The results of this work are discussed below.

Similar to the importance analysis of the initiating events discussed earlier, importance of the uncertainty in the frequency of an accident type was evaluated by calculating the risk doubling factor. Table G.4.18-5 shows the calculated factors. As seen in the table, the risk is most sensitive to the frequency of the severe combined UTOP/ULOF event. The frequency of this accident type comes almost totally from a sequence initiated by the large earthquake initiating event. Next in importance is the severe ULOF accident type. Two types of sequences contribute to this accident in the PRA model; one sequence involves EM pump trip with failure to scram and failure to coastdown, and the other includes EM pump trip with successful coastdown but with the control rods stuck so that no negative reactivity due control rod expansion can be taken credit for. The frequency of the severe ULOF is contributed almost totally by the second sequence, with the first sequence contributing on the order of 10^{-6} of the frequency. This means that the unavailability of the synchronous machines has to increase by a factor of at least 10^{7} (the doubling factor of 13 from Table G.4.18-5 times 10^{6}.) The third important accident type, combined severe UTOP/ULOF/LOSHR, is also initiated by the severe earthquake. It will take at least a factor of 1000 increase in the frequency of the remaining accident types to double the risk.

To assess the uncertainty in the common mode failure contribution to system unavailability, a comparison was made between the beta factors used in the 1986-1987 PRISM PSID PRA and those recommended by EPRI for the ALWR. Table G.4.18-6 shows the ALWR beta factors. The factors apparently were derived from the Multiple Greek Letter formulation because they reflect the effect of the extent of redundancy. The beta factors used in the 1986-1987

PRISM PSID PRA and the corresponding ALWR beta factors are shown in Table G.4.18-7. As seen in the table, the beta factors used in the 1986-1987 PRISM PRA are either the same or more conservative than those recommended for the ALWR.

Table G.4.18-5

RISK SENSITIVITY TO UNCERTAINTY IN THE FREQUENCY OF ACCIDENT TYPES

Rank	Accident Type*	Estimated Frequency (per module yr)	Allowed Frequency Increase Factor Before Risk Is Doubled
1	G4 (Combined Severe UTOP and ULOF)	2.1×10^{-8}	2.5
2	F3 (Severe ULOF)	6.6×10^{-9}	13
3	G4S (Combined G4 and LOSHR)	9.6×10^{-10}	49
4	H3 (Severe ULOHS)	7×10^{-11}	1,538
5	G3 (Combined Severe UTOP and ULOF	6×10^{11}	2×10^{4}
6	S5 (LOSHR with Degraded Core Flow)	3×10^{-11}	3×10^{3}
7	S3 (LOSHR with Normal Core Flow)	5×10^{-12}	2×10^{4}
8	P3 (Severe UTOP)	$<10^{-12}$	7×10^{8}
9	F1 (Design Basis ULOF)	$<10^{-12}$	7×10^{6}
10	All others	$<10^{-12}$	10^{9}

* See Table G.4.18-9 for definitions of accident types

Table G.4.18-6

RECOMMENDED ALWR COMMON CAUSE FACTOR

Number of Failures	Fail to Start/Run: Start	Fail to Start/Run: Run	Fail to Open/Close (or Fail to Operate): Open	Fail to Open/Close (or Fail to Operate): Close
2 of 2	1.5×10^{-1}	4×10^{-2}	1×10^{-1}	1×10^{-1}
2 of 3	5×10^{-2}	2×10^{-2}	5×10^{-2}	5×10^{-2}
3 of 3	1.5×10^{-2}	2×10^{-3}	1×10^{-2}	1×10^{-2}
2 of 4	4×10^{-2}	2×10^{-2}	2×10^{-2}	2×10^{-2}
3 of 4	5×10^{-3}	1×10^{-3}	1×10^{-3}	1×10^{-3}
4 of 4	2×10^{-3}	2×10^{-4}	2×10^{-4}	2×10^{-4}
4 of 5 or Higher	1×10^{-3}	5×10^{-4}	5×10^{-4}	5×10^{-4}
5 of 5 or Higher	5×10^{-4}	1×10^{-4}	1×10^{-4}	1×10^{-4}

Table G.4.18-7

PRISM/ALWR BETA FACTORS

PRISM System	Failure Criterion	Beta Factor Used in PRA	Recommended ALWR Beta Factor
RPS			
Sensors	3 of 4	10^{-2}	10^{-3}
Sensors Monitoring	3 of 4	0.5	5×10^{-3}
Electronics	3 of 4	10^{-3}	10^{-3}
Electronics Monitoring	3 of 4	0.5	5×10^{-3}
Setpoints	3 of 4	10^{-3}	10^{-3}
Scram Breakers	12 of 12	10^{-2}	$<5 \times 10^{-4}$
RSS			
In-vessel Initiators	6 of 6	5×10^{-2}	$<5 \times 10^{-4}$
Ex-vessel Initiators	6 of 6	1×10^{-3}	$<5 \times 10^{-4}$
Primary Flow Coastdown	3 of 4	10^{-3}	10^{-3}

The above comparison indicates the validity of the beta values used in the 1986-1987 PRIM PSID. The unavailability estimates remain relatively small, however when compared to systems performing similar safety functions in conventional LWRs. This is a result of some fundamental differences between the ALMR and conventional LWR systems operation and configuration, which reduce the significance of dependent failures resulting from common causes, human error and dependence on the supporting systems. These are:

a. Safety systems needed for reactor shutdown and decay heat removal in the ALMR are either continuously operating and monitored (RVACS) or almost continuously operating and monitored (RPS and control rods). In contrast to conventional LWRs, where the majority of safety systems are in an inactive standby mode, the ALMR safety systems are expected to show gradual rather than instantaneous system degradation, and higher availability when needed.

b. Monitoring of safety equipment, failure isolation , diagnostics of abnormal conditions and reactor protection are done automatically in the ALMR without human intervention, thus reducing the man machine interface significantly.

c. No support systems are needed for the operation of RVACS, for reactor trip by the latch mechanism, for the ultimate shutdown mechanism, for cooling the EM pumps, or for the operation of the synchronous machines. Electric power for the diverse control rod drive-in mechanism is very small, and is needed for only two minutes.

To illustrate the implications of the above differences, a comparison was made between key reliability attributes in the control rods and trip breakers of the ALMR and those of the WASH-1400 PWR. The comparison is shown in Table G.4.18-8. As seen in the table, the high level of redundancy, diversity and frequent testing of the ALMR systems increase the time between failures, and increase the system availability.

Table G.4.18-8

ALMR/PWR REACTIVITY CONTROL AND SHUTDOWN SYSTEM COMPARISON

	PWR	ALMR
Control Rods		
Number	48	6
Redundancy Level	47/48	1/6
Diversity	None	Magnetic latch + Motor Drive
MTTF		
N	48	6
(Σ 1/k)	Σ 1/k	Σ 1/k
k	k=46	k=1
Normalized MTTF	1	~50-5,000
Trip Breakers		
Number	2	4
Redundancy Level	1/2	2/4
Test Period (hrs)	720	4
Unavailability	$1/3\ (720)^2$	$<<1/3\ (4)^2$
Normalized Unavailability	1	$<<10^{-4}$

The above observations are in no way considered a substitute for detailed systems analysis covering all questions of dependencies from human interactions, systems interactions, and support systems interactions. Plans are in place to apply state of the art methodology and latest data, as the design evolves and system interfaces are well defined. Importance analysis will continue to be used to focus the effort on more urgent issues which might have a significant impact on the risk.

A concern raised in the draft SER was the low probability of 10^{-6} used in some system event tree sequences for failure to provide enough inherent reactivity feedback when needed. It should be noted that the 1986-1987

PRISM PRA risk model accounts for the failure to provide adequate inherent reactivity feedback in two parts of the model: the system event trees and the core response event trees. The system event trees account for possible structural damages or misalignment which could prevent structural components from expanding to provide the expected reactivity feedback under normal conditions. The core event trees, on the other hand, accommodate the dependence of the effectiveness of inherent reactivity feedback on the accident type and severity. Table G.4.18-9 shows the conditional probability of failing to provide enough reactivity feedback to prevent fuel/clad eutectic formation and sodium boiling which was used for different accident types and severity. Reference G.4.18-6 discusses the basis for these assignments and shows that they are conservative. The conditional probabilities are being re-evaluated to include the addition of GEMs, which enhances inherent reactivity feedback under ULOF conditions.

G.4.18.3.2 Issues Related to Core Meltdown Phenomena and Consequences

These issues involve: (1) the use of radioactivity release scenarios and fractions based on an oxide core, (2) the structural integrity of the head access area (HAA) under accident conditions, and (3) the lack of mechanistic analysis necessary for estimating accident energetics, evaluating structural capabilities, and estimating the time and form of radioactivity release. These issues are discussed below.

Metal vs Oxide Core

At the time the PRISM Project changed the reference fuel from a mixed uranium-plutonium oxide to the current metal fuel, the 1986-1987 PRISM PRA containment event trees and radioactivity release analysis were well underway and it was too late to redo the event trees, release analysis and consequence analysis. This was further delayed by major design changes which have significant impact on the source term: the addition of the GEMs which provide more inherent negative reactivity, the added capability to the reactor vessel head to accommodate an HCDA, the added capability in the reactor vessel to accommodate a whole core meltdown, and the addition of a moderate pressure, leaktight, safety class containment. Work is currently

Table G.4.18-9

CONDITIONAL PROBABILITY OF EUTECTIC FORMATION AND SODIUM BOILING USED IN THE PRISM PSID PRA

Accident		Conditional Probability: Eutectic Formation	Conditional Probability: Na Boiling Given Eutectic
F_1	(Unprotected Flow Coastdown)	10^{-2}	0
F_3	(Unprotected Loss of Flow With Failure of Flow Coastdown or Degraded Inherent Reactivity Feedback)	1.0	0.5
P_1	(Unprotected Reactivity Insertion of 0.07$ to 0.18$)	10^{-2}	0
P_2	(Unprotected Reactivity Insertion of 0.18$ to 0.36$)	5×10^{-2}	10^{-3}
P_3	(Unprotected Reactivity Insertion of >0.36$)	0.5	10^{-2}
P_4	(Unprotected Reactivity Insertion of >0.36$) With Degraded Inherent Reactivity Feedback)	0.99	10^{-2}
H_2	(Unprotected Loss of Heat Sink at Nominal Power)	10^{-2}	0
H_3	(Unprotected Loss of Heat Sink at Elevated Power)	1.0	0.5
G_3	(Combined P_2 and F_3 With P_3 and F_1)	0.7	0.1
G_4	(Combined P_3 and F_3 With P_4 and F_1)	1.0	0.9

underway to redo the core and containment event trees, redefine the release scenarios and recalculate the consequences. The work is being guided by an accident progression diagram which displays the progression of the full spectrum of possible accidents, key phenomena and uncertainties. The accident progression diagram is being used as a vehicle for discussion with experts and for tracking down the relevant R&D activities under the IFR metal fuel program.

Early results from the above work indicate that the impact on the risk estimates due to the use of the metal core may be insignificant despite the significant differences between oxide and metal fuels in terms of the release scenarios, timing, and mix of the radioactive material released. The key differences and their significance are summarized below.

a. The eutectic point of the metal fuel (725°C) is below the sodium boiling point of 883°C. Release of fission products which are volatile at the eutectic temperature (halogens and alkali metals) will most likely occur under sodium, which provides significant retention capability of the halogens. For oxide fuel, the lack of a eutectic and the high melting point (2770°C) result in a better chance of holding up volatile fission products within the cladding until sodium boil-off occurs, and the fuel is uncovered.

   The above differences are particularly significant in the protected loss of heat sink accident which was modeled as follows in the 1986-1987 PRISM PRA (Reference G.4.18-6):

   > The primary sodium heats adiabatically until boiling starts. At this point, the reactor vessel head seal is assumed to rupture, letting sodium vapor to escape to the HAA. Boiling continues until the core begins to uncover. At this point the whole core is assumed to melt down, and melt through the reactor vessel and the guard vessel. All the noble gases and volatile fission products and 0.14% of the nonvolatiles (in aerosol form) are assumed to be released to the remaining sodium and mixed uniformly with it. The melt-through of

the vessels, and the mixing of the fission products with the sodium, allow the fission products to bypass the HAA and escape through the RVACS ducts. The reactor cavity is assumed to be unlined. Sodium-concrete reaction expedites the escape of the contaminated sodium through the RVACS ducts. This scenario is believed to be conservative.

For the metal core, eutectic formation is expected before sodium boiling in the above accident. This allows early release of the fission gas and volatiles into the primary sodium. Some of the fission products will escape to the upper containment before core uncovery and vessel melt-through. This allows only a fraction to bypass the containment and escape through the RVACS ducts. Thus the hazard of early release from the metal fuel core to the sodium is reduced by containment attenuation through fallout and plateout. In effect, the metal fuel core will result in earlier release, but the total release fraction will be smaller than that of the oxide core.

b. Elemental strontium and barium are more volatile than their oxides, but elemental tellurium and ruthenium are less volatile than their oxides. In the metal fuel the fission products are expected to be in elemental form. Depending on the stoichiometry in the oxide fuel, oxides of the fission products might exist. Thus the hazard of releasing more Sr and Ba from the metal fuel is reduced by the potential for releasing less Te and Ru than the oxide fuel.

c. The high melting point and enthalpy of oxide fuel could lead to energetic molten fuel coolant interaction which results in aerosol forms of nonvolatiles and loads the retaining structures. On the other hand, metal fuel interacts exothermally with oxygen where oxide aerosols are released.

Differences between the metal and oxide fuels in the equation of state, melting and boiling points, and fission products chemical forms could lead to significantly different reactivity feedback and fuel removal mechanisms. These in turn could lead to significantly different accident scenarios, energetics and radiation source term. These differences present a major area of uncertainty in HCDA scenarios. The design changes in the ALMR to prevent these accidents and mitigate their consequences are expected to reduce the significance of this uncertainty.

Long term plans are in place at ANL to develop a version of the SAS4A code applicable to metal fuel, and to provide necessary data for estimating the source term. Some early work has been performed in the ALMR PRA to scope the problems involved. The work included parametric analysis using the SMART consequence analysis computer code (Reference G.4.18-4) which calculates early doses and early health effects at different distances and for different weather conditions. Two studies were completed so far and are discussed below.

a. To assess the significance of the higher volatility of elemental Sr and Ba in the ALMR, an importance analysis was performed to estimate the relative risks of the noble gases, halogens (I, Br), alkaline metals (Cs, Rb), Te, alkaline earths (Sr, Ba), noble metals (Ru) and the sodium coolant. The relative importance was measured by the risk from equal release fractions released at the same time and rate and under the same weather conditions. Except for the noble gases, all releases were assumed to start five hours after neutronic shutdown and to continue for 80 hours. For the noble gases, the release was assumed to occur immediately after shutdown, and to continue for one hour. For each fission product, all of the end of equilibrium cycle inventory inside the reactor vessel (including irradiated stored fuel) was assumed to be released over the release duration. SMART calculations were performed for the seven weather types in the code. The weather types range in degree of stability from A (extremely unstable) to G (extremely stable).

Table G.4.18-10 shows the results of the above analysis. The first column in the table contains the distances along the wind direction at which early dose and health effects were calculated. (SMART calculates only doses and health effects along the wind direction.) The second column contains the weather type. Only those weather types which produce risk are shown in the table. Other weather types produced zero risk. The third through the eighth columns contain the probability of early fatality of an individual stationed along the radiation cloud for each of the fission products studied. From the table it can be seen that the fission products rank as follows in increasing levels of risk: noble gases, alkaline metals, radioactive sodium, alkaline earths, halogens, Te, and noble metals. It is interesting to note that Sr and Ba (alkaline earths) produce less risk of early dose and early health effects than Te and Ru. It is also interesting to note that Ru ant Te are more volatile in oxide fuel than in metal fuel.

b. To umbrella the uncertainties resulting from lack of metal fuel data and detailed mechanistic analysis, three source terms were analyzed using the SMART code:

(1) The WASH-1400 release category PWR1 with the (3412 MWt) PWR radioisotope inventory of the SMART code library.

(2) Release category PWR1 with the ALMR end of equilibrium cycle inventory.

(3) An ALMR release category which is more consistent with the release scenario from a metal core under a hypothetical, protected indefinite loss of all decay heat removal capability (LODHR) event.

Table G.4.18-10

RISK FROM RELEASE OF 100% OF DIFFERENT RADIOISOTOPES IN THE ALMR

Distance (mi)	Weather Type	Radioisotopes/Risk						
		Noble Gases	Na	Cs	I	Sr-Ba	Te	Ru
0.25	F (Mod. Stable)	0	1.0	1.0	1.0	1.0	1.0	1.0
0.5	F	0	1.0	1.0	1.0	1.0	1.0	1.0
1.0	F	0	0	0	1.0	1.0	1.0	1.0
2.0	F	0	0	0	0	0	0.035	1.0
4.0	F	0	0	0	0	0	0	0.246
5.0	F	0	0	0	0	0	0	0.117
0.25	G (Ext. Stable)	0.57	1.0	1.0	1.0	1.0	1.0	1.0
0.5	G	0	1.0	1.0	1.0	1.0	1.0	1.0
1.0	G	0	0.09	0.041	1.0	1.0	1.0	1.0
2.0	G	0	0	0	0.039	0.04	1.0	1.0
4.0	G	0	0	0	0	0	0	1.0
5.0	G	0	0	0	0	0	0	0.248
0.25	D (Neutral)	0	0.008	0	1.0	1.0	1.0	1.0
0.25	D	0	0	0	0.021	0.01	0.065	1.0
1.0	D	0	0	0	0	0	0	1.0
2.0	D	0	0	0	0	0	0	0.122

Release category PWR1 is a single puff release starting 2.5 hours after neutronic shutdown for a duration of 0.5 hour. The category includes maximum release fractions of radioisotopes which are volatile under the accident conditions assumed in WASH-1400. In particular, the category includes a release fraction of 0.4 for Ru which is highly volatile under molten oxide fuel/ water coolant interaction. Such a release is practically impossible in the metal fuel/ sodium coolant system.

The ALMR release category for the hypothetical LODHR event includes two puffs, one from 20 to 27.5 hours and the other from 27.5 to 100 hours. The time of release at 20 hours presents a conservative estimate of the time to eutectic temperature following an adiabatic heatup of the primary sodium. It was conservatively assumed that at this temperature the fuel will melt and release all of the fission products into the primary sodium. Continued adiabatic heatup and vaporization of the primary sodium was estimated to lead to core uncovery after 75 to 80 hours. Using Castleman's release fractions from sodium (Reference G.4.18-7), it was conservatively assumed that all of the alkali metals will be released in one tenth of this time (7.5 hours). All of the noble gases and primary sodium were assumed to be released during this time. Halogens, Te, Sr, and Ba were assumed to be released in accordance with Castleman's release fractions. Other radionuclides for which no release fractions exist but which are known to be less volatile, were assumed to be released over the 7.5 hours of the first puff at a release fraction of 0.1%. This led to the first puff shown in Table G.4.18-11. The second puff was modeled to start at 27.5 hours and continue until sodium depletion. This was conservatively assumed to occur at 100 hours and to include release of the remaining inventory of halogens, Te, 10% of Sr and Ba, and 1% of the remaining radionuclides. The resulting source term is shown in Table 4.18-11. It should be noted that no containment attenuation or delay were assumed for this source term.

Table G.4.18-11

RELEASE CATEGORIES FOR BOUNDING CALCULATIONS

Case No.*	Reference	Re. Time (hrs)	Duration of Release (hrs)	Radionuclide Group Fraction Released								
				NG	I	Cs	Te	Sr-Ra	Ru	La	Ac	Na
1,2	PRW1	2.5	0.5	0.9	0.706	0.4	0.4	0.05	0.4	0.003	0.003	0.05
3	LODHR											
	First Puff	20	7.5	1.0	0.05	1.0	0.02	0.01	0.001	0.001	0.001	1.0
	Second Puff	27.5	72.5	0.	0.95	0.	0.98	0.1	0.01	0.01	0.01	0
	Total			1.	1.	1.	1.	0.11	0.011	0.011	0.011	1.0

* Case 1 - 3412 MWt PWR with PWR1 release fractions
Case 2 - 470 MWt ALMR with PWR1 release fractions
Case 3 - 470 MWt ALMR with hypothetical LODHR event

The SMART code results for the above cases are shown in Figures G.4.18-1, G.4.18-2, and G.4.18-3 for weather types F, D, and A, respectively. The figures show the probability of early fatality for an individual stationed along the radiation path. The difference in consequences between the PWR1 release for the ALMR and PWR is due to the reduced radioactivity inventory in the ALMR. It is clear that this reduction results in reducing the public at risk of radiation exposure. Using the more realistic release time assumptions of LODHR reduces the public risk even further. It should be recalled that the source term of LODHR does not take credit for any containment attenuation.

The above results indicate that the small radioactivity inventory, the fission retention capability of sodium, and its thermal capacity have a significant impact on reducing the public at risk of radiation exposure.

HAA Structural Integrity

The draft SER expresses concern over the assumptions used in the 1986-1987 PRISM PRA for the retention capability of the HAA. The ALMR design changes for accident prevention, mitigation, and containment provide retention capabilities which exceed those assumed in the 1986-1987 PRA. On-going evaluations of these changes and the IFR metal fuel program are expected to confirm this conclusion and to ensure that the risk level remains low.

Mechanistic Analysis

This issue will be addressed in future work on the ALMR PRA.

G.4.18.3.3 General PRA Methodology

This section covers the remaining three concerns of the SER; quantification and understanding the uncertainties, operator's role during recovery from accidents, and completeness of the PRA.

Uncertainty

The draft SER raised the concern that uncertainties have not been quantified or understood. Systematic procedures such as the master logic diagram of Reference G.4.18-1, and the accident progression diagram currently under development and evaluation, will be consistently applied in the ALMR Program to organize and evaluate uncertainties, and to track down R&D development needs and priorities to reduce these uncertainties. The IFR metal fuel program and the ALMR safety test are expected to furnish the needed information to confirm the ALMR safety.

Operator's Role

The draft SER raised the concern of the availability of the operator for recovery actions. Although detailed analysis of this area of the PRA is planned for the long term, importance analysis will be performed in support of the project's needs and plans to specify allowed outages for recovery. It should be noted that the control room has been relocated within the safety fence, and tornado hardened and upgraded to seismic category II. In addition, the HVAC system has been upgraded.

Completeness

A greater effort has been and will continue to be spent to ensure completeness of the PRA as the ALMR design evolves from the conceptual phase to the preliminary design and final design phases. It should be recognized, however, that with the design changes enhancing the mitigation capability of the ALMR, low probability sequences which do not bypass these mitigating provisions will be dominated by more frequent sequences which may result in equal or larger consequences. It is expected that external events, especially seismic will continue to dominate the risk as a result of the uncertainties of systems performance and interactions under such events.

G.4.18.4 References

G.4.18-1 K. A. El-Sheikh, "Probabilistic Risk Assessment of the Advanced Liquid Metal Reactor," GEFR-00873, Nov. 1989.

G.4.18-2 "Advanced Light Water Reactor Requirement Document, Vol II, Utility Requirements for Evaluating Plants, " EPRI, February 1990.

G.4.18-3 D. G. Harrison, "Advanced Reactor Severe Accident Program - Combustion Engineering External Events Integration", February 1989.

G.4.18-4 I. K. Madni, et al., "A Simplified Model for Calculating Early Offsite Consequences from Nuclear Reactor Accidents", BNL-NUREG 52153, July 1988.

G.4.18-5 "Reactor Safety Study, - An Assessment of Accidents Risks in U.S. Commercial Nuclear Power Plants," WASH-1400, US NRC, Oct. 1975.

G.4.18-6 "PRISM Preliminary Safety Information Document", App. A, GEFR-00793, November 1986.

G.4.18-7 A. W. Castleman, Jr., "LMFBR Safety 1. Fission - Product Behavior in Sodium", Nuclear Safety, Vol. II, No. 5, September - October 1970.

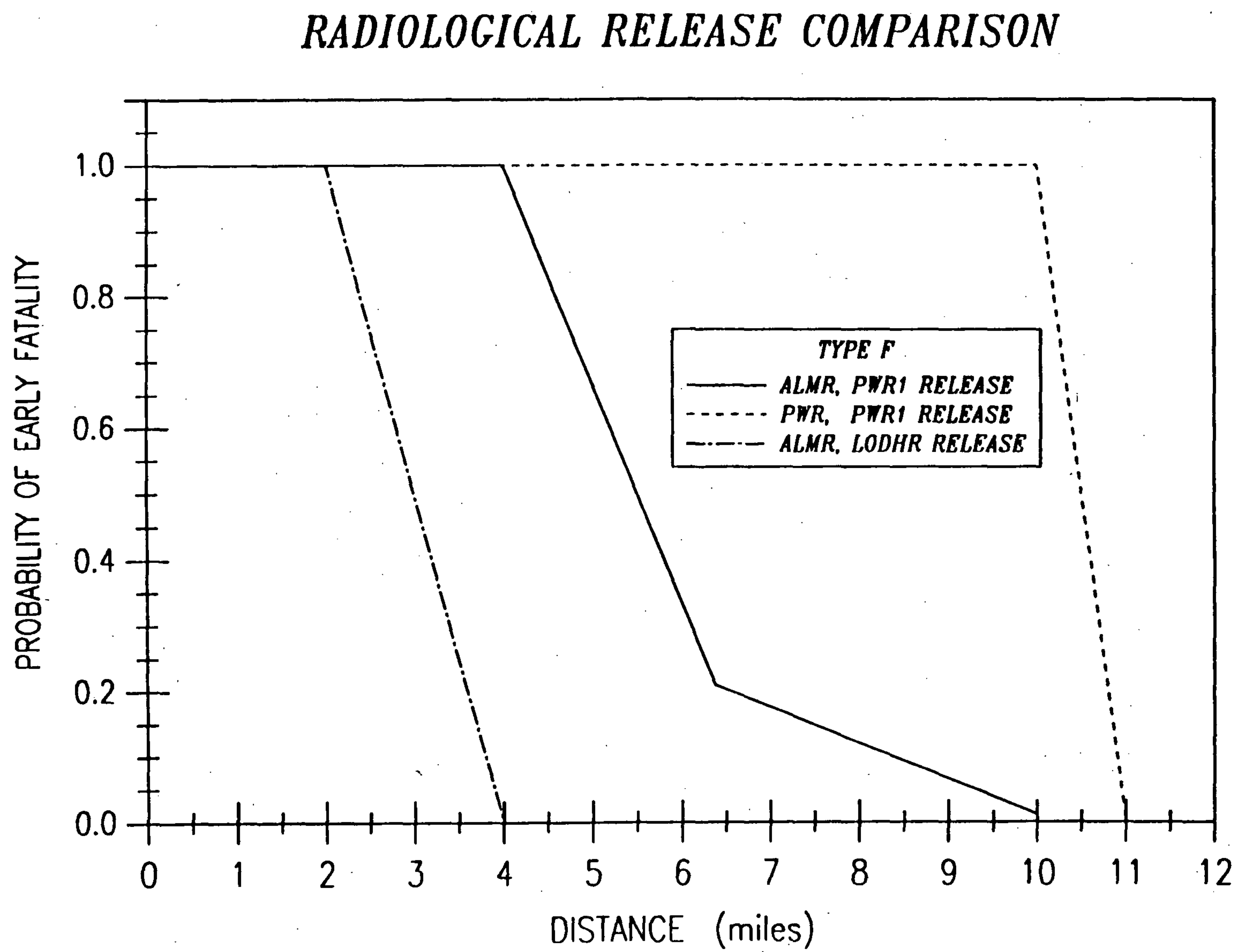


Figure G.4.18-1 SMART CODE RESULTS - MODERATELY STABLE WEATHER (TYPE F)

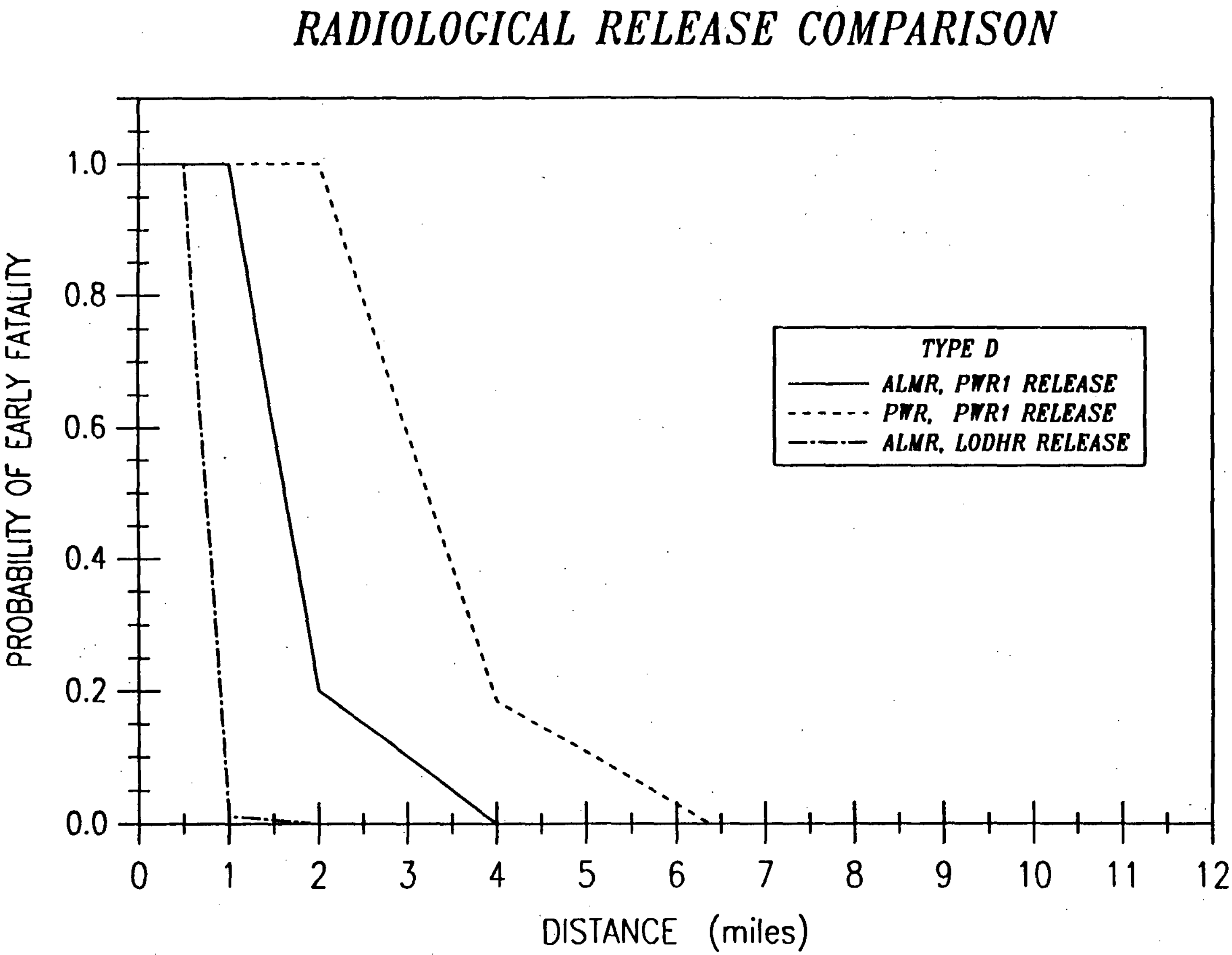


Figure G.4.18-2 SMART CODE RESULTS - NEUTRAL WEATHER (TYPE D)

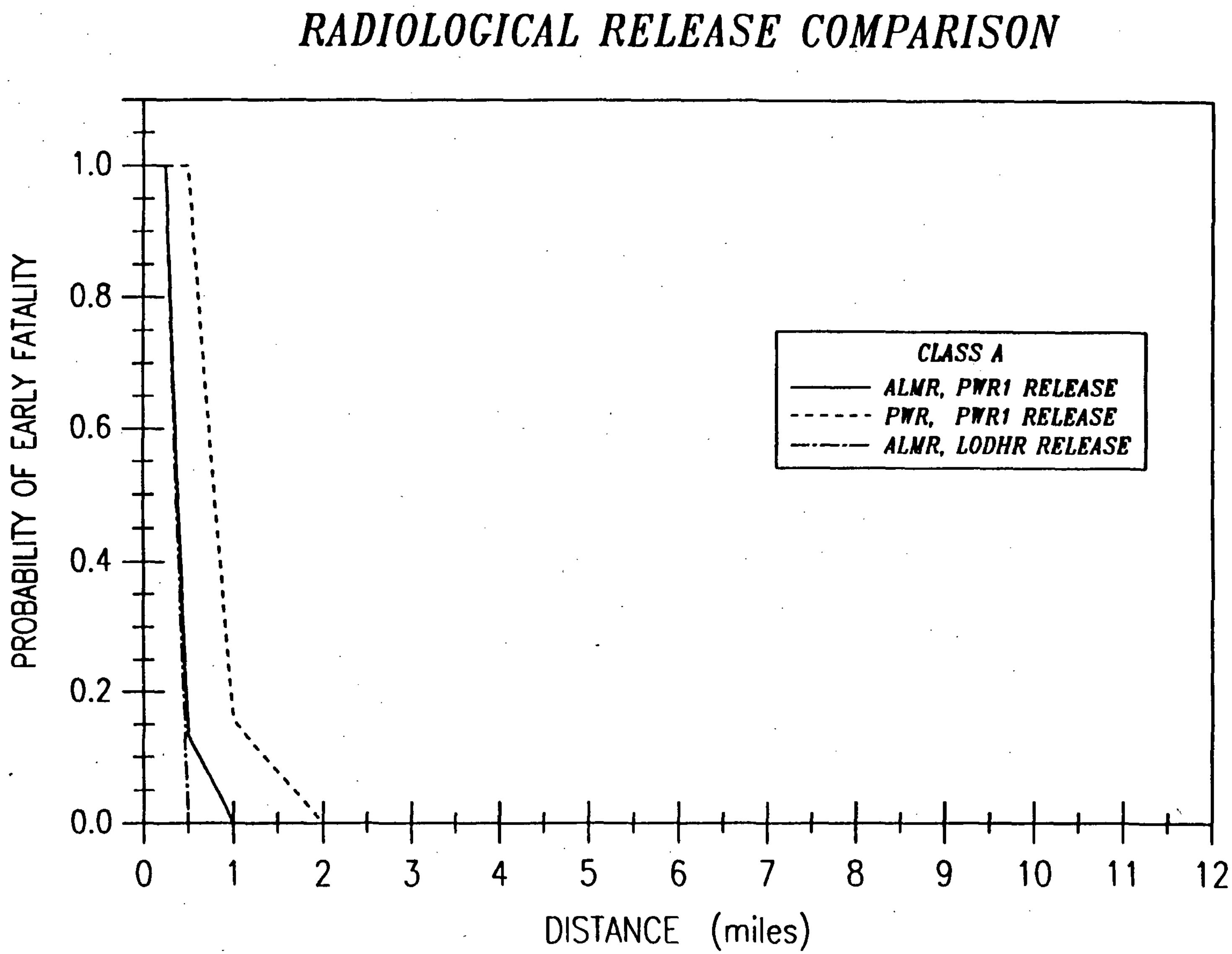


Figure G.4.18-3 SMART CODE RESULTS - EXTREMELY UNSTABLE WEATHER (TYPE A)

G.4.19 Mitigation of Severe Core Accidents

This section discusses design changes which are under consideration for enhancing the ability of the reactor to contain the consequences of a HCDA and/or a core melt accident within the primary system boundary. Preliminary analyses indicate that it may be feasible to achieve this goal, even though it is not a design or licensing requirement. However, considerable additional work needs to be completed before feasibility can be assured. Included in this additional work must be fuel testing under extreme conditions, as currently planned in Phase III of the IFR Program at ANL (see Section G.4.6.3.5).

The following discussion in this Section G.4.19 should therefore be kept in perspective: containment of the consequences of a HCDA and/or core melt accident within the primary system boundary is not a design or licensing requirement; however, attainment of this capability appears feasible; additional work is planned to determine if it actually is feasible; design changes to provide this capability may or may not be incorporated, depending on the outcome of this work compared to other feasible ways to achieve the desired degree of safety.

G.4.19.1 Introduction

The PRISM-based ALMR provides both prevention and mitigation of severe core accidents. Prevention is provided by the highly redundant reactor protection system, the strong negative reactivity feedback with rising temperatures, and the passive heat removal system. With these features, the ALMR can withstand EC-III events including the standard ATWS events and the bounding events identified by the Staff without gross fuel failure and with comfortable margins to the ASME Code design limits. Mitigation is provided by the capability of the reactor primary system boundary to contain an HCDA and a core melt. In addition, a containment dome to enclose the area above the reactor head has been added to contain any releases from the reactor head.

The staff has not identified specific EC-IV events. However, the positive sodium void reactivity coefficient has been identified by the staff and ACRS as a concern, even if it is an EC-IV event, and the issue of containment has been raised in the draft SER. These concerns are addressed in various sections of Appendix G of the PSID. Specifically, Section G.4.16, Safety Analysis, discusses the design capability to withstand EC-III bounding events without core melt or significant reactivity addition; Section G.4.6, Sodium Void, discusses the low probability and extent of possible sodium voiding; Section G.4.19, the present section, describes the capability of the primary coolant boundary to contain severe core accidents that may result from EC-IV events; and, Section G.4.1, Containment, describes the design capability to contain any release from the primary pressure boundary, should it breach.

The evaluations and conclusions regarding the primary system boundary capability to contain energetic HCDAs and slow core melt accidents are summarized in Section G.4.19.2. The HCDA capability analysis is described in Section G.4.19.3 and the core melt analysis is described in Section G.4.19.4.

G.4.19.2 Summary and Conclusions

HCDA and core melt retention capability assessments have been performed for the primary system boundary (reactor vessel, closure, IHX primary side). Analysis results show that it is feasible for the primary system boundary to contain loads and pressures from HCDAs with up to 500 MJ adiabatic work potential while meeting the ASME Code stress limits. It also appears feasible to retain a whole core fuel melt within the reactor lower internal structure and thus away from the reactor vessel while maintaining the melt support structures and the primary system boundary within the ASME Code stress and creep damage limits.

Two changes have been made to the 1986-1987 PRISM design to enhance HCDA and core melt retention capability: 1) a shear ring has been added between the fixed closure and the rotatable plug to retain the plug under impact from the sodium slug accelerated by an HCDA vapor bubble, and 2) a redundant structure maintained below the ASME Code temperature limit is placed underneath the inlet plenum to retain the core melt in case the melt leaks through any openings that may develop in the inlet plenum lower plate.

G.4.19.2.1 HCDA Evaluation Summary

<u>Magnitude of HCDA</u>

Metal core HCDAs are anticipated to have low energies because of the low melting temperature, high mobility and reduced tendency of the metal fuel to form blockages relative to oxide fuel. Based on scoping analyses, a limit of 40 MJ was stated for the total work energy of the PRISM metal core in PSID Amendment 6. However, the expected fuel behavior and low work energy remain to be demonstrated. A demonstration is planned as part of Phase III (1991-1995) of the Integral Fast Reactor (IFR) Program (Reference G.4.19-1) at the Argonne National Laboratory (ANL). The program will include (1) development of analytical models of metal fuel response to severe accident conditions, (2) ex-reactor experiments on fuel dispersal in a transient overpower event including fuel/fission product retention in sodium, (3) multi-pin transient tests in the TREAT reactor, and (4) validation of the analytical models using the ex-reactor data and the results from transient tests in TREAT, EBR-II and FFTF.

<u>Structural Evaluation</u>

HCDA loading characteristics are not yet available for the ALMR metal core. Therefore the ALMR capability evaluation was based on the HCDAs that were defined for the FFTF oxide core for a range of assumed reactivity insertion rates (50 to 175 $/sec). The work energies for these HCDAs ranged from 14 to 1100 MJ as measured in terms of the work performed by the vapor bubble generated during the HCDA if allowed to expand adiabatically to one atmosphere.

Analyses were performed for a range of pressures enveloping the available and extrapolated expansion (P-V) curves for the FFTF HCDA vapor bubbles to investigate: (1) the reactor vessel and upper internal structure (UIS) responses to the initial core pressure, (2) the closure response to the sodium slug impact, (3) the primary system boundary response to pressures following the slug impact, and (4) the primary system boundary response to the decay heat loads. The closure and UIS responses were calculated in static and dynamic inelastic finite element analyses. The responses of the remaining components were calculated using handbook formulas.

According to the analysis results, for the range of FFTF HCDAs investigated: (1) the core barrel/extension would shield the reactor vessel from the core pressures, and the UIS bottom skirt would collapse rather than transmit the high initial pressures to the closure; (2) the stresses and strains from the slug impact would be within the ASME Code stress limits and the material ductility, respectively; (3) the closure/rotatable plug relative displacements from the slug impact would be within the risers and canopy seal design capability; (4) the post-impact pressures would be within the capability of the primary system boundary components except that the canopy seal may need strengthening if the bubble is not sufficiently quenched during expansion through sodium; (5) the cover gas pressures due to the decay heat from the fission gas and the fission products would be within the ASME Code limits.

<u>Primary System Boundary Capability</u>

Based on conservative interpretation of the analysis results, the primary system boundary is concluded to have a capability of containing the FFTF 125 $/sec reactivity insertion HCDA with adiabatic work potential of ~500 MJ. The confidence in this capability is based on two factors: (1) P-V curves were used for the FFTF 125 $/sec HCDA as opposed to the larger HCDAs for which the impact loads were based on approximations. (2) The structures were predicted to remain essentially elastic under the 125 $/sec HCDA loads as opposed to relatively large inelastic strains calculated for the larger HCDAs.

In summary, the ALMR primary system boundary will contain HCDAs with work potential up to 500 MJ without a structural failure, disengagement of the rotatable plug, or sodium expulsion. Seals, including the canopy seal over the closure/plug interface, will be maintained under the slug impact. The canopy seal will also hold the residual pressures following the slug impact if the HCDA vapor bubble is quenched as expected during its expansion through sodium. Otherwise, the canopy wall will be thickened or the seal will be redesigned.

The 500 MJ work energy representing adiabatic expansion to one atmosphere is a convenient way of characterizing an HCDA. However, the HCDA damage potential also depends on the initial vapor pressure, vapor inventory, and any fuel-sodium interactions during the bubble expansion which may be different for the metal core compared to the oxide core. While the HCDA capability estimate is based on conservative analyses and large design margins, considerable work remains to be done in defining the metal core HCDAs including estimation of the post-HCDA pressures. This is planned as part of Phase III of the IFR Program (Reference G.4.19-1).

G.4.19.2.2 Core Melt Evaluation Summary

Characteristics of Core Melt

Current understanding of in-vessel retention of core melt is based on preliminary scoping analysis and experiments. Preliminary experiments at ANL have investigated the fragmentation characteristics of metal fuel. The results indicate that a very porous debris will form that should be coolable by natural convection of sodium without producing core melt. However, this needs to be demonstrated with additional analyses and experiments. The demonstration is planned to be part of Phase III (1991-1995) of the IFR Program at ANL (Reference G.4.19-1). The program will consist of ex-reactor experiments including (1) downward melt relocation in the assembly, (2) melt breakup, quench, and solidification in the sodium-filled regions under the core, (3) effect of iron in the melt compositions ranging from UFe_2 to various compositions of U-Fe-Zr, (4) the coolability of core debris accumulated on horizontal surfaces in the sodium pool, (5) fuel dispersal in a transient overpower event, and (6) the retention of fuel and fission products within the sodium.

Consequences of Core Melt

ALMR safety analyses indicate that the probability of core melt is extremely low, well below the level of the safety goals (10^{-6} per year). Preliminary tests suggest that blockages in the metal core will be coolable and would not lead to large core melts. Therefore, for the capability evaluation, a core melt was just assumed to have formed from some unknown initiating event, and the evaluation was focussed on assessing the feasibility of containing the melt within the reactor vessel. Scoping analyses were performed for a set of melt compositions and melt volumes to assess the potentials for recriticality, temperatures in the melts, the inlet plenum structure consumed by the melts, and the capability of the plenum to contain the melts.

Based on the analyses, recriticality does not appear to be a problem except if the melt consists of only the driver fuel or the driver fuel and cladding, in which case recriticality may occur when the melt slumps into the shield regions of the fuel assemblies. Addition of B_4C to this region is being considered to preclude recriticality.

Recriticality is not expected for any melt composition after the melt spreads out in the inlet plenum. Analyses of the steel that can be dissolved into the melt indicate that the melt would consume at most 1 inch of the 6 inches bottom plate of the inlet plenum, with no effect on the redundant lower core support structure underneath the plenum. Structural analyses indicate that the core melt can be retained in the lower core support structure, and the pressures from the decay heat loads can be contained in the primary system boundary with comfortable design margins against the ASME Code design limits.

Thus, the scoping analyses indicate that it should be feasible to retain a molten fuel metal core within the lower internal structures and away from the reactor vessel. Evaluations to better define the core melt compositions, geometries, and temperatures in order to validate the above conclusion are planned as part of Phase III (1991-1995) of the IFR Program (Reference G.4.19-1).

G.4.19.3 HCDA Evaluation

G.4.19.3.1 Magnitude of HCDA Loads

<u>FFTF HCDA Data Base</u>

ALMR safety analyses indicate that the probability of core melt is extremely low, well below the level of the safety goals. Preliminary scoping accident analyses have not indicated HCDA work potentials of any consequence for the ALMR metal core. Therefore, the capability evaluation was based on the HCDAs developed during the safety analyses of the FFTF oxide core which has a power rating (400 MWt) comparable to the ALMR core (471 MWt).

The FFTF analyses distinguished between HCDAs initiated by UTOP events and those initiated by ULOF events, predicting vapor bubbles with different expansion characteristics as shown in Figure G.4.19-1 (Reference G.4.19-2). The sodium vapor bubble from the UTOP-HCDA is characterized by a high initial pressure because of the large specific volume of sodium, but the pressure decreases rapidly with expansion. The fuel vapor bubbles from ULOF-HCDAs have relatively low initial pressures. However, these events may be more damaging if the initial pressures are maintained during expansion because of entrainment and vaporization of sodium by the expanding fuel as shown in Figure G.4.19-1.

The UTOP-HCDA and the 100 $/sec ULOF-HCDA with sodium entrainment were used for design capability assessment in the FFTF FSAR [Reference G.4.19-2]. While not used, larger HCDAs were defined in FFTF parametric analyses assuming arbitrarily large reactivity insertion rates as shown in Figure G.4.19-2 (Reference G.4.19-3). The total work potentials of these HCDAs as defined by the work performed by the HCDA vapor bubbles during adiabatic expansions to one atmosphere, are summarized in Table G.4.19-1.

Table G.4.19-1

FFTF HCDAS USED IN ALMR CAPABILITY EVALUATION

HCDA	Ramp Rate $/Sec	Initial Pressure, atm	Adiabatic Expansion Energy, MJ
UTOP-1	-	200	150
ULOF-1	50	3	14
-2	100	32	310
-3	125	78	510
-4	150	156	800
-5	175	266	1100

UTOP- UTOP-initiated HCDA producing a superheated sodium vapor bubble. A unique ramp rate was not specified for UTOP-1.

ULOF- ULOF-initiated HCDA producing a fuel vapor bubble. The estimates for work energy are for the fuel vapor expansion without any sodium entrainment.

HCDA Loads Used in ALMR Capability Evaluation

Parametric analyses were performed for the ALMR response to the FFTF HCDAs in Table G.4.19-1. The expansion curves in Figure G.4.19-1 were used directly to define the pressure loads for the UTOP-HCDA and the 100 and 125 $/sec ULOF-HCDAs with most damaging sodium entrainment. The evaluation for the larger HCDAs (150 and 175 $/sec) was based on approximate interpretation of the pressure-work relationship in Figure G.4.19-2. The FFTF safety analyses did not identify any valid processes which would produce appreciable bubble pressures, so the HCDAs in Figures G.4.19-1 and -2 were based on arbitrarily large envelopes assumed for the reactivity rates and reactivity worth insertions. Additional conservatism in the FFTF load calculations, and therefore in the ALMR analyses, included:

- exclusion of energy loss by deformations and flow resistances in the expansion calculations - the reflector/shield assemblies and the core barrel were assumed rigid, and the vapor was assumed to exit the core with ejection area equal to the cross-section of the entire core.

- assumption of the most damaging rate of sodium entrainment in the expanding fuel vapor bubble - less entrainment would decrease the pressure and more entrainment would quench the bubble.

- exclusion of the heat loss to the structural surfaces or the bulk sodium in the pressure calculations.

Additionally, the differences in the sodium and fuel vapor expansion characteristics in Figure G.4.19-1 were enveloped in the ALMR analyses by using the UTOP-HCDA for the initial pressure loads, and the ULOF-HCDA's with sodium entrainment for the subsequent pressure loads.

G.4.19.3.2 Loading Sequence

The sequence of HCDA load transmission to the primary system boundary is shown schematically in Figure G.4.19-3 and described below:

o <u>Initial Core Pressure (near instantaneous)</u>: The core pressure will displace the reflector/shield assemblies and the shield cylinders, and load the core barrel. Rapid deformation of the core barrel will pressurize the surrounding sodium, displace the B_4C shield cylinders, and load the flow guide. The flow guide deformations, in turn, will pressurize the annular sodium and load the reactor vessel. The UIS cylinder will transmit the pressure loads on its bottom plate from the vapor exiting the core to the rotatable plug.

o <u>Sodium Slug Impact (milliseconds)</u>: The expanding bubble will accelerate the above-core sodium against the under-head insulation plates. The impact loads will be transmitted to the rotatable plug via the UIS cylinder and insulation plate supports, by failure and impact of the insulation plates, and by direct impact of sodium.

- o <u>Post-Impact Pressure (minutes)</u>: The residual pressure following the expansion of the vapor bubble to the cover gas space will act on the entire primary system boundary including the IHX as a static pressure.

- o <u>Decay Heat (hours/days)</u>: The decay heat from the fission products released to the cover gas will build up the cover gas temperatures and pressure. The pressure will act on the entire primary system boundary including the IHX as a static pressure.

The primary system boundary responses to the four loading phases are discussed in the following four sections.

G.4.19.3.3 Initial Core Pressure

<u>Pressure Load</u>

The initial pressure is considerably higher for the sodium vapor bubble from the UTOP-HCDA compared to the fuel vapor bubbles from the ULOF-HCDA's (Figure G.4.19-1). This higher pressure was used in the evaluation. The vapor bubble was assumed to have progressed to occupy the volume equal to that of the fuel/blanket assemblies in the core region (2.32 cubic meters) which corresponds to a bubble pressure of ~100 atm or 1470 psi.

<u>Core Barrel/Reactor Vessel Response</u>

A very conservative estimate of the acoustic loads may be obtained by ignoring the energy absorption in the reflector/shield assemblies, steel and B_4C shield cylinders, flow guide and sodium. In that case, a spherical pressure wave would produce the stresses and strains shown in the following table. These stresses are within the ASME Code stress limits, and the strains are small compared to the material ductilities.

Component	Pressure, psi	Hoop Stress, psi	Inelastic Strain, %
Core Barrel (SS304)	760	22400	1.6
Reactor Vessel (SS316)	400	22400	1.0

The 1470 psi pressure, applied statically, will produce core barrel hoop stress and strain of 42,630 psi and 7.2%, respectively. The stress is within the ASME Code Level D limit and the strain is within the 10% residual tensile elongation required of in-reactor structures. Actually, the core barrel pressure will be smaller than the core pressure because of the axial pressure relief, compression of the intermediate components, and deflection of the core barrel. Even for a blocked core, the pressure load will be diffused axially by the time it reaches the core barrel. Therefore, margins to core barrel failure will be higher than implied by the above bounding stress and strain estimates. Thus, the core barrel will remain intact and shield the reactor vessel from the static pressure loads.

The 1470 psi pressure, if transmitted through the core support structure without any pressure relief from radial and axial bubble expansions, would produce an axial vessel stress of 2980 psi which is insignificant.

Upper Internal Structure

The UIS bottom plate will be subject to acoustic pressure wave from the core and direct loading from the vapor bubble. The acoustic load will be in the nature of a pulse with little energy. The direct pressure, on the other hand, may last several milliseconds depending on the time required for the vapor bubble to expand through and around the UIS bottom structure. The loading uncertainties were enveloped in the analysis by assuming a linear pressure build-up to 2400 psi in 15 milliseconds.

The UIS response was calculated in dynamic, inelastic, finite element analyses using the DYNA3D code (Reference G.4.19-4). The analysis model and results shown in Figure G.4.19-4 indicate that the UIS lower box structure will collapse because of the slots in its outer wall, and the UIS cylinder wall as a whole will bend and buckle because of the slot along its length. The UIS may momentarily transmit approximately 10^6 lbs force to the rotatable plug. However, the 2400 psi pressure ramp used in the analysis is large compared to the bubble pressure of 1470 psi and the expected loading duration, and the 10^6 lbs closure load is small compared to the slug impact loads considered in the next section. Therefore the closure loads transmitted along the UIS are acceptable.

G.4.19.3.4 Sodium Slug Impact

<u>Impact Loads</u>

The upward sodium acceleration due to the bubble pressure will be impeded by the UIS diameter changes which would force the sodium to flow in and out of the UIS cylinder. This would (1) maintain an upward force pressing the rotatable plug/shear ring assembly against the closure plate, (2) decrease the sodium impact velocity, and (3) break up the sodium column producing an incoherent impact. Beyond noting that the plug clearances will be taken up and the sodium impact loads will be transmitted without internal impacts at the shear ring assembly, these mitigating effects of the UIS interference were ignored in the analysis. The impact velocity was calculated by integrating the one-dimensional equation of motion for the sodium column moving under pressures given by the HCDA pressure-volume diagrams.

With the long core barrel extension (284 inches) compared to the sodium depth in the upper plenum (102 inches) and the sodium travel before impact (50 inches), the bubble would have barely expanded past the UIS lower end when the impact occurs. Therefore the sodium would behave like a jet rather than expanding to a vessel size slug. Accordingly, the sodium column was assumed to have the diameter of the core barrel, and the pressure on the insulation outside the UIS was based on the loss of sodium momentum. The sodium within the UIS may generate a larger impact pressure because of the confinement. This pressure was calculated using the water-hammer formula for a blocked flow in a flexible pipe ignoring the pressure relief due to sodium escape through the UIS slot and larger cylinder deformations due to the slot.

Table G.4.19-2 shows the impact velocities and pressures calculated for the various HCDAs. The velocities and pressures for the larger HCDAs (150 and 175 $/sec) in the table were obtained approximately from the expansion work characteristics in Figure G.4.19-2.

Structural Analysis

The pressures in Table G.4.19-2 were enveloped in the analyses by calculating the closure responses to a range of peak impact pressures shown in Table G.4.19-3. The variations in impact pressures with time due to the pressure wave reflections from the slug boundaries were assumed to follow the load variations calculated in the CRBRP HCDA analyses [Reference G.4.19-5]. The pressure histories obtained by scaling the CRBRP slug impact load history to the peak pressures in Table G.4.19-3 are shown in Figure G.4.19-5.

The structural responses were calculated in dynamic inelastic finite element analyses using the ANSYS code (Reference G.4.19-6). The 15-degrees analysis model shown in Figure G.4.19-6 included the rotatable plug, closure plate, a length of the reactor vessel, and the containment vessel flange, with the flange supported at its lower end. The reactor vessel was included in the model to permit an estimate of the discontinuity stresses at its attachment to the closure, and was considered to be hanging free with no loads applied at the bottom end. The closure plate and the rotatable plug were allowed relative sliding in the radial direction and relative rotation around the hoop direction. That is, the bolting system and shear ring coupling the two were assumed to provide load transfer only in the axial direction. The closure plate thickness was decreased by 50% to 6" in the IHX and pump region to allow for reduced load transfer capability at the bolts.

Table G.4.19-2

SODIUM SLUG IMPACT VELOCITIES AND PRESSURES

HCDA	Work Done at Impact, (MJ)	Impact Velocity, (in/sec)	Impact Pressure, psi Water-hammer Impact	Jet Impact
UTOP-150 MJ HCDA	54	1740	7670	235
ULOF-With Sodium Entrainment				
100 $/sec Ramp	28	1260	5540	123
125 $/sec Ramp	66	1910	8440	284
ULOF-No Sodium Entrainment				
* 50 $/sec Ramp	3	380	1680	11
*150 $/sec Ramp	100	2360	10420	435
*175 $/sec Ramp	150	2900	12760	650

* The work to slug impact for these ULOF-events without sodium entrainment, and the corresponding impact velocities and jet and water-hammer pressures, were approximated on the basis of the pressure-work relationships given in Figure G.4.19-2.

Table G.4.19-3 STRUCTURAL RESPONSE TO SLUG IMPACT LOADS

* Peak Impact Pressure, psi	Maximum Stress, psi Closure	Reactor Vessel	Support Skirt	Maximum (Closure) Inelastic Strain, %
4900	19250	14540	19250	0.15
8160	21740	19380	20390	0.15
9790	24710	18300	19270	1.30
13060	32650	20130	20610	3.00

* The peak impact pressures used in the analyses envelop the water-hammer pressures calculated for the FFTF HCDAs considered in the analysis as shown in Table G.4.19-2

Analysis Results

The maximum stresses for the closure, reactor vessel, and containment vessel support skirt are summarized in Table G.4.19-3 for all the pressure levels considered in the analyses. The vessel and support stresses remain low at all pressures while the closure stresses increase with increasing impact pressures. This reflects the bending in the IHX/pump regions modeled as a 6" plate instead of the 12" closure plate to allow for reduced load transfer through the bolts. The variation in the closure stress in this region with time is shown in Figure G.4.19-7 which shows a large elastic spring-back for the 4900 psi impact pressure while increasing permanent deformations for the higher pressure loads. This effect is summarized by the maximum closure strains included in Table G.4.19-3. The highest closure stress is still considerably below the ASME Code Level D stress limit (~49000 psi) and the strain is small compared to the material ductility (>10%).

Figure G.4.19-8 shows the deflection profiles of the structure. The rotatable plug is predicted to remain essentially flat. The bending is confined to the closure plate which would separate from the plug at the shear ring interface. The maximum values of the plug center deflection and the relative displacement and rotation at the plug/closure interface are shown Table G.4.19-4.

The shear ring design concept verified through analyses and extensive material, component, and system tests during the CRBR development [Reference G.4.19-5] can be readily dimensioned to accommodate the relative displacement and rotation shown in Table G.4.19-4. The integrity of the risers and the canopy seal welded to the risers also will not be challenged with the effects of the relative displacement and rotation confined to the riser/closure and riser/plug junctions. The local strains in the risers at the junctions were calculated to be ~1% which is small. The response of head-mounted components cannot be considered at the present stage of design definition.

TABLE G.4.19-4

DEFLECTIONS UNDER SLUG IMPACT LOADS

* Peak Pressure, psi	Rotatable Plug Center Deflection, inch	Closure / Plug Relative Motion	
		Displacement, inch	Rotation, degrees
4900	0.92	0.009	1.1
8160	1.65	0.023	1.3
9790	4.09	0.138	5.3
13060	7.70	0.470	9.7

* The peak impact pressures used in the analyses envelop the water-hammer pressures calculated for the FFTF HCDAs considered in the analysis as shown in Table G.4.19-2

G.4.19.3.5 Post-Impact Pressure

Pressure Load

Post-impact pressures in the cover gas region are not known. It is expected that the long core barrel extension (244 inches) with volume roughly equal to the cover gas volume will prevent the vapor bubble from direct communication with the cover gas while maintaining its own identity. Therefore, it is assumed that the vapor will be forced through the sodium which would quench the bubble, leaving a small residual pressure. The low specific heat of the metal fuel compared to the oxide fuel would further promote quenching. However, detailed analyses are required to verify this assumption.

Design Capacity

The capacity of the pressure boundary to withstand static pressures within the ASME Code Level D design limits are shown in Table G.4.19-5. The table also shows the capacities of the holddown bolts for the IHX, EM pump, and the fuel transfer plug. The bolt capacities are based on the material yield strength rather than the ASME Level D limits since the pressure seals for these components depend on the bolts to limit deflections. The closure capacity in the table was calculated with the ANSYS code. The capacities of the other components were based on handbook formulas.

As shown in Table G.4.19-5, the structural components and the holddown bolts have large pressure retaining capacity. Also, the number and sizes of bolts can be increased if necessary after the post-HCDA pressures and the required bolt preloads have been calculated. The holddown bolts for the rotatable plug, on the other hand, will stretch out during the slug impact. Therefore the fission gas may pressurize the canopy seal over the closure/rotatable plug coupling. The canopy seal capacity (105 psi), while believed adequate because of the expected quenching of the vapor bubble, is low compared to the other components in Table G.4.19-5. It may be necessary to thicken the canopy wall or redesign the seal after the post-HCDA pressures have been calculated.

TABLE G.4.19-5

STATIC PRESSURE CAPABILITY OF PRESSURE BOUNDARY COMPONENTS

	ASME Level D Pressure Limit
Closure	>700 psi
Reactor Vessel	715 psi
IHX Riser (ext pressure)	790 psi
Rotatable Plug Canopy Seal	105 psi
* Holddown Bolts	
IHX bolts	700 psi
EM pump bolts	510 psi
Fuel transfer plug bolts	1240 psi

* The bolt capacities are based on the material yield strength rather than the ASME stress limit in order to assure pressure retention.

G.4.19.3.6 Decay Heat

<u>Decay Heat Load</u>

The decay heat will increase the temperatures in the cover gas region. This will increase the cover gas pressure because of (1) the increased gas temperatures, (2) the addition of the volatile fission products to the cover gas, and (3) the decreased cover gas volume due to the increase in the sodium temperature and elevation. The pressure will increase the primary system boundary stresses while the increased structural temperatures will decrease the component stress limits.

Unlike the vapor bubble pressures which were based on the FFTF oxide core HCDAs, the decay heat loads for the pressure boundary capability assessment were based on the ALMR metal core. Table G.4.19-6 shows preliminary estimates of decay heat from volatile fission products which would be in the cover gas space and decay heat from 1% of the solid products which was included in the evaluation in order to envelop possible heat inputs from the sodium surface and from any products plated out on the cover gas structures. Uncertainties in the decay heat estimates in the table were enveloped in the analyses by doubling the decay heat rates shown in the table.

The temperature changes due to the decay heat loads were calculated in finite element transient heat transfer analyses assuming:

- uniform distribution of the fission products in the cover gas volume,

- decay heat load to be deposited on the structural and sodium surfaces in and around the cover gas plenum with the cover gas assumed transparent and the structures and sodium assumed opaque to the decay heat carriers,

- heat transfer between the structural and sodium surfaces through radiation with the cover gas assumed transparent, and

Table G.4.19-6 DECAY HEAT RATES

Time		Decay Heat Rate, million BTU/hour
0	seconds	10.069
25	seconds	7.2553
50	seconds	6.4707
100	seconds	7.1041
250	seconds	4.6884
1250	seconds	3.5172
3000	seconds	2.9930
0.7	days	1.1257
7.7	days	0.2214
82.7	days	0.0157

Elements:	(Br+I)	50%
	(Cs+Rb)	1%
	(Xe+Kr)	100%
	Solid Products	1%

- heat transfer out of the plenum through sodium, reactor vessel (RVACS) and closure with the sodium assumed to be a constant temperature heat sink, and the vessel and closure assumed to be radiating heat to the containment vessel and head access area, respectively.

The sodium and containment temperatures were assumed to follow the "Level D - RVACS only cooling" core exit temperature history with the peak coolant temperature normalized to the design limit of 1300°F. The sodium level was assumed to remain at -98 inches corresponding to the peak sodium temperature during the RVACS Level D transient, and the head access area temperature was assumed to remain at 200°F through the event. The reactor vessel, closure and internal structures were assumed to be at normal operation temperatures at the beginning of the event and allowed to follow the temperatures dictated by the transfer of the decay heat.

Figure G.4.19-9 shows the analysis model and the maximum structural temperatures calculated in the analyses. As shown in the model, the insulation plates were modeled by a solid structure with equivalent density and specific heat to allow for the inter-plate gaps. The radial conductivity of the equivalent structure was decreased to allow for the actual effective area of conduction and the axial conductivity was decreased to represent largely radiative rather than conductive heat transfer between the plates. The emissivities for radiative heat transfer between the plenum structures, between the plenum structures and sodium, and between the pressure boundary and the containment environment was assumed to be 0.25.

Separate lumped mass analyses, assuming 10% of the decay heat to be deposited into the cover gas with the heat transfer from the gas to the structures through convective heat transfer, indicated small temperature changes which are easily enveloped by the conservative decay heat loads and analysis assumptions made in the above analyses. Therefore, the direct heat deposit to the gas was ignored.

<u>Design Capacity</u>

Figure G.4.19-9 shows the assumed sodium temperatures and the maximum structural temperatures calculated in the heat transfer analyses. The maximum temperatures of the reactor vessel and liner occur at the sodium surface and are the same as the sodium temperatures. When calculating the pressure loads on the structures, the cover gas was assumed to be uniformly heated to the maximum calculated structural temperature. In addition to the effect of the increase in the gas temperatures, the pressure estimate included the effects of sodium expansion due to increase in the sodium temperature, and increase in the cover gas contents due to the addition of the gaseous fission products. The cover gas was calculated to reach a maximum pressure of 60 psi. This pressure is within the Level D ASME capacity for all the primary system boundary structures as shown in Table G.4.19-5.

G.4.19.3.7 Mitigation Capability

<u>Analysis Results</u>

The structural analysis results may be summarized as follows:

- <u>**Initial Core Pressure**</u>: The core barrel/extension will shield the reactor vessel from the core pressure except for acoustic pressure transmissions. The latter will produce at most 1% vessel strain for all the HCDAs considered, which is small. The UIS bottom skirt will collapse rather than transmit the high initial core pressures to the rotatable plug. The other primary system boundary components will not experience the HCDA loads until slug impact occurs.

- <u>**Sodium Slug Impact**</u>: The stresses will be within the ASME Code Level D stress limits, the strains will be within the material ductility required by the Code, and the displacements and rotations will be within the shear ring, risers, and canopy seal design capability.

  The vapor bubble will be contained within the long narrow core barrel extension during impact which would limit the bubble/sodium interface and therefore sodium entrainment compared to the FFTF geometry. However, even with the most damaging sodium entrainment assumptions used in developing the FFTF loading curves, the impact pressures used in the analyses are expected to envelop the HCDAs in Table G.4.19-1.

- <u>**Post-Impact Pressure**</u>: Post-impact pressures in the cover gas region are not known. The long core barrel extension, with volume equal to the cover gas volume, will force the vapor bubble through the sodium, quenching it rather than permitting direct expansion to the cover gas. This needs to be verified through detailed analyses.

  In any case, the post-HCDA cover gas pressures are expected to be within the capability of the primary system boundary components except for the canopy seal, which may need to be strengthened, if necessary, after the cover gas pressures have been calculated in detail.

- **Decay Heat**: The decay heat from the fission gas and fission products released by the HCDA will pressurize the cover gas to a maximum pressure of 60 psi, which is small compared to the primary system boundary capacity.

Analysis Uncertainties

According to the analysis results, the ALMR design would contain HCDAs with total adiabatic work potential up to 1100 MJ without structural failure of the primary system boundary, disengagement of the rotatable plug, or sodium expulsion from the closure seals. The conservatism used in developing the pressure loads which ignored energy losses due to flow impedances at the top of the core and in the UIS, UIS and insulation plate deformations, and heat transfer to structures and bulk sodium during the bubble expansion would further support this conclusion. However, the conservatism may be offset by the following uncertainties:

- **Sodium Slug Impact**: The expansion characteristics used in the slug impact load development for the ULOF-HCDA's beyond ULOF-2 (125 $/second reactivity insertion rate) in Table G.4.19-1 were based on approximations.

- **Post-Impact Pressure**: The pressures after the vapor bubble has expanded to the cover gas volume are not known. While the structural components would retain their integrity and the rotatable plug would not disengage, the canopy seal with ASME Level D pressure limit of 105 psi may need to be redesigned if the residual HCDA pressures are calculated to exceed its capacity.

- **Analysis**: The slug impact analyses assumed axial symmetry, with the effect of asymmetries modeled by decreasing the closure thickness by 50% and bending stiffness by 87.5% in the IHX and pump support regions. Analyses to validate this approach remain to be performed.

- **Design Uncertainties**: Components such as the control drive enclosures mounted on the rotatable plug and the IHX piping mounted on the closure have not been designed and analyzed sufficiently to assure pressure retention under large plug displacements and closure rotations at the higher HCDA loads.

Mitigation Capability

Allowing for the uncertainties, the primary system boundary was concluded to have a capability to contain HCDAs with total adiabatic work energy of 500 MJ as opposed to the 1100 MJ implied by the analyses. The 500 MJ energy level was selected on the basis of:

- **Loading Uncertainty**: 500 MJ roughly coincides with the isentropic expansion energy of the 125 $/sec ULOF-HCDA in Table G.4.19-1 which is the highest energy HCDA for which a pressure-volume diagram accounting for the effects of sodium entrainment was available. The slug impact energy for larger HCDAs used in the analyses were based on approximations.

- **Loading Sensitivity**: The peak impact pressure from the 125 $/sec HCDA with most damaging sodium entrainment was calculated to be 8410 psi. This pressure roughly corresponds with the pressures beyond which the closure strains, and the closure/rotatable plug relative displacements and rotations increase relatively rapidly as shown in Figure G.4.19-10.

Based on these factors, the ALMR primary system boundary would contain HCDAs with work potential up to 500 MJ without structural failure, disengagement of the rotatable plug, sodium expulsion, or fission gas release. However, the canopy seal design may have to be modified if the future calculations show that the vapor bubble is not quenched to sufficient degree during its expansion through the outlet plenum sodium.

While based on conservative analyses and large design margins, the 500 MJ capability estimate is also predicated on the expansion characteristics of the ULOF-initiated FFTF oxide core HCDA vapor bubbles. The initial vapor pressure, vapor inventory, and fuel-sodium interactions during the bubble expansion may be different for metal core HCDAs with comparable adiabatic work potentials. As described in Section G.4.19.2.1, the IFR program at ANL is to investigate these differences and characterize the metal core HCDAs.

G.4.19.4 Core Melt Evaluation

This section describes a number of scoping studies that have been performed to assess the feasibility of retaining a molten metal fuel core within the reactor vessel. The studies were primarily concerned with recriticality of the melt, the temperatures reached by the melt and the retaining structures, and the dissolving of iron from the retaining structures into the melt. Structural analysis of the structures retaining the melt and design modifications to enhance this capability are also included.

The results of these studies indicate that it is feasible to retain a molten metal fuel core within the reactor vessel. Work will continue towards proving this objective.

G.4.19.4.1 Scoping Analysis

The current approach is to retain the core melt in the inlet plenum region of the reactor lower internal structure, eliminating the need for a separate "core catcher" structure. The scoping analysis to date has considered six issues of the core melt scenario. They are: (1) the identification of a set of representative melt compositions and the corresponding volumes and properties of these melts, (2) the critical sizes of these melts, (3) the potential for recriticality as these melts flow from the core region to the inlet plenum, (4) the temperatures reached by the melts when they reside in the inlet plenum, (5) the amount of inlet plenum structure consumed by the melt, and (6) the capability of the inlet plenum structure to contain the melt.

Core Melt Compositions

Four cases with various amounts of fuel and clad material were evaluated. They are: (1) the driver fuel by itself, (2) the driver fuel and its adjacent clad, (3) the driver and blanket fuel, and the adjacent clad of the driver and blanket fuel, and (4) the items of number three plus the fuel and blanket gas plenum clads, the fuel and blanket duct walls, the shield rods in the fuel and blanket assemblies, and the fuel and blanket inlet modules along with the adjacent inlet plenum upper plate. The compositions of these melts along with some of the properties important to core melt studies are given in Table G.4.19-7. To get an appreciation for the possibility of the melt going critical, the radius of a bare critical sphere and the thickness of a bare infinite slab were approximated for the compositions. These dimensions are also given in Table G.4.19-7.

Recriticality

An initial concern is that the melt will go critical when it slumps into the channels between the HT-9 shield rods below the core. A scoping analysis of the driver and blanket homogeneously slumping into this region predicts a melt of 11 volume % driver fuel, 26 volume % blanket fuel and 63 volume % HT-9. A critical bare sphere or infinite slab with this composition would be 73 inches in radius or thickness, respectively. Criticality does not appear probable for this case. However, if we assume that only the driver fuel slumps into this region, the composition is 31 volume % driver fuel and 69 volume % HT-9, and the critical sphere radius or critical infinite slab thickness is only 17 inches. More detailed analyses will be performed for this situation. However, to guard against recriticality for this situation, addition of B_4C to the shield region in the lower end of the core assemblies is being considered.

After the melt reaches the inlet plenum, it is assumed to spread out over the surface of the bottom plate of the inlet plenum. The thickness of the melt layer compared with the critical thickness for the various cases listed in Table G.4.19-7 is shown on Page G.4.19-27. This comparison shows that, if the melt spreads out upon reaching the inlet plenum as expected, recriticality is not a concern.

Table G.4.19-7

PROPERTIES OF SELECTED MELTS

Melt Parameter	Case Number(1)			
	1	2	3	4
Volume, ft^3	12	18	50	143
Weight % Cladding)	0	20.5	14.1	56.2
Melt Temperature, °F	NA	2020	1692	2282
°C	NA	1100	900	1250
Volume % Driver Fuel	100	65.7	22.3	8.2
Volume % Blanket	0	0	52.8	19.5
Volume % Cladding)	0	34.2	24.9	72.3
K_∞	1.907	1.877	1.198	1.074
Migration Area, cm^2	95.47	68.24	101.26	277.92
Geometrical Buckling, B^2	0.00673	0.00923	0.00178	0.00026
Critical Thickness of Bare Slab, in.	15.1	12.9	29.3	77.7
Critical Radius of Bare Sphere, in.	15.1	12.9	29.3	77.7

(1) Case

1. Driver fuel alone
2. Driver fuel plus adjacent clad
3. Driver fuel and blanket fuel plus adjacent clad
4. Driver-fuel, adjacent plus gas plenum clad, assembly walls, ends and shield rods; blanket-fuel, adjacent plus gas plenum clad, assembly walls, ends and shield rods; the fuel and blanket inlet modules with the adjacent inlet plenum upper plate. All the steel is HT-9 except for the inlet plenum upper plate which is SS 316.

Case	Thickness of Melt, in.	Critical Thickness, in.
1	2.1	15.1
2	3.1	12.9
3	8.7	29.3
4	24.9	77.7

Assumed Scenario

Scoping analyses were made to assess the capability of retaining a melted core within the inlet plenum. The following scenario was investigated:

- o Due to degraded flow passages, the reactor hot pool is at 1300°F and decay heat is being removed only by the RVACS.

- o The driver fuel and blanket pins melt, consuming the adjacent pin cladding, and the molten metal fuel drains and collects in the core structure inlet plenum. Figure G.4.19-11 depicts this starting situation. Although the melting of the driver fuel pins alone is a more likely situation, this scenario with both driver fuel and blanket pins melting was chosen since the total decay heat is greater and the heat flux out of the inlet plenum would be larger.

- o It was assumed that it takes 1 hour after shutdown from 100% power for the molten fuel, blanket and clad mixture to collect in the inlet plenum.

- o Cases were run with and without the decay heat from the volatile fission products included in the decay heat generated in the melt. The probable situation is that these volatile fission products will leave the melt and reside in the cover gas or sodium. The results reported here are for these fission products retained in the melt. In addition, it was assumed that all the gamma ray energy was deposited in the melt along with the alpha and beta energy. This is a conservatism since a great deal of this energy would be deposited in the surrounding structures. The melt temperatures with the Xe,Kr,I,Br,Cs and Rb fission product heating removed were calculated to be over 200°F lower than the temperatures with the heating from these fission products included as reported herein.

Thermal Analysis

The thermal analysis was pseudo steady state and one dimensional. Thermal convection was assumed within the body of the melt. The heat transfer coefficient at the top of the melt, where over 90% of the heat transfer would occur, was calculated as function of the melt and sodium temperatures and the heat flux flowing from melt. A conservative heat transfer coefficient of 1000 watts/m^2-°C (5678 Btu/hr-ft^2-°F) was assumed on the lower side of the inlet plenum bottom plate.

The results of the analysis are shown in Figures G.4.19-12 and -13. Figure G.4.19-12 shows the peak melt temperature as a function of time after shutdown. Figure G.4.19-13A depicts the predicted temperature distribution in the melt and the inlet plenum bottom plate at the peak temperature time (one hour after shutdown). It can be seen that, at this time, a major portion of the bottom plate would be above 1500°F, the temperature at which the ASME code assumes it to have no strength, although in reality it would still have some structural capability. To correct this situation a backup plate of 2 inches thickness is being added under the 6 inches inlet plenum plate as shown in Figure G.4.19-14. With this configuration, the majority of the temperature gradient is taken across the 6 inches plate leaving the backup plate temperatures within the ASME Code temperature limits. The temperature distribution with this modified structure is shown in Figure G.4.19-13B.

Iron Consumption

A characteristic of metal fuel is that it will tend to dissolve iron when molten. This characteristic is a function of the uranium concentration in the melt and the temperature of the melt. The eutectic temperature is well below the melting temperature of iron, and for a range of concentrations is below the melting temperature of uranium. The iron-uranium phase diagram is shown in Figure G.4.19-15.

An additional conservatism in choosing a melt consisting of both driver and blanket pins is that this melt is uranium rich and therefore has the ability to dissolve a large quantity of iron. Figure G.4.19-15 can be used to determine the amount of iron that could be potentially dissolved by the melt at the 2185°F calculated in the previous section. Point A represents the state of the melt before dissolving any iron and point B represents the maximum condition the reaction can proceed to. The initial composition is 86 weight % uranium and the final composition is 55 weight % uranium. Using these values, ~5 ft^3 of iron could potentially be consumed. Since the area of the bottom plate of the inlet plenum is approximately 69 ft^2, on the average, approximately 0.9 inch of its surface could be consumed leaving over five inches of the plate intact.

G.4.19.4.2 Structural Analysis

The core melt will produce 1) direct loads on the inlet plenum or the lower core support structure from the accumulated melt, and 2) decay heat loads from the fission gas released to the cover gas.

<u>Direct Load</u>

The lower core support is not subject to any loads during the design duty cycle including seismic events, except for a limited number of thermal cycles of no consequence. Therefore, it will accumulate insignificant creep-fatigue damage until required to contain the core melt.

In the case of the core melt accompanied by failure of the inlet plenum to retain the melt, the loads on the lower core structure will be defined by the melt weights in Table G.4.19-7 and the temperature distributions in Figures G.4.19-12 and -13. The potential failure mode at the temperatures shown in the figure is creep-rupture which is controlled by the ASME Code creep-fatigue damage limit. The Code limits the cumulative creep-damage measured in terms of operation times at different stresses and temperatures as fractions of the allowed times at the stresses and temperatures. Figure G.4.19-16 shows the damage accumulation with time based on the temperature histories in Figure G.4.19-12 and -13 and a melt weight corresponding to Case 4 in Table G.4.19-7. The creep damage accumulates to ~0.11 in 1000 hours compared to the ASME Code limit of 1.0. The damage accumulation rate beyond 1000 hours will be low and will become insignificant as the structural temperatures and creep rate decrease.

In the case of a seismic event, the stress levels will be larger than the gravity-induced stresses by a factor of 1.25 for a 0.5g peak ground acceleration which is small compared to the time independent stress limits applicable to seismic loads. The consequences of small quantities of the melt spilling over the lower support structure side wall under seismic loads are expected to be benign and have not been evaluated.

<u>Decay Heat Load</u>

The cover gas decay heat loads and corresponding temperatures and pressures will be similar to the decay heat loads following an HCDA. The maximum cover gas pressure from these loads will be about 60 psi which is small compared to the ASME Code Level D design limits for the primary system boundary as discussed in Section G.4.19.3.6.

G.4.19.4.3 Future Work

As stated previously, the work reported here should only be considered to be scoping studies to evaluate feasibility of retaining a molten metal fuel core within the ALMR reactor vessel. Also, only a slow core melt is considered. The following paragraphs describe the additional work to be performed to demonstrate in-vessel retention of core melt in the ALMR Program and in Phase III of the IFR Program (Reference G.4.19-1).

Energetic Core Disruption

During an energetic core disruption, it is expected that some of the fuel would be ejected into the upper plenum space above the core. Preliminary experiments at ANL have investigated the fragmentation characteristics of metal fuel. The results indicate that stringers of fuel are formed, and that these would deposit to form very porous debris beds that should be easily cooled by natural convection of sodium. For the ALMR geometry, these debris beds would be expected to form on the top of the reflector/shield assemblies and top former plate which surround the driver and blanket assemblies and, if the expulsion is sufficiently energetic, upon the top of the horizontal baffle. Also, if melting of the fuel occurred under high flow conditions, the stringers could be washed into and through the IHX, and be trapped within it or settle out in the lower plenum region on the reactor vessel bottom head.

Although it is expected that the fuel will solidify and remain frozen and subcritical under these scenarios, the situation will be examined in detail as part of Phase III of the IFR Program (Reference G.4.19-1).

Thermal Analysis

The thermal scoping analysis assumed that the melt was thoroughly mixed by natural circulation with only small temperature gradients existing in the bulk of the melt. Also, arbitrary but conservative heat transfer coefficients were assigned within the melt and between the melt and the top surface of the bottom plate of the inlet plenum. Although these assumptions are not critical to the result of the analysis or its implications, they will be verified. To accomplish this and to conduct a more detailed thermal analysis of the situation requires detailed information on the thermal and hydraulic properties of the melt. Properties of the fuel melt such as its thermal conductivity, coefficient of thermal expansion, viscosity and specific heat are required to perform analyses equivalent to those which have been performed for oxide fuel melts. Such data must be obtained either theoretically or empirically before an improved thermal analysis of the melt can be conducted; this is planned as part of the IFR Program.

Recriticality Analysis

For those situations where the one dimensional scoping analysis indicates that recriticality is a concern, detailed multidimensional nuclear analysis which takes into account reflection from the surroundings and spectrum changes will be conducted. Also, if needed, the amount of B_4C for the lower axial shield and its design will be established.

Iron Consumption

Essential to showing that a metal fuel melt is retained within the reactor vessel is a better understanding of the propensity of the melt to dissolve iron in the retaining structures. In particular, the driving reactions and their dynamics must be established. Present evaluations are based on the end points and are probably overly conservative. The effect of iron in the melt composition will be determined in the IFR Program.

G.4.19.5 References

G.4.19-1 "Integral Fast Reactor Program, Program Plan (Draft)," Argonne National Laboratory, Argonne, IL, September 1989.

G.4.19-2 "Assessment of the Hypothetical Core Disruptive Accident, FFTF Final Safety Evaluation Report," Vol 7, Westinghouse Hanford Company, Richland, WA, December, 1975.

G.4.19-3 H. Cho and M. Epstein, "Work Potential Resulting From a Mechanical Disassembly of the Voided FFTF Core," ANL/RAS 74-17, Argonne National Laboratory, Argonne, IL, August 1974.

G.4.19-4 J. O. Hallquist and D. J. Benson, "DYNA3D User's Manual (Nonlinear Dynamic Analysis of Structures in Three Dimensions)," UCID-19592, Rev 2, Lawrence Livermore National Laboratory, Livermore, CA, March, 1986.

G.4.19-5 M. Todd, T. Prevenslik and G. Elder, "CRBRP Closure Head Capability for Third Level Structural Margin Loading," WARD-D-0178, Westinghouse Advanced Reactor Division, Madison, PA, June, 1977.

G.4.19-6 "ANSYS Engineering Analysis System User's Manual (for ANSYS Revision 4.4)," Swanson Analysis System, Inc, Houston, PA, May 1989.

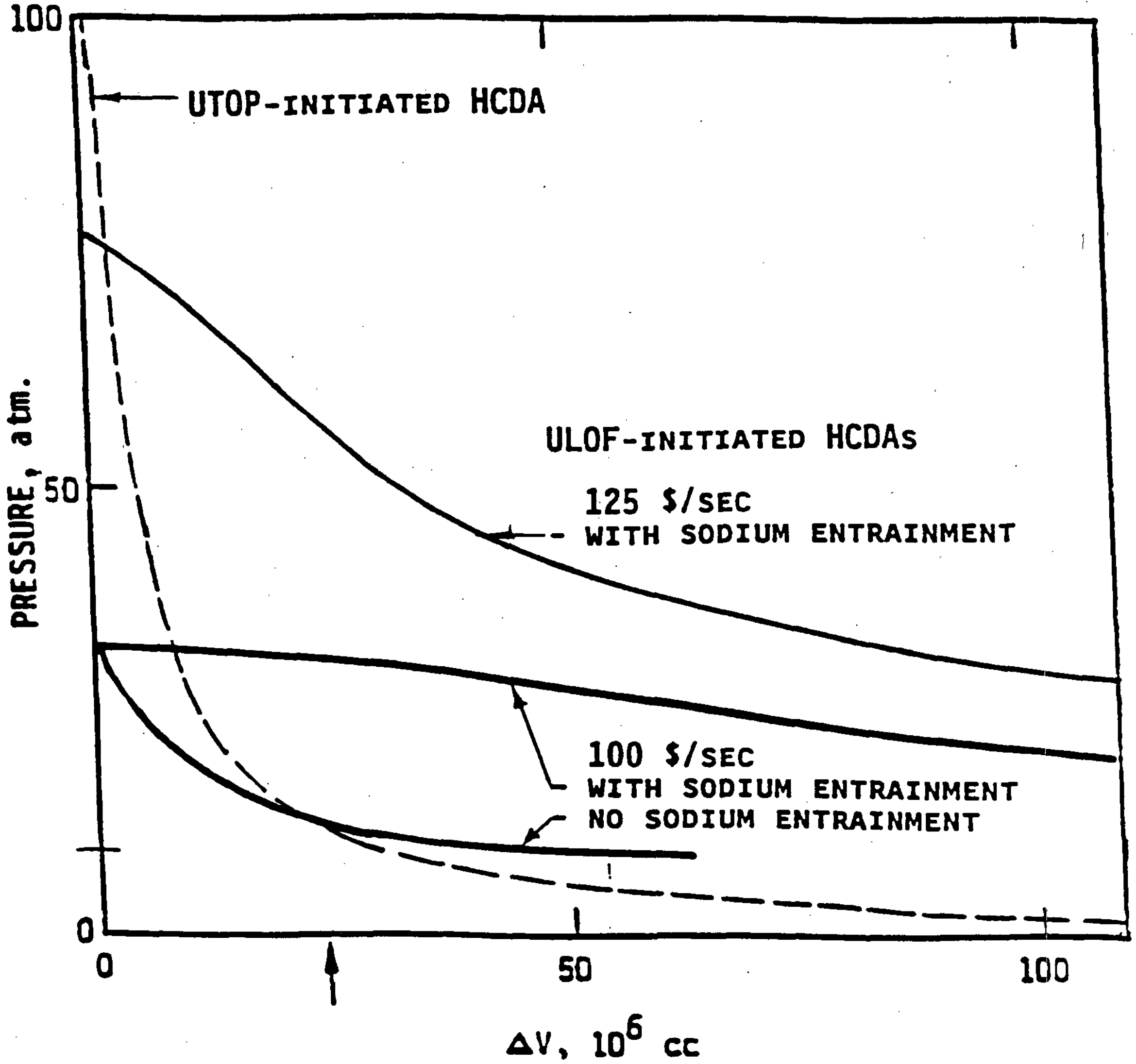


Figure G.4.19-1 FFTF HCDA PRESSURE/VOLUME DIAGRAMS

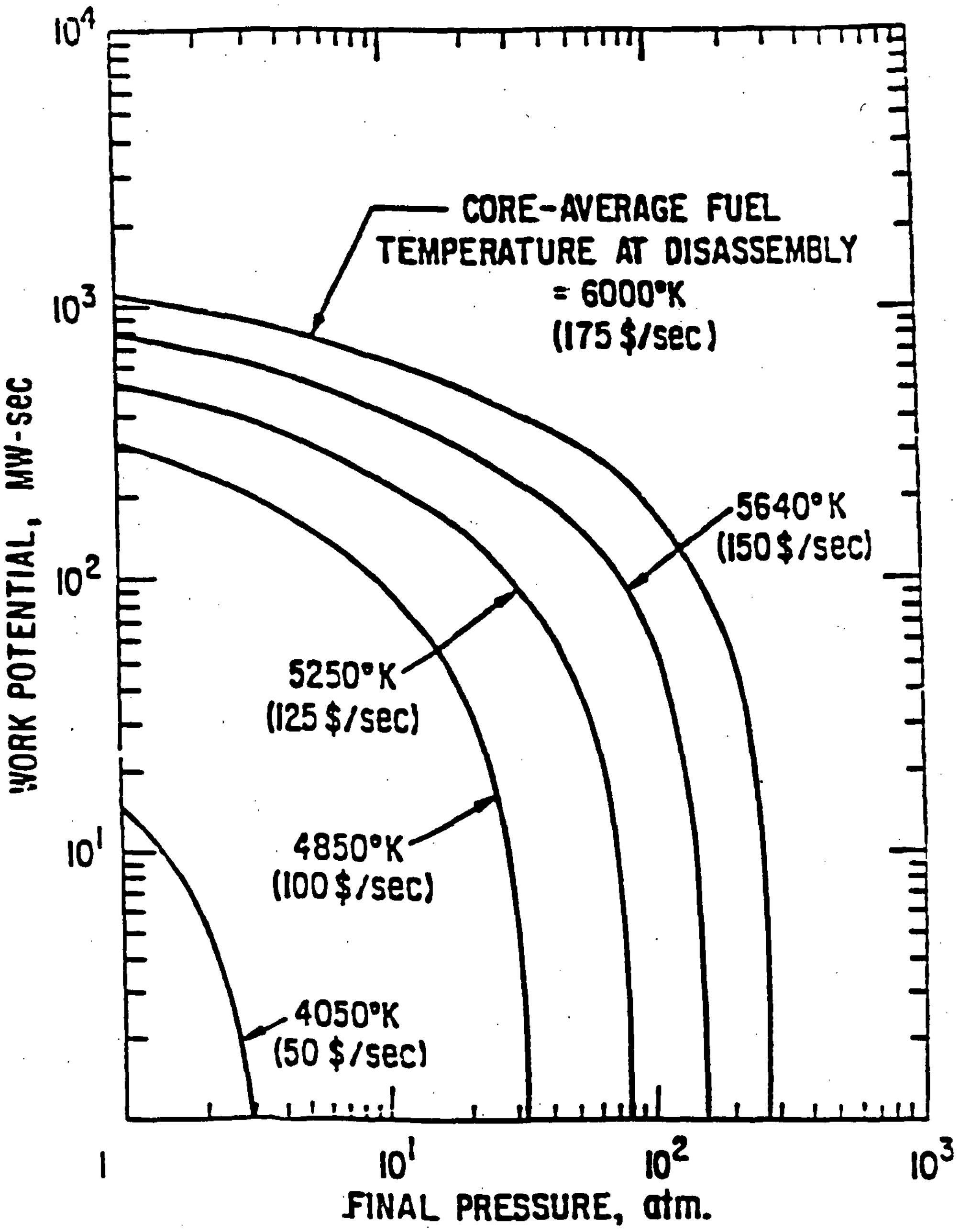


Figure G.4.19-2 FFTF PARAMETRIC ULOF-INITIATED HCDA'S

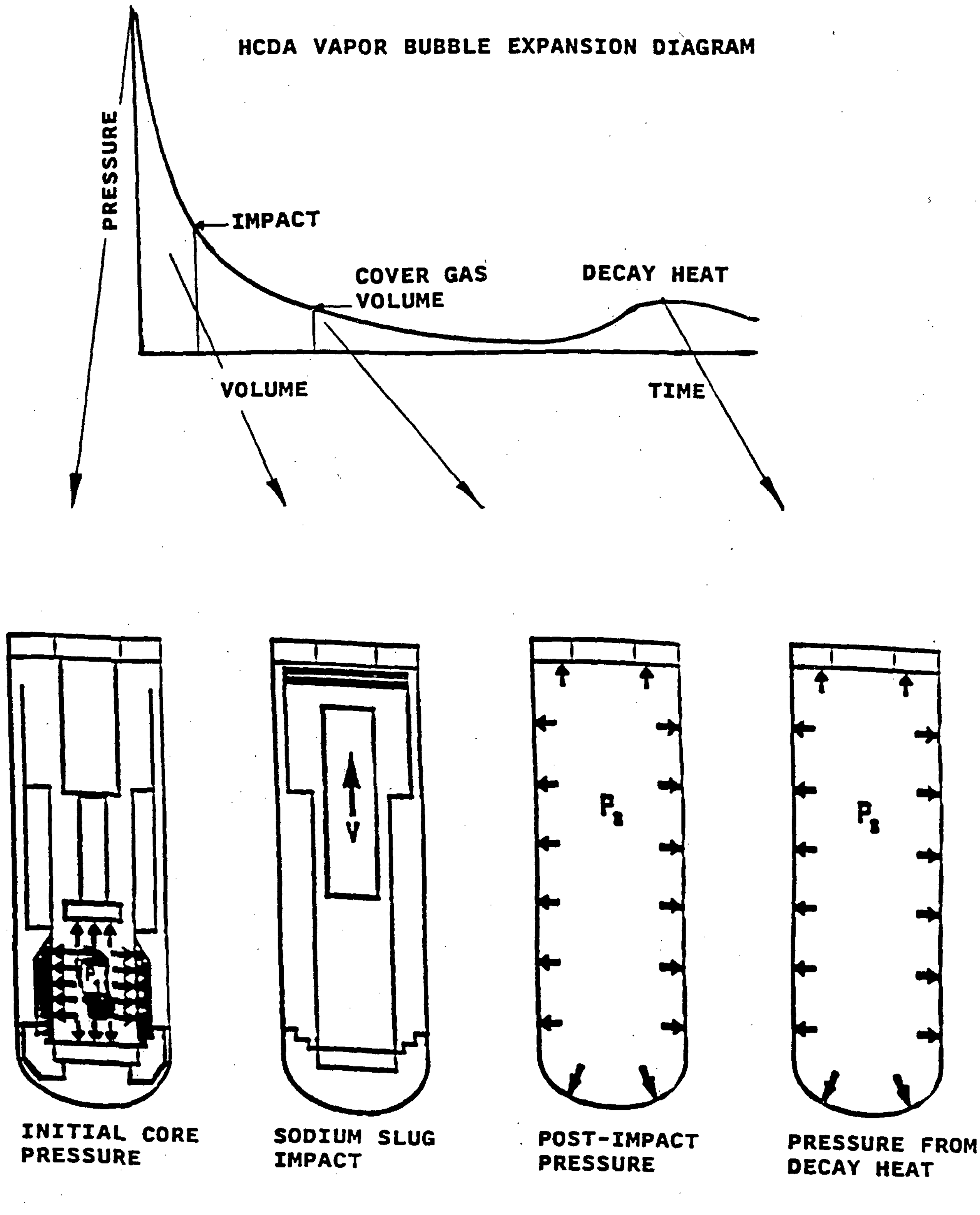


Figure G.4.19-3 HCDA LOADING SEQUENCE

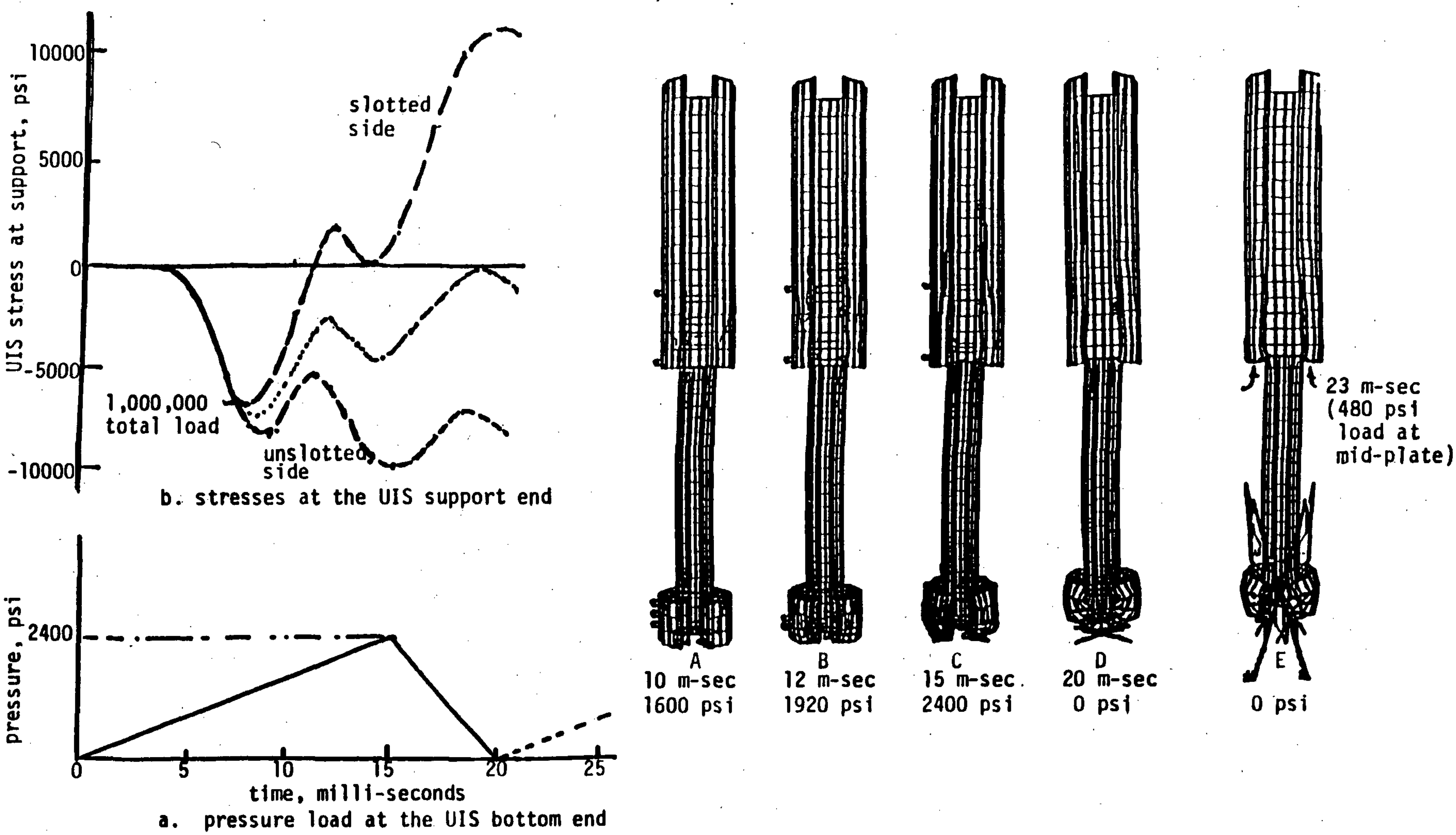


Figure G.4.19-4 UIS RESPONSE TO INITIAL CORE PRESSURE LOADS

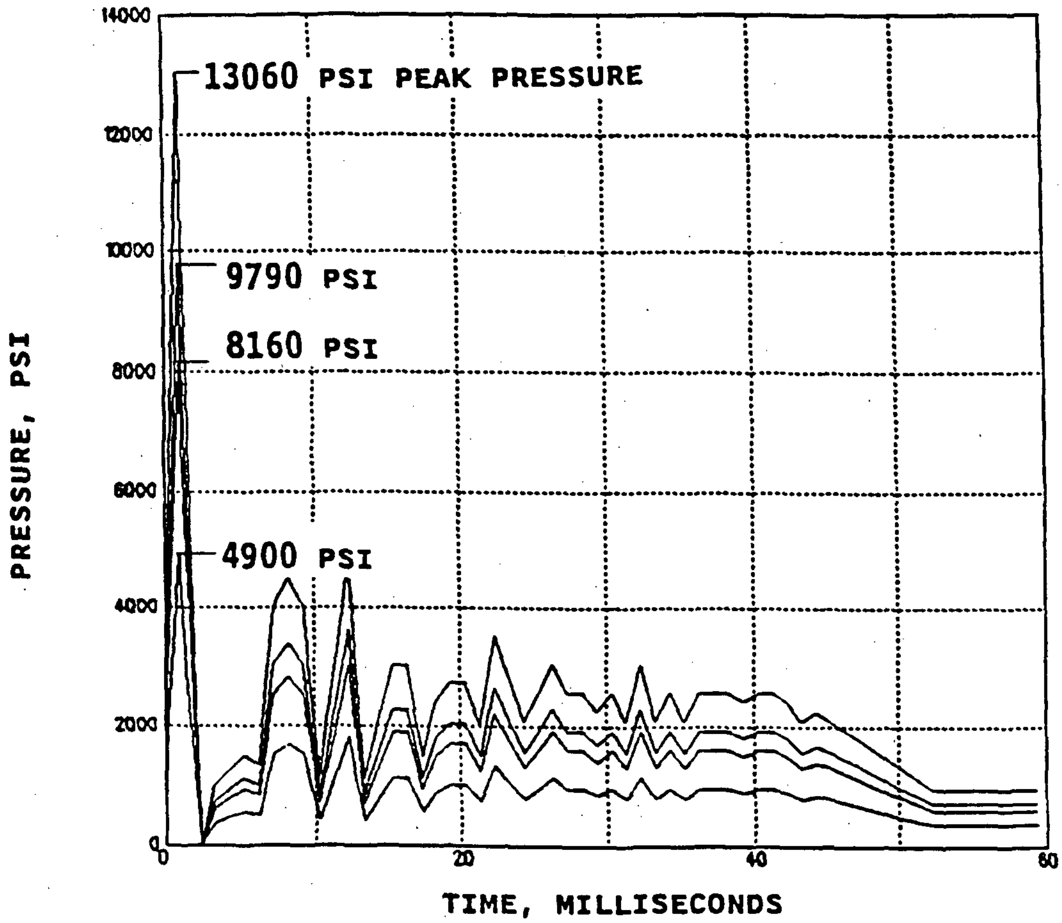


Figure G.4.19-5 SLUG IMPACT PRESSURE HISTORIES

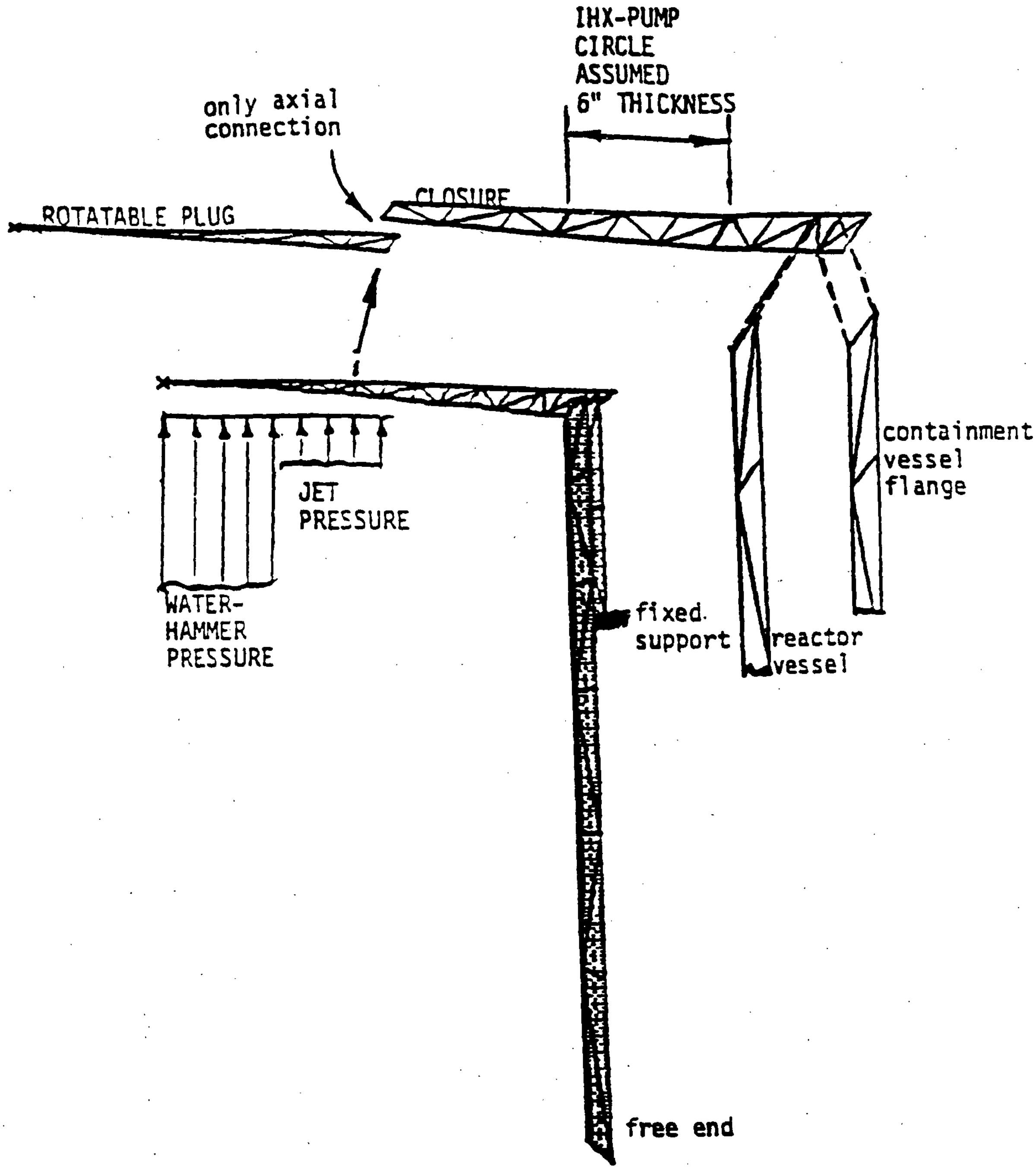


Figure G.4.19-6 SLUG IMPACT DYNAMIC ANALYSIS MODEL

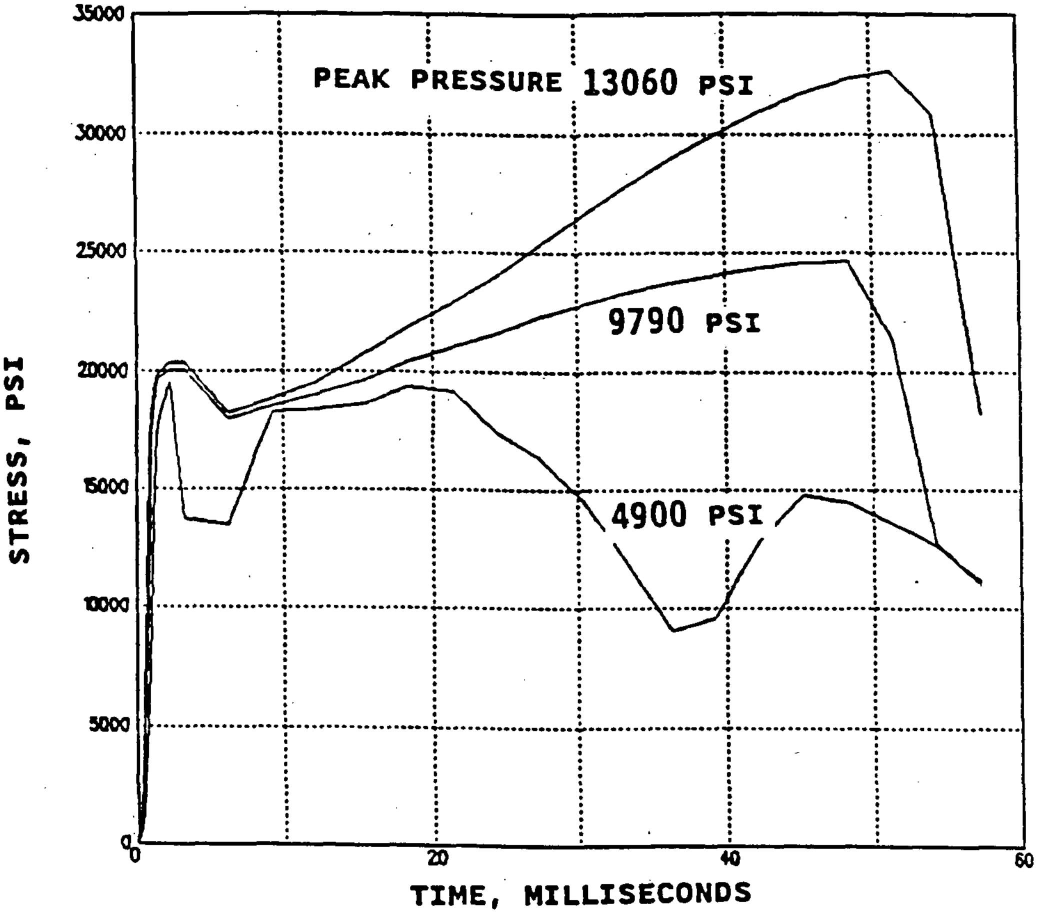


Figure G.4.19-7 MAXIMUM CLOSURE STRESS UNDER IMPACT

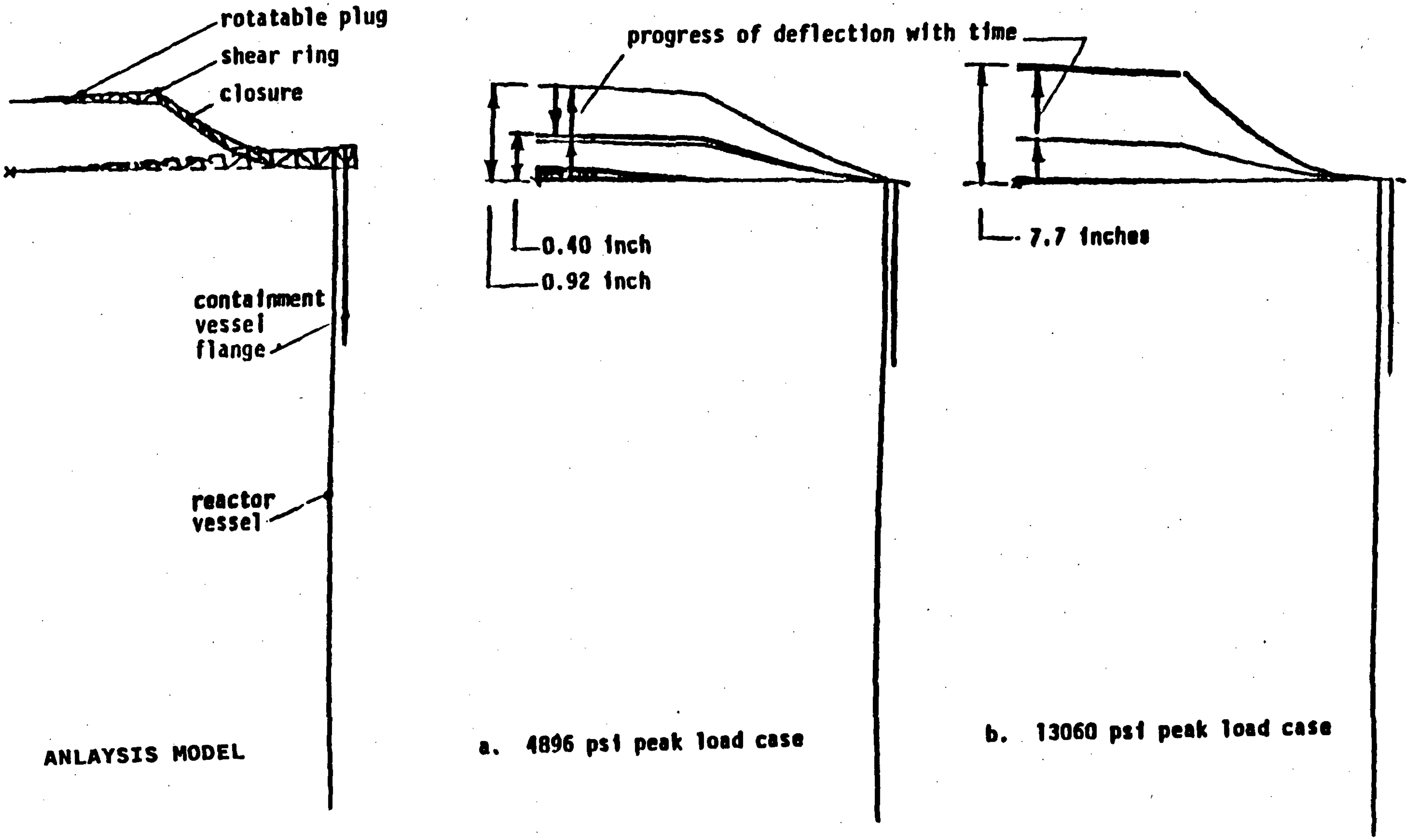


Figure G.4.19-8 DEFLECTION PROFILES UNDER SLUG IMPACT

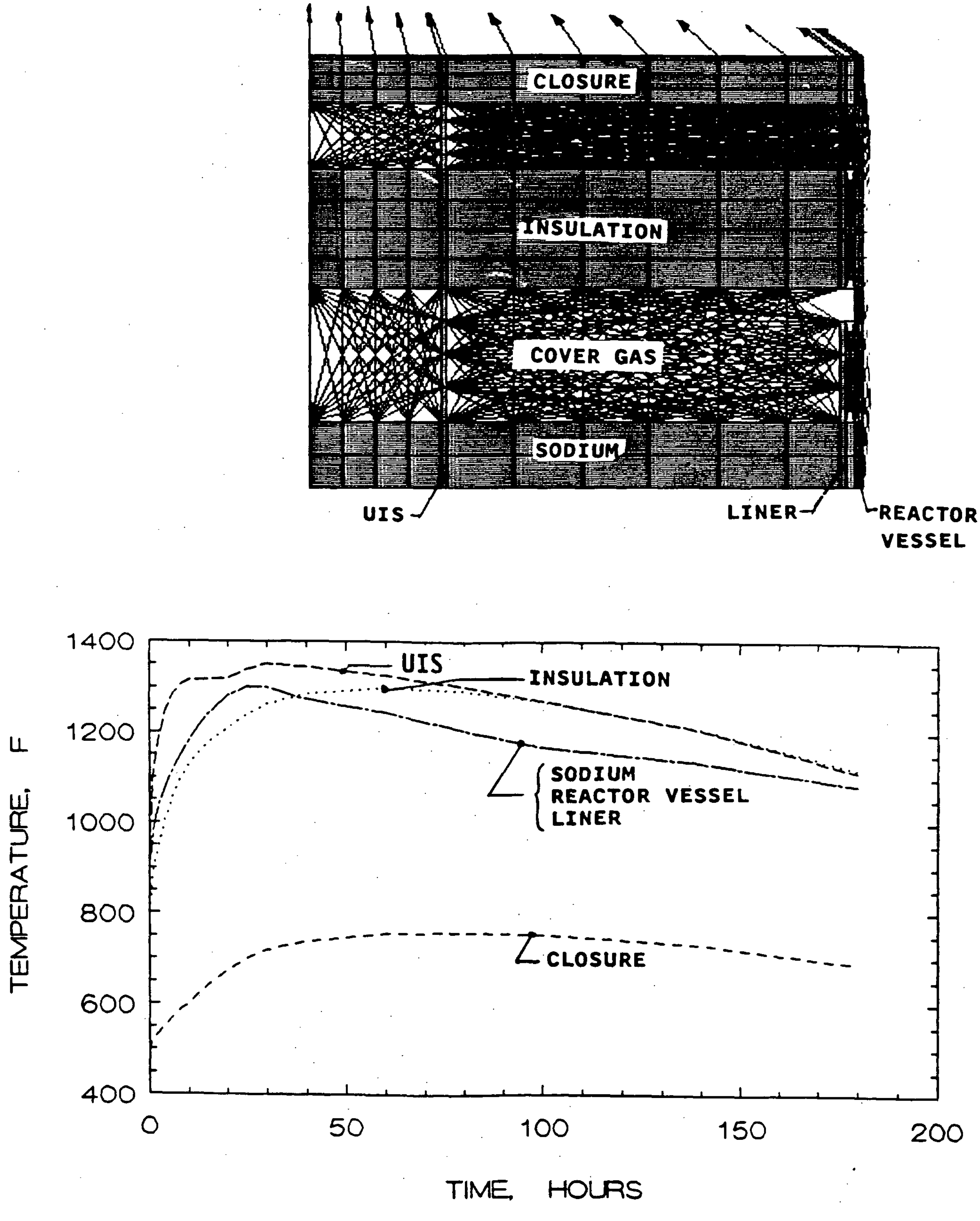


Figure G.4.19-9 MAXIMUM TEMPERATURES UNDER DECAY HEAT LOADS

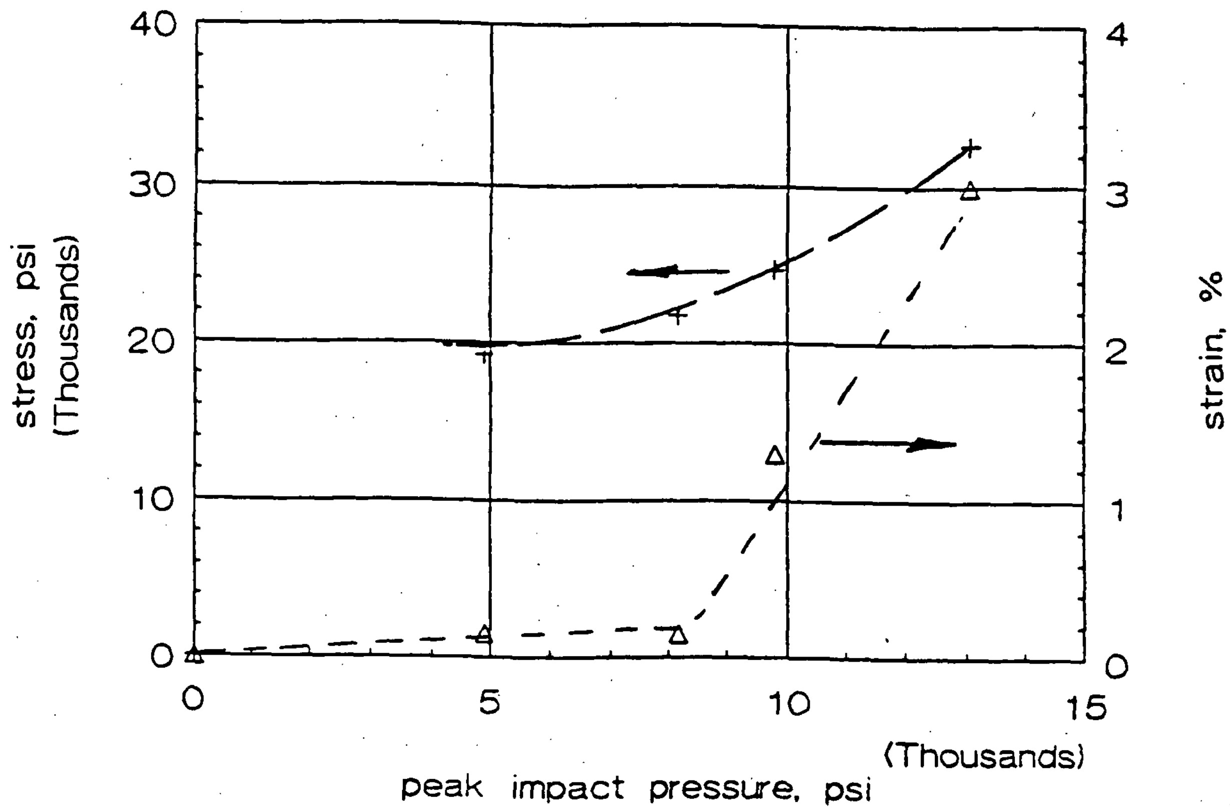


Figure G.4.19-10 CLOSURE MAXIMUM STRESS/STRAIN AS FUNCTION OF PEAK IMPACT PRESSURE

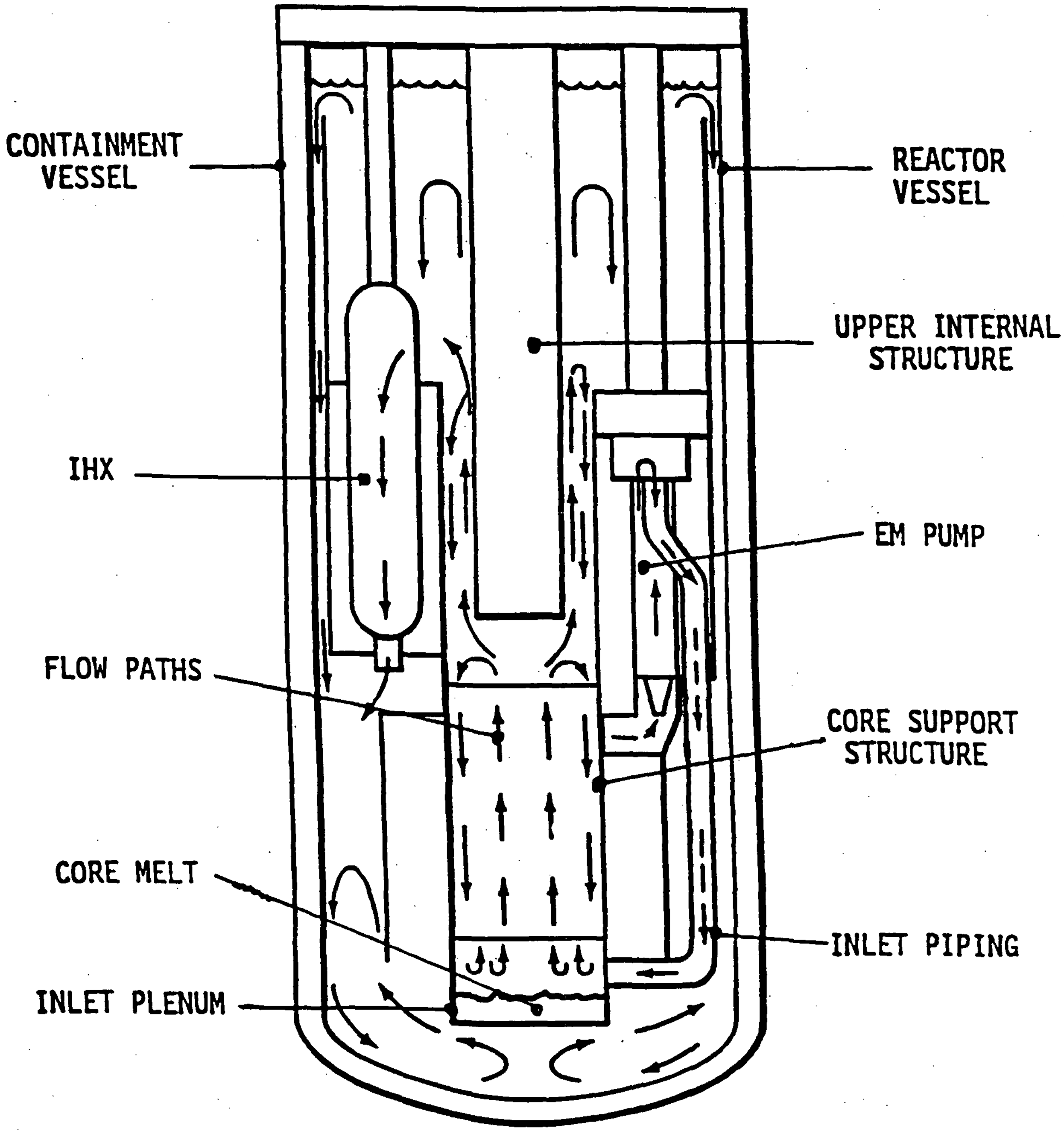


Figure G.4.19-11 CORE MELT SCENARIO

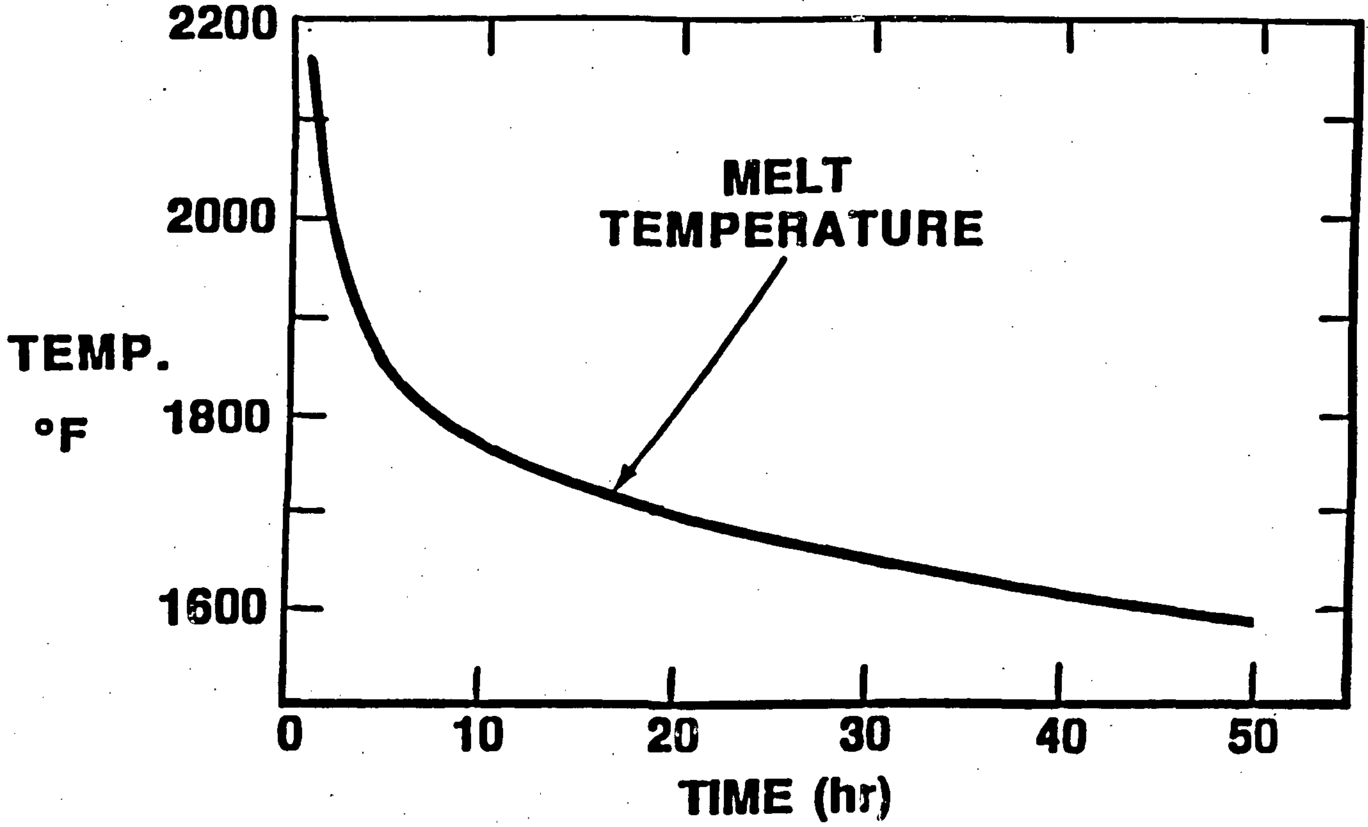
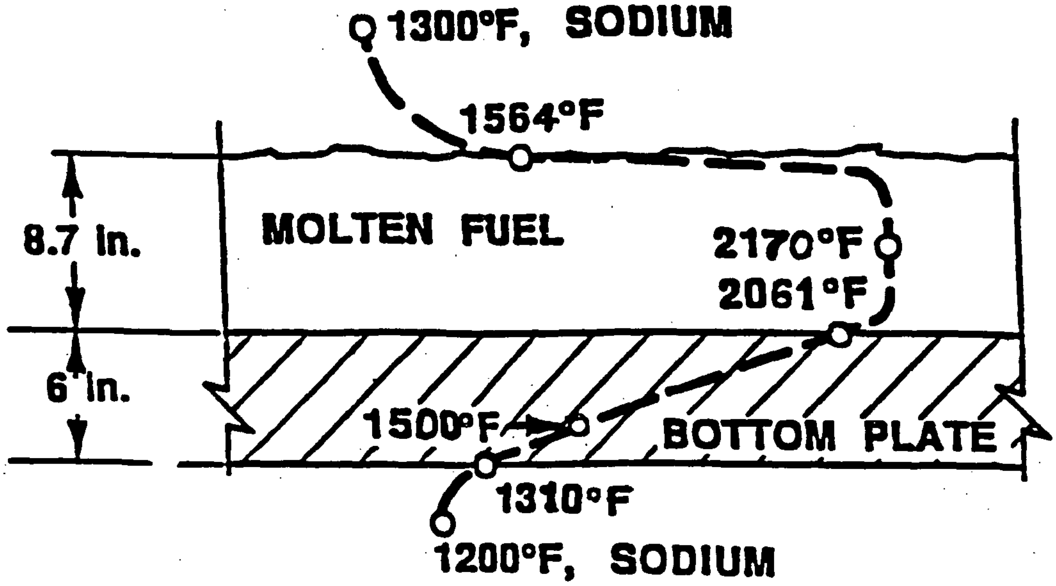
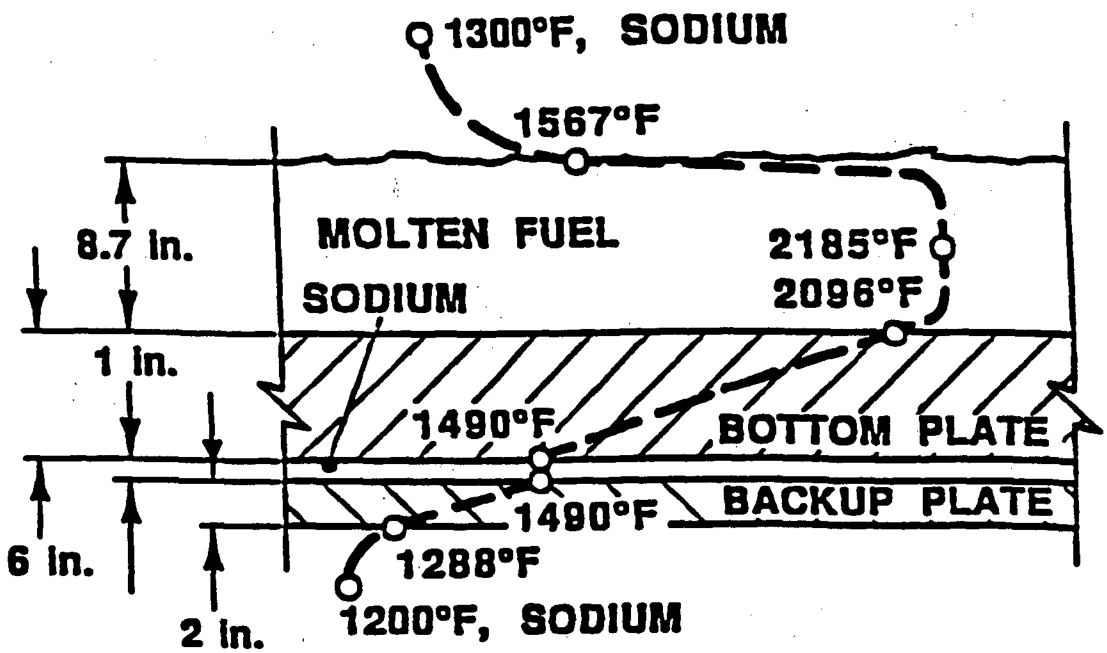


Figure G.4.19-12 MELT TEMPERATURE HISTORY



A-Without Backup Plate



B-With Backup Plate

Figure G.4.19-13 TEMPERATURE DISTRIBUTIONS IN THE MELT AND THE CORE SUPPORT BOTTOM PLATE

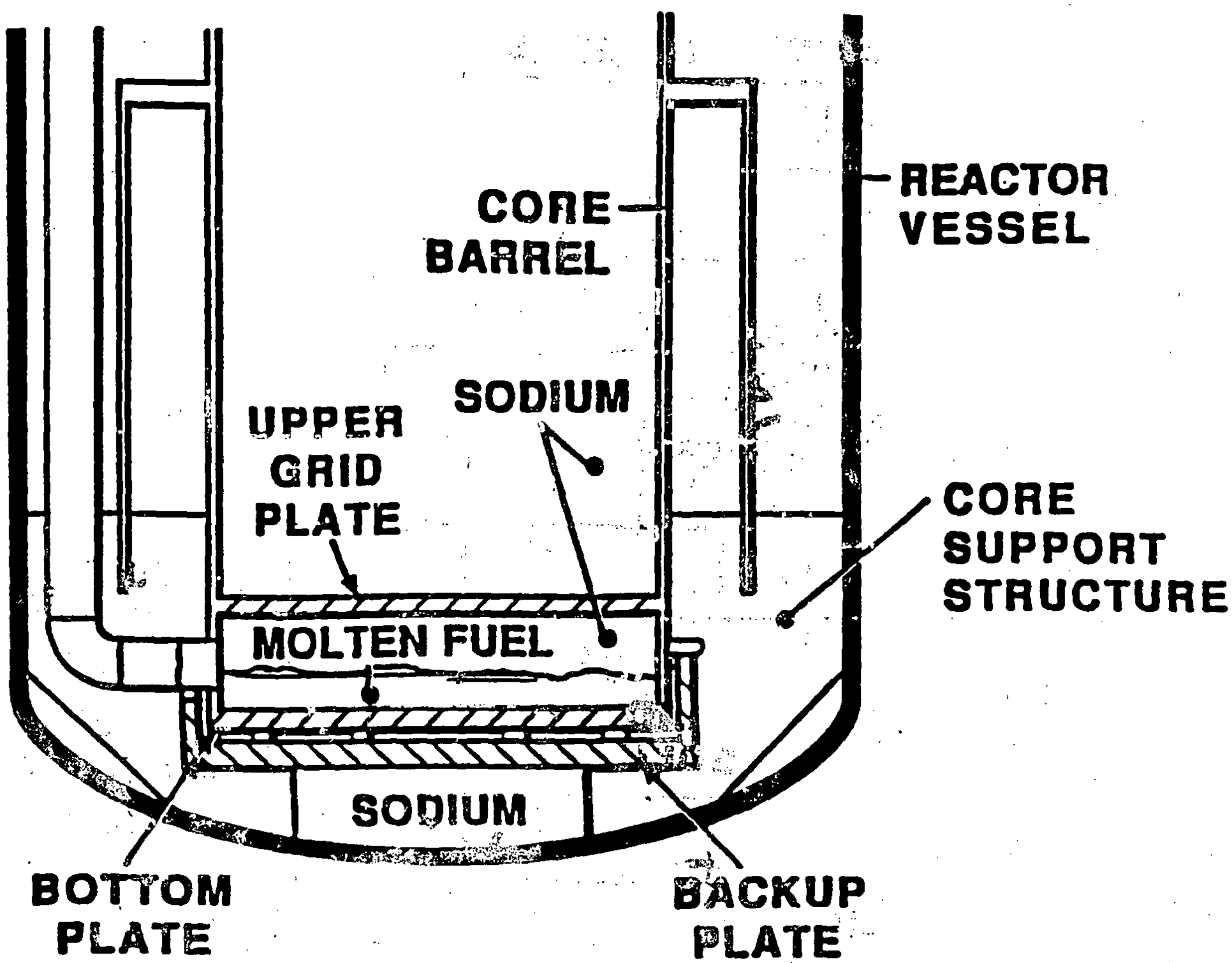


Figure G.4.19-14 MODIFIED LOWER CORE SUPPORT STRUCTURE

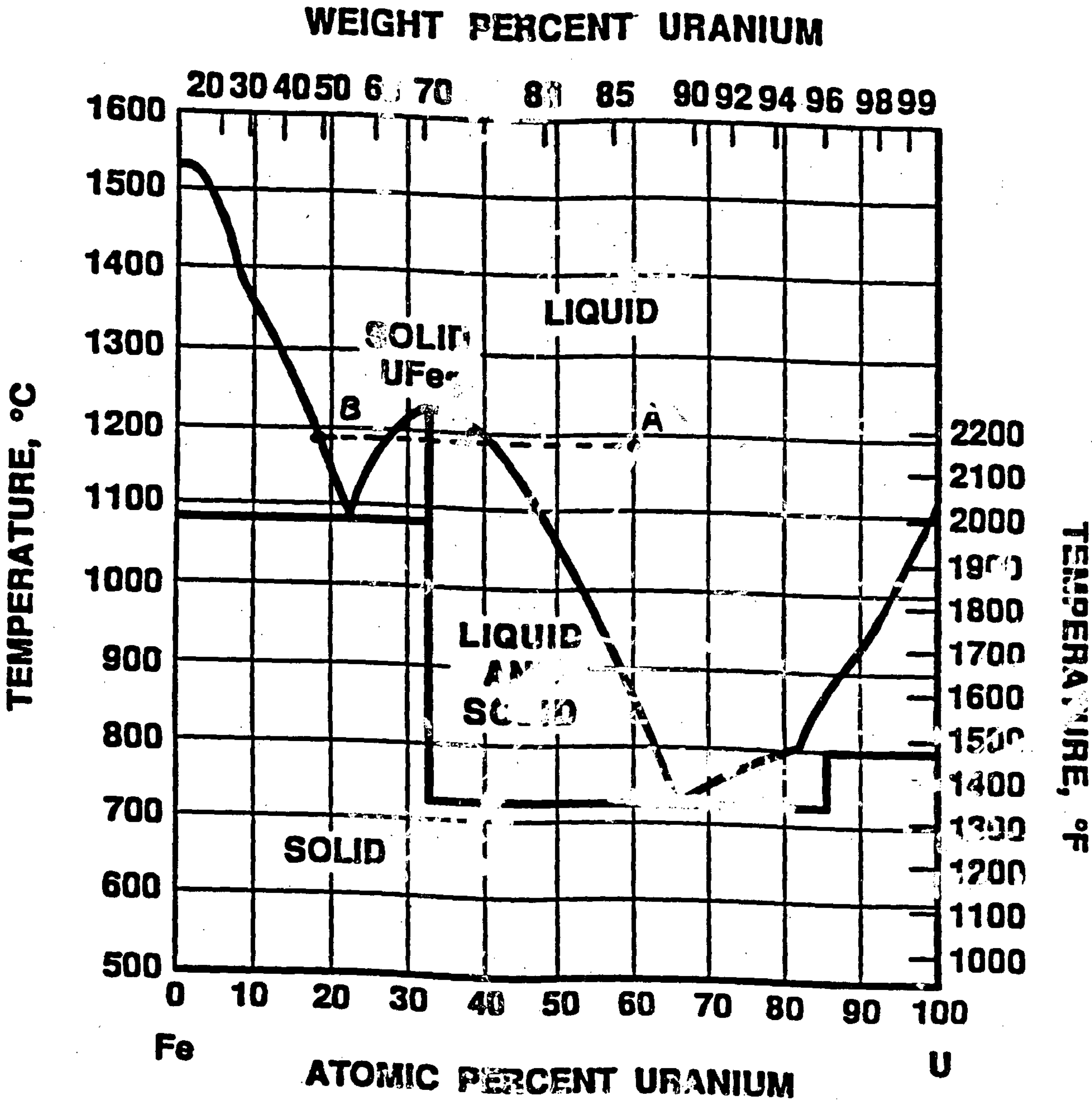


Figure G.4.19-15 IRON-URANIUM PHASE DIAGRAM

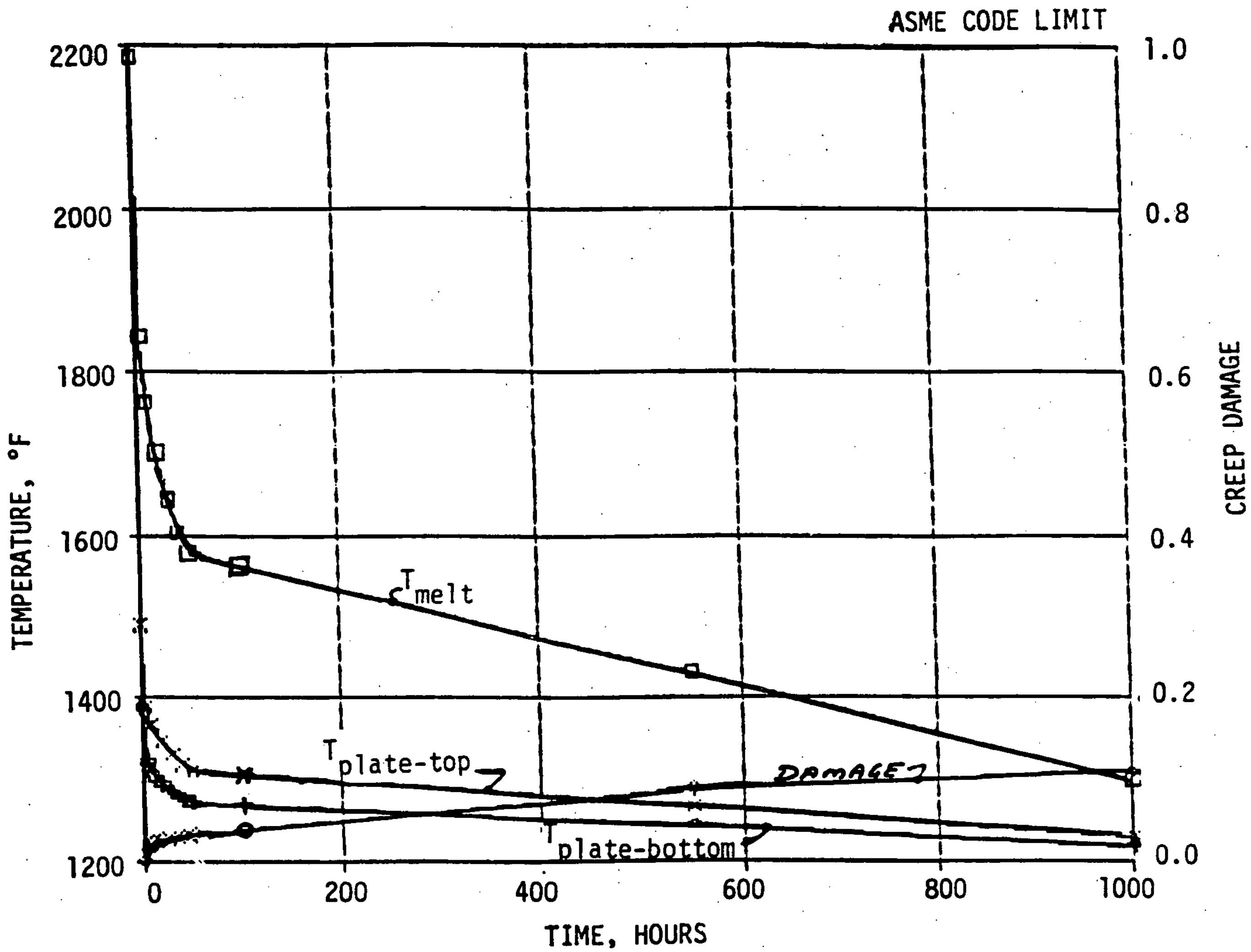


Figure G.4.19-16 LOWER CORE SUPPORT CREEP DAMAGE UNDER MELT LOADS