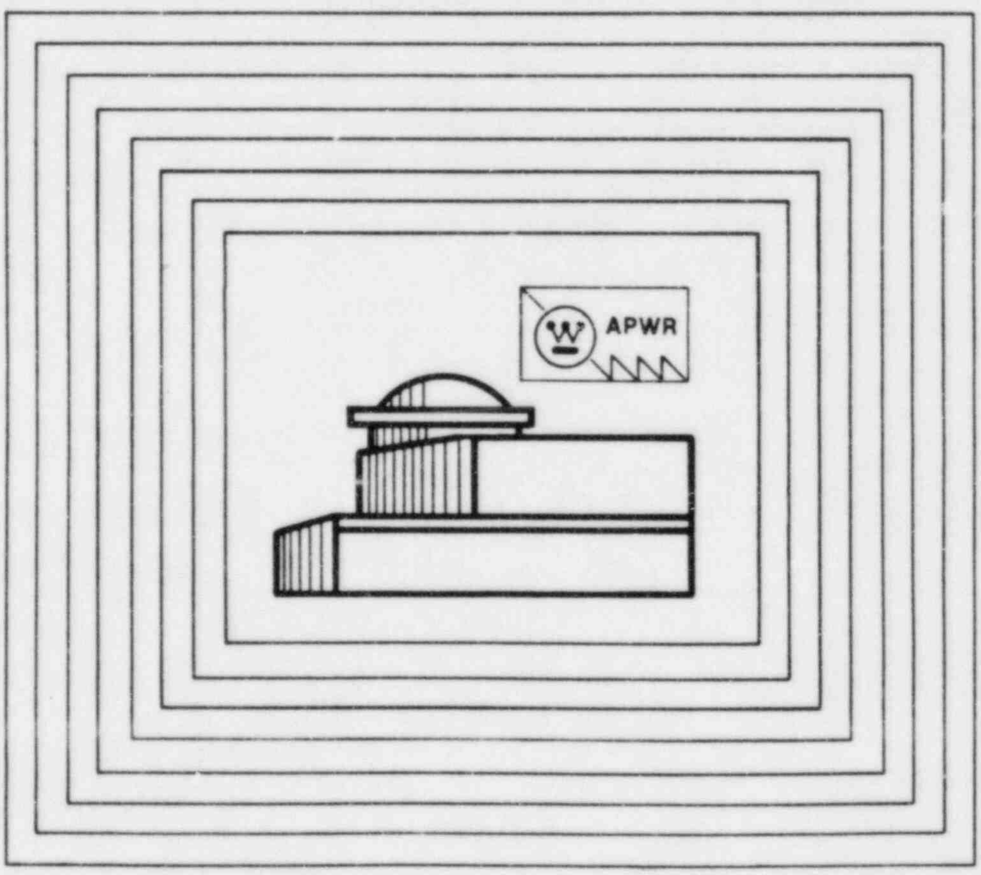
RESAR-SP/90
SECONDARY SIDE SAFEGUARDS SYSTEM/ STEAM AND POWER CONVERSION SYSTEM

WESTINGHOUSE ADVANCED PRESSURIZED WATER REACTOR



STANDARD PLANT DESIGN

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KEY TO "REFERENCE SAR SECTION STATUS" COLUMN:

Category I

Those sections which are complete and for which no additional information is to be provided for the PDA application.

Category II

Those sections which are complete insofar as providing material relevant to this system module but for which additional information will be provided in support of subsequent modules.

Category III

Those sections for which information on interfacing systems will be provided at a later date.

NA

Those sections for which categorization is not applicable. Only the section titles are included for clarity.

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1.0 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

The Westinghouse Electric Corporation (hereinafter referred to as Westinghouse) has developed this Reference Safety Analysis Report (RESAR-SP/90) for the Westinghouse Advanced Pressurized Water Reactor (WAPWR) as part of its continuing efforts toward design and licensing standardization of nuclear power plants. RESAR-SP/90 is a standard safety analysis report submitted initially for Preliminary Design Approval (PDA) in accordance with Appendix O, "Standardization of Design; Staff Review of Standard Designs," to Part 50 of Title 10 of the Code of Federal Regulations (hereinafter referred to as 10CFR). The ultimate objective is to obtain a Final Design Approval (FDA) of RESAR-SP/90 followed by a rulemaking proceeding and design certification.

1.2 GENERAL PLANT DESCRIPTION

1.2.2 Principal Design Criteria

RESAR-SP/90 is designed to comply with 10CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants." The specific applications of General Design Criteria to RESAR-SP/90 are discussed in Section 3.1 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design." Those General Design Criteria applicable to this module are listed in Section 3.1 of this module.

1.2.3 Plant Description

1.2.3.5 Secondary Side Safeguards System (SSSS)

The primary function of the Secondary Side Safeguards System is to remove heat from the core through the system generators during any plant condition when the normal secondary side systems (Main Steam and Feedwater Systems) are not available. This function is met by a combination of two secondary side systems, a Startup Feedwater System (SFWS) and an Emergency Feedwater System (EFWS).

The EFWS is a fully safety grade system while the SFWS is a control grade system. The EFWS is designed to meet all the required safety criteria as defined in this module. The SFWS, although not required to mitigate the consequences of postulated accidents, provides additional reliability and diversity of the EFWS. The SFWS also serves to minimize the number of EFWS actuations required which enhances the reliability of the EFWS. Figures 1.2-1 and 1.2-2 illustrate the general layout of the SFWS and EFWS equipment.

During normal power operation the steam generators are fed by the main feedwater pumps with steam sent to the turbine. A Startup Feedwater System (SFWS) feeds the steam generators during normal plant startup and shutdown.

Under these conditions steam from the steam generators is sent to the main condenser. The SFWS is also actuated automatically to provide feed following a reactor trip, loss of main feed, loss of offsite power, and other anticipated transients.

The Emergency Feedwater System (EFWS) consists of two totally independent and completely separated subsystems each of which receives electrical power from one of two separate safety class 1E electrical power trains. Each subsystem consists of an emergency feedwater storage tank, one motor driven emergency feedwater pump, one turbine driven emergency feedwater pump and the required piping, valves, instruments and controls necessary for system operation.

The motor driven and turbine driven pumps and the EFWST's are located in the safeguards area. The pumps in one subsystem are located on the opposite side of the reactor containment as the pumps in the other subsystem. The use of both motor driven and turbine driven pumps satisfies the requirement that the pumps be powered by diverse power sources. The turbine driven pumps are not dependent on A.C. power. When in operation, the emergency feedwater pumps take suction from the emergency feedwater storage tanks and discharge the water into the main feedline downstream of the isolation valve.

The pumps are sized such that any two of the four pumps delivering to any two of the four steam generators provides the minimum required emergency feedwater flow.

1.2.3.7 Steam and Power Conversion (SPCS)

The WAPWR nuclear steam supply system (NSSS) is designed to deliver steam via the main steam supply system to a turbine-generator unit located in the adjacent turbine generator building. It is expected that the turbine generator would be of conventional design with a gross electrical rating of approximately 1350 MW.

The turbine-generator building (or turbine-generator "island") which houses the steam and power conversion systems and equipment is designed and furnished by the plant specific applicant. The main systems/equipment are:

- o Turbine-generator unit and auxiliary systems
- o Main-steam supply system
- o Main condensors and steam dump system
- o Circulating water system
- o Condensate and main feedwater system
- o Startup feedwater system

There are major interfaces between the NSSS and the power conversation systems including the main steam and main feedwater systems which interconnect to the nuclear steam generators. All safety related (Class II) equipment in these two systems is located within the NPB.

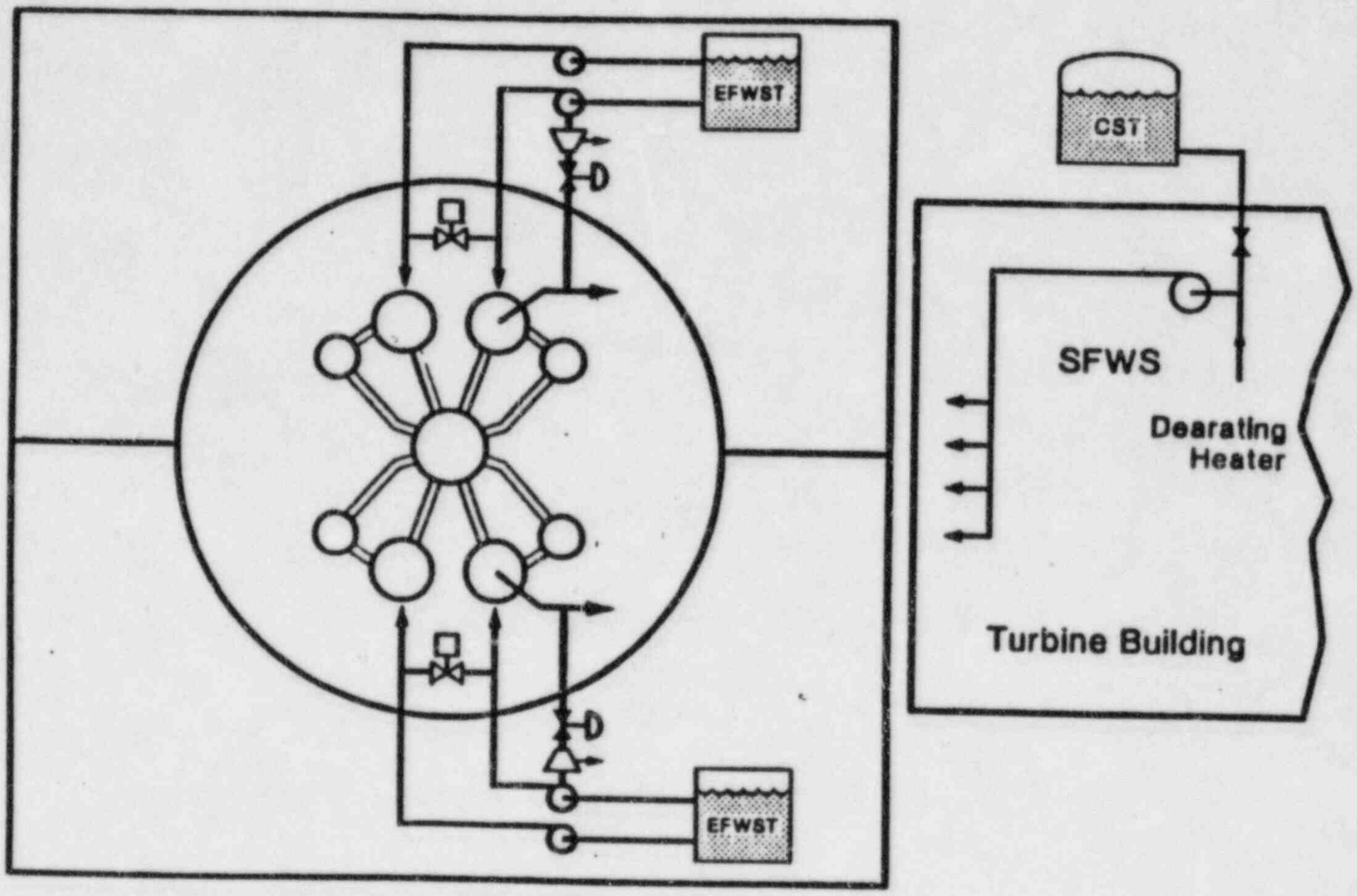

FIGURE 1.2-1 GENERAL LAYOUT OF SFWS/EFWS EQUIPMENT

(a,c)

FIGURE 1.2-2 EMERGENCY FEEDWATER SYSTEM

1.3 COMPARISON TABLES

1.3.1 Comparison With Similar Facility Designs

Table 1.3-1 presents a design comparison of the major parameters and features of the WAPWR secondary side safeguards system and steam and power conversion system with RESAR-414 (Docket No. STN-50-572; PDA-13), RESAR-3S (Docket No. STN-50-545; PDA-7), and RESAR-41 (Docket No. STN-50-480; PDA-3).

TABLE 1.3-1

DESIGN COMPARISON

Parameter or Feature	RESAR-SP/90	RESAR-414[1]	RESAR-3S[2]	RESAR-41[2]
No. of EFW pumps - M/D & flow[3], gpm	[ ](a,c)	-	2-490	3-550
- T/D & flow[3], gpm		-	1-985	1-550
No. of storage tanks - SC-3	2	-	1	1
- NNS	1	-	-	-
Total volume of safety grade water, gal.	[ ](a,c)	-	276,000	500,00
No. of SFW pumps & flow[4], gpm		-	0	0
No. of MSIVs per steam line	1	-	1	1
No. of safety valves per steam line	5	-	5	5
No. of PORVs per steam line	1	-	1	1
No. of SG overfill valves per SG	2	-	0	0
No. of MFIVs per main feed line	1	-	1	1

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(1) No docketed applications from which to obtain typical design parameters and features.
(2) Design parameters and features of typical applications.
(3) Rated flow of pump less miniflow at SG design pressure plus 3%.
(4) Rated flow of pump at hot zero power SG pressure; miniflow is isolated.

1.6 MATERIAL INCORPORATED BY REFERENCE

The WAPWR secondary side safeguards system/steam and power conversion system module incorporates, by reference, certain topical reports. The topical reports, listed in Table 1.6-1, have been filed previously in support of other Westinghouse applications.

The legend for the review status code letter follows:

A - U.S. Nuclear Regulatory Commission review complete; USNRC acceptance letter issued.

AE - U.S. Nuclear Regulatory Commission accepted as part of the Westinghouse emergency core cooling system (ECCS) evaluation model only; does not constitute acceptance for any purpose other than for ECCS analyses.

B - Submitted to USNRC as background information; not undergoing formal USNRC review.

O - On file with USNRC; older generation report with current validity; not actively under formal USNRC review.

U - Actively under formal USNRC review.

TABLE 1.6-1
MATERIAL INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	SAR Section Reference	Submitted to the NRC	Review Status
WCAP 7769	Overpressure Protection for Westinghouse Pressurized Water Reactors	1	15.2	7/5/72	U
WCAP 7907-P-A	LOFTRAN Code Description	0	15.0	10/11/72	A
WCAP 7908	FACTRAN: A FORTRAN-IV Code For Thermal Transients in a UO_2 Fuel Rod	0	15.0	9/20/72	U
WCAP 7979-P-A (P) WCAP 8028	TWINKLE: A Multi-Dimensional Neutron Kinetics Computer Code	0	15.0	1/7/75	A
WCAP 8305	LOCTA-IV Program: Loss of Coolant Transient Analysis	0	15.0	6/74	AE
WCAP 8306	SATAN-IV Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant	0	15.0	7/12/74	AE
WCAP 8330	Westinghouse Anticipated Transients Without Trip	0	15.2	9/25/74	U
WCAP 8424	Evaluation of Loss-of-Flow Accidents Caused By Power System Frequency Transients in Westinghouse PWRs	1	15.2	5/30/75	U
WCAP 8567-P(P) WCAP 8568	Improved Thermal Design Procedure	0	15.0	7/75	A
WCAP 8846-A	Hybrid B_4C Absorber Control Rod Evaluation Report	0	15.0	10/77	A

1.7 DRAWINGS AND OTHER DETAILED INFORMATION

1.7.1 Piping and Instrumentation Diagrams

Table 1.7-1 contains a list of each piping and instrumentation diagram and the corresponding SSSS/SPCS figure number. Figure 1.7-1 illustrates and defines symbols and abbreviations used in the diagrams.

Table 1.7-1

PIPING AND INSTRUMENTATION DIAGRAMS

1.8 CONFORMANCE WITH THE STANDARD REVIEW PLAN

In accordance with 10CFR50.34(g), Table 1.8-1 identifies and evaluates deviations from the acceptance criteria of those sections of the NRC Standard Review Plan (NUREG-0800) pertinent to the SSSS/SPCS.

TABLE 1.8-1

STANDARD REVIEW PLAN DEVIATIONS

SRP Acceptance Criteria	Deviation	Section

(During the licensing process, certain deviations with respect to the SRP acceptance criteria applicable to the SSSS/SPCS will be listed here as appropriate.)

2.0 SITE CHARACTERISTICS

No portion of this chapter is pertinent to the RESAR-SP/90 "Secondary Side Safeguards System/Steam and Power Conversion System" module.

3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

This section briefly outlines the General Design Criteria (GDC) applicable to the Secondary Side Safeguards System/Steam and Power Conversion System (SSSS/SPCS) of the Westinghouse Advanced Pressurized Water Reactor (WAPWR) per Title 10, Code of Federal Regulations, Part 50 (10CFR50), Appendix A, "General Design Criteria for Nuclear Power Plants". As presented in this section, each criterion is listed to denote applicability to the SSSS/SPCS (see Table 3.1-1).

A detailed discussion of the compliance of each criterion is provided in RESAR-SP/90 PDA Module 7, "Structural/Equipment Design".

Table 3.1-1

GDC APPLICABLE TO SSSS/SPCS

Criterion	Title
1	Quality standards and records
2	Design bases for protection against natural phenomena
3	Fire protection
4	Environmental and missile design bases
5	Sharing of structures, systems, and components
13	Instrumentation and controls
15	Reactor coolant system design
17	Electric power systems
18	Inspection and testing of electric power systems
19	Control room
20	Protection system functions
21	Protection system reliability and testability
22	Protection system independence
23	Protection system failure modes
24	Separation of protection and control systems
29	Protection against anticipated operational occurrences
44	Cooling water
45	Inspection of cooling water system
46	Testing of cooling water system
54	Piping systems penetrating containment
57	Closed system isolation valves

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Certain structures, components, and systems of the SSSS/SPCS are important to safety because they:

a. Assure the integrity of the reactor coolant pressure boundary.

b. Assure the capability to shut down the reactor and maintain it in a safe condition.

c. Assure the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 100.

d. Contain or may contain radioactive material.

The purpose of this section is to classify structures, systems, and components according to the importance of the item in order to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. Table 3.2-1 delineates each of the items in the plant which fall under the above-mentioned categories and the respective associated classification that the NRC, ANS, and industrial codes committees have developed. Each of the classification categories in Table 3.2-1 is addressed in the following sections.

The classification of specific piping runs and valves in these runs is provided in the system flow diagrams contained in this module. Instrumentation and electrical equipment associated with the SSSS/SPCS and which is required to shutdown the plant or mitigate an accident will be classified as 1E (or Safety Class 3 per ANS 51.1) and identified in the appropriate module.

Additional information regarding the classification of structures, components, and systems is provided in RESAR-SP/90 PDA Module 7, "Structural/Equipment Design".

3.2.1 Seismic Classification

Seismic classification criteria are set forth in 10 CFR 100 and supplemented by Regulatory Guide 1.29.

All components classified as Safety Class 1, 2, or 3 (classifications are as defined by Reference 1), are seismic Category I.

Seismic Category I structures, components, and systems are designed to withstand the Safe Shutdown Earthquake (SSE), as discussed in Section 3.7 and other applicable load combinations, as discussed in Subsection 3.8.5 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design". Seismic Category I structures are sufficiently isolated or protected from the other structures to ensure that their integrity is maintained.

3.2.2 System Quality Group Classification

The components are classified according to their importance to safety, as dictated by service and functional requirements and by the consequences of their failure. The quality assurance requirements and code requirements for the SSSS/SPCS meet the intent of Regulatory Guide 1.26.

3.2.3 Safety Classes

Table 3.2-1 lists the safety class assigned to applicable systems and components in accordance with ANS 51.1 (Ref. 1). The criteria (of Ref. 1) are used in the plant design to provide an added degree of assurance that the plant is designed, constructed, and operated without undue risk to the health and safety of the public.

3.2.4 References

1. "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." ANS-51.1, 1983.

Table 3.2-1

Classification of Structures, Systems, and Components For the SSSS/SPCS

(Sheet 1 of 2)

System/Component	Location	Quality Group[b]	Safety Class[c]	Code[d] Classification	Principal Construction Codes & Stds.	Seismic Category[e]	Quality Assurance[f]
Emergency F.W. System							
Pumps	OC[a]	C	3	3	III	1	B
Tanks (EFWST)	OC	C	3	3	III	1	B
Piping	IC/OC	B	2	2	III	1	B
	IC/OC	C	3	3	III	1	B
Valves	IC/OC	B	2	2	III	1	B
	IC/OC	C	3	3	III	1	B
Start-Up F.W. System							
Pumps	OC	D	NNS	N.A.	MFG	2	N
Tanks	OC	D	NNS	N.A.	AWWA	2	N
Valves	OC	D	NNS	N.A.	B31.1	2	N
Piping	OC	D	NNS	N.A.	B31.1	2	N
Deaerating Heater	OC	D	NNS	N.A.	MFG	2	N

(a) Outside Containment
(b) Classification Per Reg. Guide 1.26
(c) Safety Classification PER ANS-51.1
(d) ASME Section III, Division 1
N.A. - Not Applicable
(e) Classification Per Reg. Guide 1.29
(f) B-In accordance with the Q.A. Reqs. of 10 CFR 50 APP B
N - Not within Scope of 10 CFR 50 B

Table 3.2-1

Classification of Structures, Systems, and Components For the SSSS/SPCS

(Sheet 2 of 2)

System/Component	Location	Quality Group(b)	Safety Class(c)	Code(d) Classification	Principal Construction Codes & Stds.	Seismic Category(e)	Quality Assurance(f)
SG Isolation System							
MSIV	OC	B	2	2	II	1	B
Safety Valves	OC	B	2	2	II	1	B
PORV & Block Valves	OC	B	2	2	II	1	B
SG Overfill & Block	IC	B	2	2	II	1	B
Valves	IC/OC	B	2	2	II	1	B
MS Piping	IC/OC	C	3	3	III	1	B
MF Check Valves	IC	B	2	2	B31.1	1	B
MFIV	OC	B	2	2	B31.1	1	B
MF Control Valves	OC	D	NNS	N.A.	B31.1	2	N
SFW Control Valves	OC	C	3	3	B31.1	1	B
MF Piping	IC/OC	B	2	2	B31.1	1	B
	IC/OC	C	3	3	B31.1	1	B

4.0 REACTOR

No portion of this chapter is pertinent to the RESAR-SP/90 "Secondary Side Safeguards System/Steam and Power Conversion System" module.

5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

No portion of this chapter is pertinent to the RESAR-SP/90 "Secondary Side Safeguards System/Steam and Power Conversion System".

6.0 ENGINEERED SAFETY FEATURES

Engineered safety features (ESF) systems protect the public in the event of an accidental release of radioactive fission products from the reactor coolant system (RCS), particularly as the result of postulated accidents. The safety features function to localize, control, mitigate, and terminate such accidents and to maintain radiation exposure levels to the public below applicable limits and guidelines (e.g., 10CFR 100).

The emergency feedwater system (EFWS) functions as an engineered safety system because it relied upon to provide makeup water to the steam generators for decay heat removal during accidents. The steam generator isolation system (SGIS) which is the safety class portion of the main steam and main feedwater system, also functions as an engineered safety system during accidents because it is relied upon to limit and control the release of steam and possibly radiation from the steam generator.

Heat removal from the WAPWR following reactor trip and a loss of offsite power is accomplished by the operation of several systems including the secondary side safeguards system. Similar capability is required to mitigate the consequences of certain postulated piping breaks. Such heat removal involves heat transfer from the reactor to the steam generators, resulting in the production of steam which is then released to the atmosphere. In this process, it becomes necessary to supply makeup water to the steam generators. This is accomplished by the use of the Emergency Feedwater System (EFWS).

The EFWS is designed to operate when needed, using the principles of redundancy and diversity to assure that it can function under postulated accident conditions. The system is powered by electrical or steam-driven sources. The EFWS is a safety grade, Seismic Category 1, redundant system with Class 1E electric components. The EFWS meets the additional requirements of II.E.1.1 and II.E.1.2 of NUREG-0737 and NRC Branch Technical Position (BTP) ASB 10-1.

The provision of several independent flow paths for the EFWS serves to preclude the possibility of a complete loss of function due to a single event, either occurring alone, or in conjunction with the failure of an active component. The EFWS is not categorized as a high energy system, except for the section of line which connects to the main feedwater piping which is pressurized during plant operation. Note that the entire system is not pressurized during start-up, hot standby, and shutdown because the SFWS normally performs that function.

The SGIS is designed to operate when needed to assure that it can function under postulated accident conditions. The SGIS is a safety grade, Seismic Category 1, redundant system with Class 1E electrical components.

6.1 ENGINEERED SAFETY FEATURES MATERIALS

6.1.1 Materials Selection and Fabrication

Information on the selection and fabrication of the materials in the EFWS is provided in Section 6.1 of the integrated version of RESAR-SP/90.

7.0 INSTRUMENTATION AND CONTROLS

See RESAR-SP/90 PDA Module 9, "Instrumentation and Controls and Electric Power" which provides an overall discussion of the various plant instrumentation and control systems.

8.0 ELECTRIC POWER

See RESAR-SP/90 PDA Module 9, "Instrumentation and Controls and Electric Power" for a complete description and evaluation of the WAPWR onsite electric power systems.

9.0 AUXILIARY SYSTEMS

See RESAR-SP/90 PDA Module 13, "Auxiliary Systems" for a complete description and evaluation of the auxiliary systems within the WAPWR Nuclear Power Block.

10.0 STEAM AND POWER CONVERSION SYSTEM

10.1 SUMMARY DESCRIPTION

The steam and power conversion system is designed to remove heat from the reactor coolant in the four steam generators and convert it to electrical energy. The system typically includes the following subsystems and components:

- Turbine-generator
- Main steam supply system[1]
- Main condenser
- Main condenser evacuation system
- Turbine gland sealing system
- Turbine bypass system
- Circulating water system
- Condensate cleanup system
- Condensate and feedwater system[1]
- Steam generator blowdown processing system
- Emergency feedwater system
- Startup feedwater system
- Secondary liquid waste system

The turbine cycle is a closed cycle with water as the working fluid. The heat input is provided by reactor coolant in the steam generators. Work is performed by the expansion of the steam, typically in the high and low pressure turbines. Steam is condensed and waste heat is rejected by the main condenser. The condensate and feedwater systems preheat and pressurize the water and return it to the steam generators, thereby closing the cycle.

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(1) The safety class portions of the main steam supply system and the condensate and feedwater system are designated as the steam generator isolation system (SGIS).

The WAPWR nuclear power block (NPB) scope of supply includes essentially the steam generator isolation system (SGIS) and the emergency feedwater system, which are the only safety grade subsystems of the steam and power conversion system. Included is all of the piping extending through the wall separating the steam tunnel from the turbine building; including the requisite supports anchored to this wall. The only components in the turbine building which are in the NPB scope are the control valves on the main feedwater system. All remaining components, structures, systems and equipment of the steam and power conversion system are the responsibility of the plant specific applicant.

Figure 10.1-1 is a simplified flow diagram of the WAPWR NPB SGIS. Figure 10.1.2 is a simplified flow diagram of the WAPWR NPB emergency feedwater system. Table 10.1-1 gives the major design and performance data for the WAPWR NPB scope of supply. Appropriate interface criteria and design criteria, as defined in Subsection 1.1.1.3 of RESAR-SP/90 PDA Module 3, "Introduction and Site", are provided in Appendix 10A to ensure the safe and proper operation of the overall steam and power conversion system.

10.1.1 General Discussion

A summary description, the primary function and the relevance to plant safety of each of the above listed components and subsystems are provided in the following subsections.

10.1.1.1 Turbine-Generator

The turbine-generator receives high pressure steam from the main steam supply system and converts a portion of its thermal energy into electrical energy. It also supplies extraction steam and condensate therefrom for feedwater heating and steam for driving the steam generator feedwater pump turbines.

The turbine-generator and associated piping, valves and controls are located completely within the turbine building. There are no safety-related systems or components located within the turbine building. The turbine-generator serves no safety function.

The design of the turbine-generator is the responsibility of the plant specific applicant. Additional discussion of the turbine-generator is provided in Section 10.2. Interface criteria between the NPB and the turbine-generator are provided in Appendix 10A.

10.1.1.2 Main Steam Supply System

The basic function of the main steam supply system (MSSS) is to convey steam generated in the steam generators by the reactor coolant system to the turbine-generator system and auxiliary systems for power generation. In addition the safety class portion of the MSSS is designated as the steam generator isolation system, or SGIS (see Figure 10.1-1).

The MSSS also provides steam to the turbine-generator system second stage moisture separator/reheaters, the main feed pump turbines, the steam seal system for the main turbine and the feedwater pump turbines, the turbine bypass system, the auxiliary steam reboiler, the process sampling system, and the condenser spargers. The system also dissipates heat generated by nuclear steam supply system (NSSS) by means of steam dump valves to the condenser when the turbine-generator is unavailable.

The SGIS portion of the MSSS dissipates to the atmosphere heat generated by the nuclear steam supply system by means of power operated relief valves or spring loaded safety valves. The SGIS also provides steam generator overfill protection for the purpose of mitigating steam generator tube rutpure. This function is accomplished by a connection from the steam generator upper shell to the emergency water storage tank inside the containment.

The MSSS including the SGIS contains the following major components:

- o Main steam piping from the steam generator outlet steam nozzles to the main turbine stop valves,

- One main steam isolation valve (MSIV) per main steam line,
- Main steam safety valves,
- Main steam power operated relief valves, and
- Steam generator valves and piping from the steam generator upper shell to a sparger in the emergency water storage tank (EWST).

The WAPWR NPB scope of supply, shown in Figure 10.1-1 and identified as the SGIS, includes all equipment and components in the main steam tunnel and terminates at the turbine building side of the steam tunnel/turbine building wall.

Additional discussion of the MSSS is provided in Section 10.3. Interface criteria between the NPB and those portions of the MSSS outside NPB scope are provided in Appendix 10A.

10.1.1.3 Main Condenser

The main condenser is the steam cycle heat sink. During normal operation it receives and condenses main turbine exhaust steam, steam generator feedwater pumps turbine exhaust steam, and turbine bypass steam. The main condenser is also a collection point for other steam cycle miscellaneous flows, drains and vents. In addition, the main condenser is utilized as a heat sink in the initial phase of reactor cooldown during a normal plant shutdown.

The main condenser serves no safety function.

The design of the main condenser is the responsibility of the plant specific applicant. Additional discussion of the main condenser is provided in Subsection 10.4.1. Interface criteria between the NPB and the main condenser are provided in Appendix 10A.

10.1.1.4 Main Condenser Evacuation System

Main condenser evacuation is performed by the main condenser air removal system (MCARS). This system removes noncondensable gases and air from the main condenser during plant startup, cooldown, and normal operation.

The MCARS serves no safety function.

The design of the MCARS is the responsibility of the plant specific applicant. Additional discussion of the MCARS is provided in Subsection 10.4.2. Interface criteria between the NPB and the MCARS are provided in Appendix 10A.

10.1.1.5 Turbine Gland Sealing System

The turbine gland sealing system (TGSS) prevents the escape of steam from the turbine shaft/casing penetrations and valve stems and prevents air inleakage to subatmospheric turbine glands.

The TGSS serves no safety function.

The design of the TGSS is the responsibility of the plant specific applicant. Additional discussion of the TGSS is provided in Subsection 10.4.3.

10.1.1.6 Turbine Bypass System

The turbine bypass system (TBS) has the capability to bypass main steam from the steam generators to the main condenser in a controlled manner to minimize transient effects on the reactor coolant system of startup, hot shutdown and cooldown, and the step load reduction in generator load. The TBS is also called the steam dump system.

The TBS serves no safety function.

The design of the TBS is the responsibility of the plant specific applicant. Additional discussion of the TBS is provided in Subsection 10.4.4. Interface criteria between the NPB and the TBS are provided in Appendix 10A.

10.1.1.7 Circulating Water System

The circulating water system (CWS) provides cooling water for the removal of heat from the main condensers and rejects heat to the plant's ultimate heat sink.

The CWS serves no safety function.

The design of the CWS is the responsibility of the plant specific applicant. Additional discussion of the CWS is provided in Subsection 10.4.5. See the specific site's Site Addendum for descriptions of site-specific portions of the CWS.

10.1.1.8 Condensate Cleanup System

The condensate cleanup system (CCS) functions to maintain the required purity of feedwater for the steam generators by filtration to remove corrosion products and by ion exchange to remove condenser leakage impurities.

The CCS serves no safety function.

The design of the CCS is the responsibility of the plant specific applicant. Additional discussion of the CCS is provided in Subsection 10.4.6. Water chemistry requirements are discussed in Subsections 10.3.5 and Appendix 10A; Subsection 10A.3.5. Interface criteria between the NPB and the CCS are provided in Appendix 10A.

10.1.1.9 Condensate and Feedwater System

The condensate and feedwater system (CFS) functions to receive condensate from the condenser hotwells and deliver feedwater, at required pressure and temperature, to the four steam generators. In addition the safety class portion of the CFS is designated as the steam generator isolation system (see Figure 10.1-1).

The SGIS portion of the CFS consists of the main feed piping, valves and associated instrumentation and controls from the steam generator inlet nozzle to the turbine building side of the wall separating the steam tunnel and the turbine building. The main feedwater control valves and flow instruments are also part of the SGIS. All of these components are within the WAPWR NPB scope of supply.

The remaining portions of the CFS consist of the piping, valves, pumps and associated instrumentation and controls from the suction side of the condenser hotwell to, but not including, the main feedwater control valves located upstream of the steam tunnel/turbine building wall. The design of all of this equipment is the responsibility of the plant specific applicant. None of this equipment is safety-related.

Additional discussion of the CFS is provided in Subsection 10.4.7. Interface criteria between the NPB and the CFS are provided in Appendix 10A.

10.1.1.10 Steam Generator Blowdown Processing System

The steam generator blowdown processing system (SGBPS) helps to maintain the steam generator secondary side water within the prescribed chemical specifications. Heat is recovered from the blowdown and returned to the feedwater system. The blowdown is then treated to remove impurities before being returned to the condenser.

Those portions of the SGBPS which are safety-related are included in the SGIS and are within the scope of the WAPWR NPB. Included are all of the piping, valves, instrumentation and controls from the steam generator blowdown nozzle through the steam tunnel/turbine building wall, including the containment isolation valve. The remaining equipment and components, which process the blowdown are not safety-related and the design is the responsibility of the plant specific applicant.

Additional discussion of the SGBPS is provided in Subsection 10.4.8. Interface criteria between the NPB and the SGBPS are provided in Appendix 10A.

10.1.1.11 Emergency Feedwater System

In a conventional Westinghouse PWR the auxiliary feedwater (AFS) functions to supply a reliable source of water for the steam generators during normal shutdowns and accidents. The AFS functions to remove thermal energy from the reactor coolant system through the steam generator to the atmosphere. A typical AFS provides emergency water following any accident. This system may also be used following a reactor shutdown in conjunction with the condenser dump valves to cool the reactor coolant system to hot shutdown, at which temperature the residual heat removal system is brought into operation.

For the WAPWR, the above safety and control functions are performed by two secondary side systems; the emergency feedwater system (EFWS) which is discussed below and in Subsection 10.4.9, and the startup feedwater system (SFWS) which is discussed in Subsections 10.1.1.12 and 10.4.10, and in Appendix 10A.

The EFWS functions to provide feedwater to the steam generators following transients or accidents such as reactor trip, loss of main feed, steam or feed line breaks, steam generator tube ruptures, and any time the main and startup feedwater systems are not available.

The EFWS consists of two identical subsystems, each of which receives electrical power from one of two separate safety class 1E electrical power trains. Each subsystem consists of an emergency feedwater storage tank (EFWST), one motor driven emergency feedwater pump, one turbine driven emergency feedwater pump and the required piping, valves, instruments and controls necessary for system operation (see Figure 10.1-2).

The EFWS is a safety related system and is entirely within the scope of the WAPWR NPB. Interface criteria between the EFWS and the plant specific applicant's scope of supply is provided in Appendix 10A.

Additional discussion of the EFWS is provided in Subsection 10.4.9.

10.1.1.12 Startup Feedwater System

During normal power operation the steam generators are fed by the main feedwater pumps with the steam produced being sent to the turbine. For the WAPWR, a startup feedwater system (SFWS) is included to feed the steam generators during normal plant startup and shutdown. Under these conditions, steam from the steam generators is sent to the main condenser. The SFWS is also actuated automatically to provide feedwater following a reactor trip, loss of main feed, loss of offsite power, and other anticipated transients.

The SFWS is a control grade system which, although not required to mitigate the consequences of postulated accidents, provides additional reliability and diversity to the EFWS. The SFWS also serves to minimize the number of EFWS actuations required which reduces the probability of EFWS failures.

The major components of the SFWS is one motor driven feedwater pump and associated pipes and valves. The normal suction source is anticipated to be a deaerating heater in the CFS, however, that portion of the CFS is not in the scope of the NPB. Refer to Appendix 10A for interface criteria. The components are located in the turbine building.

As indicated above, the SFWS performs no safety-related function. The design of the SFWS is the responsibility of the plant specific applicant.

Additional discussion of the SFWS is provided in Subsection 10.4.10. Detailed design criteria and interface criteria between the NPB and the SFWS are provided in Appendix 10A.

10.1.1.13 Secondary Liquid Waste System

The secondary liquid waste system (SLWS) functions to process condensate demineralizer regeneration wastes and potentially radioactive liquid waste collected in the turbine building.

The SLWS serves no safety function.

The design of the SLWS is the responsibility of the plant specific applicant. Additional discussion of the SLWS is provided in Subsection 10.4.11.

(a,c)

FIGURE 10.1-1 THE STEAM GENERATOR ISOLATION SYSTEM

(a,c)

FIGURE 10.1-2 EMERGENCY FEEDWATER SYSTEM

10.2 TURBINE GENERATOR

The turbine-generator (T-G) receives high pressure steam from the main steam system and converts a portion of the thermal energy into electrical energy. The T-G also supplies extraction steam and condensate therefrom for feedwater heating and steam for driving the steam generator feedwater pump turbines.

The design of the T-G is the responsibility of the plant specific applicant. The design must be compatible with the interface criteria given in Appendix 10A; Section 10A.2.

10.2.1 Design Bases

10.2.1.1 Safety Design Bases

The T-G serves no safety function and has no safety design basis.

10.2.1.2 Power Generation Design Bases

See the plant specific applicant's safety analysis report for a discussion of T-G power generation design bases.

10.2.2 System Description

The T-G system including its associated piping, valves and controls is located completely within the turbine building. There are no safety-related systems or components located within the turbine building, hence any failures associated with the T-G system will not affect any safety-related equipment.

See the plant specific applicant's safety analysis report for a description of the T-G system, its components, and system operation.

10.2.3 Turbine Disk Integrity

See the plant specific applicant's safety analysis report for a discussion of turbine disk integrity.

10.2.4 Evaluation

For the WAPWR there is no significant radioactive contaminants present in the T-G system during normal operation. No radiation shielding is required for the T-G system. Even in the event of a large primary-to-secondary steam generator leak, the T-G system will not become contaminated to the extent that access is precluded.

A discussion of the radiological aspects of primary-to-secondary leakage is included in RESAR-SP/90 PDA Module 12, "Waste Management".

See the plant specific applicant's safety analysis report for further details of the T-G system evaluation.

10.3 MAIN STEAM SUPPLY SYSTEM

The basic function of the main steam supply system (MSSS) is to convey steam generated in the steam generators by the reactor coolant system to the turbine-generator system and auxiliary systems for power generation. In addition the safety class portion of the MSSS is designated as the steam generator isolation system, or SGIS (see Figure 10.1-1).

The MSSS also provides steam to the turbine-generator system second stage moisture separator/reheaters, the main feed pump turbines, the steam seal system for the main turbine and the feedwater pump turbines, the turbine bypass system, the process sampling system, and the condenser spargers. The system also dissipates heat generated by nuclear steam supply system (NSSS) by means of steam dump valves to the condenser when the turbine-generator is unavailable. The SGIS portion of the MSSS dissipates to the atmosphere heat generated by the nuclear steam supply system by means of the power operated relief valves or spring loaded safety valves. The SGIS also provides steam generator overfill protection for the purpose of mitigating steam generator tube rupture. This function is accomplished by a connection from the steam generator upper shell to the emergency water storage tank inside containment.

The WAPWR NPB scope of supply, shown in Figure 10.3-1 and identified as the SGIS, includes all equipment and components in the main steam tunnel and terminates on the turbine building side of the steam tunnel/turbine building wall. The design of all remaining equipment and components is the responsibility of the plant specific applicant and must be compatible with the interface criteria given in Appendix 10A, Section 10A.3.

10.3.1 Design Bases

10.3.1.1 Safety Design Bases

The SGIS which is the WAPWR NPB scope of supply constitutes the safety related portion of the MSSS and is required to function following a DBA, and to achieve and maintain the plant in a safe shutdown condition.

SAFETY DESIGN BASIS ONE - The SGIS portion of the MSSS is protected from the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, and external missiles (GDC-2).

SAFETY DESIGN BASIS TWO - The SGIS portion of the MSSS is designed to remain functional after a SSE or to perform its intended function following postulated hazards of fire, internal missile, or pipe break (GDC-3 and 4).

SAFETY DESIGN BASIS THREE - Component redundancy is provided so that safety functions can be performed, assuming a single active component failure coincident with the loss of offsite power (GDC-34).

SAFETY DESIGN BASIS FOUR - The SGIS is designed so that the active components are capable of being tested during plant operation. Provisions are made to allow for inservice inspection of components at appropriate times specified in the ASME Boiler and Pressure Vessel Code, Section XI.

SAFETY DESIGN BASIS FIVE - The SGIS uses design and fabrication codes consistent with the quality group classification assigned by Regulatory Guide 1.26 and the seismic category assigned by Regulatory Guide 1.29. The power supply and control functions are in accordance with Regulatory Guide 1.32.

SAFETY DESIGN BASIS SIX - The SGIS provides for isolation of the secondary side of the steam generator to deal with leakage or malfunctions and to isolate nonsafety-related portions of the MSSS.

SAFETY DESIGN BASIS SEVEN - The SGIS provides means to dissipate heat generated in the reactor coolant system during hot shutdown and cooldown (GDC-34).

SAFETY DESIGN BASIS EIGHT - The SGIS provides an assured source of steam to operate the turbine-driven emergency feedwater pumps for reactor cooldown under emergency conditions and for shutdown operations (GDC-34).

10.3.1.2 Power Generation Design Bases

POWER GENERATION DESIGN BASIS ONE - The MSSS is designed to deliver steam from the steam generators to the turbine-generator system for a range of flows and pressures varying from warmup to rated conditions. The system provides means to dissipate heat during plant step load reductions and during plant startup. It typically also provides steam to:

a. The turbine-generator system second stage reheaters
b. The main feed pump turbines
c. The steam seal system
d. The turbine bypass system
e. The auxiliary steam reboiler
f. The process sampling system
g. Condenser spargers

See the plant specific applicant's safety analysis report for additional MSSS power generation design bases.

10.3.2 System Description

The MSSS including the SGIS conveys steam from the steam generators to the turbine-generator system. The system consists of main steam piping, power-operated relief valves, safety valves, and main steam isolation valves. The turbine bypass system is discussed in detail in Subsection 10.4.4. The piping and instrumentation diagram for the SGIS which is the WAPWR NPB scope of supply is shown in Figure 10.3-1 (Sheets 1 through 4). Also included in this figure is the WAPWR NPB portion of the condensate and feedwater system which is discussed in Subsection 10.4.7. Table 10.3-1 lists pertinent SGIS design data.

10.3.2.1 Component Description

Codes and standards applicable to the WAPWR NPB portion of the MSSS are listed in Table 3.2-1. The WAPWR scope of supply is designed and constructed in accordance with quality group B and seismic category I requirements.

10.3.2.1.1 Main Steam Piping

Saturated steam from the four steam generators is conveyed to the turbine-generator by four 32 inch O.D. lines. The lines are sized for a pressure drop of 25 psi from the steam generators to the turbine stop valves at turbine manufacturer's guaranteed conditions. That portion of the main steam piping from the steam generator outlet nozzles, through the steam tunnel, and terminating on the turbine building side of the steam tunnel/turbine building wall is supplied by Westinghouse. The remaining piping is the responsibility of the plant specific applicant (see Figure 10.3-1).

Each of the main steam lines is anchored at the containment wall and has sufficient flexibility to provide for relative movement of the steam generators due to thermal expansion.

Each line is equipped with:

a. One steam generator power operated relief valve and block valve,

b. Five spring loaded safety valves,

c. One main steam isolation valve and associated bypass isolation valve, and

d. One low point drain, which is piped by the plant specific applicant to the condenser through a drain valve.

All of these valves are supplied by Westinghouse.

All main steam branch process line connections are made downstream of the isolation valves with the exception of the line to the power operated atmospheric relief valve, connections for the safety valves, lines to the emergency feedwater pump turbine, and low point drains and high point vents.

Each steam generator outlet nozzle contains a flow restrictor consisting of several venturis, with a total throat area of approximately [ ] square feet (a,c) to limit flow in the event of an MSLB.

See the applicant's safety analysis report for a description of the main steam piping outside the scope of the WAPWR NPB.

10.3.2.1.2 Power Operated Atmospheric Relief Valve

A power operated atmospheric relief valve is installed on the main steam piping of each steam generator outside containment. The valve discharges to the atmosphere through a silencer. The four valves are installed to provide for controlled removal of reactor decay heat during normal or post accident reactor cooldown when the main steam isolation valves are closed or the turbine bypass system is not available. The valves will pass sufficient flow at all pressures to achieve a 50°F per hour plant cooldown rate. The maximum actual capacity of the relief valve at design pressure is limited to reduce the magnitude of a reactor transient if one valve would inadvertently open and remain open.

The atmospheric relief valves are fast acting solenoid operated, carbon steel, 6 inch, [ ] pound globe valves powered by Class 1E DC sources. The valves (a,c) automatically open and close on high steam line pressure; 2 out of 4 logic. The capability for remote manual valve operation is provided in the main control room and at the shutdown control panel.

10.3.2.1.3 Steam Generator Power Operated Relief Valve Block Valve

In series with each power operated atmospheric relief valve is a block valve. (a,c) The block valve is a motor operated 6 inch [ ] pound globe valve powered by Class 1E AC power sources. These valves provide the capability of isolating the PORV in case it is leaking. For the case where the PORV fails to open the block valve is automatically closed on a low-low steam line pressure; 2 out of 4 logic.

In addition these block valves can be positioned to provide throttling of the steam release to enhance the operability of the SGIS during these situations where the condenser dump is not available.

10.3.2.1.4 Safety Valves

The spring-loaded main steam safety valves provide overpressure protection in accordance with the ASME Section III code requirement for the secondary side of the steam generators and the main steam piping. There are five valves installed in each main steam line. Table 10.3-1 identifies the valves, their set pressure, and capacities. The valves discharge directly to the atmosphere via vent stacks. The maximum actual capacity of the safety valves at the design pressure is limited to reduce the magnitude of a reactor transient if one of the valves would open and remain open.

10.3.2.1.5 Main Steam Isolation Valves and Bypass Valves

One MSIV and associated bypass valve are installed in each of the four main steam lines outside the containment and downstream of the safety valves. The MSIVs are installed to prevent uncontrolled blowdown from more than one steam generator. The valves isolate the nonsafety-related portions from the safety-related portions of the system. The valves are gate valves with pneumatic/hydraulic operators. Stored energy for closing is supplied by high pressure gas. The MSIV is maintained in a manually open position by high pressure hydraulic fluid. For emergency closure, redundant solenoids are energized which cause the high pressure hydraulic fluid to be dumped to a fluid reservoir. The redundant electrical solenoids are energized from separate Class 1E sources. The valves are designed to close in less than 5 seconds against the flows associated with line breaks on either side of the valve, assuming the most limiting normal operating conditions prior to occurrence of the break.

The main steam bypass valve is used when the MSIVs are closed to permit warming of the main steam lines prior to startup. The bypass valves are air operated globe valves and are normally closed. For emergency closure, either of two separate solenoids, when de-energized, will result in valve closure. Electrical solenoids are energized from a separate Class 1E source.

10.3.2.1.6 Steam Generator Overfill Control Valves

Each steam generator has a line from the upper shell to the EWST (IRC) containing two parallel isolation valves. These steam generator overfill valves are normally closed fast acting, solenoid operated, globe valves. These 3 inch, carbon steel, [ ] pound valves are powered by Class 1E DC (a,c)
sources and fail closed on loss of DC power.

The capacity of one valve is sufficient to prevent steam generator overfill assuming the rupture of one steam generator tube and maximum EFWS flow. The valves automatically open and close on a high steam generator level signal based on a 2 out of 4 logic.

10.3.2.1.7 Steam Generator Overfill Block Valves

Each steam generator has an overfill block valve that is in series with the two overfill control valves. These overfill block valves are normally open, motor operated gate valves. These 4 inch, carbon steel, [ ] pound valves are (a,c)
powered by Class 1E AC sources.

10.3.2.2 System Operation

NORMAL OPERATION - At low plant power levels, the MSSS typically supplies steam to the steam generator feedwater pump turbines, the auxiliary steam reboiler, and the turbine steam seal system. At high power levels, these components are typically supplied from turbine extraction steam. Steam is typically supplied to the second stage steam reheaters in the T-G system when the T-G load exceeds 15 percent.

If a large, rapid reduction in T-G load occurs, steam is bypassed (40 percent of VWO) directly to the condenser via the turbine bypass system. The system is capable of accepting a 50 percent load rejection without reactor trip and a full load rejection without lifting safety valves. If the turbine bypass system is not available, steam is vented to the atmosphere via the power operated relief valves (PORV) and the safety valves, as required.

EMERGENCY OPERATION - In the event that the plant must be shut down and offsite power is lost, the MSIV and other valves (except to the emergency feedpump turbines) associated with the main steam lines are closed. The PORV may be employed to remove decay heat and to lower the steam generator pressure to achieve cold shutdown. If the steam generator PORV for an individual main steam line is not operable, the associated safety valves will provide overpressure protection. The remaining PORVs are sufficient to achieve cold shutdown.

In the event that a DBA occurs which results in a steam line isolation signal (i.e., large steam line break), the MSIV automatically closes. Steam is automatically provided to each emergency feedwater pump turbine from one steam line upon low-low level in two steam generators. The closure of three out of four MSIVs will ensure that no more than one steam generator can supply a postulated break. In addition, closure of the HP turbine steam stop and steam control valves prevents uncontrolled blowdown of more than one steam generator following a postulated main steam line break inside the containment. Coordinated operation of the emergency feedwater system (refer to Subsection 10.4.9) and PORV or safety valve may be employed to remove decay heat.

10.3.3 Safety Evaluation

Safety evaluations are numbered to correspond to the safety design bases of Section 10.3.1.1.

SAFETY EVALUATION ONE - The SGIS portion of the MSSS is located in the reactor external building. This building is designed to withstand the effects of earthquakes, tornadoes, hurricanes, floods, external missiles, and other appropriate natural phenomena. Sections 3.3, 3.4, 3.5, 3.7 and 3.8 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design" provide the bases for the adequacy of the structural design of these buildings.

SAFETY EVALUATION TWO - The SGIS portion of the MSSS is designed to remain functional after a SSE. Subsection 3.7.2 and Section 3.9 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design" provide the design loading conditions that are considered.

SAFETY EVALUATION THREE - As indicated by Table 10.3-2, no single failure will compromise the system's safety functions. All vital power can be supplied from either onsite or offsite power systems, as described in Chapter 8.0 of RESAR-SP/90 PDA Module 9, "I&C and Electric Power".

SAFETY EVALUATION FOUR - The MSSS is initially tested consistent with the programs given in Chapter 14.0 of RESAR-SP/90 PDA Module 14, "Initial Test Program" and Chapter 14 of the plant specific applicant's safety analysis report. Periodic inservice functional testing is done in accordance with Subsection 10.3.4.

Section 6.6 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design" provides the ASME Boiler and Pressure Vessel Code, Section XI requirements that are appropriate for the MSSS.

SAFETY EVALUATION FIVE - Section 3.2 delineates the quality group classification and seismic category applicable to the SGIS portion of this system and supporting systems. Table 10.3-1 shows that the components meet the design and fabrication codes given in Section 3.2. All the power supplies and controls necessary for SGIS functions of the MSSS are Class 1E, as described in Chapters 7.0 and 8.0 of RESAR-SP/90 PDA Module 9, "I&C and Electric Power".

SAFETY EVALUATION SIX - Redundant power supplies and power trains operate the MSIVs to isolate safety and nonsafety-related portions of the system. Branch lines upstream of the MSIV contain normally closed, power-operated relief valves which open and close on steam line pressure. The atmospheric relief valves fail closed on loss of DC power, and the safety valves provide the overpressure protection.

Accidental releases of radioactivity from the MSSS are minimized by the negligible amount of radioactivity in the system under normal operating conditions. Additionally, the steam generator overfill control valves provide the capability of reducing accidental releases following a steam generator tube rupture.

Detection of radioactive leakage into and out of the system is facilitated by main steam line radiation monitoring, area radiation monitoring, process radiation monitoring and steam generator blowdown sampling.

SAFETY EVALUATION SEVEN - Each main steam line is provided with safety valves that limit the pressure in the line to preclude overpressurization and remove stored energy. Each line is provided with a power operated relief valve to permit reduction of the main steam line pressure and remove stored energy to achieve an orderly shutdown. The emergency feedwater system, which is described and evaluated in Subsection 10.4.9, provides makeup to the steam generators consistent with the steaming rate.

SAFETY EVALUATION EIGHT - The steam line to each emergency feedwater pump turbine is connected to one main steam line upstream of the MSIV. This arrangement ensures a supply of steam to this turbine when the steam generators are isolated. The emergency feedwater system is described and evaluated in Subsection 10.4.9.

10.3.4 Inspection and Testing Requirements

10.3.4.1 Preservice Valve Testing

The set pressures of the safety valves are individually checked d ıng initial startup either by bench testing or with a pneumatic test device. A pneumatic test device is attached to the valve stem. The pneumatic pressure is applied until the valve seat just lifts, as indicated by the steam noise. Combination of the steam pressure and pneumatic pressure with calibration data furnished by the valve manufacturer verifies the set pressure.

The set pressure of each PORV is checked by checking the accuracy of the pressure instruments which control it.

The MSIVs are checked for closing time prior to initial startup.

10.3.4.2 Preservice System Testing

Preoperational testing is described in Chapter 14.0 of RESAR-SP/90 PDA Module 14, "Initial Test Program" and Chapter 14 of the plant specific applicant's safety analysis report.

The SGIS is designed to include the capability for testing through the full operational sequence that brings the system into operation for reactor shutdown and for MSLB accidents, including operation of applicable portions of the protection system and the transfer between normal and standby power sources.

The safety related components of the system, i.e., v[illegible] piping and instrumentation, are designed and located to permit preservice and inservice inspections to the extent practicable.

10.3.4.3 Inservice Testing

The performance, and structural and leaktight integrity of all system components are demonstrated by continuous operation.

The redundant actuator power trains of each MSIV are subjected to the following tests:

a. Closure time - The valves are checked for closure time at each refueling.

b. On-line operability - While the MSSS is in operation, the operability of the actuator system is checked periodically by exercising the valve to approximately 90 percent of full open.

Additional discussion of inservice inspection of ASME Code Class 2 and 3 components is contained in Section 6.6 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design".

10.3.5 Secondary Water Chemistry (PWR)

10.3.5.1 Chemistry Control Basis

Steam generator secondary side water chemistry control is typically accomplished by:

a. A close control of the feedwater chemistry to limit the amount of impurities that can be introduced into the steam generator.

b. The capability of a continuous blowdown of the steam generators to reduce concentrating effects of the steam generator.

c. Chemical addition to establish and maintain an environment that minimizes system corrosion.

d. Post-construction cleaning of the feedwater system.

e. Minimizing feedwater oxygen content prior to entry into the steam generator by deaeration in the hotwell and a deaerating heater.

f. The capability of continuous demineralization and filtration of the condensate system through full flow, deep bed condensate demineralizers.

Secondary water chemistry is based on the all volatile treatment (AVT) method. This method employs the use of volatile additives to maintain system pH and to scavenge dissolved oxygen present in the feedwater. Ammonia is added to establish and maintain alkaline conditions in the feed train. Although ammonia is volatile and will not concentrate in the steam generator, it will reach an equilibrium level which will establish an alkaline condition in the steam generator.

Hydrazine is added to scavenge dissolved oxygen present in the feedwater. Hydrazine also tends to promote the formation of a protective oxide layer on metal surfaces by keeping these layers in a reduced chemical state.

Both ammonia and hydrazine can be injected continuously at the discharge headers of the condensate pumps and are added, as necessary, for chemistry control.

Operating chemistry guidelines for secondary steam generator water have been developed using EPRI guidelines. Maintaining secondary water chemistry is the responsibility of the plant specific applicant. See Subsection 10.4.6 for a discussion of the condensate cleanup system, the design of which is the responsibility of the plant specific applicant. See Appendix 10A, Subsection 10A.4.6, which provides the secondary water chemistry interface criteria which the plant specific applicant must meet.

10.3.5.2 Corrosion Control Effectiveness

Alkaline conditions in the feedtrain and the steam generator reduce general corrosion at elevated temperatures and tend to decrease the release of soluble corrosion products from metal surfaces. These conditions promote formation of a protective metal oxide film and thus reduce the corrosion products released into the steam generator.

Hydrazine also promotes formation of a metal oxide film by the reduction of ferric oxide to magnetite. Ferric oxide may be loosened from the metal surfaces and be transported by the feedwater. Magnetite, however, provides an adhesive, protective layer on carbon steel surfaces. Hydrazine also promotes formation of protective metal oxide layers on copper surfaces. Removal of oxygen from the secondary waters is also essential in reducing corrosion. Oxygen dissolved in water causes general corrosion that can result in pitting of ferrous metals, particularly carbon steel. Oxygen is removed from the steam cycle condensate in the main condenser deaerating section. Additional oxygen protection is obtained by chemical injection of hydrazine into the condensate steam. Maintaining a residual level of hydrazine in the feedwater ensures that any dissolved oxygen not removed by the main condenser is scavenged before it can enter the steam generator.

The presenee of free hydroxide (OH) can cause rapid corrosion (caustic stress corrosion) if it is allowed to concentrate in a local area. Free hydroxide is avoided by maintaining proper pH control and by minimizing impurity ingress into the steam generator.

AVT control is a technique whereby both soluble and insoluble solids are kept at a minimum within the steam generator. This is accomplished by maintaining strict surveillance over the possible sources of feedtrain contamination (e.g., main condenser cooling water leakage, air inleakage, and subsequent corrosion product generation in the low pressure drain system, etc.). Solids are also excluded, as discussed above, by injecting only volatile chemicals to establish conditions that reduce corrosion and, therefore, reduce transport of corrosion products into the steam generator.

In addition to minimizing the sources of contaminants entering the steam generator, condensate demineralizers are used, and a continuous blowdown from the steam generators is employed to limit the concentration of contaminants. With the low solids level that results from employing the above procedures, the accumulation of scale and deposits on steam generator heat transfer surfaces and internals is limited. Scale and deposit formations can alter the thermal hydraulic performance in local regions which creates a mechanism that allows impurities to concentrate and thus possibly cause corrosion. The effect of this type of corrosion is reduced by limiting the ingress of solids into the steam generator and limiting their buildup.

The chemical additives, because they are volatile, do not concentrate in the steam generator and do not represent chemical impurities that can themselves cause corrosion.

See Appendix 10A, Subsection 10A.4.6 for the specific AVT chemistry guidelines.

10.3.6 Steam and Feedwater System Materials

10.3.6.1 Fracture Toughness

Compliance with fracture toughness requirements of ASME III, Articles NC-2300 and ND-2300 for WAPWR NPB MSSS scope of supply is discussed in Section 6.1 of the integrated RESAR-SP/90 PDA document.

10.3.6.2 Material Selection and Fabrication

All pipe, flanges, fittings, valves, and other piping material for WAPWR NPB MSSS scope of supply conform to the referenced ASME, ASTM, ANSI or MSS-SP Code.

Expendable materials that may come into contact with metallic surfaces do not contain the following as a basic and essential chemical constituent: copper, zinc, lead, mercury, cadmium, and other low melting point metals, their alloys and/or compounds.

The following code requirements apply:

	Stainless Steel	Carbon Steel
Pipe	ANSI B36.19	ANSI B36.10
Fittings	ANSI B16.9, B16.11 or B16.28	ANSI B16.9, B16.11 or B16.28
Flanges	ANSI B16.5	ANSI B16.5

The following ASME Material Specifications apply specifically:

ASME SA-155 GR KCF 70 Class 1 (impact tested)
ASME SA-155 GR KCF 70 Class 1
ASME SA-106, GR C (impact tested)
ASME SA-106, GR, B
ASME SA-106, GR, B (normalized)
ASME SA-234 GR WPB
ASME SA-234 GR WPBW (Mfd from gr 70 plate)
ASME SA-234 GR WPC
ASME SA-105
ASME SA-193 GR B7
ASME SA-194 GR 2H

ASME SA-216 GR WCB

ASME SA-333 GR 6 (impact tested)

ASME SA-420 GR WPL6 (impact tested)

ASME SA-508 Class 1 (impact tested)

ASME SA-312, TP 304

ASME SA-403, WP-304

ASME SA-403, WP-304 W

ASME SA-182, F-304

ASME SA 672 GR C70

Compliance with the following Regulatory Guides is discussed in Section 6.1 of the integrated RESAR-SP/90 PDA document:

Regulatory Guide 1.31 - Control of Stainless Steel Welding

Regulatory Guide 1.36 - Nonmetallic Thermal Insulation for Austenitic Stainless Steel

Regulatory Guide 1.37 - Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants

Regulatory Guide 1.44 - Control of the Use of Sensitized Stainless Steel

Regulatory Guide 1.50 - Control of Preheat Temperatures for Welding of Low-Alloy Steels

Regulatory Guide 1.71 - Welder Qualification for Areas of Limited Accessibility

Regulatory Guide 1.85 - Material Code Case Acceptability - ASME Section III, Division I

TABLE 10.3-1 (Sheet 1 of 2)

MAIN STEAM SUPPLY SYSTEM DESIGN DATA
(W APWR NPB Scope of Supply)

Main Steam Piping (Safety-Related Portion)

Total design flowrate at 1,000 psia and 0.25 percent moisture, lb/hr	17,080,000
Number of lines	4
O.D., in.	32
Design pressure, psia	[ ](a,c)
Design temperature, F	
Design code	ASME Section 3, Class 2
Seismic design	Category I

Main Steam Isolation Valves

Number per main steam line	1
Type	Gate valve with pneumatic/ hydraulic actuator. (Closed by self-contained high pressure gas.)
Size, in.	32
Closing time, seconds	5
Design code	ASME Section 3, Class 2
Seismic design	Category I

Main Steam Power-operated Relief Valves

Number per main steam line	1
Type	Globe valves with solenoid operator, fail closed.
Size, in.	6
Normal set pressure, psig	[ ](a,c)
Maximum capacity (each) at 1300 psia, lb/hr	
Minimum capacity (each) at 100 psia, lb/hr	
Design code	ASME Section 3, Class 2
Seismic design	Category I

Main Steam Safety Valves

Number of main steam line	5
Type	Pressure actuated safety/relief valves
Size, in.	6
Design code	ASME Section 3, Clas 2
Seismic design	Category I

TABLE 10.3-1 (Sheet 2 of 2)

MAIN STEAM SUPPLY SYSTEM DESIGN DATA
(W APWR NPB Scope of Supply)

Main Steam Safety Valves (Cont.)

Number	Set Pressure (psig)	Capacity at 3-Percent Accumulation (lb/hr)	
1			(a,c)
2			
3			
4			
5			

TABLE 10.3-2 (Sheet 1 of 9)

SGIS (Including MSSS, CFS, SGBPS)
SINGLE ACTIVE FAILURE ANALYSIS

Component	Failure Mode	Plant Mode	Effect on System Operation	Failure Detection
1. Main Steam Line A Power Operated Relief Valve 1-PCV-1864, normally closed solenoid operated globe valves. (Typical of all four steam generators.)	Fails to open when steam pressure exceeds setpoint.	1a. All.	Steam pressure is limited to 110% of design by the safety valves (1-9730A, 1-9731A, 1-9732A, 1-9733A). Cooldown capability is maintained with the other steam generators, or with the relief path to the EWST via 1-9783A and 1-9784A.	Steam generator p indication, PORV position indicati
	Fails to close after opening, when steam pressure falls below closure setpoint.	1b. All.	Blowdown path will be automatically isolated by the PORV block valve, 1-9727A. No adverse impact on system operation.	Steam generator pressure, PORV pc indication, block position indicati
	Opens spuriously.	1c. All shutdown.	Blowdown path will be automatically isolated by the PORV block valve, 1-9727A. No adverse impact on system operation.	Steam generator pressure, PORV pc indication, block position indicati
	Opens spuriously.	1d. At power.	Operator must manually isolate PORV block valve to stop blowdown. No adverse impact on system operation.	PORV position indication, block valve position indication.

TABLE 10.3-2 (Sheet 2 of 9)

SGIS (Including MSSS, CFS, SGBPS)
SINGLE ACTIVE FAILURE ANALYSIS

Component	Failure Mode	Plant Mode	Effect on System Operation	Failure Detection
2. Steam generator PORV block/flow control valve 1-9727A, normally open motor operated globe valve. (Typical of all four steam generators.)	Fails to open on demand.	2a. All.	Failure effects identical to case 1a. above.	PORV block valve position indicati steam generator p instrumentation.
	Fails to limit flow on demand.	2b. All.	Blowdown can be isolated by the steam generator PORV. Heat removal can be controlled with other steam generator PORV block/control valves, or by on/off operation of the PORV.	PORV position indication, PORV valve position indication.
	Closes spuriously.	2c. All.	Failure effects identical to case 1a. above.	PORV block valve position indicati steam generator p instrumentation.
3. Steam generator A main steam line safety valve 1-9730A. (Analogous for 1-9731A, 1-9732A, 1-9733A, 1-9734A.) (Typical of all four steam generators.)	Fails to close after opening.	3a. All	Steam release to atmosphere cannot be stopped. The operator can mitigate the problem be closing the main and emergency feedwater lines to the affected steam generator, and closing the main steam isolation valves.	Steam line pressure will dec

TABLE 10.3-2 (Sheet 3 of 9)

SGIS (Including MSSS, CFS, SGBPS)
SINGLE ACTIVE FAILURE ANALYSIS

Component	Failure Mode	Plant Mode	Effect on System Operation	Failure Detectio
4. Steam Generator A overfill control valve 1-9783A, normally closed, fail closed solenoid operated globe valve. (Analogous for 1-9784A.) (Typical of all four steam generators.)	Fails to open when steam generator level exceeds high level setpoint.	4a. Steam generator tube rupture, steam generator A.	Overfill control valve 1-9784A is a redundant, parallel valve which will open and prevent overfilling of the steam generator. Redundancy is reduced. No adverse impact on system safety function.	Valve position indication.
	Fails to close after opening, when steam generator level below closure setpoint.	4b. Steam generator tube rupture, steam generator A.	Operator can manually close block valve 1-9780A, stopping the blowdown. No impact on system safety function.	Valve position indication, bloc position indicat Steam generator
	Opens spuriously.	4c. All shutdown.	Operator can manually close block valve 1-9780A. No adverse impact on system operation.	Steam generator level valve posi indication, bloc position indication, EWST temperture, EWST level.
	Opens spuriously.	4d. All at power.	Operator can manuall close block valve 1-9780A. No adverse impact on system operation.	Valve position indication, block valve position indication, EWST temperature, EWST level.

TABLE 10.3-2 (Sheet 4 of 9)

SGIS (Including MSSS, CFS, SGBPS)
SINGLE ACTIVE FAILURE ANALYSIS

Component	Failure Mode	Plant Mode	Effect on System Operation	Failure Detection Met
5. Steam Generator A overfill control valve block valve 1-9780A, normally open motor operated gate valve. (Typical of all four steam generators.)(1)	Fails to close on demand.	5a. All.	Overfill control valve can be closed. No impact on system safety function. Redundancy is reduced.	PORV position indication, PORV bloc valve position indication.

Note (1): This valve is normally open, with power locked out. Spurious closure is considered to be incredible.

SGIS (Including MSSS, CFS, SGBPS)
SINGLE ACTIVE FAILURE ANALYSIS

Component	Failure Mode	Plant Mode	Effect on System Operation	Failure Detection
6. Steam generator A Main Steam Isolation Valve 1-9740A, normally open fail closed hydraulically/ pneumatically operated valve. (Analagous for the other three steam generators.)	Fails to close on demand.	6a. Steam line break between steam generator A and 1-9740A.	No impact on system safety function. The faulted steam generator A will blow down through the break, but the other three steam generators will be isolated by their associated MSIV's.	Valve position indication.
		6b. Steam line break downstream of 1-9704A. (Faulted main steam system).	Failure of 1-9740A will allow steam generator A to blow down, but the other three steam generators will be isolated from the break by their associated MSIV's.	Valve position indication, build' temperature/humid' pressure instrumer
		6c. Steam line break between steam generator B and its associated MSIV. (Steam generators C and D are analogous.)	Faulted steam generator B will blow down out the break but the other three steam generators will be isolated from the break by the MSIV associated with steam generator B. The other steam generators must be prevented from blowing down to the	Valve position indication.

SGIS (Including MSSS, CFS, SGBPS)
SINGLE ACTIVE FAILURE ANALYSIS

Component	Failure Mode	Plant Mode	Effect on System Operation	Failure Detection
6. Steam generator A Main Steam Isolation Valve 1-9740A, normally open fail closed hydraulically/ pneumatically operated valve. (Analagous for the other three steam generators.) (cont.)			turbine by closure of the turbine stop valves and/or turbine throttle valves which are high quality, redundant non-safety grade valves. In addition, all other major flow paths from the main steam system (e.g. motive steam to the turbine driven main feedwater pumps; the steam dump line) must be isolated.	

TABLE 10.3-2 (Sheet 7 of 9)

SGIS (Including MSSS, CFS, SGBPS)
SINGLE ACTIVE FAILURE ANALYSIS

Component	Failure Mode	Plant Mode	Effect on System Operation	Failure Detection Me
7. Steam generator A Main Feedwater Isolation Valve 1-9757A, normally open fail closed pneumatically/ hydraulically operated valve. (Typical of all four steam generators.)	Fails to close on demand.	7a. Feed line break, steam generator A.	No impact on plant safety. The emergency feedwater system will function normally to provide adequate feedwater to the other three steam generators.	Valve position indication, main feedwater flow indication.
		7b. Steam line break, steam generator A.	Overfilling of steam generator A must be prevented by use of the main feedwater throttle valve or by stopping the main feedwater pump (by stopping the flow of motive steam to the pump turbine). Although both the main feedwater throttle valve and the steam supply valve are control grade, this redundancy is sufficient to ensure that the feedwater can be stopped.	Valve position indication, main feedwater flow indication, steam generator A level instrumentation.

TABLE 10.3-2 (Sheet 8 of 9)

SGIS (Including MSSS, CFS, SGBPS)
SINGLE ACTIVE FAILURE ANALYSIS

Component	Failure Mode	Plant Mode	Effect on System Operation	Failure Detection Me
7. S/G A Main Feedwater Isolation Valve (cont.)	Fails to close on demand. (cont.)	7c. Feed line break, steam generator B. (Typical of C or D as well.)	Overfilling and overcooling of steam generator A must be prevented as described above, by closing the main feedwater throttle valve or by stopping the main feedwater pump.	Valve position indication, main feedwater flow indication, steam generator A level instrumentation.
		7d. Steam line break, steam generator B. (Typical of C or D as well.)	Overfilling and overcooling of steam generator A must be prevented as described above, by closing the main feedwater throttle valve or by stopping the main feedwater pump.	Valve position indication, main feedwater flow indication, steam generator A level instrumentation.
		7e. Excessive feedwater incident; i.e., a spurious movement of the control grade main feedwater throttle valve to full open.	Overfilling and overcooling of steam generator A must be prevented by stopping the main feedwater pumps.	Valve position indication, main feedwater flow indication, steam generator A level instrumentation.

TABLE 10.3-2 (Sheet 9 of 9)

SGIS (Including MSSS, CFS, SGBPS)
SINGLE ACTIVE FAILURE ANALYSIS

Component	Failure Mode	Plant Mode	Effect on System Operation	Failure Detection Me
8. Steam generator blowdown isolation valve 1-9773A, normally open motor operated gate valve. (Analogous for series valve 1-9774A.) (Typical of all four steam generators.)	Fails to close on demand.	8a. All.	None. Redundant (series) isolation valve 1-9974A will close, effecting isolation.	Valve position indication.

8 | 7 | 6 | 5

H

WESTINGHOUSE PROPRIETARY CLASS 2

G

F

E

D

C

B

WAPWR-SSSS/SPCS

8 | 7 | 6 | 5

4 3 2 1 (a,c)

G F E D C B

Figure 10.3-1
WAPWR Steam Generator Isolation
System Flow Diagram
(Sheet 1 of 4)

SEPTEMBER, 1984

4 3 2 1

8 | 7 | 6 | 5

WESTINGHOUSE PROPRIETARY CLASS 2

H

G

F

E

D

C

B

WAPWR-SSSS/SPCS

8 | 7 | 6 | 5

4 3 2 1

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Figure 10.3-1
APWR Steam Generator Isolation
System Flow Diagram
(Sheet 2 of 4)

SEPTEMBER, 1984

4 3 2 1

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H
G
F
E
D
C
B

WAPWR-SSSS/SPCS

4	3	2	1	(a,c)

G

F

E

D

C

B

Figure 10.3-1
WAPWR Steam Generator Isolation
System Flow Diagram
(Sheet 3 of 4)

SEPTEMBER, 1984

4	3	2	1

WESTINGHOUSE PROPRIETARY CLASS 2

WAPWR-SSSS/SPCS

4 3 2 1 (a,c)

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Figure 10.3-1
WAPWR Steam Generator Isolation
System Flow Diagram
(Sheet 4 of 4)

SEPTEMBER, 1984

4 3 2 1

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10.4 OTHER FEATURES OF STEAM AND POWER CONVERSION SYSTEM

This section provides discussions of each of the principal design features of the remaining portions of the steam and power conversion within the WAPWR NPB scope of supply. See the plant specific applicant's safety analysis report for a discussion of those areas outside the WAPWR NPB scope.

10.4.1 Main Condenser

The main condenser is the steam cycle heat sink. During normal operation, it receives and condenses main turbine exhaust steam, steam generator feedwater pump turbine exhaust steam, and turbine bypass steam. The main condenser is also a collection point for other steam cycle miscellaneous flows, drains, and vents. In addition, the main condenser is utilized as a heat sink in the initial phase of reactor cooldown during a normal plant shutdown.

The design of the main condenser is the responsibility of the plant specific applicant. The design must be compatible with the interface criteria given in Appendix 10A; Subsection 10A.4.1.

10.4.1.1 Design Bases

10.4.1.1.1 Safety Design Bases

The main condenser serves no safety function and has no safety design basis.

10.4.1.1.2 Power Generation Design Bases

See the plant specific applicant's safety analysis report for a discussion of the main condenser power generation design bases.

10.4.1.2 System Description

The main condenser is located in pits below the turbine building operating floor and is supported above the turbine building foundation.

See the plant specific applicant's safety analysis report for a description of the main condenser, its components, and system operation.

10.4.2 Main Condenser Evacuation System

Main condenser evacuation is performed by the main condenser air removal system (MCARS). This system removes noncondensable gases and air from the main condenser during plant startup, cooldown, and normal operation.

The design of the MCARS is the responsibility of the plant specific applicant. The design of the MCARS must be compatible with the interface criteria given in Appendix 10A; Subsection 10A.4.2.

10.4.2.1 Design Bases

10.4.2.1.1 Safety Design Bases

The MCARS serves no safety function and has no safety design bases.

10.4.2.1.2 Power Generation Design Bases

See the plant specific applicant's safety analysis report for a discussion of the MCARS power generation design bases.

10.4.2.2 System Description

See the plant specific applicant's safety analysis report for a description of the MCARS, its components, and system operation.

10.4.3 Turbine Gland Sealing System

The turbine gland sealing system (TGSS) prevents the escape of steam from the turbine shaft/casing penetrations and valve stems and prevents air inleakage to subatmospheric turbine glands.

The design of the TGSS is the responsibility of the plant specific applicant.

10.4.3.1 Design Bases

10.4.3.1.1 Safety Design Bases

The TGSS serves no safety function and has no safety design bases.

10.4.3.1.2 Power Generation Design Bases

See the plant specific applicant's safety analysis report for a discussion of the TGSS power generation design bases.

10.4.3.2 System Description

See the plant specific applicant's safety analysis report for a discussion of the TGSS, its components, and operation.

10.4.4 Turbine Bypass System

The turbine bypass system (TBS) has the capability to bypass main steam from the steam generators to the main condenser in a controlled manner to minimize transient effects on the reactor coolant system of startup, hot shutdown and cooldown, and step load reductions in generator load. The TBS is also called the steam dump system.

The design of the TBS is the responsibility of the plant specific applicant. The design of the TBS must be compatible with the interface criteria given in Appendix 10A; Subsection 10A.4.4.

10.4.4.1 Design Bases

10.4.4.1.1 Safety Design Basis

The TBS serves no safety function and has no safety design basis.

10.4.4.1.2 Power Generation Design Basis

See the plant specific applicant's safety analysis report for a discussion of TBS power generation design bases.

10.4.4.2 System Description

See the plant specific applicant's safety analysis report for a discussion of the TBS, its components, and operation.

10.4.5 Circulating Water System

The circulating water system (CWS) provides cooling water for the removal of heat from the main condensers and rejects heat to the plant's ultimate heat sink.

The design of the CWS is the responsibility of the plant specific applicant.

10.4.5.1 Design Bases

10.4.5.1.1 Safety Design Bases

The CWS serves no safety function and has no safety design bases.

10.4.5.1.2 Power Generation Design Bases

See the plant specific applicant's safety analysis report for a discussion of the CWS power generation design bases.

10.4.5.2 System Description

See the plant specific applicant's safety analysis report for a description of the CWS, its components, and operation.

10.4.6 Condensate Cleanup System

The condensate cleanup system (CCS) functions to maintain the required purity of feedwater for the steam generators by filtration to remove corrosion products and by ion exchange to remove condenser leakage impurities.

The design of the CCS is the responsibility of the plant specific applicant. The design must be compatible with the interface criteria given in Appendix 10A; specifically the chemistry specifications given in Subsection 10A.4.6.

10.4.6.1 Design Bases

10.4.6.1.1 Safety Design Bases

The CCW serves no safety function and has no safety design bases.

10.4.6.1.2 Power Generation Design Bases

See the plant specific applicant's safety analysis report for a discussion of CCS power generation design bases.

10.4.6.2 System Description

See the plant specific applicant's safety analysis report for a description of the CCS, its components, and operation.

10.4.7 Condensate and Feedwater System

The condensate and feedwater system (CFS) functions to receive condensate from the condenser hotwells and deliver deaerated feedwater, at required pressure and temperature, to the four steam generators.

The safety-related portions of the CFS are designated the SGIS and consist of the main feed piping, valves and associated instrumentation and controls from the steam generator inlet nozzle to the turbine building side of the wall separating the steam tunnel and the turbine building. The main feedwater control valve and flow meter are also in the NPB scope. All of these components are within the WAPWR NPB scope of supply.

The remaining portions of the CFS typically consist of the piping, valves, deaerating feedwater heater, pumps and associated instrumentation and controls from the suction side of the condenser hotwell to, and including, the main feedwater control valves located upstream of the steam tunnel/turbine building wall. The design of all of this equipment, with the exception of the main feedwater control valves, is the responsibility of the plant specific applicant, and must be compatible with the interface criteria given in Appendix 10A; Subsection 10A.4.7. None of this equipment is safety related.

10.4.7.1 Design Bases

10.4.7.1.1 Safety Design Bases

The SGIS portion of the CFS, as defined above, is required to function following a DBA, and to achieve and maintain the plant in a safe shutdown condition.

SAFETY DESIGN BASIS ONE - The SGIS portion of the CFS is protected from the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, and external missiles (GDC-2).

SAFETY DESIGN BASIS TWO - The SGIS portion of the CFS is designed to remain functional after an SSE or to perform its intended function following postulated hazards of fire, internal missiles, or pipe break (GDC-3 and 4).

SAFETY DESIGN BASIS THREE - Safety functions can be performed, assuming a single active component failure coincident with the loss of offsite power (GDC-34).

SAFETY DESIGN BASIS FOUR - The SGIS portion of the CFS is designed such that the active components are capable of being tested during plant operation. Provisions are made to allow for inservice inspection of components at appropriate times specified in the ASME Boiler and Pressure Vessel Code, Section XI.

SAFETY DESIGN BASIS FIVE - The SGIS portion of the CFS is designed and fabricated to codes consistent with the quality group classification assigned by Regulatory Guide 1.26 and the seismic category assigned by Regulatory Guide 1.29. The power supply and control functions are in accordance with Regulatory Guide 1.32.

SAFETY DESIGN BASIS SIX - For a main feedwater line break inside the containment or an MSLB, the SGIS portion of the CFS is designed to limit high energy fluid to the broken loop and to provide a path for addition of emergency feedwater to the three intact loops.

SAFETY DESIGN BASIS SEVEN - For a main feedwater line break upstream of the main feedwater isolation valve (outside of the containment), the SGIS portion of the CFS is designed to prevent the blowdown of any steam generator and to provide a path for the addition of emergency feedwater.

SAFETY DESIGN BASIS EIGHT - The SGIS portion of the CFS is designed to provide a path to permit the addition of emergency feedwater for reactor cooldown under emergency shutdown conditions (GDC-34).

SAFETY DESIGN BASIS NINE - The SGIS, in conjunction with the steam generators, is designed to prevent water hammer.

10.4.7.1.2 Power Generation Design Bases

See the plant specific applicant's safety analysis report for a discussion of CFS power generation design bases.

10.4.7.2 System Description

See the plant specific applicant's safety analysis report for a description of that portion of the CFS outside the WAPWR NPB scope.

10.4.7.2.1 Component Description

Codes and standards applicable to the WAPWR NPB SGIS portion of the CFS are listed in Table 3.2-1. With the exception of main feed control and control bypass valves, the WAPWR NPB CFS scope of supply is designed and constructed in accordance with quality group B and seismic category I requirements. See Figure 10.3-1 for the piping and instrumentation diagram for the Westinghouse scope of CFS supply. Design data for the CFS are provided in Table 10.4-1. Safety related feedwater piping materials are discussed in Subsection 10.3.6.

10.4.7.2.1.1 Main Feedwater Piping

Feedwater is supplied to the four steam generators by four 16-inch carbon steel lines. Each of the lines is anchored at the containment wall and has sufficient flexibility to provide for relative movement of the steam generators due to thermal expansion.

10.4.7.2.1.2 Main Feedwater Isolation Valves

One main feedwater isolation valve (MFIV) is installed in each of the four main feedwater lines outside the containment and downstream of the feedwater control valve. The MFIVs are installed to prevent uncontrolled blowdown from more than one steam generator in the event of a feedwater pipe rupture in the turbine building. The main feedwater check valve provides backup isolation. The MFIVs isolate the nonsafety-related portions from the safety-related portions of the system. In the event of a secondary cycle pipe rupture inside the containment, the MFIV limits the quantity of high energy fluid that enters the containment through the broken loop and provides a pressure boundary for the controlled addition of emergency feedwater to the three intact loops. The valves are bi-directional gate valves with pneumatic/hydraulic operators. Stored energy for closing is supplied by high pressure gas. The MFIV is maintained in a manually open position by high pressure hydraulic fluid. For emergency closure, redundant solenoids are energized, which cause the high pressure hydraulic fluid to be dumped to the fluid reservoir. The redundant electrical solenoids are energized from separate Class 1E sources.

10.4.7.2.1.3 Main Feedwater Control Valves and Control Bypass Valves

The MF control valves are air-operated globe valves which automatically control feedwater between 20 percent and full power. The bypass control valves are air-operated globe valves, which are used during startup up to 25 percent power. The MF control valves and bypass control valves are located in the turbine Building.

In the event of a secondary cycle pipe rupture inside the containment, the main feedwater control valve (and associated bypass valve) provide a diverse backup to the MFIV to limit the quantity of high energy fluid that enters the containment through the broken loop. For emergency closure, either of two separate solenoids, when de-energized, will result in valve closure. The valves also close on loss of air pressure. Electrical solenoids are energized from separate Class 1E sources.

10.4.7.2.1.4 Main Feedwater Check Valves

The main feedwater check valves are located inside the containment, upstream of the emergency feedwater connection. In the event of a secondary cycle pipe rupture, inside the containment, the main feedwater check valves provide a diverse backup to the MFIV to ensure the pressure boundary of the three intact loops.

10.4.7.2.1.5 Startup Feedwater (SFW) Control Valves

The SFW control valves are air-operated, glove valves that close on loss of air pressure or solenoid power. These valves are normally closed and they modulate to automatically control SFW flow when that system is operating. These valves also serve to isolate SFW flow whenever the emergency feedwater system (EFWS) is operating, for which a solenoid is de-energized which results in valve closure.

10.4.7.2.2 System Operation

See the applicant's safety analysis report for a discussion of the operation of those portions of the CFS outside the NPB scope. The following is a discussion of the operation of the WAPWR NPB portion of the system.

10.4.7.2.2.1 Startup and Shutdown Operation

During startup and shutdown operation the main feedwater system is not utilized. Instead the startup feedwater pump provides feedwater heated to approximately 250°F. The source of heated water is anticipated to be a deaerating heater in the main feedwater system, not part of the NPB. The heated feedwater minimizes thermal stresses on the feedwater piping and steam generator feedwater nozzles.

10.4.7.2.2.2 Normal Operation

At very low power levels, feedwater is supplied by the motor-driven startup feedwater pump. Once sufficient steam pressure has been established, a main feedwater pump turbine is started, and from this low power level, to approximately 20-percent power, feedwater flow is under the control of the feedwater bypass control valves and their control system.

At approximately 20-percent power, feedwater flow is controlled by the main feedwater control valves, and the main feedwater pump turbine speed is automatically controlled. The control system utilizes measurements of steam generator steam flow, feedwater pressure, and steam pressure to produce this signal. The pump speed is increased or decreased in accordance with the speed signal by modulating the flow of steam admitted to the pump turbine drivers.

The feedwater flow to each steam generator is controlled to maintain a programmed water level in the steam generator. The feedwater controllers regulate the feedwater control valves and feedwater pump speed by continuously comparing steam generator water level with the programmed level and feedwater flow with the pressure-compensated steam flow signal.

Ten-percent step load and 5-percent per minute ramp changes are accommodated without major effect in the CFS. The system is capable of accepting a 50-percent step load rejection.

10.4.7.2.2.3 Emergency Operation

In the event that the plant must be shut down or trips, then the main feedwater flow may be isolated by closing of the main feedwater control valves and isolating steam to the main feedwater pump turbines. In this case the MFIV remains open and the SFWS is actuated to provide a supply of controlled, heated feedwater to the steam generator.

Should a more severe event occur, or a DBA, then the MFIV is also closed and the EFWS actuated. Steam is relieved to the condenser if available or to the atmosphere by the SGIS portion of the MSSS.

10.4.7.3 Safety Evaluation

Safety evaluations are numbered to correspond to the safety design bases of Subsection 10.4.7.1.1.

SAFETY EVALUATION ONE - The safety-related portion of the CFS is located in the reactor external building. This building is designed to withstand the effects of earthquakes, tornadoes, hurricanes, floods, external missiles, and other appropriate natural phenomena. Sections 3.3, 3.4, 3.5, 3.7 and 3.8 of

SAFETY EVALUATION NINE - Several features are employed to prevent significant flow instabilities due to steam void collapse (i.e., water hammer) in the main feedwater line. The features also reduce the severity of thermal transients
(a,c) on the SG MFW nozzle and piping. [

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10.4.7.4 Tests and Inspections

10.4.7.4.1 Preservice Valve Testing

The MFIVs and feedwater control valves are checked for closing time prior to initial startup.

10.4.7.4.2 Preoperational System Testing

Preoperational testing of the CFS is performed as described in Chapter 14.0 of the plant specific applicant's safety analysis report and Chapter 14.0 of RESAR-SP/90 PDA Module 14, "Initial Test Program".

10.4.7.4.3 Inservice Inspections

The performance and structural and leaktight integrity of all system components are demonstrated by continuous operation.

The redundant actuator power trains of each MFIV are subjected to the following tests:

a. Closure time - The valves are checked for closure time at each refueling.

RESAR-SP/90 PDA Module 7, "Structural/Equipment Design" provide the bases for the adequacy of the structural design of these buildings.

SAFETY EVALUATION TWO - The SGIS portion of the CFS is designed to remain functional after a SSE. Subsection 3.7.2 and Section 3.9 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design" provide the design loading conditions that were considered.

SAFETY EVALUATION THREE - The SGIS safety functions are accomplished by redundant means, as indicated by Table 10.4-2. No single failure will compromise the system's safety functions. All vital power can be supplied from either onsite or offsite power systems, as described in Chapter 8.0 of RESAR-SP/90 PDA Module 9, "I&C and Electric Power".

SAFETY EVALUATION FOUR - Preoperational testing of the SGIS portion of the CFS is performed as described in Chapter 14.0 of the plant specific applicant's safety analysis report and Chapter 14.0 of RESAR-SP/90 PDA Module 14, "Initial Test Program". Periodic inservice functional testing is done in accordance with Subsection 10.4.7.4.

Section 6.6 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design" provides the ASME Boiler and Pressure Vessel Code Section XI requirements that are appropriate for the CFS.

SAFETY EVALUATION FIVE - Section 3.2 delineates the quality group classification and seismic category applicable to the safety-related portion of this system. Table 10.4-1 shows that the components meet the design and fabrication codes given in Section 3.2. All the power supplies and controls necessary for safety-related functions of the MSSS are Class 1E, as described in Chapters 7.0 and 8.0 of RESAR-SP/90 PDA Module 9, "I&C and Electric Power".

SAFETY EVALUATION SIX - For a main feedwater line break inside the containment or an MSLB, the MFIVs located in the steam tunnel, and the main feedwater control valves and the MFW pump turbine steam isolation valves located in the turbine building are automatically closed upon receipt of a feedwater isolation signal or low-low steam generator level signal. For each intact loop, the MFIV and main feedwater control valve and associated redundant isolation of the steam generator blown line will close, forming a pressure boundary to permit emergency feedwater addition. The emergency feedwater system is described in Subsection 10.4.9.

SAFETY EVALUATION SEVEN - For a main feedwater line break upstream of the MFIV, the MFIVs are supplied with redundant power supplies and power trains to ensure their closure to isolate safety and nonsafety-related portions of the system. Branch lines downstream of the MFIVs contain normally closed, power operated valves which close on a feedwater isolation signal. These valves fail closed on loss of power.

Releases of radioactivity from the SGIS due to the main feedwater line break are minimal because of the negligible amount of radioactivity in the system under normal operating conditions. Additionally, following a steam generator tube rupture, the main steam isolation system provides controls for reducing accidental releases, as discussed in Section 10.3. Detection of radioactive leakage into and out of the system is facilitated by area radiation monitoring (discussed in RESAR-SP/90 PDA Module 11, "Radiation Protection"), process radiation monitoring (discussed in RESAR-SP/90 PDA Module 12, "Waste Management"), and steam generator blowdown sampling (discussed in Subsection 10.4.8).

SAFETY EVALUATION EIGHT - In the event of loss of the steam generator feedwater pumps, or other situations which may result in a loss of main feedwater, the feedwater isolation signal will automatically isolate the main feedwater system and permit the addition of startup feedwater to allow a controlled reactor heat removal and cooldown. In the case of a malfunction of the SFWS or a more severe plant condition, the emergency feedwater system is actuated to ensure controlled reactor heat removal and cooldown with emergency procedures. The emergency feedwater system is described and evaluated in Subsection 10.4.9.

SAFETY EVALUATION NINE - Several features are employed to prevent significant flow instabilities due to steam void collapse (i.e., water hammer) in the main feedwater line. The features also reduce the severity of thermal transients
(a,c) on the SG MFW nozzle and piping. [

]

10.4.7.4 <u>Tests and Inspections</u>

10.4.7.4.1 Preservice Valve Testing

The MFIVs and feedwater control valves are checked for closing time prior to initial startup.

10.4.7.4.2 Preoperational System Testing

Preoperational testing of the CFS is performed as described in Chapter 14.0 of the plant specific applicant's safety analysis report and Chapter 14.0 of RESAR-SP/90 PDA Module 14, "Initial Test Program".

10.4.7.4.3 Inservice Inspections

The performance and structural and leaktight integrity of all system components are demonstrated by continuous operation.

The redundant actuator power trains of each MFIV are subjected to the following tests:

a. Closure time - The valves are checked for closure time at each refueling.

b. On-line operability - While the CFS is in operation, the operability of the valve is checked periodically by exercising the valve to approximately 90 percent of full open.

Additional discussion of inservice inspection of ASME Code Class 2 and 3 components is presented in Section 6.6 of RESAR-SP/90 PDA Module 7 "Structural/Equipment Design".

10.4.7.5 Instrumentation Applications

The main feedwater instrumentation is designed to facilitate automatic operation, remote control, and continuous indication of system parameters.

The feedwater flow to each steam generator is controlled by a flow control system to maintain a programmed water level in the steam generator. The feedwater controllers regulate the feedwater control valves by continously comparing the feedwater flow and steam generator water level with the programmed level and the pressure-compensated steam flow signal.

See the plant specific applicant's safety analysis report for a discussion of additional CFS instrumentation applications.

10.4.8 Steam Generator Blowdown Processing System

The steam generator blowdown processing system (SGBPS) helps to maintain the steam generator secondary side water within the prescribed chemical specifications. Heat is recovered from the blowdown and returned to the feedwater system. The blowdown is then treated to remove impurities before being returned to the condenser.

Those portions of the SGBPS which are safety-related are within the scope of the WAPWR NPB. Included are all of the piping, valves, and controls from the steam generator blowdown nozzle through the steam tunnel/turbine building wall, including the containment isolation valves. The remaining equipment and components, which process the blowdown are not safety-related and the design is the responsibility of the plant specific applicant. These portions of the

SGBPS design must be compatible with the interface criteria given in Appendix 10A; Subsection 10A.4.8.

10.4.8.1 Design Bases

10.4.8.1.1 Safety Design Basis

Refer to Section 10.3 for a discussion of safety design basis, safety evaluations and FMEA of the safety related portions of the SGBPS which is part of the SGIS.

10.4.8.1.2 Power Generation Design Basis

See the plant specific applicant's safety analysis report for a discussion of the SGBPS power generation design bases.

10.4.9 Emergency Feedwater System

In a conventional Westinghouse PWR the auxiliary feedwater (AFS) functions to supply a reliable source of water for the steam generators during normal shutdowns and accidents. The AFS functions to remove thermal energy from the reactor coolant system through the steam generator to the atmosphere. A typical AFS provides emergency water following any accident. This system may also be used following a reactor shutdown in conjunction with the condenser dump valves to cool the reactor coolant system to hot shutdown, at which temperature the residual heat removal system is brought into operation.

For the WAPWR, the above safety and control functions are performed by two secondary side systems; the emergency feedwater system (EFWS) which is discussed below and the startup feedwater system (SFWS) which is discussed in Subsection 10.4.10 and in Appendix 10A.

The EFWS functions to provide feedwater to the steam generators following transients or accidents such as reactor trip, loss of main feed, steam or feed line breaks, steam generator tube ruptures, and any time the main and startup feedwater systems are not available.

SAFETY DESIGN BASIS FIVE - The EFWS is designed and fabricated consistent with the quality group classification assigned by Regulatory Guide 1.26 and the seismic category assigned by Regulatory Guide 1.29. The power supply and control functions are in accordance with Regulatory Guide 1.32.

SAFETY DESIGN BASIS SIX - The capability to isolate components or piping is provided, if required, so that the EFWS safety function will not be compromised. This includes isolation of components to deal with leakage or malfunctions and to isolate portions of the system that may be directing flow to a broken secondary side loop.

SAFETY DESIGN BASIS SEVEN - The EFWS has the capacity to be operated locally as an alternate, redundant means of feedwater control, in the unlikely event that the control room must be evacuated.

SAFETY DESIGN BASIS EIGHT - The EFWS provides feedwater to maintain sufficient steam generator level to ensure heat removal from the reactor coolant system in order to achieve a safe shutdown following a main feedwater line break, a main steamline break, or an abnormal plant situation requiring shutdown. The EFWS is capable of delivering full flow when required, after detection of any accident requiring emergency feedwater.

10.4.9.1.2 Power Generation Design Bases

The EFWS has no power generation design bases. The condensate and feedwater system is designed to provide a continuous feedwater supply to the steam generators during normal plant operation. The SFWS is designed to provide a continuous feedwater supply to the steam generators during startup and shutdown. Refer to Subsections 10.4.7 and 10.4.10, respectively.

10.4.9.2 System Description

10.4.9.2.1 Component Description

Codes and standards applicable to the EFWS are listed in Table 3.2-1. The EFWS is designed and constructed in accordance with quality groups B and C and seismic category I requirements.

The EFWS consists of two identical subsystems, each of which receives electrical power from one of two separate safety class 1E electrical power trains. Each subsystem consists of an emergency feedwater storage tank (EFWST), one motor driven emergency feedwater pump, one turbine driven emergency feedwater pump and the required piping, valves, instruments and controls necessary for system operation (see Figure 10.4-1).

The EFWS is a safety related system and is entirely within the scope of the WAPWR NPB. See Appendix 10A, Subsection 10A.4.9 for interface criteria between the EFWS and plant specific applicant's scope of supply.

Design data for the components of the EFWS are provided in Table 10.4-1.

10.4.9.1 Design Bases

10.4.9.1.1 Safety Design Bases

SAFETY DESIGN BASIS ONE - The EFWS is protected from the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, and external missiles (GDC-2).

SAFETY DESIGN BASIS TWO - The EFWS is designed to remain functional after an SSE or to perform its intended function following a postulated hazard, such as fire, internal missile, or pipe break (GDC-3 and 4).

SAFETY DESIGN BASIS THREE - The safety functions can be performed, assuming a single active component failure coincident with the loss of offsite power. The system requirements are also met with a complete loss of AC power (GDC-34).

SAFETY DESIGN BASIS FOUR - The EFWS is designed so that the active components are capable of being tested during plant operation. Provisions are made to allow for inservice inspection of components at appropriate times specified in the ASME Boiler and Pressure Vessel Code, Section XI.

10.4.9.2.1.1 Motor Driven and Turbine Driven Emergency Feedwater Pumps

Two motor driven and two turbine driven, multi-stage centrifugal pumps are provided, one motor driven and one turbine driven pump in each subsystem, which take suction from their respective emergency feedwater storage tank and discharge via EFWS piping into the steam generators. The pumps are provided with flanged suction and discharge connections.

The pumps are sized so that two pumps can deliver the minimum required emergency feedwater flow following a main feedline rupture to at least two effective steam generators within 1 minute after system actuation, with the steam generators at a pressure equal to the lowest set safety relief valve plus the accumulation of the valve.

The pumps are normally started automatically by a signal from the reactor protection system. However, controls are also provided which allow the pumps to be started and stopped manually.

An orificed miniflow line is provided from each pump discharge back to the emergency feedwater storage tank. The orfice is sized to provide the required amount of recirculation flow for pump protection in the event the discharge flow paths are isolated. In addition flow paths are provided in the pump package that allows pumped fluid to flow through heat exchangers which are used to cool the turbine bearing oil and the pump bearing oil.

Runout protection for the pumps is provided by cavitating venturies in the EFWS piping. These venturies are sized to choke flow at[ ]gpm. Since one (a,c)
pump can discharge through two cavitating venturies in parallel, the pump discharge flow will be choked at[ ]gpm.
(a,c)

The emergency feedwater pump turbines are non-condensing and exhaust to the atmosphere. The turbines are controlled by integral governor valves and are capable of supplying the required power to the attached pumps such that each pump is able to supply its required feedwater flow.

The steam supply line to each turbine is fitted with a pneumatically-operated steam admission valve which is normally closed but arranged to fail open on loss of air or electric power. The steam admission valve utilizes both A and B powered actuation trains to operate redundant solenoid valves, either of which can vent the air from the operator to open the admission valve and start the pump.

10.4.9.2.1.2 Emergency Feedwater Storage Tank

Two emergency feedwater storage tanks are provided, one in each subsystem. The tanks are safety grade, seismically qualified and protected from missiles, fire, etc. The tanks contain a sufficient quantity of condensate quality water to allow the following:

1. To allow a cold shutdown using only safety grade equipment even with extended hot standby and cooldown times (8 hour hot, 6 hour cooldown). One reactor coolant pump is assumed to operate.

2. To allow indefinite hot standby operation using only safety grade equipment. An indefinite time is satisfied by having 1 day of water stored in the EFWST's and taking credit for offsite water supplies in that time frame or use of primary side feed and bleed operations.

In addition, quantities of water are provided in the tank to allow for inaccuracies in the tank level indicators, overflow and pumpout margins, and spill allowance.

Each tank is provided with redundant level indicators with high, low, and low-low level alarms. The maximum permissible temperature of the water in the

(a,c) emergency feedwater storage tank is [ ]

10.4.9.2.1.3 Cavitating Venturies

Four cavitating venturies are provided in the EFWS, one in each system discharge line to the steam generators. These venturies are sized to cavitate and choke the emergency feedwater flow in each discharge line at [ ] gpm. The cavitating venturies serve several purposes: (a,c)

- o In the event of a steamline or feedline rupture, the associated emergency feedwater pump(s) will discharge to a reduced pressure. In such cases the cavitating venturies will choke the flow in each discharge line to [ ] gpm and thereby prevent the pumps from being damaged by runout. (a,c)

- o Also in the event of a steamline rupture, with all pumps operating, the cavitating venturies prevent an excessive flow of emergency feedwater to the steam generators which could cause an unacceptably high cooldown rate of reactor coolant system components.

- o By limiting the emergency feedwater flow to the steam generators in the short term (i.e., before operator action can be assumed), the cavitating venturies prevent the steam generators from being filled solid with water and prevent the attendant problem of steam line flooding.

- o In the event of a steamline rupture inside containment, the cavitating venturies limit the EFWS contribution to the mass and energy released to the containment

During normal EFWS flowrates the venturies will not cavitate and the permanent head loss caused by these devices will be considerably less than that of an equivalent orifice.

10.4.9.2.1.4 Motor-Operated Valves

EFW Pump Steam Isolation Valve

A normally open motor-operated (M/O) gate valve is provided on the steam line each turbine driven (T/D) EFW pump. The valves are located ORC and are provided to isolate leaks and allow for maintenance on the T/D pump steam admission valves. Control is provided from the main control panel (MCP).

EFW Pump Separation Valves

Two normally open motor-operated gate valves are located IRC in a cross-over line between EFW pumps 1 and 4; two similar valves are provided for pumps 2 and 3. The valves allow the flow from one EFW pump to feed 2 steam generators.

With a faulted steam generator (SG), a pressure differential signal is generated that closes these valves in pairs. If SG A is faulted then the valves servicing EFW pumps 1 and 4 are closed; valves servicing EFW pumps 2 and 3 are unaffected. The action prevents the spill of 2 EFW pumps and reduces the probability of a SG not being fed.

10.4.9.2.1.5 Air-Operated Valves

EFW Pump Control Valves

A motor-operated globe control valve is located outside reactor containment (ORC) in the discharge of each EFW pump. These valves are normally full open. These valves are modulated by the operator to control the flow of EFW allowing the SG level to be maintained in the narrow range. These valves also provide the capability of isolating the containment and prevent EFW pump spill. The electrical power supply is Class 1E. In addition, a hand control is provided on each valve.

EFW T/D Pump Steam Admission Valves

Air-operated globe isolation valves are located ORC in the steam lines to the T/D EFW pumps. These valves are normally closed and fail open. Any EFW T/D pump start signal opens these valves. Note that two solenoids are provided for each valve to allow the use of train A and B actuation signals.

10.4.9.2.1.6 Check Valves

Check valves are located ORC in the discharge of each EFW pump to prevent reverse flow through a failed EFW pump.

Check valves are also located inside reactor containment (IRC) at the connection to the main feed lines. These valves prevent the flow of high temperature and pressure main feedwater from entering the EFWS when it is not operating.

10.4.9.2.1.7 Manual Valves

Normally open manual valves are located in the suction, discharge, and miniflow lines of each EFW pump for maintenance. Normally closed manual valves are also located in the ORC cross-connection of the EFWS. All of these valves are locked in position.

10.4.9.2.1.8 Piping

Piping is provided as shown on Figure 10.4-1 to enable the EFWS to meet its intended functions. Schedule 40 carbon steel pipe is used in the suction of the EFW pumps and schedule 160 carbon steel pipe is used in the discharge of the pumps. Schedule 80 carbon steel pipe is used in the steam line to the T/D EFW pumps.

10.4.9.2.2 System Operation

10.4.9.2.2.1 Normal Operation

During normal plant power operation the EFWS does not operate, but is in standby mode ready to provide emergency feedwater if needed. During normal startup, hot standby, and cooldowns the startup feedwater system (see Subsection 10.4.10) provides automatically controlled feedwater to the SG and again the EFWS does not operate. Also following most plant transients such as reactor trip and loss of main feedwater the SFWS is automatically started. The EFWS provides backup capability for both normal startup/shutdown and for reactor trips in case there is a malfunction in the SFWS.

10.4.9.2.2.2 Accident Operation

In the event the SFWS fails to function properly following a plant transient or in case of a more severe event such as a feedline break the EFWS is automatically started. The only action required to initiate flow of emergency feedwater is to energize the AC motor of the M/O EFW pumps and/or to open the fail open steam admission value of the T/D EFW pumps; all the other valves are already open.

The system is designed to deliver feedwater at the minimum required flowrate to 2 or 4 SG (depending on the event) and to continue this delivery for an indefinite period of time. After approximately 30 minutes the operator must take certain actions such as to isolate the faulted SG, or to throttle or isolate the SG to reduce loss of EFW. In the longer term, the operator must either cool the plant to initiate primary safeguards closed loop cooling (RHR) or to secure other water supplies. In the extremely unlikely case that neither of these alternatives can be accomplished then the primary side safeguards open loop cooling mode (feed and bleed via EWST) must be utilized (see RESAR-SP/90 PDA Module 1, "Primary Side Safeguards System").

If the initiating problem can be resolved then the SFWS can be started and the EFWS stopped. Later the plant can brought back to power using normal procedures.

10.4.9.3 Safety Evaluation

Safety evaluations are numbered to correspond to the safety design bases in Section 10.4.9.1.1.

SAFETY EVALUATION ONE - The EFWS is located in the safeguards area outside reactor containment. This structure is designed to withstand the effects of earthquakes, tornadoes, hurricanes, floods, external missiles, and other appropriate natural phenomena. Section 3.8 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design" provides the bases for the adequacy of the structural design of the reactor external building.

SAFETY EVALUATION TWO - The EFWS is designed to remain functional after an SSE. Section 3.8 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design" provides the design loading conditions that were considered.

Breaks in seismic Category I piping are not postulated during a seismic event. Thus an MSLB or MFLB inside containment or in the steam tunnel is not postulated following a seismic event and the design of the exhaust line does not enter into the evaluation of these breaks. If the seismic event causes an MSLB in the turbine building, the physical separation between the turbine building and the safeguards area ensures that none of the EFWS equipment will be adversely affected by jet impingment or extreme environmental conditions. Therefore, even if an MSIV fails to close resulting in the blowdown of one steam generator, at least three emergency feedwater pumps will deliver to three intact steam generators. All single failures are bounded by those discussed in Table 10.3-2.

SAFETY EVALUATION THREE - Complete redundancy is provided and, as indicated by Table 10.4-2, no single failure will compromise the system's safety functions. All vital power can be supplied from either onsite or offsite power systems, as described in Chapter 8.0 of RESAR-SP/90 PDA Module 9, "I&C and Electric Power".

The turbine-driven pumps are energized by steam drawn from the main steam lines between the containment penetrations and the main steam isolation valves. All valves and controls necessary for the function of the turbine-driven pump are independant of AC power, and are either fail open air operated valves (steam admission valves) or mechanical/hydraulic valves (steam control valves). Turbine bearing lube oil is circulated by an integral shaft-driven pump. Turbine and pump bearing oil is cooled by pumped emergency feedwater.

SAFETY EVALUATION FOUR - The EFWS is initially tested with the program given in Chapter 14.0 of RESAR-SP/90 PDA Module 14 "Initial Test Program".

Section 6.6 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design" provides the ASME Boiler and Pressure Vessel Code, Section XI requirements that are appropriate for the EFWS.

SAFETY EVALUATION FIVE - Section 3.2 delineates the quality group classification and seismic category applicable to this system. Table 10.4-1 shows that the components meet the design and fabrication codes given in Section 3.2. All the power supplies and control function necessary for safe function of the EFWS are Class 1E, as described in Chapters 7.0 and 8.0 of RESAR-SP/90 PDA Module 9, "I&C and Electric Power".

SAFETY EVALUATION SIX - As discussed in Subsections 10.4.9.2 and 10.4.9.5, adequate instrumentation and control capability is provided to permit the plant operator to quickly identify and isolate the emergency feedwater flow to a broken secondary side loop.

SAFETY EVALUATION SEVEN - The EFWS can be controlled from either the main control room or the shutdown control panel. Refer to Section 7.4 of RESAR-SP/90 PDA Module 9, "I&C and Electric Power" for the control description.

SAFETY EVALUATION EIGHT - The EFWS provides a means of pumping sufficient feedwater to prevent damage to the reactor following a main feedwater line break inside the containment, or a main steamline break incident, as well as to cool down the reactor coolant system at a rate of 50°F per hour to a temperature of 350°F, at which point the residual heat removal portion of the integrated safeguards system can operate. Pump capacities, as shown in Table 10.4-1, and start times are such that these objectives are met. Restriction venturis located in the pump discharge lines limit the flow to the broken loop so that adequate cooldown flow can be provided to the other steam generators for removal of reactor decay heat and so that containment design pressure is not exceeded. Pump discharge head is sufficient to establish the minimum necessary flowrate against a steam generator pressure corresponding to the lowest pressure setpoint of the main steam safety valves. The maximum time period required to start the electric motors and the steam turbine which drive the emergency feedwater pumps is chosen so that sufficient flowrates are established within the required time for primary system protection.

The accidents that result in emergency feedwater system (EFWS) and steam generator isolation system (SGIS) actuation are categorized as follows and are evaluated in Chapter 15.0:

A. <u>Increase in Heat Removal by the Secondary System</u>

1. Inadvertent opening of a steam generator power operated atmospheric steam relief or safety valve (Condition II Fault).

2. Steam system piping failure (Condition III Fault).

B. <u>Decrease in Heat Removal by the Secondary System</u>

1. Feedwater system piping failure (Condition IV Fault).

2. Loss of Non-Emergency AC Power to the Station Auxiliaries (Condition II Fault).

3. Loss of Normal Feedwater Flow (Condition II Fault)

C. <u>Decrease in Reactor Coolant System (RCS) Inventory</u>

1. Steam generator tube rupture (Condition IV Fault).

2. Loss-of-coolant accident (LOCA) from a spectrum of postulated RCS piping failures (Condition IV Fault).

Each of these accidents results in generation of a safety injection (SI) signal, ECCS operation, and initiation of other safeguards automatic actions including the EFWS and SGIS for mitigation within acceptable criteria.

<u>Increase in Heat Removal by the Secondary System</u>

A number of events have been postulated which could result in an increase in heat removal from the RCS by the secondary system. Detailed analyses of these events are presented in Section 15.1. Those events which result in EFWS, SGIS, and ECCS actuation are:

A. Inadvertent Opening of a Steam Generator Relief or Safety Valve

The most severe core conditions resulting from an accidental depressurization of the main steam system are associated with an inadvertent opening of a single steam dump, relief, or safety valve. Refer to Subsection 15.1.4 for a detailed description of this accident, including acceptance criteria and analytical results.

For this accident, the EFWS, SGIS and ECCS are actuated upon generation of an SI signal. The high head safety injection pumps function to inject a borated water solution from the EWST into the reactor vessel. The borated water solution does not provide sufficient negative reactivity to maintain the reactor below criticality. The core is ultimately shut down by the termination of the uncontrolled cooling and the departure from nucleate boiling (DNB) design basis is met.

The EFWS and the SGIS function to provide a means of controlled decay heat removal to cool the RCS to a point at which the RHRS can operate. The volume of water contained in the dedicated EFWSTs is sufficient to supply water for 8 hours of hot standby followed by 6 hours of plant cooldown, for flow spilled through a pipe break for 30 minutes at the rate limited by the cavitating venturi, and for the heat generated by a reactor coolant pump for 14 hours after reactor trip.

B. Steam System Piping Failure

The most severe core conditions resulting from a steam system piping failure are associated with a rupture of a steam line which occurs at zero power. Effects of smaller piping failures at higher power levels are bounded by the rupture at zero power. Refer to Subsection 15.1.5 for a detailed description of this accident, including acceptance criteria and analytical results.

For this accident, the EFWS, SGIS and ECCS function as described in Paragraph A above for the inadvertent opening of a steam generator relief or safety valve. However, this piping failure constitutes a more severe cooldown transient. The negative reactivity provided by operation of the high head safety injection pumps is not sufficient to prevent the reactor from returning to criticality during the transient. The core is ultimately shut down by the borated water solution and by termination of the uncontrolled cooldown, and the DNB design basis is met.

Decrease in Heat Removal by the Secondary System

A number of events have been postulated which could result in a decrease in heat removal from the RCS by the secondary system. Detailed analyses of these events are presented in Section 15.2. Those events which result in, SGIS and ECCS actuation are:

A. Feedwater System Piping Failure

The most severe core conditions resulting from a feedwater system piping failure are associated with a rupture of a feed line at full power. Depending on break size and power level, a feedwater system pipe failure could cause either an RCS cooldown transient or RCS heatup transient. Only the RCS heatup transient is evaluated a a feedwater system pipe failure, since the spectrum of cooldown transients is bounded by the steam system pipe failure analyses. The heatup transient effects of smaller piping failures at reduced power levels are bounded by the feed line rupture at full power. Refer to Subsection 15.2.8 for a detailed description of this accident, including acceptance criteria and analytical results.

For this event, the EFWS and SGIS function to ensure the availability of sufficient feedwater to the unaffected steam generators and controlled steam release from the steam generators so that:

a. No substantial overpressurization of the RCS occurs (less than 110 percent of design pressures); and

b. Sufficient liquid in the RCS is maintained so that the core remains in place and geometrically intact with no loss of core cooling capability.

For this accident, the EFWS and ECCS are actuated upon generation of an SI signal and the HHSI pumps inject the borated water solution from the EWST into the RCS vessel. The high head safety injection pump flow functions to increase RCS inventory to ensure that sufficient inventory exists to keep the core covered with water. Since the accident is characterized by a heatup transient, the borated water solution from the EWST is not required to control core reactivity. Subsequent to stabilizing plant conditions and satisfying termination criteria, the operator initiates plant shutdown operations. The volume of water required by the EFWS is as described in the previous Paragraph A, above.

B. Loss of Non-Emergency AC Power to Station Auxiliaries

C. Loss of Normal Feedwater

Diversity is provided in the SSSS (and EFWS) in the type and number of pumps, power supplies, piping arrangements and pump and valve controls so that any single failure in the EFWS, loss of non-emergency AC power to the station auxiliaries, or loss of normal (start-up feedwater system) feedwater will not negate the ability of the EFWS to perform its safety function with any coincident, single active mechanical failure.

In accordance with Branch Technical Position (BTP) ASB 10-1, the EFWS provides redundant and diverse means of supplying water to the steam generators for cooling the RCS under emergency conditions.

Decrease in RCS Inventory

A number of events have been postulated which could result in a decrease in RCS inventory. Detailed analyses of these events are presented in Section 15.6. Those events which represent a safety function requirement for the EFWS and accordingly result in actuation of this system as well as the ECCS are:

A. Steam Generator Tube Rupture

Although a steam generator tube rupture is an accident which results in a decrease in RCS inventory, severe core conditions are not associated with this event. The accident analyzed is a complete severance of a single steam generator tube that occurs at power with the reactor coolant contaminated with fission products, corresponding to continuous operation with a limited amount of defective fuel rods. Effects of smaller breaks are bounded by the complete severance. Refer to Subsection 15.6.3 for a detailed description of this accident, including acceptance criteria and analytical results.

For this accident, the EFWS and ECCS are actuated upon generation of an SI signal and the pumps inject the borated water solution from the EWST into the reactor vessel downcomer. The high head safety injection pump flow functions to replace RCS inventory that is being lost through the ruptured steam generator tube, provide a heat sink which helps absorb decay heat, and repressurize the RCS.

The EFWS functions to provide a decay heat removal capability in each of the steam generators to cool the RCS to a point at which the RHRS can operate. In addition, EFWS make up to the steam generators is a mitigating feature considered in reducing post-accident fission product release to the environment. Subsequent to stabilizing plant conditions and satisfying plant conditions, the operator initiates plant shutdown operations.

The SGIS functions with the EFWS to remove decay heat and to contain activity that may enter the SG from the RCS. The SG overflow valves prevent the possibility of water overfilling the SG.

B. Loss-of-Coolant Accident

A LOCA is defined as a rupture of the RCS piping or branch piping which results in a decrease in RCS inventory that exceeds the flow capability of the normal make up system. Ruptures which result in break flow within the capability of the normal make up system will not result in decreasing RCS pressure and EFWS/ECCS actuation. For breaks less than a 0.375-inch diameter hole, the normal make up system can maintain RCS pressure and permit the operator to execute an orderly shutdown.

For the purpose of evaluation, the spectrum of postulated piping breaks in the RCS is divided into major pipe breaks (large break) and minor pipe breaks (small breaks). The large break is defined as a rupture with a total cross-sectional area of 1 ft^2 or greater. The small break is defined as a rupture with a total cross-sectional area less than 1 ft^2 but larger than the 0.375-inch diameter hole. Refer to Subsection 15.6.4

of RESAR-SP/90 PDA Module 1, "Primary Side Safeguards System" for a detailed description of this accident, including acceptance criteria and analytical results.

For this accident, the EFWS and ECCS are actuated upon receipt of an SI signal. Once actuated, the ECCS mitigates the spectrum of LOCA accidents; following completion of core reflood (large break) or core recovery (small break), the ECCS continues to supply water to the RCS for long-term cooling. The EFWS functions to provide the necessary decay heat removal function in each of the steam generators.

For the instance of the rod ejection accident, core damage may occur. The EFWS continues to (a) provide the decay heat removal function and (b) provides make up to each steam generator to maintain coverage of the S/G tubes. This tube bundle coverage is also a mitigating feature considered in the post-accident radiological dose evaluation and environmental release. Refer to Subsection 15.6.4 of RESAR-SP/90 PDA Module 1, "Primary Side Safeguards System" for a detailed description of this accident, including acceptance criteria and analytical results.

10.4.9.4 Tests and Inspections

Preoperational testing is described in Chapter 14.0 of RESAR-SP/90 PDA Module 14, "Initial Test Program". The performance and structural and leaktight integrity of system components is demonstrated by periodic operation. The EFWS has the capability for being tested during normal plant operation; the EFW pumps and critical valves can be started/stopped or open/closed. In this case valves are simply cycled open/closed and the EFW pumps run on mini-flow with the path to the SG closed.

Should it be required because of pump maintenance the EFW pumps can also be run at full flow without depressurizing the SG by opening a bypass of the mini-flow orifices.

The safety-related components, i.e., pumps, valves, piping, and turbine, are designed and located to permit preservice and inservice inspection.

10.4.9.5 <u>Instrumentation Applications</u>

The EFWS instrumentation is designed to facilitate automatic operation and remote control of the system and to provide continuous indication of system parameters.

10.4.9.5.1 Pressure

<u>EFW Pump Suction Pressure</u>

Each EFW pump has a local pressure guage mounted on the suction pipe. This instrument is used to monitor pump suction conditions and ensure proper NPSH during test and post maintenance operation.

<u>EFW Pump Discharge Pressure</u>

Each EFW pipe has a pressure guage mounted in the discharge piping. Readout is provided in the MCP and local at the pipe. This instrument is used to monitor pump performance during all modes of operation.

10.4.9.5.2 Flow

<u>EFW Pump Mini-Flow</u>

A flow meter is provided in the mini-flow lines of each EFW pump with readout in the MCP and local at the pump. These instruments monitor pump performance during testing and other conditions.

<u>EFW Pump Flow</u>

A flow meter is located in the discharge of each EFW pump IRC with readout in the MCP and the shutdown control panel (SCP). This instrument provides the prime indication of EFWS operation and is used during normal and post-accident conditions. The SCP readout is provided for MCP evacuation situations.

10.4.9.5.3 Temperature

EFW Pump Temperature

A temperature instrument is located in the discharge of each EFW pump just outside of the containment with readout on the MCP. These instruments are provided to detect significant backleakage from the main feed lines and alert the operators to possible steam binding of the EFW pumps.

10.4.9.5.4 Level

EFW Storage Tank Level

Redundant level instruments are provided on each EFW storage tank with readout on the MCP and the SCP. Multiple alarms are provided to alert the operator to potential overflow, violation of Technical Specification levels and inadequate EFW pump suction conditions. The SCP readout is provided for control room evacuation purposes.

10.4.10 Startup Feedwater System

During normal power operation the steam generators are fed by the main feedwater pumps with the steam produced being sent to the turbine. For the WAPWR, a startup feedwater system (SFWS) is included to feed the steam generators during normal plant startup and shutdown. Under these conditions, steam from the steam generators is sent to the main condenser. The SFWS is also actuated automatically to provide feedwater following a reactor trip, loss of main feed, loss of offsite power, and other anticipated transients.

The SFWS is a control grade system which, although not required to mitigate the consequences of postulated accidents, provides additional reliability and diversity to the EFWS. The SFWS also serves to minimize the number of EFWS actuations required which reduces the probability of EFWS failures.

As indicated above, the SFWS performs no safety-related function. The design of the SFWS is the responsibility of the plant specific applicant. The design must be compatible with the design criteria and interface criteria given in Appendix 10A; Subsection 10A.4.10.

10.4.10.1 Design Bases

10.4.10.1.1 Safety Design Bases

The SFWS serves no safety function and has no safety design basis.

10.4.10.1.2 Power Generation Design Bases

See the plant specific applicant's safety analysis report for a discussion of SFWS power generation design bases.

10.4.10.2 System Description

The major components of the SFWS is one motor driven feedwater pump and associated pipes and valves. The normal suction source is anticipated to be a deaerating heater in the CFS, however, that portion of the CFS is not shown in the scope of the NPB. The components are located in the turbine building.

See the plant specific applicant's safety analysis report for additional description of the SFWS, its components, and system operation. Refer to Appendix 10A., Subsection 10A.4.10, for interface criteria.

10.4.11 Secondary Liquid Waste System

The secondary liquid waste system (SLWS) functions to process condensate demineralizer regeneration wastes and potentially radioactive liquid waste collected in the turbine building.

The SLWS is not a safety-related system. The design of the SLWS is the responsibility of the plant specific applicant.

10.4.11.1 <u>Design Bases</u>

10.4.11.1.1 Safety Design Bases

The SLWS serves no safety function and has no safety design basis.

10.4.11.1.2 Power Generation Design Bases

See the plant specific applicant's safety analysis report for a discussion of the SLWS power generation design bases.

10.4.11.2 System Description

See the plant specific applicant's safety analysis report for a description of the SLWS, its components, and system operation.

TABLE 10.4-1

DESIGN DATA FOR OTHER FEATURES OF STEAM

AND POWER CONVERSION SYSTEM

(WAPWR NPB Scope of Supply)

CONDENSATE AND FEEDWATER SYSTEM

Main Feedwater Piping (Safety-Related Portion)	
Design (VWO) flowrate, total, lb/hr	17,520,000
Number of lines	4
Nominal size, in.	16
Design pressure, psig	[ ] (a,c)
Design temperature, F	
Design code	ANS 2
Seismic Design	1
Feedwater Isolation Valves	
Number per main feedwater line	1
	Normally open, fail-closed hydraulic/pneumatically operated gate valve; closed by self-contained high pressure gas
Nominal Size, in	16
Closing time, sec	5
Body design pressure, psig	[ ] (a,c)
Design temperature, F	
Design code	ANS 2
Seismic design	1
Feedwater Control Valves	
Number per main feedwater line	1
	Normally open, fail-closed, air operated glove valve
Nominal Size, in	16
Design code	NNS

TABLE 10.4-1 (Continued)

DESIGN DATA FOR OTHER FEATURES OF STEAM
AND POWER CONVERSION SYSTEM
(WAPWR NPB Scope of Supply)

EMERGENCY FEEDWATER SYSTEM

Emergency Feedwater Pumps (Motor-Driven)	
Number	2
Type - pump	Horizontal, multi-stage, barrel
- driver	AC motor
Design flow/head	[ ] (a,c)
Runout flow/head	
Shutoff head (max.)	
Minimum flow	
Maximum power	
Suction design pres/temp.	
Discharge design pres./temp.	
Material	Carbon steel
Safety class - pump	SC-3
- motor	1E

Emergency Feedwater Pumps (Turbine-Driven)	
Number	2
Type - pump	Horizontal, multi-stage, barrel
- driver	With integral steam turbine drive
Design flow/head	[ ] (a,c)
Runout flow/head	
Shutoff head (max.)	
Minimum flow	
Suction design pres/temp.	
Discharge design pres./temp.	
Material	Carbon steel
Safety class - pump	SC-3
- turbine	SC-3

TABLE 10.4-1 (Continued)

DESIGN DATA FOR OTHER FEATURES OF STEAM

AND POWER CONVERSION SYSTEM

(WAPWR NPB Scope of Supply)

TANKS

Emergency Feedwater Storage Tank

Number	2
Type	Concrete vaults
Volume - useable[1]	[ ](a,c)
- total	
Design - temp.	
- pressure	
Material	Concrete with SS liner
Safety class	SC-3

Cavitating Venturies

Number	4
Design pressure	[ ](a,c)
Design temperature	
Design flow	
Choke flow	
Pressure loss at design flow	
Material	Carbon Steel
Safety class	SC-2

(1) Useable volume is above pump suction connection and below overflow level.

TABLE 10.4-2

EMERGENCY FEEDWATER SYSTEM SINGLE ACTIVE FAILURE ANALYSIS

Component	Failure Mode	Plant Mode	Effect on System Operation	Failure Detection Method
1. Turbine driven EFW Pump No. 4 (Pump No. 2 is analogous)	Fails to deliver working fluid	1a. Faulted steam generator D; steam line break.	EFW Pump No. 4 does not deliver since it has failed. High differential pressure between steam generators A and D causes EFW pump 1/4 crosstie isolation valves (1-9956A, 1-9957A) to close, preventing any flow from feeding to faulted steam generator D. Total flow into the steam generators will be 925 GPM.	Low flow indication on 1-FT-7523; low level on steam generator D level instrumentation.
		1b. Faulted steam generator D; feed line break.	EFW Pump No. 4 does not deliver since it has failed. High differential pressure between steam generators A and D causes EFW Pump 1/4 crosstie isolation valves (1-9956A, 1-9957A) to close, which allows EFW Pump 1 (M/D) to deliver to steam generator A. Total flow into the steam generators will be 925 GPM.	Low flow indication on 1-FT-7523; low level on steam generator D level instrumentation.
		1c. Faulted steam generator A; steam line break.	High differential pressure between steam generators A and D causes EFW Pump 1/4 crosstie isolation valves (1-9956A, 1-9957A) to close, so EFW Pump No. 1 (M/D) cannot deliver to steam generator D. Flow from EFWP	Low flow indication on 1-FT-7523; low level on steam generator D level instrumentation.

TABLE 10.4-2 (Continued)

EMERGENCY FEEDWATER SYSTEM SINGLE ACTIVE FAILURE ANALYSIS

Component	Failure Mode	Plant Mode	Effect on System Operation	Failure Detection Method
1. Turbine driven EFW Pump No. 4 (Cont.)	Fails to deliver working fluid (Cont.)		No. 1 to faulted steam generator A is limited to 450 GPM by the cavitating venturi. Total flow to the intact steam generators will be 750 GPM.	
		1d. Faulted steam generator A; feed line break.	Flow from EFW Pump No. 1 spills out the break, but is limited to 450 GPM so the pump is not damaged. Total flow into the steam generators is 750 GPM.	Low flow indication on 1-FT-7523; high flow indication on 1-FT-7524; low level from steam generators A and D level instrumentation.
		1e. Faulted Steam Generator B, steam line break. (Steam generator C is analogous.)	High differential pressure between steam generators B and C causes EFW pump 2/3 crosstie isolation valves (1-9956B, 1-9957B) to close, which prevents EFWP No.3 from delivering to faulted steam generator C. Feedflow from EFWP No. 2 to steam generator B is limited to 450 GPM by cavitating venturi. Total feedwater flow 925 GPM to steam generators A, C and D.	Valve position indication steam generator B and C level and pressure indication; high flow indication on 1-FT-7525.
		1f. Faulted Steam Generator B, feed line break. (Steam generator C is analogous.)	High differential pressure between steam generators B and C causes EFW pump 2/3 crosstie isolation valves to close, which allows EFWP	Valve position indication; steam generator B and C level and pressure indication; low flow indication on 1-FT-7525.

TABLE 10.4-2 (Continued)

EMERGENCY FEEDWATER SYSTEM SINGLE ACTIVE FAILURE ANALYSIS

Component	Failure Mode	Plant Mode	Effect on System Operation	Failure Detection Method
1. Turbine driven EFW Pump No. 4 (Cont.)	Fails to deliver working fluid (Cont.)		No. 3 to deliver to steam generator C. Flow from EFWP No. 2 out the break is limited to 450 GPM, so pump is not damaged and is capable of being manually realigned and used. Total feedwater flow 925 GPM to steam generators A, C and D.	
		1g. All non-faulted steam generator cases.	No feed flow from EFWP 4 due to pump failure. Total feedwater flow reduced to 1300 GPM to four intact steam generators.	Low pump discharge pressur on 1-PT-7515; flow indicat on 1-FT-7523 and 1-FT-7524
2. Motor driven EFW Pump No. 1 (Pump No. 3 is analogous)	Fails to deliver working fluid	2a. Faulted steam generator D; steam line break.	EFW Pump No. 4 will not deliver, or will deliver at reduced rate because of loss of motive steam to the pump turbine; in the worst case, injection flow to steam generator D will be limited to 450 GPM by the cavitating venturi. Feedwater flow will be 750 GPM total into two intact steam generators.	Low flow indication on 1-FT-7524; flow indication from 1-FT-7523; low level on steam generator A and D level instrumentation.
		2b. Faulted steam generator D; feed line break.	EFW Pump No. 4 will not deliver due to loss of motive steam to the pump turbine. Feedwater flow will be a total of 750 GPM to two intact steam generators.	Low flow indication on 1-FT-7523 and 1-FT-7524; low level indication on steam generator A and D level instrumentation.

TABLE 10.4-2 (Continued)

EMERGENCY FEEDWATER SYSTEM SINGLE ACTIVE FAILURE ANALYSIS

Component	Failure Mode	Plant Mode	Effect on System Operation	Failure Detection Method
2. Motor driven EFW Pump No. 1 (Cont.)	Fails to deliver working fluid (Cont.)	2c. Faulted steam generator A; steam line break.	High differential pressure between steam generators A and D causes EFW Pump 1/4 crosstie isolation valves (1-9956A, 1-9957A) to close. Feedwater flow will be a total of 1125 GPM to three intact steam generators.	Low flow indication on 1-FT-7524; low level on steam generator A level instrumentation.
		2d. Faulted steam generator A; feed line break.	High differential pressure between steam generators A and D causes EFW Pump 1/4 crosstie isolation valves (1-9956A, 1-9957A) to close. Feedwater flow will be a total of 1125 GPM to three intact steam generators.	Low flow indication on 1-FT-7524; steam generato A low level indication; valve position indication.
		2e. Faulted Steam Generator B, steam line break. (Steam generator C is analogous.)	High differential pressure between steam generators B and C causes EFW pump 2/3 crosstie isolation valves close, (1-9956B, 1-9957B) to which prevents EFWP No.3 from delivering to faulted steam generator B. Feedflow from EFWP No. 2 to steam generator B is limited to 450 GPM by cavitating venturi. Feedwater flow will be 750 GPM total to three intact steam generators.	Valve position indication; steam generator B and C level and pressure indicat

TABLE 10.4-2 (Continued)

EMERGENCY FEEDWATER SYSTEM SINGLE ACTIVE FAILURE ANALYSIS

Component	Failure Mode	Plant Mode	Effect on System Operation	Failure Detection Method
2. Motor driven EFW Pump No. 1 (Cont.)	Fails to deliver working fluid	2f. Faulted Steam Generator B, feed line break. (Steam generator C is analogous.)	High differential pressure between steam generators B and C causes EFW pump 2/3 crosstie isolation valves to close, which allows EFWP No. 3 to deliver to steam generator C. Flow from EFWP No. 2 out the break is limited to 450 GPM, so pump is not damaged and is capable of being manually realigned and used. Total feedwater flow is 925 GPM to steam generators A, C and D.	Valve position indication; low level in steam generat B; pressure indication fro steam generators B and C.
		2g. All non-faulted steam generator cases.	Total feedwater flow will be 925 GPM into three intact steam generators.	Low pump discharge pressur on 1-PT-7516; flow indicat on 1-FT-7523 and 1-FT-7524
3. EFW Pump 1/4 crosstie isolation valve 1-9956A; normally open motor operated gate valve. (1-9956B is analgous).	Fails to close on demand.	Faulted steam generator.	Reduces redundancy of valves used to isolate faulted steam generator. No effect on system operation or safety function, since isolation will be provided by 1-9957A.	Valve position indication.
	Closes spuriously.	All.	Eliminates capability for one EFW pump to feed two steam generators; no effect on system operation unless multiple failures are considered.	Valve position indication.

TABLE 10.4-2 (Continued)

EMERGENCY FEEDWATER SYSTEM SINGLE ACTIVE FAILURE ANALYSIS

Component	Failure Mode	Plant Mode	Effect on System Operation	Failure Detection Method
4. EFW Pump 1/4 crosstie isolation valve 1-9957A; normally open motor operated gate valve. (1-9957B is analgous,.	Fails to close on demand.	Faulted steam generator.	Reduces redundancy of valves used to isolate faulted steam generator. No effect on system operation or safety function, since isolation will be provided by 1-9957A.	Valve position indication.
	Closes spuriously.	All.	Eliminates capability for one EFW pump to feed two steam generators; no effect on system operation unless multiple failures are considered.	Valve position indication.
5. Turbine driven EFWP steam admission valve 1-9921A; fail open air operated valve. (1-9921B is analogous.)	Fails to open on demand.	All.	Prevents turbine driven emergency feedwater pump number 4 from operating. Effect of failure on injection flow bounded by failure mode 1 above.	Valve position indication.

TABLE 10.4-2 (Continued)

EMERGENCY FEEDWATER SYSTEM SINGLE ACTIVE FAILURE ANALYSIS

Component	Failure Mode	Plant Mode	Effect on System Operation	Failure Detection Met[illegible]
6. Turbine driven EFW Pump No. 4 steam isolation valve 1-9922A; normally open motor operated gate valve. (1-9922B is analogous).	Fails to close on demand	6a. Steam generator tube rupture, steam generator D.	Ability to isolate the coolant release to atmosphere is compromised.	Valve position indica[illegible]
		6b. All, including steam generator tube rupture.	Redundancy of ability to prevent steam generator overfill is decreased. Injection flow can still be terminated by closing 1-9944A and 1-9956A or 1-9957A. Since 1-9922A is direct current powered and 1-9944A, 1-9956A, and 1-9957A are alternating current powered, no single failure can prevent stopping of injection flow.	Valve position indica[illegible]
	Closes spuriously.	All.	Effect identical to failure of turbine driven EFW Pump No. 4; failure case 1 above.	Valve position indica[illegible]
7. EFW Pump No. 4 throttle valve 1-9944A; normally open motor operated globe valve.	Fails to close on demand.	Faulted steam generator cases.	Eliminates ability to isolate injection flow to faulted steam generator. Spilled flow will be limited to 450 GPM by the cavitating venturi, and effect on system safety function is covered under case 2 above.	Valve position indica[illegible]
(1-9944B is analogous.)	Closes spuriously.	All.	Effect identical to failure of EFW Pump No. 4; described by case 1 above.	Valve position indication.

TABLE 10.4-2 (Continued)

EMERGENCY FEEDWATER SYSTEM SINGLE ACTIVE FAILURE ANALYSIS

Component	Failure Mode	Plant Mode	Effect on System Operation	Failure Detection Method
8. EFW Pump No. 1 throttle valve 1-9945A; normally open motor operated globe valve. (1-9945B is analogous.)	Fails to close on demand.	Faulted steam generator cases.	Eliminates ability to isolate injection flow to faulted steam generator. Spilled flow will be limited to 450 GPM by the cavitating venturi, and effect on system safety function is covered under case 2 above.	Valve position indication.
	Closes spuriously.	All.	Effect identical to failure of EFW Pump No. 4; described by case 1 above.	Valve position indication.
9. Electrical train A (Train B is analogous).	Fails to provide alternating current electrical power. (Direct current power and all instrumentation and actuation signals are maintained by safety grade batteries.)	9a. Steam line break, steam generator D.	Motor driven EFW Pump No. 1 will not operate due to electrical train failure. Flow from EFW Pump No. 4 to steam generator D cannot be throttled since 1-9944A is a train A valve, but will be limited to 450 GPM by the cavitating venturi. In addition, the pump can be stopped by closing 1-9922A, which is a battery operated valve. EFW Pumps Nos. 2 and 3 will provide 750 GPM to steam generators B and C, and this flowrate is completely controllable.	Valve position indication.

TABLE 10.4-2 (Continued)

EMERGENCY FEEDWATER SYSTEM SINGLE ACTIVE FAILURE ANALYSIS

Component	Failure Mode	Plant Mode	Effect on System Operation	Failure Detection Method
9. Electrical train A (Cont.)	Fails to provide alternating current electrical power. (Cont.)	9b. Feed line break, steam generator D.	Motor driven EFW Pump No. 1 will not operate due to electrical train failure. Turbine driven EFW PumpNo. 4 will not operate due to loss of steam supply to pump turbine from faulted steam generator. EFW Pumps Nos. 2 and 3 will provide 750 GPM to steam generators B and C.	
		9c. Steam line break, steam generator A.	Motor driven EFW Pump No. 1 will not operate due to electrical train failure. EFW pump 1/4 crosstie isolation valve 1-9956A will not close due to train failure, but redundant valve 1-9957A is a train B valve and will close, thus preventing EFWP 4 from delivering to steam generator A. Ability to throttle flow from EFWP 4 to steam generator D will be lost, but flow will be limited to 450 GPM by the cavitating venturi, or the pump can be stopped by closing 1-9922A which is a battery operated valve. Total flow will be less than 1350 GPM to three steam generators.	

TABLE 10.4-2 (Continued)

EMERGENCY FEEDWATER SYSTEM SINGLE ACTIVE FAILURE ANALYSIS

Component	Failure Mode	Plant Mode	Effect on System Operation	Failure Detection Method
9. Electrical train A (Cont.)	Fails to provide alternating current electrical power.	9d. Feed line break, steam generator A.	Motor driven EFW Pump No. 1 will not operate due to electrical train failure. EFW pump 1/4 crosstie isolation valve 1-9956A will not close due to train failure, but redundant valve 1-9957A is a train B valve and will close, thus stopping EFWP 4 from pumping out the break and allowing it to deliver to steam generator D.	
		9e. Steam line break, steam generator B. (Steam generator C is analogous.)	Motor driven EFW Pump No will not operate due to electrical train failure. EFW pump 2/3 crosstie isolation valve 1-9956B will not close due to train failure, but redundant valve 1-9957B is a train B valve and will close, thus preventing EFWP 3 from delivering to the faulted generator. EFWP 2 can be stopped normally. Flow to the intact steam generators can be controlled with EFWP No. 3 throttle valve 1-9944B or by stopping EFWP No.4.	

TABLE 10.4-2 (Continued)

EMERGENCY FEEDWATER SYSTEM SINGLE ACTIVE FAILURE ANALYSIS

Component	Failure Mode	Plant Mode	Effect on System Operation	Failure Detection Method
9. Electrical train A (Cont.)	Fails to provide alternating current electrical power (Cont.)	9f. Feed line break, steam generator B. (Steam generator C is analogous.)	Motor driven EFW Pump No. 1 will not operate due to electrical train failure. EFW pump 2/3 crosstie isolation valve 1-9956B will not close due to train failure, but redundant valve 1-9957B is a train B valve and will close, thus allowing EFWP No. 3 to deliver to steam generator C. Feed flow to the steam generators will be 925 GPM.	
10. EFWS signal train A. (Train B signal is analogous.)	Fails to occur.	All.	Motor driven EFW Pump No. 1 is not actuated. Turbine driven EFW Pump No. 4 will start because the air operated steam admission valve (1-9921A) recieves both train A and train B actuation signals. EFW Pump Nos. 2 and 3 also both start.	

4 3 2 1 (a,c)

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Figure 10.4-1
WAPWR Emergency Feedwater System
Flow Diagram
(Sheet 1 of 2)

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Figure 10.4-1

WAPWR Emergency Feedwater System
Flow Diagram
(Sheet 2 of 2)

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APPENDIX 10A

STEAM AND POWER CONVERSION SYSTEM - INTERFACES

This appendix provides NPB interfaces for the design of those areas of the steam and power conversion system which are the plant specific applicant's responsibility.

10A.1 SUMMARY DESCRIPTION

10A.1.1 Steam and Power Conversion System Parameters

A summary of pertinent steam and power conversion system parameters is as follows:

Thermal Input, MWt (100% power)	3816
Pressure, psia (100% power)	1000
Temperature, °F (100% power)	saturated
Flow rate per steam generator, 10^6 lb/hr	17.04

10A.1.2 General Design Requirements

The plant specific applicant will provide necessary systems for the utilization of the NPB's nuclear steam supply system energy produced. These systems include those portions of the secondary cycle and its supporting systems extending from the turbine building side of the steam tunnel/turbine building wall through the turbine-generator and back to this wall.

10A.2 TURBINE GENERATOR

The design of the turbine-generator (T-G) is the responsibility of the plant specific applicant. The design must meet the following interface criteria:

1. The T-G load change characteristics shall be designed to be compatible with the NPB instrumentation and control system which coordinates T-G and reactor operation (see RESAR-SP/90 PDA Module 9, "I&C and Electric Power" for further discussion).

2. The T-G orientation shall be as shown on Figure 1.2-1 and 1.2-2 of RESAR-SP/90 PDA Module 3, "Introduction and Site".

3. The T-G shall be designed to permit periodic testing of steam valves important to overspeed protection, emergency overspeed trip circuits, and several other trip circuits under load.

4. The T-G shall be designed to permit unlimited access to all levels of the turbine areas under all operating conditions.

5. The T-G unit and all associated piping, valves, and controls shall be located completely within the turbine building.

6. The T-G unit shall be designed to trip following receipt of the following trip signals related to reactor operation

   a) reactor trip
   b) manual trip from control room
   c) low condenser vacuum

10A.3 MAIN STEAM SUPPLY SYSTEM

The design of the nonsafety-related portions of the MSSS is the responsibility of the plant specific applicant. The design must meet the following interface criteria:

1. The applicant's portion of the MSSS piping shall be compatible with the interface connections from the SGIS as shown in Figure 10.3-1.

2. The applicant's portion of the MSSS shall be capable of accommodating the steam delivered from the SGIS consistent with the parameters given in Table 10.3-1.

10A.4 OTHER FEATURES OF STEAM AND POWER CONVERSIONS SYSTEM

10A.4.1 Main Condenser

The design of the main condenser is the responsibility of the plant specific applicant. The design must meet the following interface criteria:

1. The main condenser shall be capable of accommodating up to 40 percent of the VWO main steam flow which is bypassed directly to the condenser by the turbine bypass system.

2. The main condenser shall provide the surge volume required for the condensate and feedwater system.

10A.4.2 Main Condenser Evacuation System

Main condenser evacuation is performed by the main condenser air removal system (MCARS). The design of the MCARS is the responsibility of the plant specific applicant. The design must meet the following interface criterion:

1. A deaerating feedwater heater is recommended for feedwater oxygen removal so that the feedwater oxygen content will not exceed 5 ppb under any normal operating condition. It shall provide for feedwater heating so that the minimum feedwater temperature will be at least 450°F, at full power, under any normal operating condition.

10A.4.3 Turbine Gland Sealing System

There are no interface criteria necessary between the WAPWR NPB and the turbine gland sealing system.

10A.4.4 Turbine Bypass System

The design of the turbine bypass system (TBS) is the responsibility of the plant specific applicant. The design must meet the following interface criteria:

1. The TBS must have the capacity to bypass 40 percent of the VWO main steam flow to the main condenser.

2. The TBS must be designed to bypass steam to the main condenser during plant startup and to permit a normal manual cooldown of the reactor coolant system from a hot shutdown condition to a point consistent with the initiation of operation of the residual heat removal portion of the integrated safeguards system.

3. The TBS shall permit a 50 percent electrical step-load reduction without reactor trip. The system shall also allow a turbine and reactor trip from full power without lifting the main steam relief and safety valves.

10A.4.5 Circulating Water System

There are no interface criteria necessary between the WAPWR NPB and the circulating water system.

10A.4.6 Condensate Cleanup System

The design of the condensate cleanup system (CCS) is the responsibility of the plant specific applicant. The design must meet the following interface criteria:

1. The CCS makeup water, including treated water from the steam generator blowdown processing system, shall meet the requirements given in Table 10A.4-1.

2. The CCS shall have the capability of maintaining the steam side water chemistry within the requirements given in Table 10A.4-2.

10A.4.7 Condensate and Feedwater System

As discussed in Subsection 10.4.7, the design of essentially all of the non-safety related portions of the condensate feedwater system (CFS) is the responsibility of the plant specific applicant. The design must meet the following interface criteria:

1. The applicant's portion of the CFS shall be capable of maintaining feedwater flow following a 50 percent step reduction in electrical load.

2. The applicant's portion of the CFS shall be designed to provide a continuous feedwater supply at required pressure, temperature, and dissolved oxygen concentration conditions to the WAPWR NPB portion of the CFS (i.e., the SGIS) for delivery to the four steam generators under anticipated steady-state and transient conditions.

3. The applicant's portion of the CFS shall be compatible with the interface connections to the SGIS as shown in Figure 10.3-1.

4. The applicant's portion of the CFS shall have the capability of adding hydrazine and an amine at the condensate pump discharge with assurance that each feedwater train receives equal chemical addition. The system shall also have the capability of adding hydrazine and ammonium hydroxide to the steam generators via the main feedwater pumps.

5. The applicant's portion of the CFS shall have appropriate controls to trip the feedwater pumps upon any one of the following:

   Both SGFP turbines are tripped upon any one of the following:

   a. High-high level in any one steam generator

b. Feedwater isolation signal from the engineered safety features actuation system

c. Any condition which actuates safety injection

d. Trip of all condensate pump motors

e. High feedwater system pressure.

10A.4.8 Steam Generator Blowdown Processing System

The design of the nonsafety-related portions of the SGBPS is the responsibility of the plant specific applicant. The design must meet the following interface criteria:

1. The applicant's portion of the SGBPS shall be designed to ensure that blowdown treatment is compatible with the condensate and feedwater to ensure an effective secondary system water chemistry control program as discussed in Subsections 10.3.5 and 10A.4.6.

2. The applicant's portion of the SGBPS shall be designed to accommodate continuous flows up to 43,500 pounds per hour (nominally 90 gpm) per steam generator, while returning to the feedwater system a sizable portion of the heat removed from the steam generators.

3. During normal operation without primary-to-secondary leakage, the applicant's portion of the SGBPS shall be designed to process blowdown to meet the chemical composition limits for release to the environment or for return to the condenser hotwell/condensate storage tank.

4. Portions of the applicant's portion of the SGBPS shall use design and fabrication codes consistent with quality group D (augmented) as assigned by Regulatory Guide 1.143 for radioactive waste management systems.

5. The applicant's portion of the SGBPS shall be compatible with the interface connections to the SGIS as shown in Figure 10.3-1.

10A.4.9 Emergency Feedwater System

The EFWS is entirely within the WAPWR NPB scope of supply. However, the plant specific applicant must provide the following:

1. The applicant shall supply an alternate emergency feedwater source. This may be a single tank (including the condensate storage tank) or reservoir, or several tanks and/or reservoirs depending on plant site layout. The alternate water source is not safety grade, but must contain sufficient condensate quality water to allow the plant to be maintained in hot standby conditions, with one reactor coolant pump operating for two days after a reactor trip. The maximum permissible water temperature in the alternate water supply shall be 120°F.

10A.4.10 Startup Feedwater System

The design of the SFWS is the responsibility of the plant specific applicant. The design must meet the following design criteria[1]:

1. The SFWS shall be comprised of the following components[2]:

    a. One motor driven feedwater pump
    b. Piping and valves as required.

2. All components of the SFWS shall be located in the turbine building.

3. Appropriate instrumentation shall be provided in order to monitor proper operation of the system.

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(1) See subsection 1.1.1.3 of RESAR-SP/90 PDA Module 3, "Introduction and Site" for the definition of design criteria.

(2) See Figure 10A.4-1 for a schematic of a suggested SFWS configuration.

4. The SFWS shall include a safety grade isolation valve, a safety grade flowmeter, and a flow control valve.

5. The SFWS shall have the ability of automatically delivering 250° water to the steam generators. It is anticipated that this will be accomplished by using a deaerating heater as the normal suction supply. The SFWS pumps shall also be capable of taking suction from the condensate storage tank, or from the main condenser hotwell.

6. The SFWS shall be designed to automatically start and to provide heated feedwater following a reactor trip, loss of main feed, loss of offsite power, and other anticipated transients.

7. The SFWS components shall be sized to remove sufficient core heat for all Condition I and II events so that, under normal circumstances, the EFWS should not normally be actuated for emergency (Condition III) and faulted (Condition IV) events. In order to accomplish this the SFWS shall deliver 900 gpm of water with a steam generator pressure of 1220 psig.

8. The SFWS shall be compatible with the interface connections to the SGIS and EFWS as shown in Figures 10.3-1 and 10.4-1, respectively.

9. The SFWS safety class shall be NNS per Table 3.2-1.

10A.4.11 Secondary Liquid Waste System

There are no interface criteria necessary between the WAPWR NPB and the secondary liquid waste system.

TABLE 10A.4-1

GUIDELINES FOR CONDENSATE STORAGE AND EMERGENCY FEEDWATER TANKS

Specific conductivity (μS/cm @ [illegible]°C)	≤ 0.1[a]
Silica (ppb)	≤ 10
Dissolved oxygen (ppb)	≤ 100
Total organic carbon (ppb)	≤ 100
Hydrazine in aux. feedwater	≥ 3x $[O_2]$

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[a] Prior to the introduction of additives or adsorption of carbon dioxide.

TABLE 10A.4-2

AVT CHEMISTRY GUIDELINES FOR
STEAM GENERATOR BLOWDOWN, CONDENSATE AND FEEDWATER

Chemistry Parameters	Cold Shutdown/ Wet Layup Blowdown	Heatup to < 5% Power Blowdown	Power Operation Condensate	Power Operation Feedwater	Power Operation Blowdown
Cation Conductivity @ 25°C, μs/cm	NA	≤ 2.0	≤ 0.15	≤ 0.2	< 0.8
Sodium, ppb	≤ 1000	≤ 100	<< 1	NA	≤ 20
Chloride, ppb	≤ 1000	≤ 100	NA	NA	≤ 20
Sulfate, ppb	≤ 1000	≤ 100	NA	NA	≤ 20
Hydrazine, ppm	75-200	NA	NA	≥ 0.02	NA
Dissolved Oxygen, ppb	NA	≤ 5	< 10	≤ 5	NA
SiO_2, ppb	NA	≤ 1000	NA	NA	≤ 300
Fe, ppb	NA	NA	NA	≤ 20	NA
Cu, ppb	NA	NA	NA	≤ 2	NA
Total Organic Carbon, ppb	≤ 100	NA	< 100	NA	NA
Suspended Solids, ppb	NA	NA	NA	NA	< 1000
Blowdown Rate	NA	NA	NA	NA	< 0.5% FW
Ferrous Systems					
pH @ 25°C	9.8-10.5	≥ 9.0	NA	9.3-9.6	9.0-9.5
Specific Conductivity @ 25°C, μS/cm	NA	≥ 1	NA	5.2-10	< 3.0
Ammonia, ppb	As pH requires	≥ 60	NA	700-2000	> 250
Ferrous/Copper Systems					
pH @ 25°C	9.8-10.5	8.5-9.2	NA	8.8-9.2	8.5-9.0
Specific Conductivity @ 25°C, μS/cm	NA	≥ 1	NA	1.8-4.2	> 1.0
Ammonia, ppb	As pH requires	≥ 60	NA	150-500	> 60

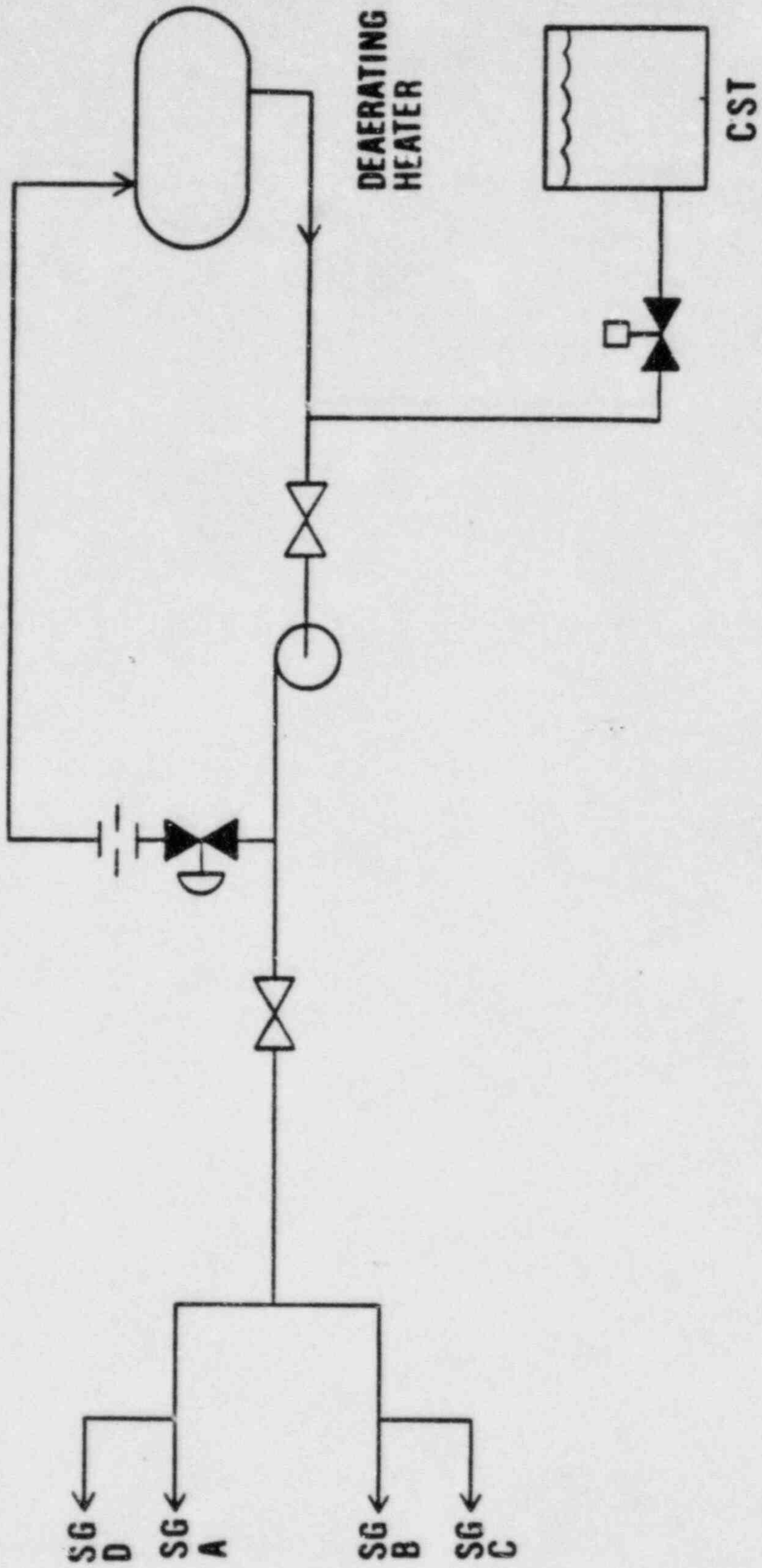


FIGURE 10A.4-1 STARTUP FEEDWATER SYSTEM
(SUGGESTED ARRANGEMENT)

11.0 RADIOACTIVE WASTE MANAGEMENT

No portion of this chapter is pertinent to the RESAR-SP/90 "Secondary Side Safeguards System/Steam and Power Conversion System" module.

12.0 RADIATION PROTECTION

No portion of this chapter is pertinent to the RESAR-SP/90 "Secondary Side Safeguards System/Steam and Power Conversion System" module.

13.0 CONDUCT OF OPERATIONS

See the applicant's safety analysis report for a discussion of "Conduct of Operations".

14.0 INITIAL TEST PROGRAM

No portion of this chapter is pertinent to the RESAR-SP/90 "Secondary Side Safeguards System/Steam and Power Conversion System" module.

15.0 ACCIDENT ANALYSES

15.0.1 General

This chapter addresses the representative initiating events listed on Table 15-1 of Regulatory Guide 1.70, Revision 3, the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants", as they apply to a Westinghouse pressurized water reactor.

Certain items of Table 15-1 in the guide warrant comment, as follows:

1. Items 1.3 and 2.1 - There are no pressure regulators in the Nuclear Steam Supply System (NSSS) pressurized water reactor (PWR) design whose malfunction or failure could cause a steam flow transient.

2. Item 6.2 - No instrument lines from the reactor coolant pressure boundary in the NSSS PWR design penetrate the Containment. (For the definition of the Reactor Coolant System boundary, refer to Section 5, ANSI-N18.2, "Nuclear Safety Criteria for the Design of Stationary PWR Plants," 1973.)

15.0.2 Classification of Plant Conditions

Since 1970 the ANS classification of plant conditions has been used to divide plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

1. Condition I: Normal Operation and Operational Transients.
2. Condition II: Faults of Moderate Frequency.
3. Condition III: Infrequent Faults.
4. Condition IV: Limiting Faults.

The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk to the public and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur. Where applicable, reactor trip system and engineered safeguards functioning is assumed to the extent allowed by considerations, such as the single failure criterion, in fulfilling this principle.

15.0.2.1 Condition I - Normal Operation and Operational Transients

Condition I occurrences are those which are expected frequently or regularly in the course of normal plant operation, refueling, and maintenance. As such, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Inasmuch as Condition I occurrences occur frequently or regularly, they must be considered from the point of view of affecting the consequences of fault conditions (Conditions II, III and IV). In this regard, analysis of each fault condition described is generally based on a conservative set of initial conditions corresponding to adverse conditions which can occur during Condition I operation.

Typical Condition I events are as follows:

1. Steady state and shutdown operations

   a. Mode 1 - Power operation (> 5 to 100 percent of rated thermal power).

   b. Mode 2 - Startup ($K_{eff} \geq 0.99$, $\leq$ 5 percent of rated thermal power).

   c. Mode 3 - Hot standby ($K_{eff} < 0.99$, $T_{avg} \geq 350°F$).

   d. Mode 4 - Hot shutdown ($K_{eff} < 0.99$, $200°F \leq T_{AVG} \leq 350°F$).

e. Mode 5 - Cold Shutdown (K_{eff} < 0.99, T_{avg} < 200°F).

f. Mode 6 - Refueling ($K_{eff} \leq 0.95$, $T_{avg} \leq 140$°F).

2. Operation with permissible deviations

Various deviations which may occur during continued operation as permitted by the plant Technical Specifications must be considered in conjunction with other operational modes. These include:

a. Operation with components or systems out of service (such as power operation with a reactor coolant pump out of service).

b. Radioactivity in the reactor coolant, due to leakage from fuel with cladding defects and other sources.

1) Fission products
2) Corrosion products
3) Tritium

c. Operation with steam generator primary-to-secondary leakage up to the maximum allowed by the Technical Specifications.

d. Testing as required by the Technical Specifications.

3. Operational transients

a. Plant heatup and cooldown (up to 100°F/hour for the reactor coolant system; 200°F/hour for the pressurizer during cooldown and 100°F/hour for the pressurizer during heatup).

b. Step load changes (up to ± 10 percent).

c. Ramp load changes (up to 5 percent/minute).

d. Load rejection up to and including design full load rejection transient.

15.0.2.2 Condition II - Faults of Moderate Frequency

At worst, a Condition II fault results in a reactor trip with the plant being capable of returning to operation. By definition, these faults (or events) do not propagate to cause a more serious fault, i.e., Condition III or IV events. In addition, Condition II events are not expected to result in fuel rod failure or reactor coolant system or secondary system overpressurization.

The following faults are included in this category:

1. Feedwater system malfunctions causing a reduction in feedwater temperature (Subsection 15.1.1 of this module).

2. Feedwater system malfunctions causing an increase in feedwater flow (Subsection 15.1.2 of this module).

3. Excessive increase in secondary steam flow (Subsection 15.1.3 of this module).

4. Inadvertent opening of a steam generator relief or safety valve causing a depressurization of the main steam system (Subsection 15.1.4 of this module).

5. Loss of external load (Subsection 15.2.2 of this module).

6. Turbine trip (Subsection 15.2.3 of this module).

7. Inadvertent closure of main steam isolation valves (Subsection 15.2.4 of this module).

8. Loss of condenser vacuum and other events resulting in turbine trip (Subsection 15.2.5 of this module).

9. Loss of nonemergency A-C power to the station auxiliaries (Subsection 15.2.6 of this module).

10. Loss of normal feedwater flow (Subsection 15.2.7 of this module).

11. Partial loss of forced reactor coolant flow (Subsection 15.3.1 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System").

12. Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition (Subsection 15.4.1 of RESAR-SP/90 PDA Module 5, "Reactor System").

13. Uncontrolled rod cluster control assembly bank withdrawal at power (Subsection 15.4.2 of RESAR-SP/90 PDA Module 5, "Reactor System").

14. Control rod misalignment - Dropped full length assembly, dropped full length assembly bank, or statically misaligned full length assembly (Subsection 15.4.3 of RESAR-SP/90 PDA Module 5, "Reactor System" and RESAR-SP/90 PDA Module 9, "I&C and Electric Power").

15. Startup of an inactive reactor coolant loop at an incorrect temperature (Subsection 15.4.4 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System").

16. Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant (Subsection 15.4.6 of RESAR-SP/90 PDA Module 13, "Auxiliary Systems").

17. Inadvertent operation of emergency core cooling system during power operation (Subsection 15.5.1 of RESAR-SP/90 PDA Module 1, "Primary Side Safeguards System").

18. Chemical and volume control system malfunction that increases reactor coolant inventory (Subsection 15.5.2 of RESAR-SP/90 PDA Module 13, "Auxiliary Systems").

19. Inadvertent opening of a pressurizer safety or relief valve (Subsection 15.6.1 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System").

20. Failure of small lines carrying primary coolant outside containment (Subsection 15.6.2 of RESAR-SP/90 PDA Module 1, "Primary Side Safeguards System").

15.0.2.3 Condition III - Infrequent Faults

By definition, Condition III occurrences are faults which may occur very infrequently during the life of the plant. They will be accommodated with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude immediate resumption of the operation. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion area boundary. A Condition III fault will not, by itself, generate a Condition IV fault or result in a consequential loss of function of the reactor coolant system or containment barriers. The following faults are included in this category:

1. Minor steam system piping failures (Subsection 15.1.5 of this module).

2. Complete loss of forced reactor coolant flow (Subsection 15.3.2 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System").

3. Control rod misalignment - Single rod cluster control assembly withdrawal at full power (Subsection 15.4.3 of RESAR-SP/90 PDA Module 5, "Reactor System").

4. Inadvertent loading and operation of a fuel assembly in an improper position (Subsection 15.4.7 of RESAR-SP/90 PDA Module 5, "Reactor System").

5. Loss of reactor coolant from small ruptured pipes or from cracks in large pipes, which actuate the emergency core cooling system (Subsection 15.6.4 of RESAR-SP/90 PDA Module 1, "Primary Side Safeguards System").

6. Waste gas system failure (Subsection 15.7.1 of RESAR-SP/90 PDA Module 12, "Waste Management").

7. Radioactive liquid waste system leak or failure (atmospheric release) (Subsection 15.7.2 of RESAR-SP/90 PDA Module 12, "Waste Management").

8. Liquid containing tank failure (Subsection 15.7.3 of RESAR-SP/90 PDA Module 12, "Waste Management").

15.0.2.4 Condition IV - Limiting Faults

Condition IV occurrences are faults which are not expected to occur, but are postulated because their consequences would include the potential for release of significant amounts of radioactive material. They are the most drastic which must be designed against and represent limiting design cases. Plant design must be such as to preclude a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR 100. A single Condition IV fault must not cause a consequential loss of required functions of systems needed to mitigate the consequences of the fault including those of the emergency core cooling system and containment. The following faults have been classified in this category:

1. Steam system piping failure (Subsection 15.1.5 of this module).

2. Feedwater system pipe break (Subsection 15.2.8 of this module).

3. Reactor coolant pump rotor seizure (locked rotor) (Subsection 15.3.3 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System").

4. Reactor coolant pump shaft break (Subsection 15.3.4 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System").

5. Spectrum of rod cluster control assembly ejection accidents (Subsection 15.4.8 of RESAR-SP/90 PDA Module 5, "Reactor System").

6. Steam generator tube failure (Subsection 15.6.3 of this module).

7. Loss-of-coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary (Subsection 15.6.4 of RESAR-SP/90 PDA Module 1, "Primary Side Safeguards System").

8. Fuel handling accident (Subsection 15.7.4 of RESAR-SP/90 PDA Module 12, "Waste Management").

15.0.3 Optimization of Control Systems

A control system automatically maintains prescribed conditions in the plant even under a conservative set of reactivity parameters with respect to both system stability and transient performance. For each mode of plant operation, a group of optimum controller setpoints is determined. In areas where the resultant setpoints are different, compromises based on the optimum overall performance are made and verified. A consistent set of control system parameters is derived satisfying plant operational requirements throughout the core life and for various levels of power operation.

The system setpoints are derived by an analysis of the following control systems: rod control, steam dump, steam generator level, pressurizer pressure and pressurizer level.

15.0.4 Plant Characteristics and Initial Conditions Assumed in the Accident Analyses

15.0.4.1 Design Plant Conditions

Table 15.0-1 gives the guaranteed nuclear steam supply system thermal power output which is assumed in analyses performed in this report. This power output includes the thermal power generated by the reactor coolant pumps and is consistent with the license application rating described in Chapter 1.0. Allowances for errors in the determination of the steady-state power level are made as described in Subsection 15.0.4.2. The values of pertinent plant parameters utilized in the accident analyses are given in Table 15.0-2. The thermal power values used for each transient analyzed are given in Table 15.0-3.

15.0.4.2 Initial Conditions

For most accidents which are DNB limited, nominal values of initial conditions are assumed. The allowances on power, temperature, and pressure noted above are determined on a statistical basis and are included in the limit DNBR, as described in WCAP-8567 (Reference 1). This procedure is known as the "Improved Thermal Design Procedure," and is discussed more fully in Section 4.4 of RESAR-SP/90 PDA Module 5, "Reactor System".

For accidents which are not DNB limited, or for which the Improved Thermal Design Procedure is not employed, initial conditions are obtained by adding the maximum steady state errors to rated values. The following conservative steady state errors were assumed in the analysis:

1. Core power	± 2% allowance for calorimetric error
2. Average reactor coolant system temperature	± 4°F allowance for controller deadband and measurement error
3. Pressurizer pressure	± 30 psi allowance for steady-state fluctuations and measurement error.

Table 15.0-3 summarizes initial conditions and computer codes used in the accident analysis, and shows which accidents employed a DNB analysis using the Improved Thermal Design Procedure.

15.0.4.3 Power Distribution

The limiting conditions occurring during reactor transients are dependent on the core power distribution. The design of the core and the control system minimizes adverse power distribution through the placement of control rods and operating methods. In addition, the core power distribution is continuously monitored by the integrated protection system as described in RESAR-SP/90 PDA Module 9, "I&C and Electric Power" and the Technical Specifications. Audible alarms will be activated in the control room whenever the power distribution exceeds the limits assumed as initial conditions for the transients presented in this chapter.

For transients which may be DNB limited both the radial and axial peaking factors are of importance. The core thermal limits illustrated in Figure 15.0-1 are based on a reference axial power shape. The low DNBR reactor trip setpoint is automatically adjusted for axial shapes differing from the reference shape by the method described in Section 4.4 of RESAR-SP/90 PDA Module 5, "Reactor System" and also described in RESAR-SP/90 PDA Module 9, "I&C and Electric Power". The radial peaking factor $F_{\Delta H}$ increases with decreasing power and with increasing rod insertion. The increase in $F_{\Delta H}$ resulting from decreasing reactor power and increased rod insertion is accounted for in the low DNBR reactor trip through measurement of power and control rod position.

For transients which may be overpower limited, the total peaking factor F_q is of importance. F_q is continuously monitored through the high Kw/ft reactor trip as described in RESAR-SP/90 PDA Module 9, "I&C and Electric Power" and the Technical Specifications to assure that the limiting overpower conditions are not exceeded.

For overpower transients which are slow with respect to the fuel rod thermal time constant, fuel rod thermal evaluations are determined as discussed in Section 4.4 of RESAR-SP/90 PDA Module 5, "Reactor System". Examples of this are the uncontrolled boron dilution incident, which lasts many minutes, and the excessive load increase incident, which reaches equilibrium without causing a reactor trip. For overpower transients which are fast with respect to the fuel rod thermal time constant (for example, the uncontrolled rod cluster control assembly bank withdrawal from subcritical and rod cluster control assembly ejection incidents, which result in a large power rise over a few seconds), a detailed fuel heat transfer calculation is performed.

Although the fuel rod thermal time constant is a function of system conditions, fuel burnup, and rod power, a typical value at beginning-of-life for high power rods is approximately 5 seconds.

15.0.5 Reactivity Coefficients Assumed in the Accident Analyses

The transient response of the reactor system is dependent on reactivity feedback effects, in particular the moderator temperature coefficient and the Doppler power coefficient. These reactivity coefficients and their values are discussed in detail in this module.

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values whereas, in the analysis of other events, conservatism requires the use of small reactivity coefficient values. Some analyses, such as loss of reactor coolant from cracks or ruptures in the reactor coolant system, do not depend highly on reactivity feedback effects. The values used for each accident are given in Table 15.0-3. Reference is made in that table to Figure 15.0-2 which shows the upper and lower bound Doppler power coefficients as a function of power, used in the transient analysis. The justification for use of conservatively large vs. small reactivity coefficient values are treated on an event-by-event basis. Conservative combinations of parameters are used for a given transient to bound the effects of core life, although these combinations may not represent possible realistic situations.

15.0.6 Rod Cluster Control Assembly Insertion Characteristics

The negative reactivity insertion following a reactor trip is a function of the position vs. time of the rod cluster control assemblies and the variation in rod worth as a function of rod position. With respect to accident analyses, the critical parameter is the time of insertion up to the dashpot entry or approximately 85% of the rod cluster travel. For all accidents the insertion time to dashpot entry is conservatively taken as 3.4 seconds. The normalized rod cluster control assembly position vs. time assumed in accident analyses is shown in Figure 15.0-3.

Figure 15.0-4 shows the fraction of total negative reactivity insertion vs. normalized rod position for a core where the axial distribution is skewed to the lower region of the core. An axial distribution which is skewed to the lower region of the core can arise from an unbalanced xenon distribution. This curve is used to compute the negative reactivity insertion vs. time following a reactor trip which is input to all point kinetics core models used in transient analyses. The bottom skewed power distribution itself is not an input into the point kinetics core model.

There is inherent conservatism in the use of Figure 15.0-4 in that it is based on a skewed flux distribution which would exist relatively infrequently. For cases other than those associated with unbalanced xenon distributions, significant negative reactivity would have been inserted due to the more favorable axial distribution existing prior to trip.

The normalized rod cluster control assembly negative reactivity insertion vs. time is shown in Figure 15.0-5. The curve shown in this figure was obtained from Figures 15.0-3 and 15.0-4. A total negative reactivity insertion following a trip of 4% $\Delta\rho$ is assumed in the transient analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available as shown in Section 4.3 of RESAR-SP/90 PDA Module 5, "Reactor System".

The normalized rod cluster control assembly negative reactivity insertion vs. time curve for an axial power distribution skewed to the bottom (Figure 15.0-5) is used for those transient analyses for which a point kinetics core model is used. Where special analyses required use of three-dimensional or axial one-dimensional core models, the negative reactivity insertion resulting from the reactor trip is calculated directly by the reactor kinetics code and is not separable from the other reactivity feedback effects. In this case, the rod cluster control assembly position vs. time (Figure 15.0-3) is used as code input.

15.0.7 Trip Points and Time Delays to Trip Assumed in Accident Analyses

A reactor trip signal acts to open eight trip breakers, two per channel set, feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the rod cluster control assemblies which then fall by gravity into the core. There are various instrumentation delays associated with each trip function, including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanism. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and begin to fall. Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in Table 15.0-4.

Reference is made in Table 15.0-4 to the low DNBR trips shown in Figure 15.0-1. These figures present the allowable reactor power as a function of the coolant loop inlet temperature and primary coolant pressure for N loop operation (4-loop operation), for the design flow and power distribution, as described in Section 4.4 of RESAR-SP/90 PDA Module 5, "Reactor System".

The boundaries of operation defined by the low DNBR trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions trip would occur well within the area bounded by these lines. The DNB lines represent the locus of conditions for which the DNBR equals the limit value of 1.62. All points below and to the left of a DNB line for a given pressure have DNBR greater than the limit value with the assumed axial and radial power distributions. The diagram shows that the DNB design basis is not violated for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line at any point.

The area of permissible operation (power, pressure and temperature) is bounded by the combination of reactor trips: high neutron flux (fixed setpoint); high pressure (fixed setpoint); low pressure (fixed setpoint); low DNBR (variable setpoint); high kw/ft (fixed setpoint).

The limit value, which was used as the DNBR limit for all accidents analyzed with the Improved Thermal Design Procedure (see Table 15.0-3), is conservative compared to the actual design DNBR value required to meet the DNB design basis is discussed in Section 4.4 of RESAR-SP/90 PDA Module 5, "Reactor System".

The difference between the limiting trip point assumed for the analysis and the nominal trip point represents an allowance for instrumentation channel error and setpoint error. Nominal trip setpoints are specified in the plant Technical Specifications. During plant startup tests, it is demonstrated that actual instrument time delays are equal to or less than the assumed values. Additionally, protection system channels are calibrated and instrument response times determined periodically in accordance with the plant Technical Specifications.

15.0.8 Instrumentation Drift and Calorimetric Errors - Power Range Neutron Flux

The instrumentation drift and calorimetric errors used in establishing the power range high neutron flux setpoint are presented in Table 15.0-5. The calorimetric error is the error assumed in the determination of core thermal power as obtained from secondary plant measurements. The total ion chamber current (sum of the multiple sections) is calibrated (set equal) to this measured power on a periodic basis.

The secondary power is obtained from measurement of feedwater flow, feedwater inlet temperature to the steam generators and steam pressure. High accuracy instrumentation is provided for these measurements with accuracy tolerances much tighter than those which would be required to control feedwater flow.

15.0.9 Plant Systems and Components Available for Mitigation of Accident Effects

The Westinghouse nuclear steam supply system (NSSS) is designed to afford power protection againts the possible effects of natural phenomena, postulated environmental conditions, and the dynamic effects of the postulated accident.

In addition, the design incorporates features which minimize the probability and effects of fires and explosions. Chapter 17.0 of the RESAR-SP/90 integrated PDA document will discuss the quality assurance program which is implemented to ensure that the plant will be designed, constructed, and operated without undue risk to the health and safety of the general public. The incorporation of these features, coupled with the reliability of the design, ensures that the normally operating systems and components listed in Table 15.0-6 will be available for mitigation of the events discussed in Chapter 15. In determining which systems are necessary to mitigate the effects of these postulated events, the classification system of ANSI-N18.2-1973 is utilized. The design of "systems important to safety" (including protection systems) is consistent with IEEE 379-1972 and Regulatory Guide 1.53 in the application of the single failure criterion.

In the analysis of the Chapter 15 events, the operation of the non-safety-related rod control system, other than the reactor trip portion of the control rod drive system (CRDS), is considered only if that action results in more severe consequences. No credit is taken for control system operation if that operation mitigates the results of an accident. For some accidents, the analysis is performed both with and without control system operation to determine the worst case. The pressurizer heaters are not assumed to be energized during any of the Chapter 15 events.

15.0.10 Fission Product Inventories

15.0.10.1 Inventory in the Core

The time dependent fission product inventories in the reactor core are calculated by the ORIGEN code(10) using a data library based on ENDF/B-IV.(11) Core inventories are shown in Table 15.0-7.

The fission product radiation sources considered to be released from the fuel to the containment following a maximum credible accident are based on the assumptions stated in TID-14844(3): 100 percent of the noble gases and 50 percent of the halogens.

15.0.10.2 Inventory in the Fuel Pellet Clad Gap

The radiation sources associated with a gap activity release accident are based on the assumption that the fission products in the space between the fuel pellets and the cladding of all fuel rods in the core are released as a result of cladding failure.

The gap activities were determined using the model suggested in Regulatory Guide 1.25. Specifically, 10 percent of the iodine and noble gas activity (except Kr-85, I-127, and I-129, which are 30 percent) is accumulated in the fuel clad gap. The gap activities are shown in Table 15.0-7.

15.0.10.3 Inventory in the Reactor Coolant

Reactor coolant iodine concentrations for the Technical Specification limit of 1 μCi/gm of dose equivalent (D.E.) I-131 and for the assumed pre-accident iodine spike concentration of 60 μCi/gm of D.E. I-131 are presented in Table 15.0-8. Reactor coolant noble gas concentrations based on 1 percent fuel defects are presented in Table 15.0-9. Iodine appearance rates in the reactor coolant, for normal steady state operation at 1 μCi/gm of D.E. I-131, and for an assumed accident initiated iodine spike are presented in Table 15.0-10.

15.0.11 <u>Residual Decay Heat</u>

15.0.11.1 Total Residual Heat

Residual heat in a subcritical core is calculated for the loss-of-coolant accident per the requirements of Appendix K, 10 CFR 50.46, as described in References 5 and 6. These requirements include assuming infinite irradiation time before the core goes subcritical to determine fission product decay energy. For all other accidents, the same models are used except that fission product decay energy is based on core average exposure at the end of the equilibrium cycle.

15.0.12 Computer Codes Utilized

Summaries of some of the principal computer codes used in transient analyses are given below. Other codes, in particular, very specialized codes in which the modeling has been developed to simulate one given accident, such as those used in the analysis of the reactor coolant system pipe rupture (Section 15.6), are summarized in their respective accident analyses sections. The codes used in the analyses of each transient are listed in Table 15.0-3.

15.0.12.1. FACTRAN

FACTRAN calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod and the transient heat flux at the surface of the cladding using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, and density). The codes uses a fuel model which exhibits the following features simultaneously:

1. A sufficiently large number of radial space increments to handle fast transients such as rod ejection accidents.

2. Material properties which are functions of temperature and a sophisticated fuel-to-clad gap heat transfer calculation.

3. The necessary calculations to handle post DNB transient: film boiling heat transfer correlations, Zircaloy-water reaction and partial melting of the materials.

FACTRAN is further discussed in Reference 7.

15.0.12.2 LOFTRAN

The LOFTRAN program is used for studies of transient response of a pressurized water reactor system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by a model containing reactor vessel, hot and cold leg piping, steam generators (tube and shell sides) and the pressurizer. The pressurizer heaters, spray, relief and safety valves are also considered in the program. Point model neutron kinetics, and reactivity effects of the moderator, fuel, boron and rods are included. The secondary side of the steam generator utilizes a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The reactor protection system is simulated to include reactor trips on neutron flux, low DNBR, high linear power (kW/ft), high and low pressure, low flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control and pressurizer pressure control. ECCS, including the accumulators, is also modeled.

LOFTRAN is a versatile program which is suited to both accident evaluation and control studies as well as parameter sizing.

LOFTRAN also has the capability of calculating the transient value of DNBR based on the input from the core limits illustrated in Figure 15.0-1. The core limits represent the minimum value of DNBR as calculated for typical, small thimble, large thimble, corner or side cell.

LOFTRAN is further discussed in Reference 8.

15.0.12.3 TWINKLE

The TWINKLE program is a multi-dimensional spatial neutron kinetics code, which was patterned after steady state codes presently used for reactor core design. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two or three dimensions. The code uses six delayed neutron groups and contains a detailed multi-region fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2000 spatial points, and performs its own steady state initialization. Aside from basic cross section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion. Various edits are provided, e.g., channelwise power, axial offset, enthalpy, volumetric surge, pointwise power, and fuel temperatures.

The TWINKLE Code is used to predict the kinetic behavior of a reactor for transients which cause a major perturbation in the spatial neutron flux distribution.

TWINKLE is further described in Reference 9.

15.0.12.4 THINC

The THINC Code is described in Section 4.4 of RESAR-SP/90 PDA Module 5, "Reactor System".

15.0.13 REFERENCES

1. H. Chelemer, et al., "Improved Thermal Design Procedure", WCAP-8567-P (Proprietary), July 1975, and WCAP-8568 (Non-Proprietary) July, 1975.

2. J. Skaritka, ed., "Hybrid B_4C Absorber Control Rod Evaluation Report", WCAP-8846-A, October 1977.

3. J. J. DiNunno et al., "Calculation of Distance Factors for Power and Test Reactor Sites", TID-14844, March 1962.

4. M. E. Meek and B. R. Rider, "Compilation of Fission Product Yields", NEDO-12154-1, General Electric Corporation, January 1974.

5. F. M. Bordelon et al., "SATAN-VI Program: Comprehensive Space Time Dependent Analysis of Loss-of-Coolant", WCAP-8306, June 1974.

6. F. M. Bordelon et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis", WCAP-8305, June 1974.

7. C. Hunin, "FACTRAN, A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod", WCAP-7908, June 1972.

8. T. W. T. Burnett et al., "LOFTRAN Code Description", WCAP-7907-P-A, April, 1984.

9. D. H. Risher, Jr., and R. F. Barry, "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code": WCAP-7979-P-A (Proprietary) January 1975, and WCAP-8028-A, (Non-Proprietary), January 1975.

10. Bell, M. J., "ORIGEN - The ORNL Isotope Generation and Depletion Code," ORNL-4628, May 1973.

11. "ORIGEN Yields and Cross Sections - Nuclear Transmutation and Decay Data From END F/B-IV", Radiation Shielding Information Center, Oak Ridge National Laboratory, RSIC-DLC-38, September 1975.

TABLE 15.0-1

NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

	N-Loop Operation
Reactor core thermal power output (MWt)*	3800
Thermal power generated by the reactor coolant pumps (MWt)	16
Guaranteed nuclear steam supply system thermal power output (MWt)	3816

* Radiological consequences based on 3565 (MWt) power level.

TABLE 15.0-2

VALUES OF PERTINENT PLANT PARAMETERS
UTILIZED IN ACCIDENT ANALYSES*

	N-Loop Operation
Thermal output of nuclear steam supply system (MWt)	3816
Reactor core thermal power output (MWt)	3800
Core inlet temperature (°F)	560.8
Reactor coolant average temperature (°F)	592.6
Reactor coolant system pressure (psia)	2250
Reactor coolant flow per loop (gpm)	97900
Total reactor coolant flow (10^6 lb/hr)	145.0
Total steam flow from NSSS (10^6 lb/hr)	17.14
Steam pressure at steam generator outlet (psia)	1024
Maximum steam moisture content (%)	0.25
Feedwater temperature at steam generator inlet (°F)	450
Average core heat flux (Btu/hr-ft^2)	162960

* For accident analyses using the improved thermal design procedure.

TABLE 15.0-2a

VALUES OF PERTINENT PLANT PARAMETERS
UTILIZED IN ACCIDENT ANALYSES*

	N-Loop Operation
Thermal output of nuclear steam supply system (MWt)	3816
Reactor core thermal power output (MWt)	3800
Core inlet temperature (°F)	560.7
Reactor coolant average temperature (°F)	592.9
Reactor coolant system pressure (psia)	2250
Reactor coolant flow per loop (gpm)	96900
Total reactor coolant flow (10^6 lb/hr)	143.5
Total steam flow from NSSS (10^6 lb/hr)	17.14
Steam pressure at steam generator outlet (psia)	1024
Maximum steam moisture content (%)	0.25
Feedwater temperature at steam generator inlet (°F)	450
Average core heat flux (Btu/hr-ft^2)	162960

* For accident analyses not using the improved thermal design procedure.

TABLE 15.0-3

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Faults	Computer Codes Utilized	Kinetic Parameters Assumed: Delayed Neutron Fraction	Moder. Density (Δρ/gm/cc)	Doppler	DNB Correlation	Improved Thermal Design Proced.	Initial NSSS Thermal Power Output (MWt)	Reactor Vessel Coolant Flow (gpm)	Vessel Average Temp. (°F)	Press. Pressure (psia)	Press. Water Volume (ft3)	Feedwater Temp. (°F)
15.1 Increase in Heat Removal by the Secondary System												
Feedwater System Malfunction Causing an Increase in Feedwater Flow	LOFTRAN	.0075	.44	Maximum	NA	Yes	3816	397000	590.55	2250	1545	450
Excessive Increase in Secondary Steam Flow	LOFTRAN	.0044 .0075	0.0 .44	Minimum & Maximum	NA	Yes	3816	397000	590.55	2250	1545	450
Accidental Depressurization of the Main Steam System	LOFTRAN	.0055	Figure 15.1-11	Figure 15.1-15	W-3	No	0.0	387600	567	2250	680	80
Steam System Piping Failure	LOFTRAN	.0055	Figure 15.1-11	Figure 15.1-15	W-3	No	0.0	387600	567	2250	680	80
15.2 Decrease in Heat Removal by the Secondary System												
Loss of External Electrical Load and/or Turbine Trip	LOFTRAN	.0044 .0075	.44 0.0	Maximum & Minimum	NA	NA	3968	387600	547	2220	1545	450
Loss on Non-Emergency A-C Power to the Station Auxiliaries	LOFTRAN	.0075	0.0	Minimum	NA	NA	3892	387600	597	2280	1675	450
Loss of Normal Feedwater Flow	LOFTRAN	.0075	0.0	Minimum	NA	NA	3892	387600	597	2280	1675	450
Feedwater System Pipe Break	LOFTRAN	.0075	0.43	Maximum	NA	NA	3892	387600	597	2280	1675	450

TABLE 15.0-3 (Con't)

Faults	Computer Codes Utilized	Delayed Neutron Fraction	Moder. Density (Δρ/gm/cc)	Doppler	DNB Correlation	Improved Thermal Design Proced.	Initial NSSS Thermal Power Output (MWt)	Reactor Vessel Coolant Flow (gpm)	Vessel Average Temp. (°F)	Press. Pressure (psia)	Press. Water Volume (ft3)	Feedwater Temp. (°F)
	Kinetic Parameters Assumed											
15.3 Decrease in Reactor Coolant System Flow Rate												
Partial and Complete Loss of Forced Reactor Coolant Flow	(See RESAR-SP/90 PDA Module 4, "Reactor Coolant System")											
Reactor Coolant Pump Shaft Seizure (locked rotor	(See RESAR-SP/90 PDA Module 4, "Reactor Coolant System")											
15.4 Reactivity and power distribution anomalies												
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition.	(See RESAR-SP/90 PDA Module 5, "Reactor System")											
Uncontrolled Rod Cluster Assembly Bank Withdrawal at Power	(See RESAR-SP/90 PDA Module 5, "Reactor System")											
Control Rod Mis-alignment	(See RESAR-SP/90 PDA Module 5, "Reactor System")											
Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature	(See RESAR-SP/90 PDA Module 4, "Reactor Coolant System")											

TABLE 15.0-3 (Con't)

Kinetic Parameters Assumed

Faults	Computer Codes Utilized	Delayed Neutron Fraction	Moder. Density (Δρ/gm/cc)	Doppler	DNB Correlation	Improved Thermal Design Proced.	Initial NSSS Thermal Power Output (MWt)	Reactor Vessel Coolant Flow (gpm)	Vessel Average Temp. (°F)	Press. Pressure (psia)	Press. Water Volume (ft3)	Feedwater Temp. (°F)
Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	(See RESAR-SP/90 PDA Module 13, "Auxiliary Systems")											
Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	(See RESAR-SP/90 PDA Module 5, "Reactor System")											
Spectrum of Rod Cluster Control Assembly Ejection Accidents	(See RESAR-SP/90 PDA Module 5, "Reactor System")											
15.5 Increase in Coolant Inventory												
Inadvertent Operation of ECCS During Power Operation	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA
15.6 Decrease in Reactor Coolant Inventory												
Inadvertent Opening of a Pressurizer Safety or Relief Valve	(See RESAR-SP/90 PDA Module 4, "Reactor Coolant System")											

* Reference Figure 15.0-2. Maximum refers to lower curve and minimum refers to upper curve.
NA - Not applicable.
BOC - Beginning of cycle
EOC - End of cycle

TABLE 15.0-4

TRIP POINTS AND TIME DELAYS TO TRIP
ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Point Assumed in Analysis	Time Delays (sec)
Power range high neutron flux, high setting	118%	0.5
Power Range high neutron flux, low setting	35%	0.5
Power range neutron flux, high negative rate	*	*
High neutron flux, P-8	85%	0.5
Low DNBR	Variable, see Figure 15.0-1	6.0**
High pressurizer pressure	2410 psig	2.0
Low pressurizer pressure	1836 psia	2.0
Low reactor coolant flow (from loop flow detectors)	87% loop flow	1.0
RCP underspeed	92% or nominal speed	0.6
Turbine trip	Not applicable	2.0
Safety injection reactor trip	Not applicable	2.0
Low steam generator level		
High steam generator level - produces feedwater isolation and turbine trip		

* See RESAR-SP/90 PDA Module 9, "I&C and Electric Power"

** Total time delay (including RTD time response and trip circuit channel electronics delay) from the time the temperature in the coolant loops exceeds the trip setpoint until the rods are free to fall.

TABLE 15.0-5

DETERMINATION OF MAXIMUM OVERPOWER TRIP POINT - POWER RANGE NEUTRON FLUX CHANNEL - BASED ON NOMINAL SETPOINT CONSIDERING INHERENT INSTRUMENT ERRORS

Variable	Accuracy of Measurement of Variable (% error)	Effect on Thermal Power Determination (% error) (Estimated)	Effect on Thermal Power Determination (% error) (Assumed)
Calorimetric errors in the measurement of secondary system thermal power:			
Feedwater temperature	± 0.5		
Feedwater pressure (small correction on enthalpy)	± 0.5	0.3	
Steam pressure (small correction on enthalpy	± 2		
Feedwater flow	± 1.25	1.25	
Assumed calorimetric error (% of rated power)			± 2(a)
Axial power distribution effects on total ion chamber current			
Estimated error (% rated power)		3	
Assumed error (% of rated power)			± 5(b)
Instrumentation channel drift and setpoint reproducibility			
Estimated error (% or rated power)		1	
Assumed.error (% of rated power)			± 2 (c)
Total assumed error in setpoint (a) + (b) + (c)			± 9

TABLE 15.0-5 (Con't)

	Percent Rated Power
Nominal Setpoint	109
Maximum overpower trip point assuming all individual errors are simultaneously in the most adverse direction.	118

TABLE 15.0-6

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
15.1 Increase in Heat Removed by the Secondary System				
Feedwater System Malfunction Causing an Increase in Feedwater Flow	Power range high flux, high steam generator level, manual, low DNBR, high kw/ft	High steam generator level-produced feedwater isolation and turbine trip	Feedwater isolation valves	NA
Excessive Increase Secondary Steam Flow	Power range high flux, manual, low DNBR, high kw/ft	NA	Pressurizer self-actuated safety valves; steam generator safety valves	NA
Accidental Depressurization of the Main Steam System	Low pressurizer pressure, manual, SIS	Low pressurizer pressure, low compensated steam line pressure, Hi-1 containment pressure, manual, low 4 T_{cold}	Feedwater isolation valves, steamline stop valves	Emergency feedwater system; Safety Injection System
Steam System Piping Failure	SIS, low pressurizer pressure, manual	Low pressurizer pressure, low compensated steamline pressure, Hi-1 containment pressure, manual, low 4 T_{cold}	Feedwater isolation valves, steamline stop valves	Emergency feedwater system; Safety Injection System
15.2 Decrease in Heat Removal by the Secondary System				
Loss of External Electrical Load/ Turbine Trip	High pressurizer pressure, low DNBR, low steam generator level, manual	Low steam generator level	Pressurizer safety valves, steam generator	Emergency feedwater system

TABLE 15.0-6 (Con't)

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
Loss of Non-Emergency A-C Power to the Station Auxiliaries	Steam generator low level, manual	Steam generator low level	Steam generator safety valves	Emergency feedwater system
Loss of Normal Feedwater Flow	Steam generator low level, manual	Steam generator low level	Steam generator safety valves	Emergency feedwater system
Feedwater System Pipe Break	Steam generator low level, high pressurizer pressure, SIS, manual low DNBR	Hi-1 containment pressure, steam generator low level, low compensated steamline pressure	Steamline isolation valves, feedline isolation, pressurizer safety valves steam generator safety valves	Emergency feedwater system, Safety Injection System
15.3 Decrease in Reactor Coolant System Flow Rate				
Partial and Complete Loss of Forced Reactor Coolant Flow	Low flow, low RCP speed, manaul	NA	Steam generator safety valves	NA
Reactor Coolant Pump Shaft Seizure (Locked Rotor)	Low flow, manual	NA	Pressurizer safety valves, steam generator safety valves.	NA
15.4 Reactivity and Power Distribution Anomalies				
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or low Power Startup Condition	Power range high flux (low s.p.), manual	NA	NA	NA

TABLE 15.0-6 (Con't)

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	Power range high flux, Hi pressurizer pressure, manual, low DNBR	NA	Pressurizer safety valves, steam generator safety valves	NA
Control Rod Misalignment	Power range negative flux rate, manual	NA	NA	NA
Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature	Power range high flux, P-8, manual	NA	NA	NA
Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Source range high flux, power range high flux, manual, low DNBR, high kw/ft	NA	Low insertion limit annunciators for boration, VCT outlet isolation valves	NA
Spectrum of Rod Cluster Control Assembly Ejection Accidents	Power range high flux, high positive flux rate, manual	NA	NA	NA
15.5 Increase in Reactor Coolant Inventory				
Inadvertent Operation of ECCS During Power Operation	NA	NA	NA	NA

TABLE 15.0-6 (Con't)

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
15.6 Decrease in Reactor Coolant Inventory .				
Inadvertent Opening of a Pressurizer Safety or Relief Valve	Pressurizer low pressure, manual, low DNBR	Low pressurizer pressure	NA	Safety Injection System
Steam Generator Tube Rupture	Reactor trip system	Engineered Safety Features Actuation System	Service Water System, Component Cooling Water System, steam generator safety valves, PORVs, SG overfill control valves and steamline stop valves	Emergency Core Cooling System, Emergency Feedwater System, Emergency Power Systems
Loss of Coolant Accident from Spectrum of Postulated Piping Breaks within the System	Reactor trip system	Engineered Safety Features Actuation System	Service Water System, Component Cooling Water System, Steam Generator Safety Valves	Emergency Core Cooling System, Emergency Feedwater System, Containment Heat Removal System, Emergency Power System

TABLE 15.0-7

FUEL AND ROD GAP INVENTORIES, CORE (Ci)[(a)]

Isotope	Core Fuel	Core Gap[(b)]
I-131	1.0E + 7	1.0E + 6
I-132	1.5E + 8	1.5E + 7
I-133	2.1E + 8	2.1E + 7
I-134	2.3E + 8	2.3E + 7
I-135	2.0E + 8	2.0E + 7
Kr-83m	1.3E + 7	1.3E + 6
Kr-85m	2.9E + 7	2.9E + 6
Kr-85	7.0E + 5	2.1E + 5
Kr-87	5.2E + 7	5.2E + 6
Kr-88	7.5E + 7	7.5E + 6
Kr-89	9.3E + 7	9.3E + 6
Xe-131m	7.5E + 5	7.5E + 4
Xe-133m	3.1E + 7	3.1E + 6
Xe-133	2.0E + 8	2.0E + 7
Xe-135m	4.3E + 7	4.3E + 6
Xe-135	4.5E + 7	4.5E + 6
Xe-138	1.7E + 8	1.7E + 7
I-127	3.0 kg	0.90 kg
I-129	12.2 kg	3.7 kg

a. Three-region equilibrium cycle core at end of life. The three regions have operated at a specific power of 40.03 MWt per metric ton of uranium for 300, 600, and 900 effective full power days, respectively.

b. Gap activity is assumed to be 10 percent of core activity for all isotopes except Kr-85, I-127, and I-129, whose gap activities are assumed to be 30 percent of their core activities (Regulatory Guide 1.25 assumption).

TABLE 15.0-8

REACTOR COOLANT IODINE CONCENTRATIONS FOR

1 μCi/GRAM AND 60 μCi/GRAM OF DOSE EQUIVALENT I-131

Nuclide	Reactor Coolant Concentration (Ci/gm) 1 μCi/gm D.E. I-131	60 μ Ci/gm D.E. I-131
I-131	0.76	45.6
I-132	0.76	45.6
I-133	1.14	68.4
I-134	0.195	11.7
I-135	0.63	37.8

TABLE 15.0-9

REACTOR COOLANT NOBLE GAS SPECIFIC ACTIVITY BASED ON ONE PERCENT DEFECTIVE FUEL

Nuclide	Activity (μc/gram)
Kr-85m	2.0
Kr-85	7.3
Kr-87	1.3
Kr-88	3.6
Xe-131m	2.2
Xe-133m	1.7×10^{1}
Xe-133	2.7×10^{2}
Xe-135m	4.8×10^{-1}
Xe-135	7.2
Xe-138	6.4×10^{-1}

TABLE 15.0-10

IODINE APPEARANCE RATES IN THE REACTOR COOLANT (Curies/sec)

	*Equilibrium Appearance Rates Due to Fuel Defects	**Appearance Rates Due to an Accident Initiated Iodine Spike
I-131	3.4×10^{-3}	1.7
I-132	1.8×10^{-2}	9.0
I-133	7.2×10^{-3}	3.6
I-134	1.1×10^{-2}	5.5
I-135	6.8×10^{-3}	3.4

* Based on RCS concentration of 1 μCi/gm of dose equivalent I-131.

** 500 x equilibrium appearance rate.

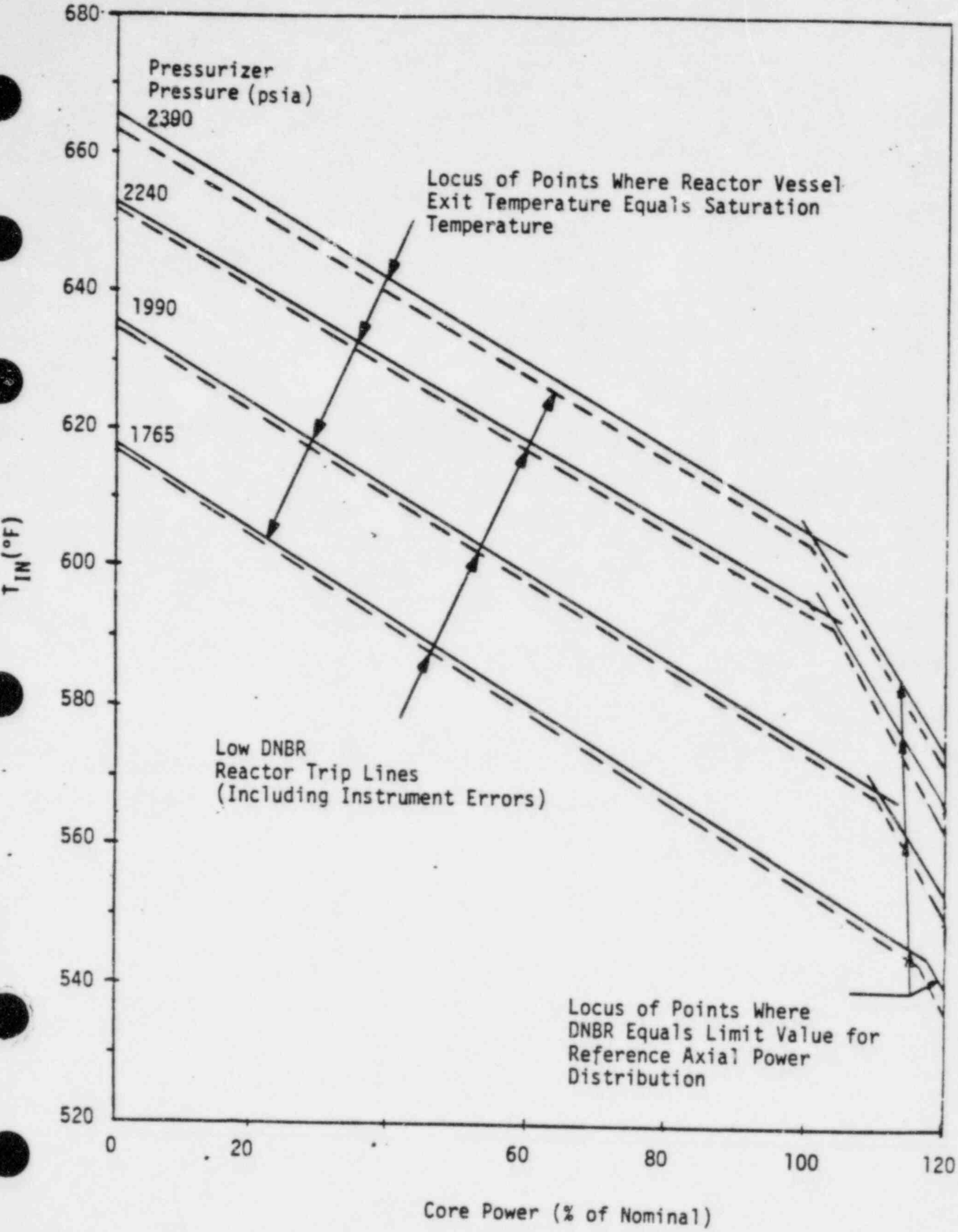
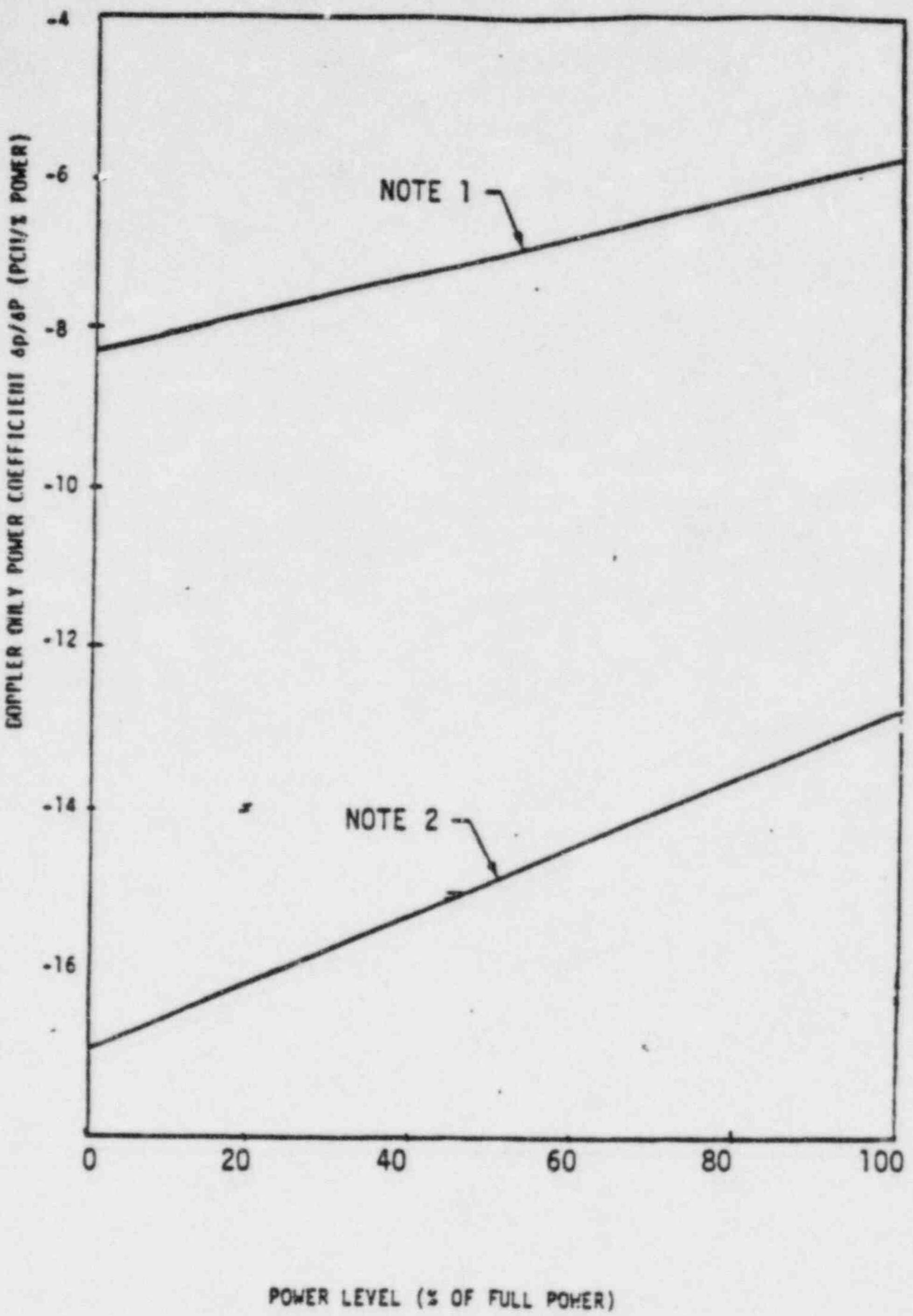


Figure 15.0-1 Illustration of Core Thermal Limits and DNB Protection (N Loop Operation)



Note 1 - Upper Curve, Least Negative Doppler Only Power Defect = -6.95% Δρ (0 to 100% Power)

Note 2 - Lower Curve, Most Negative Doppler Only Power Defect = -1.49% Δρ (0 to 100% Power)

FIGURE 15.0-2 DOPPLER POWER COEFFICIENT USED IN ACCIDENT ANALYSIS

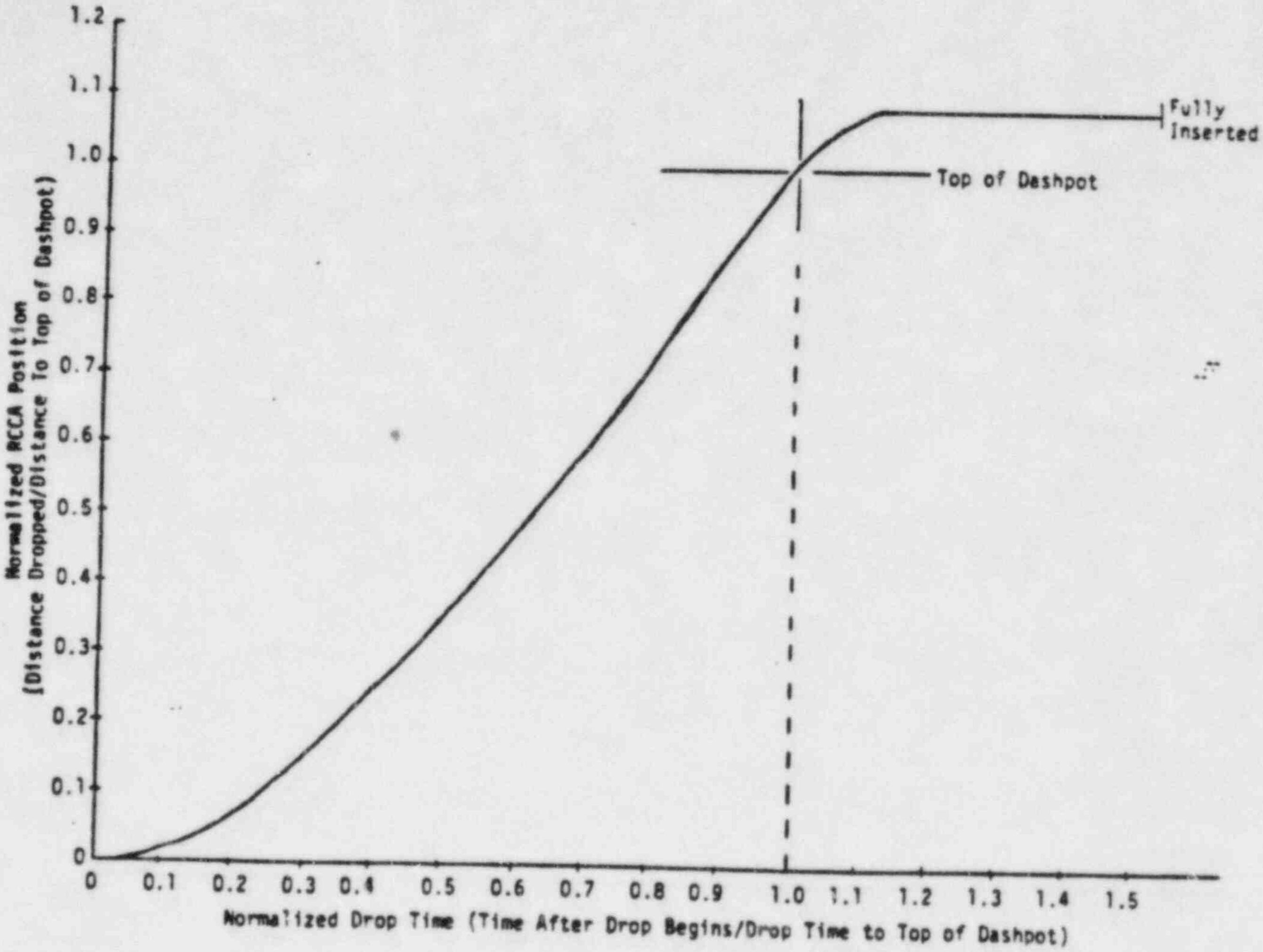


FIGURE 15.0-3 RCCA POSITION VS. TIME TO DASHPOT

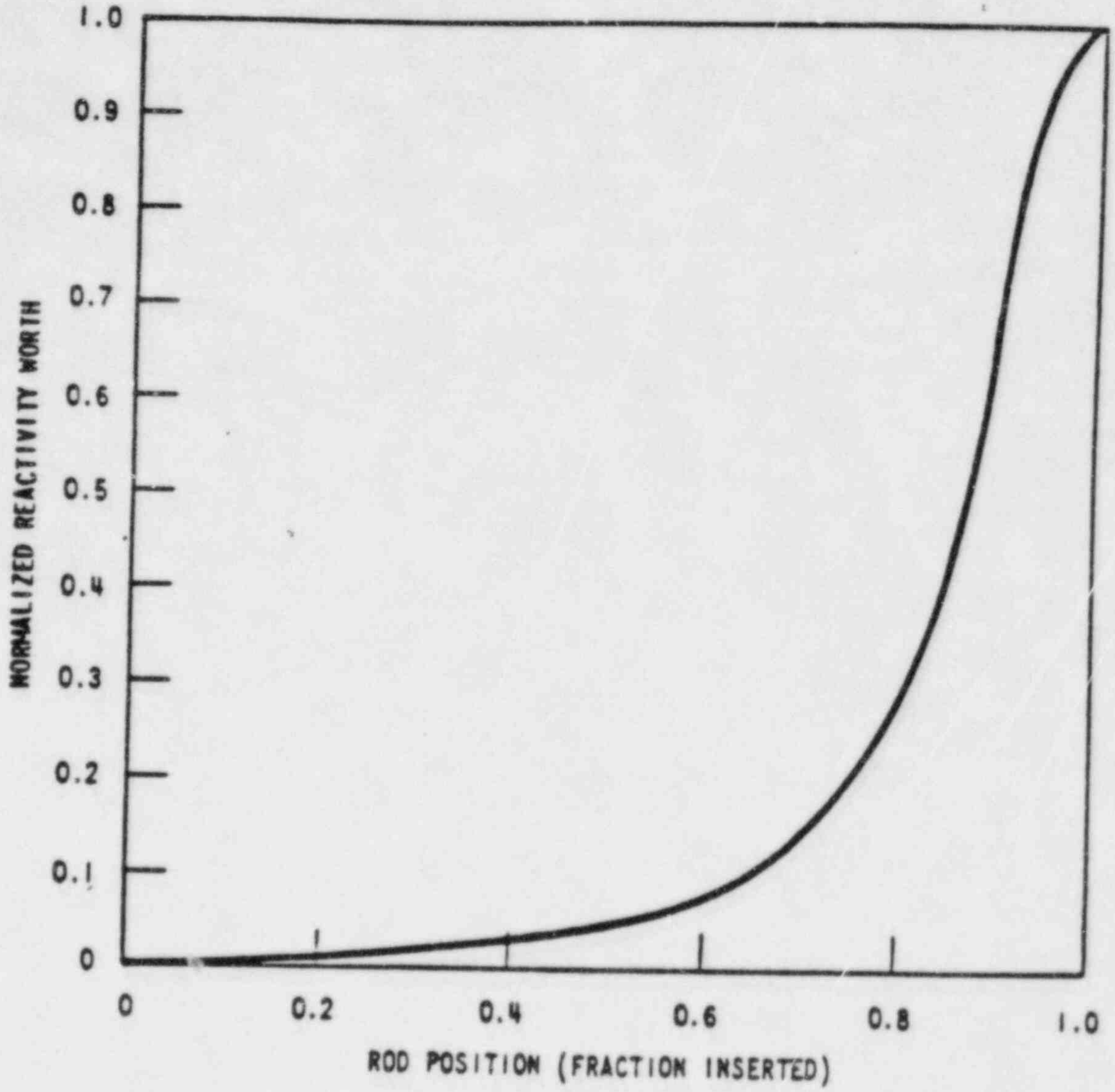


FIGURE 15.0-4 NORMALIZED RCCA REACTIVITY WORTH VS. FRACTION INSERTION

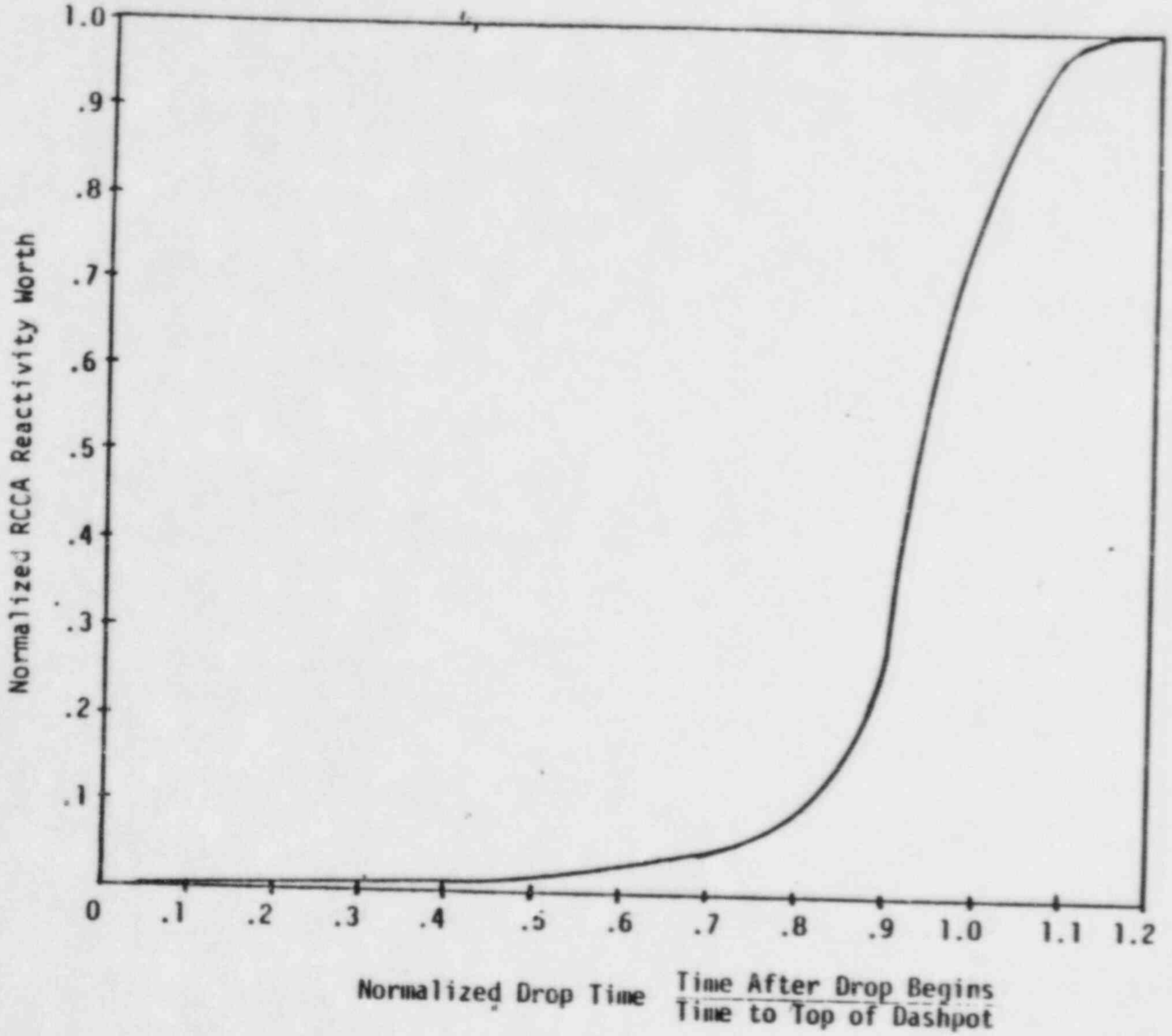


FIGURE 15.0-5 NORMALIZED RCCA BANK REACTIVITY WORTH VS. NORMALIZED DROP TIME

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15.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

A number of events have been postulated which could result in an increase in heat removal from the reactor coolant system by the secondary system. Analyses are presented for several such events which have been identified as limiting cases.

Discussions of the following reactor coolant system cooldown events are presented:

1. Feedwater system malfunction causing a reduction in feedwater temperature (Subsection 15.1.1).

2. Feedwater system malfunction causing an increase in feedwater flow (Subsection 15.1.2).

3. Excessive increase in secondary steam flow (Subsection 15.1.3).

4. Inadvertent opening of a steam generator relief or safety valve causing a depressurization of the main steam system (Subsection 15.1.4).

5. Spectrum of steam system piping failures inside and outside containment (Subsection 15.1.5).

The above are considered to be ANS Condition II events, with the exception of a major steam system pipe break, which is considered to be an ANS Condition IV event (Subsection 15.0.2).

15.1.1 Feedwater System Malfunctions Causing a Reduction in Feedwater Temperature

15.1.1.1 Identification of Causes and Accident Description

Reductions in feedwater temperature will cause an increase in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and of the reactor coolant system (RCS). The high neutron flux trip, low DNBR trip, and high Kw/ft trip prevent any power increase which could lead to a DNBR less than the limit value.

A reduction in feedwater temperature may be caused by the accidental opening of a feedwater heater bypass valve which diverts flow around a portion of the feedwater heaters. In the event of an accidental opening of the bypass valve, there is a sudden reduction in feedwater inlet temperature to the steam generators. At power, this increased subcooling will create a greater load demand on the reactor coolant system.

With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in reactor coolant system temperature and, thus, a reactivity insertion due to the effects of the negative moderator temperature coefficient of reactivity. However, the rate of energy change is reduced as load and feedwater flow decrease so the transient is less severe than the full power case.

The net effect on the reactor coolant system due to a reduction in feedwater temperature is similar to the effect of increasing secondary steam flow, i.e., the reactor will reach a new equilibrium condition at a power level corresponding to the new steam generator ΔT.

A decrease in normal feedwater temperature is classified as an ANS Condition II event, fault of moderate frequency. (See Subsection 15.0.2).

The protection available to mitigate the consequences of a decrease in feedwater temperature is the same as that for an excessive steam flow increase, as discussed in Subsection 15.0.9 and listed in Table 15.0-6.

15.1.1.2 Analysis of Effects and Consequences

This transient is analyzed by computing conditions at the feedwater pump inlet following opening of the heater bypass valve. These feedwater conditions are then used to perform a heat balance through the high pressure heaters. This heat balance gives the new feedwater conditions at the steam generator inlet.

The following assumptions are made:

1. Plant initial power level corresponding to guaranteed NSSS thermal output.

2. Simultaneous actuation of a low-pressure heater bypass and isolation of one string of low-pressure feedwater heaters.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.4.

Results

Opening of a low-pressure heater bypass valve causes a reduction in feedwater temperature which increases the thermal load on the primary system. The calculated reduction in feedwater temperature is less than 60°F, resulting in an increase in heat load on the primary system of less than 10 percent of full power. The increased thermal load, due to opening of the low-pressure heater bypass valve, thus would result in a transient very similar (but of reduced magnitude) to that presented in Subsection 15.1.3 for an excessive increase in secondary steam flow incident, which evaluates the consequences of a 10 percent step load increase. Therefore, the results of this analysis are not presented.

15.1.1.3 Radiological Consequences

There will be no radiological consequences associated with a decrease in feedwater temperature event, and activity is contained within the fuel rods and reactor coolant system within design limits.

15.1.1.4 Conclusions

The decrease in feedwater temperature transient is less severe than the increase in feedwater flow event (Subsection 15.1.2), and the increase in secondary steam flow event (Subsection 15.1.3). Based on results presented in Subsections 15.1.2 and 15.1.3, the applicable acceptance criteria for the decrease in feedwater temperature event have been met. There are no radiological consequences of this event.

15.1.2 Feedwater System Malfunctions Causing An Increase In Feedwater Flow

15.1.2.1 Identification of Causes and Accident Description

Additions of excessive feedwater will cause an increase in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and of the reactor coolant system. The high neutron flux trip, low DNBR trip, and high Kw/ft trip prevent any power increase which could lead to a DNBR less than the limit value.

An example of excessive feedwater flow would be a full opening of a feedwater control valve due to a feedwater control system malfunction or an operator error. At power this excess flow causes a greater load demand on the Reactor Coolant System due to increased subcooling in the steam generator. With the plant at no-load conditions, the addition of an excess of feedwater may cause a decrease in Reactor Coolant System temperature and, thus, a reactivity insertion due to the effects of the negative moderator coefficient of reactivity.

Continuous addition of excessive feedwater is prevented by the steam generator high level trip, which closes all feedwater control and isolation valves and trips the main feedwater pumps.

An increase in normal feedwater flow is classified as an ANS Condition II event, a fault of moderate frequency (see Subsection 15.0.2).

Plant systems and equipment which are available to mitigate the effects of the accident, are discussed in Subsection 15.0.9 and listed in Table 15.0-6.

15.1.2.2 Analysis of Effects and Consequences

Method of Analysis

The excessive heat removal due to a feedwater system malfunction transient is analyzed by using the detailed digital computer code LOFTRAN (Reference 1). This code simulates a multi-loop system, the neutron kinetics, the pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

A control system malfunction or operator error is assumed to cause a feedwater control valve to open fully. Two cases are analyzed as follows:

1. Accidental opening of one feedwater control valve with the reactor just critical at zero load conditions assuming a conservatively large negative moderator temperature coefficient.

2. Accidental opening of one feedwater control valve with the reactor in automatic control at full power.

The reactivity insertion rate following a feedwater system malfunction is calculated with the following assumptions:

1. For the feedwater control valve accident at full power, one feedwater control valve is assumed to malfunction resulting in a step increase to 200 percent of nominal feedwater flow to one steam generator.

2. For the feedwater control valve accident at zero load conditions, a feedwater control valve malfunction occurs which results in an increase in flow to one steam generator from zero to 175 percent of the nominal full load value.

3. For the zero load condition, feedwater temperature is at a conservatively low value of 50°F.

4. No credit is taken for the heat capacity of the Reactor Coolant System and steam generator thick metal in attenuating the resulting plant cooldown.

5. The feedwater flow resulting from a fully open control valve is terminated by a steam generator high level trip signal which closes all feedwater control and isolation valves, trips the main feedwater pumps, and trips the turbine.

The at-power accident is analyzed with the Improved Thermal Design Procedure as described in WCAP-8567. Initial operating conditions are assumed at values consistent with steady-state operation.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.4.

Normal reactor coolant systems and engineered safety feature systems are not required to function. The reactor protection system may function to trip the reactor due to overpower or high steam generator water level conditions. No single active failure will prevent operation of the reactor protection system. A discussion of ATWT considerations is presented in Reference 2.

Results

The calculated sequence of events for this accident are shown in Table 15.1-1.

In the case of an accidental full opening of one feedwater control valve with the reactor at zero power and the above mentioned assumptions, the maximum reactivity insertion rate is less than the maximum reactivity insertion rate analyzed in Subsection 15.4.1 of RESAR-SP/90 PDA module 5, "Reactor System" and, therefore, the results of the analysis are not presented here. It should be noted that if the incident occurs with the unit just critical at no-load conditions, the reactor may be tripped by the power range high neutron flux trip (low setting) set at approximately 25 percent of nominal full power.

The full power case (manual rod control) gives the largest reactivity feedback and results in the greatest power increase. Assuming the reactor to be in the automatic control mode results in a slightly less severe transient. The rod control system is not required to function for an excessive feedwater flow event.

When the steam generator water level in the faulted loop reaches the high level setpoint, all feedwater control and isolation valves and pump discharge valves are automatically closed and the main feedwater pumps are tripped. This prevents continuous addition of feedwater. In addition, a reactor trip and turbine trip are initiated.

Transient results, (see Figures 15.1-1 and 15.1-2), show the core heat flux, pressurizer pressure, Tave and DNBR as well as the increase in nuclear power and loop ΔT associated with the increased thermal load on the reactor. The DNBR does not drop below the limit value.

Following reactor trip and feedwater isolation, the plant will approach a stabilized condition at hot standby. Normal plant operating procedures may then be followed. The operating procedures would call for operator action to control reactor coolant system boron concentration and pressurizer level using the CVCS and to maintain steam generator level through control of the main or auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of thirty minutes following reactor trip.

Since the power level rises during the excessive feedwater flow incident, the fuel temperatures will also rise until after reactor trip occurs. The core heat flux lags behind the neutron flux response due to the fuel rod thermal time constant, hence the peak value does not exceed 118 percent of its normal value (i.e., the assumed high neutron flux trip point). The peak fuel temperature will thus remain well below the fuel melting temperature.

The transient results show that DNB does not occur at any time during the excessive feedwater flow incident; thus, the ability of the primary coolant to remove heat from the fuel rod is not reduced. The fuel cladding temperature, therefore, does not rise significantly above its initial value during the transient.

15.1.2.3 Radiological Consequences

There are minimal radiological consequences from this event. The high level signal causes a reactor and turbine trip and heat is removed from the secondary system through the steam generator power relief or safety valves. Since no fuel damage is postulated to occur from this transient, the radiological consequences will be less severe than the steamline break accident analyzed in Subsection 15.1.5.3.

15.1.2.4 Conclusions

The results of the analysis show that the DNBR's encountered for an excessive feedwater addition at power are at all times above the limit value; hence, the DNB design basis as described in Section 4.4 of RESAR-SP/90 PDA module 5, "Reactor System", is met. Additionally, it has been shown that the reactivity insertion rate which occurs at no-load conditions following excessive feedwater addition is less than the maximum value considered in the analysis of the rod withdrawal from a subcritical condition analysis. The radiological consequences of this event will be less than the steam line break accident analyzed in Subsection 15.1.5.3.

15.1.3 Excessive Increase In Secondary Steam Flow

15.1.3.1 Identification of Causes and Accident Description

An excessive increase in secondary system steam flow (excessive load increase incident) is defined as a rapid increase in steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The reactor control system is designed to accommodate a 10 percent step load increase or a 5 percent per minute ramp load increase in the range of 15 to 100 percent of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor protection system. Steam flow increases greater than 10 percent are analyzed in Subsections 15.1.4 and 15.1.5 (there are no pressure regulators whose malfunction could cause a steam flow transient).

This accident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam dump control or turbine speed control.

During power operation, steam dump to the condenser is controlled by reactor coolant condition signals, i.e., high reactor coolant temperature indicates a need for steam dump. A single controller malfunction does not cause steam dump; an interlock is provided which blocks the opening of the valves unless a large turbine load decrease or a turbine trip has occurred.

Protection for an excessive load increase accident is provided by the following reactor protection system signals:

1. Low DNBR.
2. High Kw/ft.
3. Power range high neutron flux.
4. Low pressurizer pressure.

An excessive load increase incident is considered to be an ANS Condition II event, fault of moderate frequency (See Subsection 15.0.2).

15.1.3.2 Analysis of Effects and Consequences

<u>Method of Analysis</u>

This accident is analyzed using the LOFTRAN Code (Reference 1). The code simulates neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and feedwater system. The code computes pertinent plant variables including temperatures, pressures, and power level.

Four cases are analyzed to demonstrate the plant behavior following a 10 percent step load increase from rated load. These cases are as follows:

1. Reactor control in manual with minimum moderator reactivity feedback.

2. Reactor control in manual with maximum moderator reactivity feedback.

3. Reactor control in automatic with minimum moderator reactivity feedback.

4. Reactor control in automatic with maximum moderator reactivity feedback.

For the minimum moderator feedback cases, the core has the least negative moderator temperature coefficient of reactivity and, therefore, the least inherent transient capability. For the maximum moderator feedback cases, the moderator temperature coefficient of reactivity has its highest absolute value. This results in the largest amount of reactivity feedback due to changes in coolant temperature. For all the cases, the least negative Doppler-only power coefficient curve of Figure 15.0-2 was used.

A conservative limit on the turbine valve opening is assumed, and all cases are studied without credit being taken for pressurizer heaters. Initial operating conditions are assumed at values consistent with steady-state operation. This accident is analyzed with the Improved Thermal Design Procedure as described in WCAP-8567. Uncertainties in initial conditions of reactor power, pressure, and reactor coolant system temperature are included in the limit DNBR as described in the WCAP.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.4.

Normal reactor control systems and engineered safety feature systems are not required to function. The reactor protection system is assumed to be operable; however, reactor trip is not encountered for most cases due to the error allowances assumed in the setpoints. No single active failure will prevent the reactor protection system from performing its intended function.

The cases which assume automatic rod control are analyzed to ensure that the worst case is presented. The automatic function is not required.

Results

The calculated sequence of events for the excessive load increase incident is shown on Table 15.1-1.

Figures 15.1-3 through 15.1-6 illustrate the transient with the reactor in the manual control mode. As expected, for the minimum moderator feedback case, there is a slight power increase, and the average core temperature shows a large decrease. This results in a DNBR which increases above its initial value. For the maximum moderator feedback, manually controlled case there is a much larger increase in reactor power due to the moderator feedback. A reduction in DNBR is experienced but DNBR remains above the limit value.

Figures 15.1-7 through 15.1-10 illustrate the transient assuming the reactor is in the automatic control mode. Both the minimum and maximum moderator feedback cases show that core power increases, thereby increasing the coolant average temperature and pressurizer pressure above their initial value. For both of these cases, the minimum DNBR remains above the limit value.

For all cases, the plant rapidly reaches a stabilized condition at the higher power level. Normal plant operating procedures would then be followed to reduce power. If the reactor trips, operating procedures would call for operator action to control reactor coolant system boron concentration and pressurizer level using the CVCS and to maintain steam generator level through control of the main or auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of thirty minutes following reactor trip.

The excessive load increase incident is an overpower transient for which the fuel temperatures will rise. Reactor trip does not occur for most of the cases analyzed, and the plant reaches a new equilibrium condition at a higher power level corresponding to the increase in steam flow.

Since DNB does not occur at any time during the excessive load increase transients, the ability of the primary coolant to remove heat from the fuel rod is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

15.1.3.3 Radiological Consequences

There will be no radiological consequences associated with this event and activity is contained within the fuel rods and reactor coolant system within design limits.

15.1.3.4 Conclusions

The analysis presented above shows that for a 10 percent step load increase, the DNBR remains above the limit value; thus the DNB design basis as described in Section 4.4 of RESAR-SP/90 PDA module 5, "Reactor System", is met. The plant reaches a stabilized condition rapidly following the load increase.

15.1.4 Inadvertent Opening of A Steam Generator Relief or Safety Valve Causing A Depressurization of The Main Steam System

15.1.4.1 Identification of Causes and Accident Description

The most severe core conditions resulting from an accidental depressurization of the main steam system result from an inadvertent opening of a single steam dump, relief, or safety valve. The analyses performed assuming a rupture of a main steamline are given in Subsection 15.1.5.

The steam release as a consequence of this accident results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the reactor coolant system causes a reduction in coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity.

The analysis is performed to demonstrate that the following criterion is satisfied: assuming a stuck rod cluster control assembly, with offsite power available, and assuming a single failure in the engineered safety features, there will be no consequential damage to the core or reactor coolant system after reactor trip for a steam release equivalent to the spurious opening, with failure to close, of the largest of any single steam dump, relief, or safety valve.

Accidental depressurization of the secondary system is classified as an ANS Condition II event (See Subsection 15.0.2).

The following systems provide the necessary protection against an accidental depressurization of the main steam system:

1. Safety injection actuation from any of the following:

   a. Excessive cooldown protection (low T_{cold} or low steamline pressure)

   b. Low pressurizer pressure

   c. High-1 containment pressure

2. A reactor trip from 1) DNB protection (low DNBR or high neutron flux), 2) low pressurizer pressure, or 3) safety injection signal.

3. Redundant isolation of the main feedwater lines: sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the main feedwater valves following reactor trip, a safety injection signal will rapidly close all feedwater control valves and backup feedwater isolation valves, trip the main feedwater pumps, and close the feedwater pump discharge valves.

4. Trip of the fast-acting steamline stop valves (designed to close in less than 5 seconds) on:

   a. Excessive cooldown protection (low T_{cold} or low steamline pressure)

   b. High negative steam pressure rate in any loop

   c. High-2 containment pressure

Systems and equipment which are available to mitigate the effects of the accident are also discussed in Subsection 15.0.9 and listed in Table 15.0-6.

15.1.4.2 Analysis of Effects and Consequences

Method of Analysis

The following analyses of a secondary system steam release are performed for this section:

1. A full plant digital computer simulation using the LOFTRAN Code (Reference 1) to determine reactor coolant system temperature and pressure, during cooldown, and the effect of safety injection.

2. Analyses to determine that there is no damage to the core or reactor coolant system.

The following conditions are assumed to exist at the time of a secondary steam system release:

1. End-of-life shutdown margin at no-load, equilibrium xenon conditions, and with the most reactive rod cluster control assembly stuck in its fully withdrawn position. Operation of rod cluster control assembly banks during core burnup is restricted in such a way that addition of positive reactivity in a secondary system steam release accident will not lead to a more adverse condition than the case analyzed.

2. A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive rod cluster control assembly in the fully withdrawn position. The variation of the coefficient with temperature and pressure is included. The K_{eff} versus temperature at 1000 psi corresponding to the negative moderator temperature coefficient used is shown in Figure 15.1-11.

3. Minimum capability for injection of concentrated boric acid solution corresponding to the most restrictive single failure in the safety injection system. This corresponds to the flow delivered by two safety injection pumps delivering to the RCS at separate nozzle locations. Low concentration boric acid must be swept from the safety injection lines downstream of the emergency water storage tank prior to the delivery of boric acid (2500 ppm) to the reactor coolant loops. This effect has been allowed for in the analysis.

4. The case studied is a steam flow of 269 lb/sec at 1200 psia from one steam generator with offsite power available. This is the maximum capacity of any single steam dump, relief, or safety valve. Initial hot shutdown conditions at time zero are assumed since this represents the most conservative initial condition. Cases analyzed in Subsection 15.1.3, excessive increase in secondary steam flow, bound a failure of a steam generator steam dump, safety, or relief valve from full power.

5. In computing the steam flow, the Moody Curve (Reference 3) for FL/D = 0 is used.

6. Perfect moisture separation in the steam generator is assumed.

Results

The calculated time sequence of events for this accident is listed in Table 15.1-1.

The results presented are a conservative indication of the events which would occur assuming a secondary system steam release since it is postulated that all of the conditions described above occur simultaneously.

Figures 15.1-13 and 15.1-14 show the transient results for a steam flow of 269 lb/sec at 1200 psia from one steam generator.

The assumed steam release is typical of the capacity of any single steam dump, relief, or safety valve.

Safety injection is initiated automatically by low pressurizer pressure. Operation of two SI pumps is assumed. Boron solution at 2500 ppm enters the reactor coolant system providing sufficient negative reactivity to prevent core damage. The cooldown for the case shown in Figures 15.1-13 and 15.1-14 is more rapid than the case of steam release from all steam generators through one steam dump, relief, or safety valve. The calculated transient is quite conservative with respect to cooldown, since no credit is taken for the energy stored in the system metal other than that of the fuel elements or the energy stored in other steam generators. Since the transient occurs over a period of about 5 minutes, the neglected stored energy will have a significant effect in slowing the cooldown.

Following blowdown of the faulted steam generator, the plant can be brought to a stabilized hot standby condition through control of emergency feedwater flow and safety injection flow as described by plant operating procedures. The operating procedures would call for operator action to limit reactor coolant system pressure and pressurizer level by terminating safety injection flow and to control steam generator level and reactor coolant system temperature using the emergency feedwater system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of thirty minutes following safety injection actuation.

15.1.4.3 Radiological Consequences

The inadvertent opening of a single steam dump relief or safety valve can result in steam release from the secondary system. If steam generator leakage exists coincident with the failed fuel conditions, some activity will be released. (The activity release and dose is provided on a plant specific basis).

15.1.4.4 Conclusions

The analysis shows that the criteria stated earlier in this section are satisfied. For an accidental depressurization of the main steam system, the minimum DNBR remains well above the limiting value and no system design limits are exceeded. (The radiological consequences of this event are found on a plant specific basis).

15.1.5 Spectrum of Steam System Piping Failure Inside and Outside Containment

15.1.5.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steamline would result in an initial increase in steam flow which decreases during the accident as the steam pressure decreases. The energy removal from the reactor coolant system causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. If the most reactive rod cluster control assembly (RCCA) is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid delivered by the safety injection system.

The analysis of a main steamline rupture is performed to demonstrate that the following criteria are satisfied:

1. Assuming a stuck RCCA, with or without offsite power, and assuming a single failure in the engineered safety features, the core remains in place and intact. Radiation doses do not exceed the guidelines of 10CFR100.

2. Although DNB and possible cladding perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that the DNB design basis is met as stated in Section 4.4 of RESAR-SP/90 PDA module 5, "Reactor System", for any rupture assuming the most reactive RCCA assembly stuck in its fully withdrawn position.

A major steamline rupture is classified as an ANS Condition IV event (See Section 15.0.2).

The rupture of a major steamline is the most limiting cooldown transient and, thus, is analyzed at zero power with no decay heat. Decay heat would retard the cooldown thereby reducing the return to power. A detailed analysis of this transient with the most limiting break size, a double ended rupture, is presented here.

The following functions provide the necessary protection for a steamline rupture:

1. Safety injection system actuation from any of the following:

   a. Excessive cooldown protection (low T_{cold} or low steamline pressure)

   b. Low pressurizer pressure.

   c. High-1 containment pressure.

2. A reactor trip from 1) DNB protection (low DNBR or high neutron flux), 2) high linear heat flux, 3) low pressurizer pressure, or 4) safety injection signal.

3. Redundant isolation of the main feedwater lines: sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the main feedwater valves following reactor trip, a safety injection signal will rapidly close all feedwater control valves and backup feedwater isolation valves, and trip the main feedwater pumps.

4. Trip of the fast acting steamline stop valves (designed to close in less than 5 seconds) on:

    a. Excessive cooldown protection (low T_{cold} or low steamline pressure)

    b. High negative steam pressure rate in any loop

    c. High-2 containment pressure.

Fast-acting isolation valves are provided in each steamline that will fully close within 5 seconds of actuation following a steamline isolation signal from the integrated protection system. An additional delay of 2.0 seconds is included for sensor and protection system delays. For breaks downstream of the isolation valves, closure of all valves would completely terminate the blowdown. For any break, in any location, no more than one steam generator would experience an uncontrolled blowdown even if one of the isolation valves fails to close. A description of steamline isolation is included in Chapter 10.0.

Table 15.1-2 lists the equipment required in the recovery from a high energy steamline rupture. Not all equipment is required for any one particular break, since it will vary depending upon postulated break location and details criteria. Design criteria and methods of protection of safety-related equipment from the dynamic effects of postulated piping ruptures are provided in Section 3.6 of RESAR-SP/90 PDA module 7, "Structural/Equipment Design".

15.1.5.2 Analysis of Effects and Consequences

Method of Analysis

The analysis of the steam pipe rupture has been performed to determine:

1. The core heat flux and reactor coolant system temperature and pressure resulting from the cooldown following the steamline break. The LOFTRAN Code (Reference 1) has been used.

2. The thermal and hydraulic behavior of the core following a steamline break. A detailed thermal and hydraulic digital-computer code, THINC, has been used to determine if DNB occurs for the core conditions computed in Item 1 above.

The following conditions were assumed to exist at the time of a main steamline break accident:

1. End-of-life shutdown margin at no-load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position: operation of the control rod banks during core burnup is restricted in such a way that addition of positive reactivity in a steamline break accident will not lead to a more adverse condition than the case analyzed.

2. A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position: the variation of the coefficient with temperature and pressure has been included. The K_{eff} versus average coolant temperature at 1000 psi corresponding to the negative moderator temperature coefficient used is shown in Figure 15.1-11. (The effect of power generation in the core on overall reactivity is shown in Figure 15.1-15).

   The core properties associated with the sector nearest the affected steam generator and those associated with the remaining sectors were conservatively combined to obtain average core properties for reactivity feedback calculations. Further, it was conservatively assumed that the core power distribution was uniform. These two conditions cause underprediction of the reactivity feedback in the high power region near the stuck rod. To verify the conservatism of this method, the reactivity, as well as the power distribution, was checked for the limiting conditions for the cases analyzed. This core analysis considered the Doppler reactivity from the high fuel temperature near the stuck RCCA, moderator feedback from the high water enthalpy near the stuck RCCA, power redistribution and nonuniform core inlet temperature effects. For cases in which steam generation occurs in the high flux regions of the core, the effect of void formation was also included. It was determined that the reactivity employed in the kinetics analysis was always larger than the reactivity calculated including the above local effects for the conditions. These results verify conservatism: underprediction of negative reactivity feedback from power generation.

3. Minimum capability for injection of boric acid (2500 ppm) solution corresponding to the most restrictive single failure in the safety injection portion of the emergency core cooling system (ECCS). The ECCS consists of three systems: 1) the high pressure passive accumulators, 2) the low pressure passive accumulators, and 3) the high head safety injection system. Only the high head system is modeled for the steamline break accident analysis.

   The modeling of the SI system in LOFTRAN is described in Reference 1. The flow corresponds to that delivered by two SI pumps delivering full flow to the RCS. No credit has been taken for the low concentration borated water, which must be swept from the lines downstream of the HHSI pumps prior to the delivery of boric acid to the reactor coolant system.

   For the cases where offsite power is assumed, the sequence of events in the SI system is the following. After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the SI pump starts. In 12 seconds, the pump is assumed to be at full speed. The volume containing the low concentration borated water is swept into the core before the 2500 ppm borated water reaches the core. This delay, described above, is inherently included in the modeling.

   In cases where offsite power is not available, a 10 second delay to start the standby diesel generators in addition to the time necessary to start the safety injection equipment (mentioned above) is included.

4. Design value of the steam generator heat transfer coefficient including allowance for fouling factor.

5. Since the steam generators are provided with integral flow restrictors with a 1.4 ft^2 throat area, any rupture with a break area greater than 1.4 ft^2, regardless of location, would have the same effect on the nuclear steam supply system (NSSS) as the 1.4 ft^2 break. The following cases have been considered in determining the core power and reactor coolant system transients:

   a. Complete severance of the pipe, with the plant initially at no-load conditions, full reactor coolant flow with offsite power available.

   b. Case (a) with loss of offsite power simultaneous with the steamline break and initiation of the safety injection signal. Loss of offsite power results in reactor coolant pump coastdown.

6. Power peaking factors corresponding to one stuck RCCA and nonuniform core inlet coolant tempertures are determined at end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck control assembly during the return to power phase following the steamline break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck RCCA assembly. The power peaking factors depend upon the core power, temperature, pressure, and flow, and, thus, are different for each case studied.

   The core parameters used for each of the two cases correspond to values determined from the respective transient analysis.

   Both cases above assume initial hot shutdown conditions at time zero since this represents the most pessimistic initial condition. Should the reactor be just critical or operating at power at the time of a steamline break, the reactor will be tripped by the normal overpower protection system when power level reaches a trip point. Following a trip at power the reactor coolant system contains more stored energy than at no-load, the average coolant temperature is higher than at no-load, and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steamline break before the no-load conditions of reactor coolant system temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis which assumes no-load condition at time zero.

7. In computing the steam flow during a steamline break, the Moody Curve (Reference 3) for FL/D = 0 is used.

8. Perfect moisture separation in the steam generator is assumed.

9. Feedwater addition aggravates cooldown accidents like the steamline rupture. Therefore, the maximum feedwater flow is assumed. All the main and emergency feedwater pumps are assumed to be operating at full capacity when the rupture occurs, even though the plant is assumed to be in a hot standby condition. Full main and emergency feedwater flow is maintained for five seconds following the receipt of a feedwater isolation signal from the integrated protection system following safety injection actuation. An additional 2.0 second delay is added for sensor and protection system delays. During the first 5 seconds following the start of the transient, a feedwater isolation signal is generated (to close both the feedwater control and the feedwater isolation valves) to be sent to redundant valves with a 5 second closure time.

Results

The calculated sequence of events for all cases analyzed is shown on Table 15.1-1.

The results presented are a conservative indication of the events which would occur assuming a steamline rupture since it is postulated that all of the conditions described above occur simultaneously.

Core Power and Reactor Coolant System Transient

Figures 15.1-16 through 15.1-18 show the reactor coolant system transient and core heat flux following a main steamline rupture (complete severance of a pipe) at initial no-load conditions (case a). Offsite power is assumed available so that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator. Should the core be critical at near zero power when the rupture occurs, the initiation of safety injection by low steamline pressure will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the fast acting isolation valves in the steamlines by high containment pressure signals or by low steam line pressure signals. Even with the failure of one valve, release is limited to no more than 10 seconds for the other steam generators while the one generator blows down. The steamline stop valves are designed to be fully closed in less than 5 seconds from receipt of a closure signal.

As shown in Figure 15.1-18, the core attains criticality with the RCCA's inserted (with the design shutdown assuming one stuck RCCA) before boron solution at 2500 ppm enters the reactor coolant system. A peak core power less than the nominal full power value is attained.

Figures 15.1-19 through 15.1-21 show the response of the salient parameters for case (b), which corresponds to the case discussed above with additional loss of offsite power at the time the safety injection signal is generated. The safety injection system delay time includes 10 seconds to start the standby diesel generator and 12 seconds to start the safety injection pump and open the valves. Criticality is achieved later and the core power increase is slower than in the similar case with offsite power available. The ability of the emptying steam generator to extract heat from the reactor coolant system is reduced by the decreased flow in the reactor coolant system. The peak power remains well below the nominal full power value.

It should be noted that following a steamline break only one steam generator blows down completely. Thus, the remaining steam generators are still available for dissipation of decay heat after the initial transient is over. In the case of loss of offsite power this heat is removed to the atmosphere via the steamline safety valves.

Following blowdown of the faulted steam generator, the plant can be brought to a stabilized hot standby condition through control of the emergency feedwater flow and safety injection flow as described by plant operating procedures. The operating procedures would call for operator action to limit reactor coolant system pressure and pressurizer level by terminating safety injection flow and to control steam generator level and reactor coolant system coolant temperature using the emergency feedwater system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of thirty minutes following safety injection actuation.

The ability of the intact steam generators to remove residual energy from the reactor coolant system in the long term is demonstrated by the major rupture of a main feedwater line. The steamline break is less limiting with respect to cooldown without offsite power because temperatures are much lower, all of the emergency feedwater can be delivered to the steam generators, and the steam blowdown leaves a higher water inventory than the feedline blowdown. The feedline rupture demonstrates that the intact steam generators and emergency feedwater provide sufficient heat sink to remove long term heat following the transient.

Margin to Critical Heat Flux

A DNB analysis was performed for all of these cases. It was found that the DNB design basis as stated in Section 4.4 of RESAR-SP/90 PDA module 5,"Reactor System", was met for all cases.

15.1.5.3 Radiological Consequences

The evaluation of the radiological consequences of a postulated main steam line break (MSLB) outside containment assumes that the reactor has been operating with a small percent of defective fuel and leaking steam generator tubes for sufficient time to establish equilibrium concentrations of radionuclides in the reactor coolant and in the secondary coolant.

Following the rupture, auxiliary feedwater to the faulted loop is isolated and the steam generator is allowed to steam dry. Hence, radioiodines carried from the primary coolant to the generator via leaking tubes are assumed to be released directly to the environment. Iodines released from the generators in the intact loops via the steam line safety or power-operated relief valves are assumed to be mixed with the secondary coolant and partitioned between the generator liquid and steam before release to the environment.

15.1.5.3.1 Analytical Assumptions

The major assumptions and parameters used in the analysis are itemized in Table 15.1-3.

15.1.5.3.1.1 Source Term Calculations. The concentration of nuclides in the primary and secondary system, prior to the accident are determined as follows:

A. The iodine concentrations in the reactor coolant will be based upon preaccident and accident initiated iodine spikes.

   1. Accident Initiated Spike

      The reactor trip associated with the MSLB creates an iodine spike in the primary system which increases the iodine release rate from the fuel to the primary coolant to a value 500 times greater than the release rate corresponding to the maximum equilibrium primary system iodine concentration of 1 μCi/g of dose equivalent (DE) I-131. The duration of the spike, 3.3 hours, is sufficient to increase the initial RCS I-131 inventory by a factor of 100.

2. Preaccident Spike

A reactor transient has occurred prior to the MSLB and has raised the primary coolant iodine concentration from 1 μCi/g to 60 μCi/g of DE I-131.

B. The noble gas concentrations in the primary coolant are based on 1-percent defective fuel.

C. The secondary coolant activity is based on the DE of 0.1 μCi/g of I-131.

15.1.5.3.1.2 Mathematical Models Used in the Analysis. Mathematical models used in the analysis are described in the following sections:

A. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A.

B. The atmospheric dispersion factors used in the analysis were calculated based on typical onsite meteorological measurement programs. The dispersion factors are provided in Table 15A-2.

C. The thyroid inhalation dose and total body gamma immersion doses to a receptor at the exclusion area boundary and outer boundary of the low population zone were analyzed using the models described in Appendix 15A.

15.1.5.3.1.3 Identification of Leakage Pathways and Resultant Leakage Activity

For evaluating the radiological consequences due to a postulated MSLB, the activity released from the affected steam generator is released directly to the environment. The unaffected steam generators are assumed to continually discharge steam and entrained activity via the safety and relief valves up to the time initiation of the RHR system can be accomplished.

All activity is released to the environment with no consideration given to radioactive decay or cloud depletion by ground deposition during transport to the exclusion area boundary and low population zone. Hence, the resultant radiological consequences represent the most conservative estimate of the potential integrated dose due to the postulated MSLB.

15.1.5.3.2 Identification of Uncertainties and Conservative Elements in the Analysis

A. Reactor coolant activities are based on the technical specification limit of 1.0-μCi/g I-131 DE with extremely large iodine spike values, resulting in equivalent concentrations many times greater than the reactor coolant activities based on 0.12-percent defective fuel associated with normal operating conditions.

B. The noble gas activities are based on 1-percent defective fuel which cannot exist simultaneously with 1.0-μCi/g I-131. For iodines, 1-percent defects would be approximately three times the technical specification limit.

C. A 1-gal/min steam generator primary-to-secondary leakage is assumed, which is significantly greater than that anticipated during normal operation. Furthermore, it was conservatively assumed that 0.35-gal/min leakage goes to the affected steam generator.

D. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that the assumed meteorological conditions would be present during the course of the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meteorological conditions assumed, are conservative.

15.1.5.3.3 Conclusions

15.1.5.3.3.1 Filter Loadings. The only ESF filtration system considered in the analysis which limits the consequences of the MSLB is the control room filtration system.

Integrated activity on the control room filters has been evaluated for the more limiting loss-of-coolant accident (LOCA) analysis, as discussed in Subsection 15.6 of RESAR-SP/90 PDA Module 1, "Primary Sides Safeguards Systems". Since the control room filters are capable of accommodating the potential design basis LOCA fission product iodine loadings, there will be sufficient capacity to accommodate any fission product loading due to a postulated MSLB.

15.1.5.3.3.2 Dose to Receptor at the Exclusion Area Boundary and Low Population Zone Outer Boundary. The potential radiological consequences resulting from the occurrence of a postulated MSLB have been conservatively analyzed using assumptions and models described. The total-body gamma dose due to immersion from direct radiation and the thyroid dose due to inhalation have been analyzed for the 0- to 2-h dose at the exclusion area boundary and for the duration of the accident (0 to 8 h) at the low population zone outer boundary. The results are listed in Table 15.1-4. The resultant doses are small fractions of the guideline values of 10 CFR 100.

15.1.6 References

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907, June 1972. Also supplementary information in letter from T. M. Anderson, NS-TMA-1802, May 26, 1978 and NS-TMA-1824, June 16, 1978.

2. "Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August 1974.

3. Moody, F. M., "Transactions of the ASME, Journal of Heat Transfer" Figure 3, page 134, February 1965.

TABLE 15.1-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE AN INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

ACCIDENT	EVENT	TIME (sec) N LOOP
Excessive feedwater flow at full load	One main feedwater control valve fails fully open	0.0
	High steam generator water level signal generated	43.6
	Turbine trip occurs due to high steam generator level	46.1
	Minimum DNBR occurs	46.5
	Reactor trip occurs	48.1
	Feedwater isolation valves close	50.6
Excessive Increase in Secondary Steam Flow		
1. Manual Reactor Control (Minimum moderator feedback	10% step load increase	0.0
	Equilibrium conditions reached (approximate time only)	150

TABLE 15.1-1 (Continued)

ACCIDENT	EVENT	TIME (sec) N LOOP
2. Manual Reactor Control (Maximum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate time only	70
3. Automatic Reactor Control (Minimum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reach (approximate time only)	200
4. Automatic Reactor Control (Maximum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reach (approximate time only)	70
Accidental depressurization of the main steam system	Inadvertent opening of one main steam safety or relief valve	0.0

TABLE 15.1-1 (Continued)

ACCIDENT	EVENT	TIME (sec) N LOOP
	Pressurizer empties	290.0
	2500 ppm boron reaches core	350.0
Steam System Piping Failure		
1. Case a (Plant initially at no load with offsite power)	Steamline ruptures	0.0
	Pressurizer empty	18.5
	Criticality attained	45.0
	2500 ppm boron reaches core	30.0
2. Case b (Same as Case a Except for loss of off-site power)	Steamline ruptures	0.0
	Criticality attained	54.0
	2500 ppm boron reaches core	76.0

TABLE 15.1-2

EQUIPMENT REQUIRED FOLLOWING A RUPTURE OF A MAIN STEAMLINE

SHORT TERM (REQUIRED FOR MITIGATION OF ACCIDENT)	HOT STANDBY	REQUIRED FOR COOLDOWN
Reactor trip and safeguards actuation channels including sensors, circuitry, and processing equipment (the protection circuits used to trip the reactor on low DNBR, high Kw/ft, and low reactor coolant pump speed may be excluded).	Emergency feedwater system including pumps, water supply, and system valves and piping (this system must be placed in service to supply water to operable steam generators no later than 10 minutes after the incident).	Steam generator power-operated relief valves (can be manually operated locally).
		Controls for defeating automatic safety injection actuation during a cooldown and depressurization.
Safety injection system including the pumps, the refueling water storage tank, and the systems valves and piping.	Reactor containment ventilation cooling units.	Residual heat removal system including pumps, heat exchanger and system valves and piping necessary to cool and maintain the reactor coolant system in a cold shutdown condition.
	Capability for obtaining a reactor coolant system sample.	
Standby diesel generators and Class 1E power distribution equipment.		

TABLE 15.1-2 (Continued)

EQUIPMENT REQUIRED FOLLOWING A RUPTURE OF A MAIN STEAMLINE

SHORT TERM (REQUIRED FOR MITIGATION OF ACCIDENT)	HOT STANDBY	REQUIRED FOR COOLDOWN
Service water and reactor plant component cooling water system.		
Containment spray system equipment.		
Emergency feedwater system including pumps, water supplies, piping, valves.		
Pressurizer and main steam safety valves.		
Circuits and/or equipment required to trip the main feedwater pumps.		
Main feedwater isolation valves (trip closed feature).		

TABLE 15.1-2 (Continued)

EQUIPMENT REQUIRED FOLLOWING A RUPTURE OF A MAIN STEAMLINE

SHORT TERM (REQUIRED FOR MITIGATION OF ACCIDENT	HOT STANDBY	REQUIRED FOR COOLDOWN
Bypass feedwater control valves (trip closed feature).		
Main steamline stop valves (trip closed feature).		
Main steamline stop valve bypass valves (trip closed feature).		
Steam generator blowdown isolation valves (automatic closure feature).		
Batteries (Class 1E).		
Control room air conditioning.		

TABLE 15.1-2 (Continued)

EQUIPMENT REQUIRED FOLLOWING A RUPTURE OF A MAIN STEAMLINE

SHORT TERM (REQUIRED FOR MITIGATION OF ACCIDENT	HOT STANDBY	REQUIRED FOR COOLDOWN
Control room equipment must not be damaged to an extent where any equipment will be spuriously actuated or any of the equipment contained elsewhere in this list cannot be operated.		
Emergency lighting.		
Post-Accident Monitoring System.		
Containment Atmosphere Recirculation System.		
ESFA and SI cubicle unit coolers.		

TABLE 15.1-2a

BALANCE OF PLANT ASSUMPTIONS USED IN
MAJOR RUPTURE OF A MAIN STEAM LINE

Item	Value Used in Analysis
Steam Line Stop Valve Closure Time	5.0 sec
Feedwater Isolation Valve Closure Time	5.0 sec

Table 10.1-1 gives the interface requirements for steam line and feedwater isolation valves.

TABLE 15.1-3

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A MAIN STEAM LINE BREAK

I. Source Data	
a. Core power level, MWt	3876
b. Total steam generator tube leakage, gpm	1
c. Reactor coolant iodine activity:	
1. Accident Initiated Spike	Initial activity equal to the dose equivalent of 1.0 μCi/gm of I-131 with an assumed iodine spike that increases the rate of iodine release into the reactor coolant by a factor of 500. See Tables 15A-5 and 15A-6.
2. Pre-Accident Spike	An assumed pre-accident iodine spike, which has resulted in the dose equivalent of 60 μCi/gm of I-131 in the reactor coolant. See Table 15A-5.
d. Reactor coolant noble gas activity (both cases)	Based on one percent defective fuel. See Table 15A-7.
e. Secondary system initial activity	Dose equivalent of 0.1 μCi/gm of I-131.
f. Reactor coolant mass, grams	3.3×10^8
g. Secondary coolant mass (4 generators), grams	2.0×10^8
h. Offsite power	Lost after Trip
i. Primary-to-secondary leakage duration	8 hours
j. Species of iodine	100 percent elemental
II. Atmospheric Dispersion Factors	See Table 15A-2

TABLE 15.1-3 (Sheet 2)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A MAIN STEAM LINE BREAK

III. Activity Release Data for the steam generator in the faulted loop	
a. Primary to secondary leak-rate (gpm)*	0.35
b. Steam released	
0 - 10 min (lb)	230,000
10 min - 8.0 hr	1,000
c. Iodine Partition Factor	1
IV. Activity Release Data for the Steam Generators in the Intact Loops	
a. Primary to secondary leak rate (gpm)*	0.65
b. Steam Released	
0 - 2 hr (lb)	530,000
2 - 8 hr	1,500,000
c. Iodine partition factor	100

V. Activity Released to the Environment

a. Accident Initiated Spike

Isotope	0-2 h (Ci)	2-8 h (Ci)
I-131	7.0	26.5
I-132	17.0	53.8
I-133	12.2	50.3
I-134	7.0	11.5
I-135	8.9	37.2

b. Pre-Accident Spike

Isotope	0-2 h (Ci)	2-8 h (Ci)
I-131	9.1	16.0
I-132	7.4	4.0
I-133	13.3	21.0
I-134	1.4	0.2
I-135	7.0	8.0

*Based on water at 590°F, 2250 psia

TABLE 15.1-3 (Sheet 3)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A MAIN STEAM LINE BREAK

c. Noble gases - both cases

Isotope	0-2 h (Ci)	2-8 h (Ci)
Xe-131m	0.5	2.1
Xe-133m	5.5	16.0
Xe-133	87.0	254.0
Xe-135m	0.03	1.0×10^{-4}
Xe-135	2.0	5.0
Xe-138	0.04	3.0×10^{-4}
Kr-85m	0.4	0.9
Kr-85	1.8	7.0
Kr-87	0.2	0.1
Kr-88	0.6	1.1

TABLE 15.1-3 (Sheet 4)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A MAIN STEAM LINE BREAK

RADIOLOGICAL CONSEQUENCES	Doses (rem)
Case 1 - Accident Initiated Iodine Spike	
Exclusion area boundary (0 to 2 h) Thyroid	0.9
Low population zone outer boundary (8 h) Thyroid	1.0
Case 2 - Preaccident Iodine Spike	
Exclusion area boundary (0 to 2 h) Thyroid	1.1
Low population zone outer boundary (8 h) Thyroid	0.79
Both Cases - Whole-Body Gamma	
Exclusion area boundary (0 to 2 h)	2.7×10^{-4}
Low population zone outer boundary (8 h)	2.2×10^{-4}

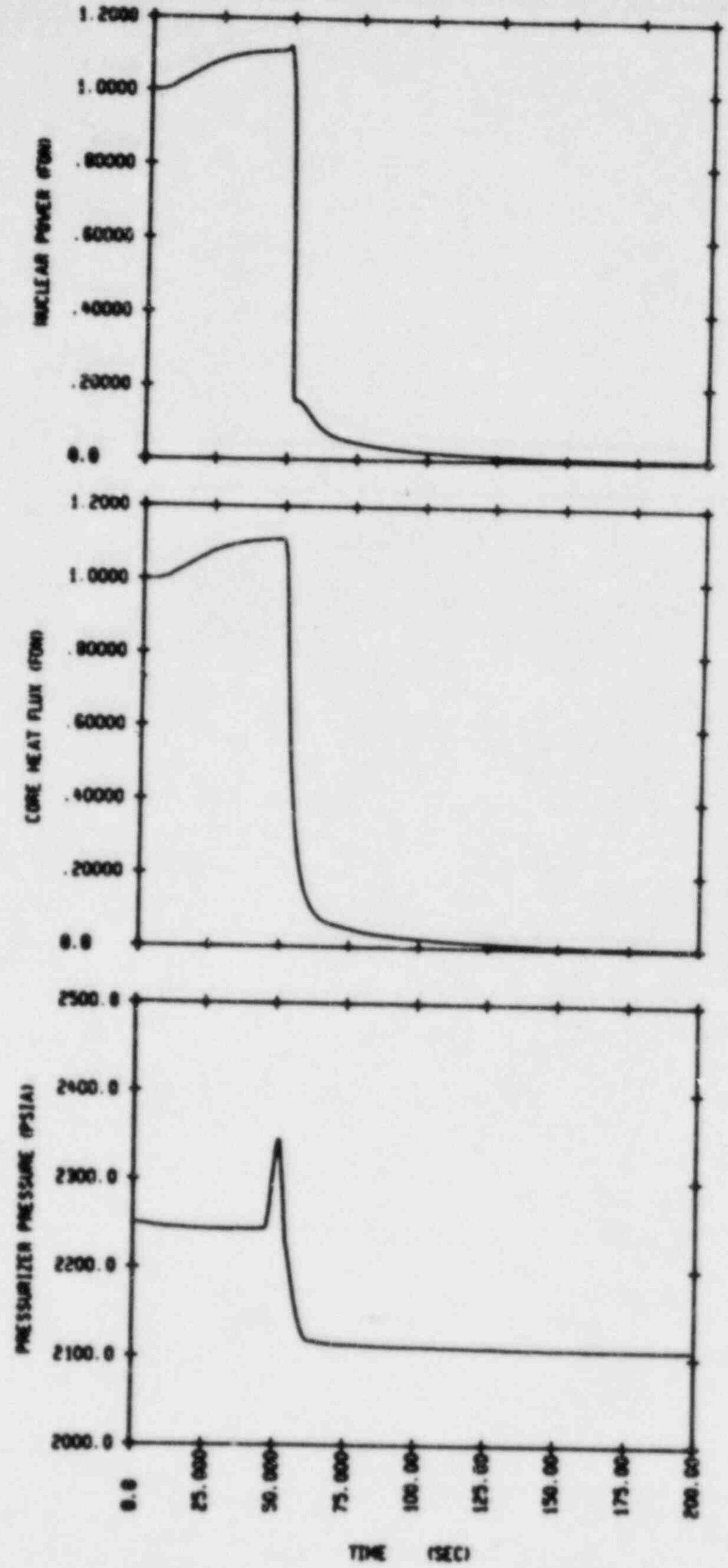

FIGURE 15.1-1 TRANSIENTS FOR FEEDWATER CONTROL VALVE MALFUNCTION

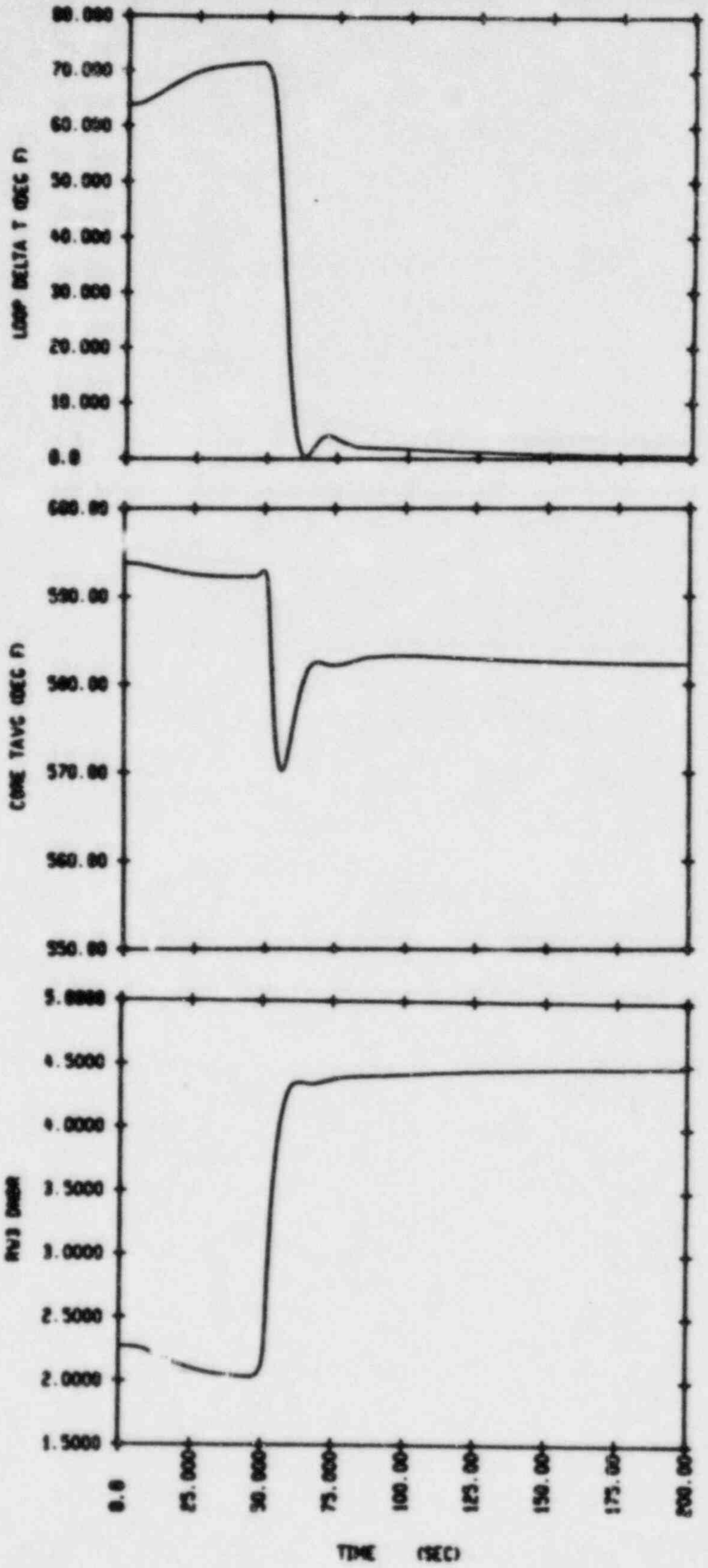

FIGURE 15.1-2 TRANSIENTS FOR FEEDWATER CONTROL VALVE MALFUNCTION

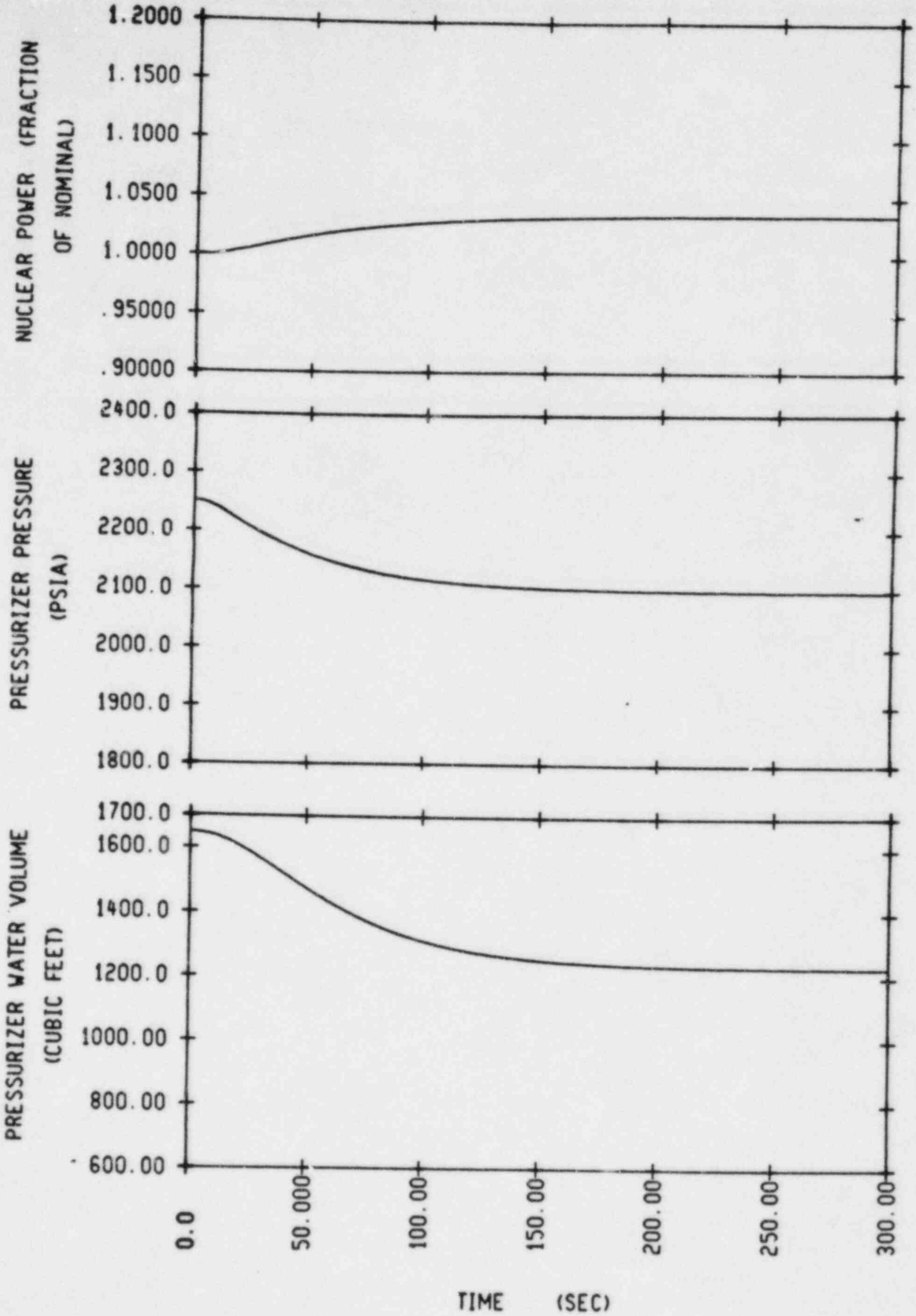

FIGURE 15.1-3 TEN PERCENT STEP LOAD INCREASE, MINIMUM MODERATOR FEEDBACK, MANUAL REACTOR CONTROL

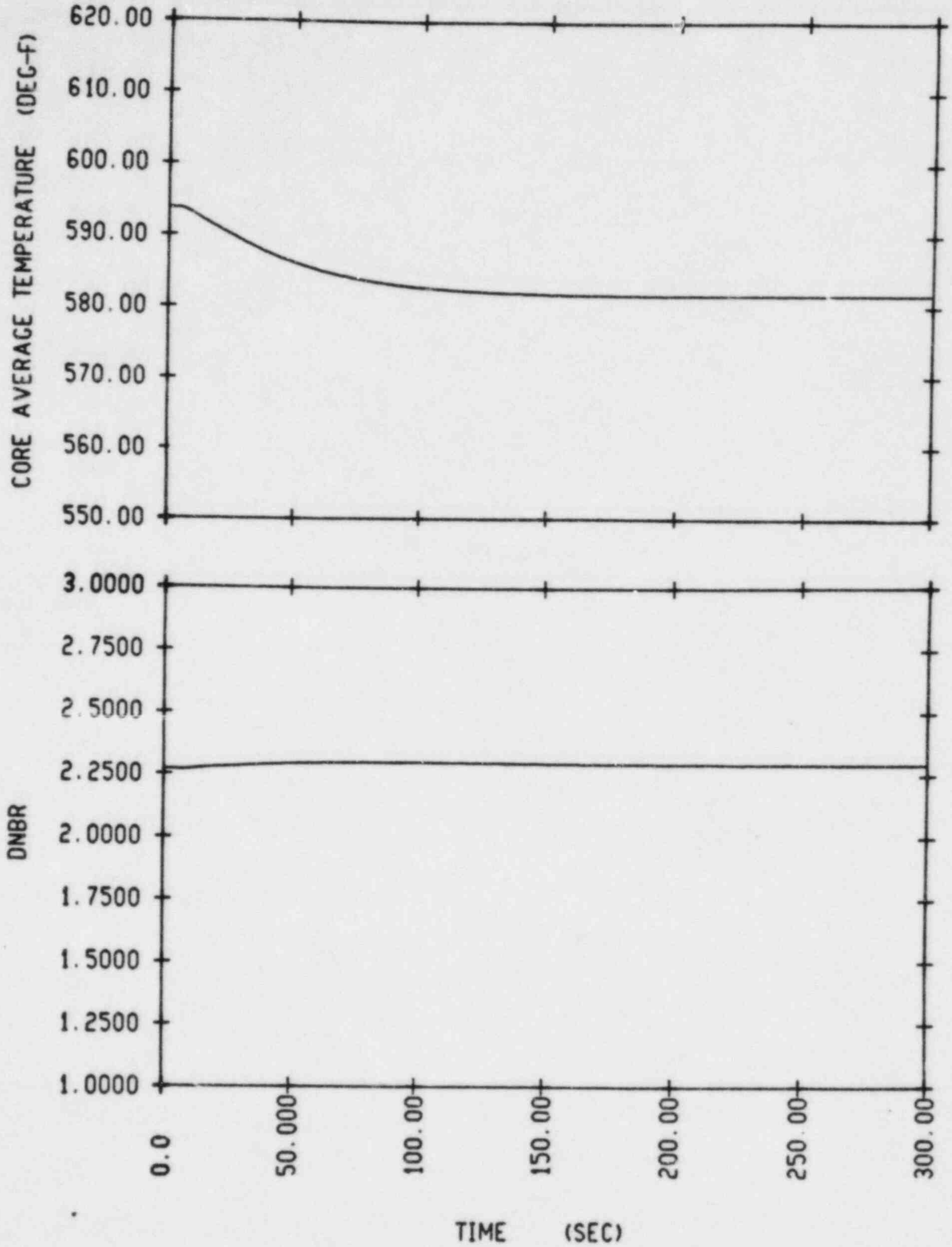

FIGURE 15.1-4 TEN PERCENT STEP LOAD INCREASE, MINIMUM MODERATOR FEEDBACK, MANUAL REACTOR CONTROL

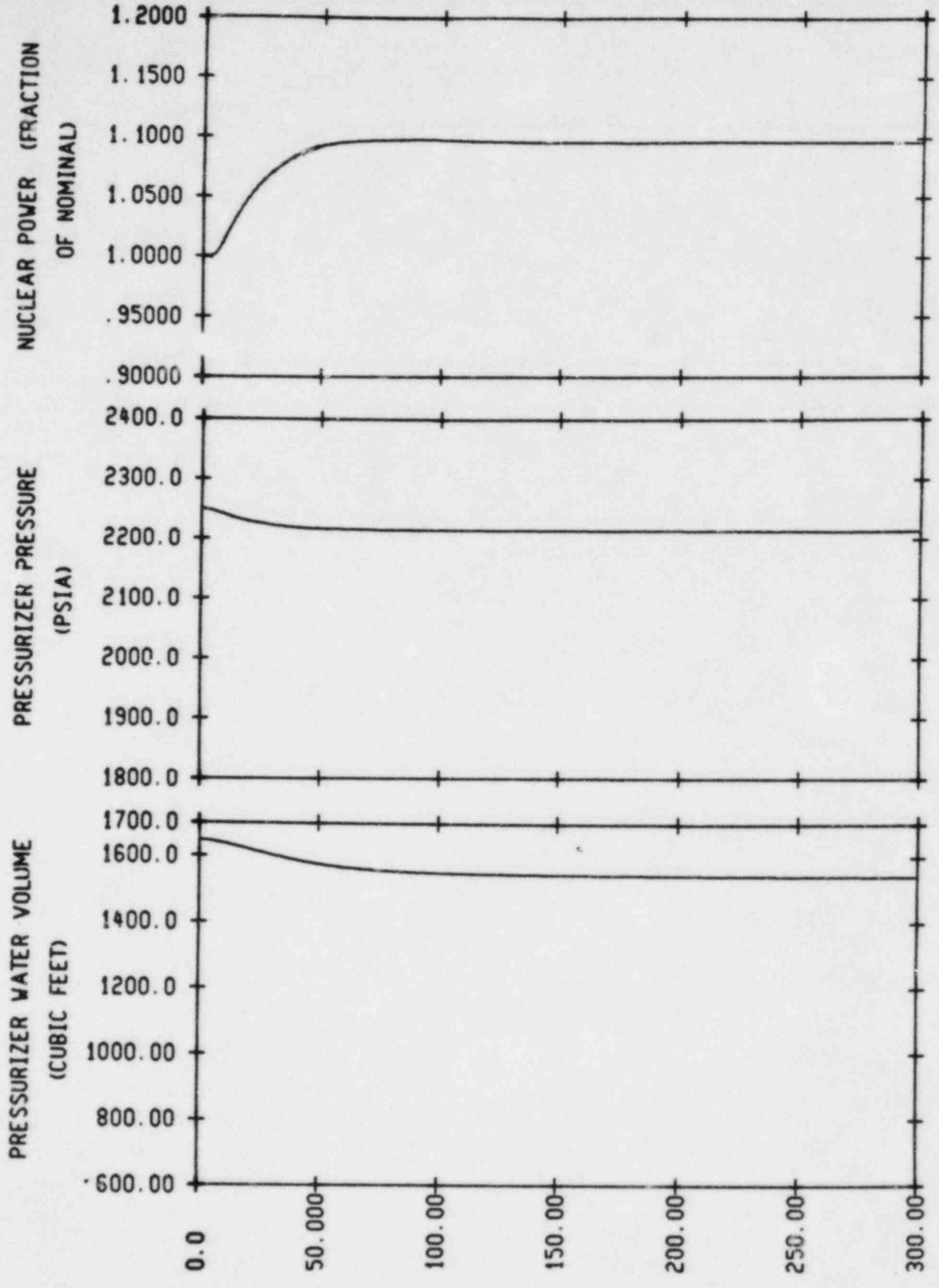

FIGURE 15.1-5 TEN PERCENT STEP LOAD INCREASE, MAXIMUM MODERATOR FEEDBACK, MANUAL REACTOR CONTROL

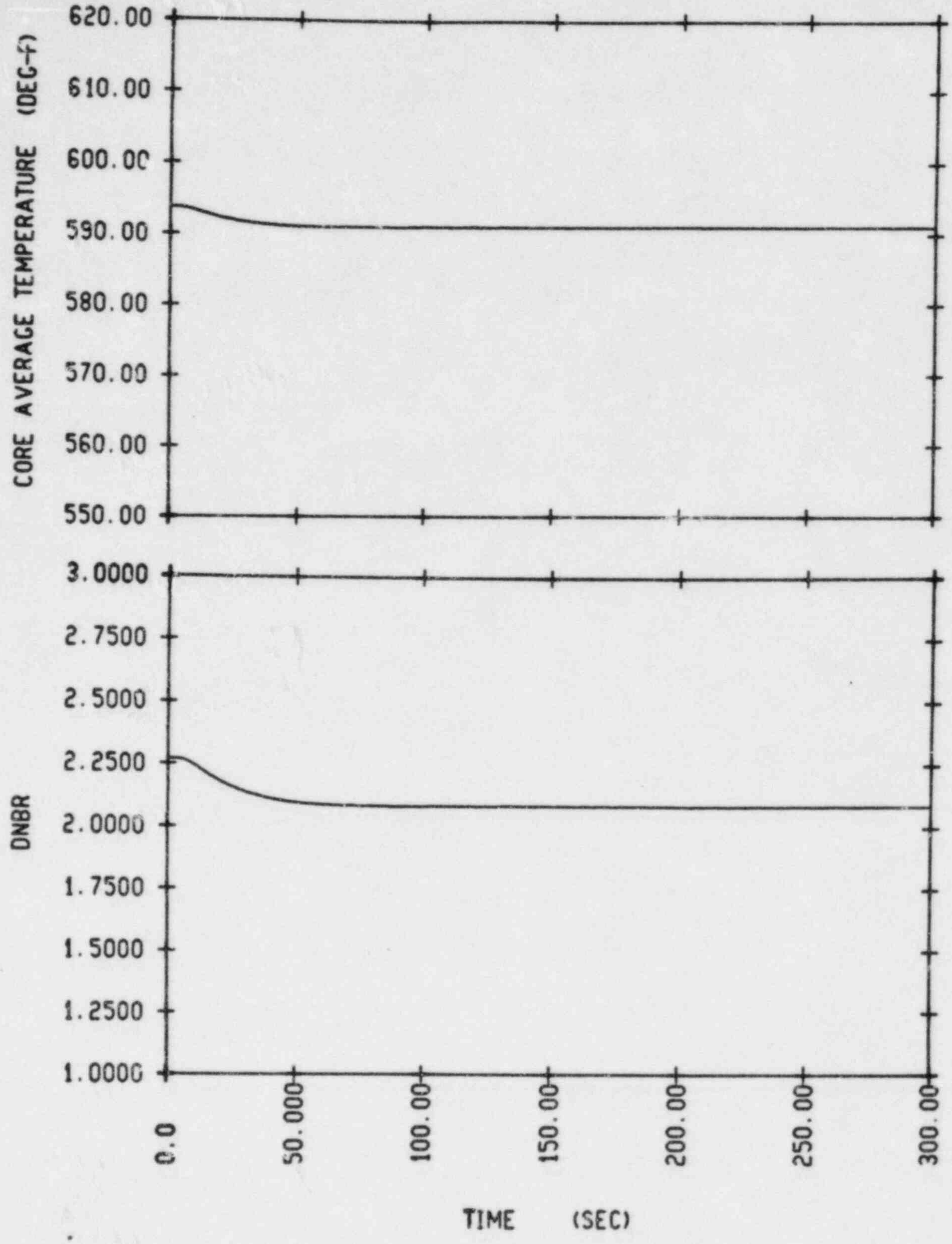

FIGURE 15.1-6 TEN PERCENT STEP LOAD INCREASE, MAXIMUM MODERATOR FEEDBACK, MANUAL REACTOR CONTROL

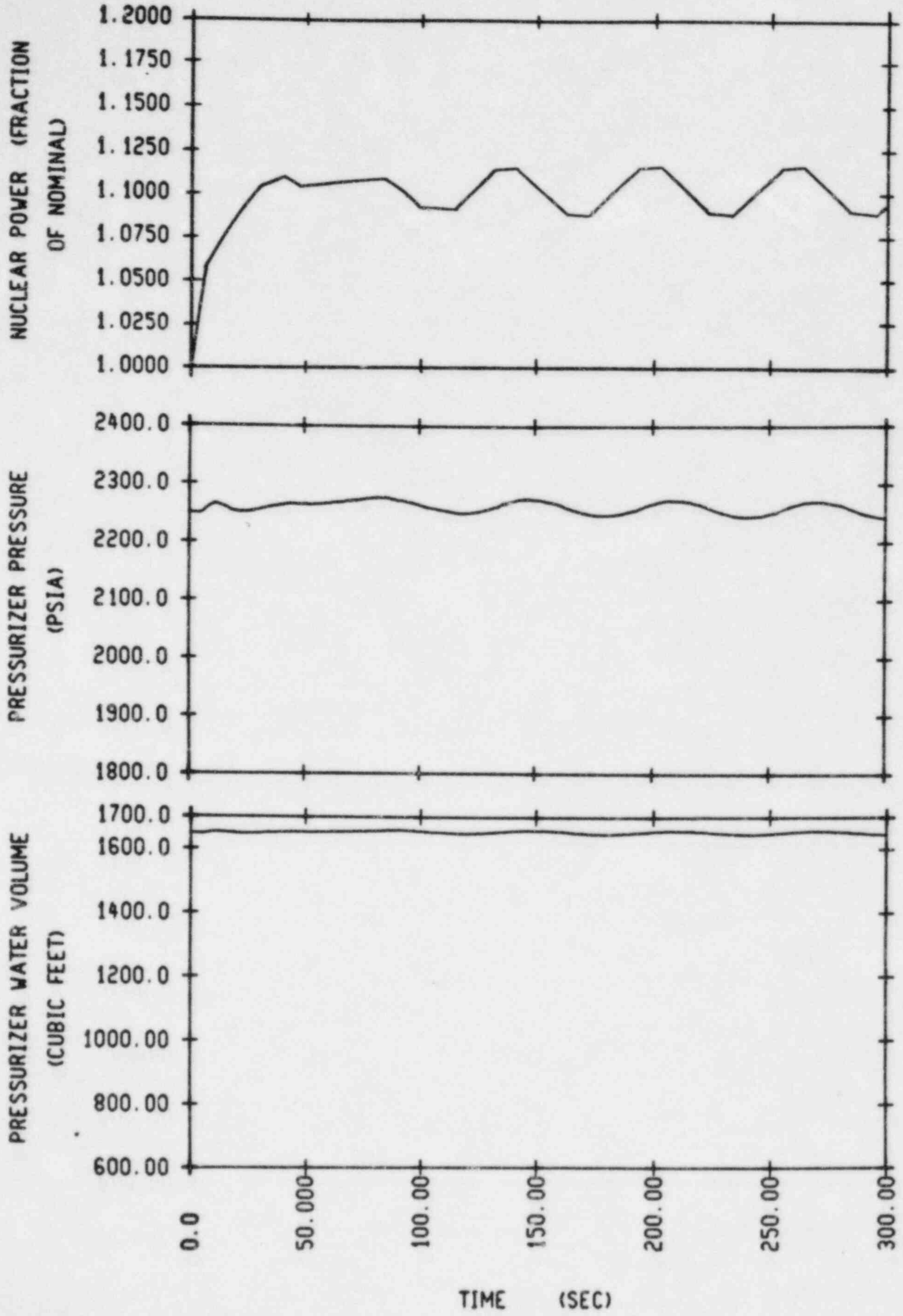

FIGURE 15.1-7 TEN PERCENT STEP LOAD INCREASE, MINIMUM MODERATOR FEEDBACK, AUTOMATIC REACTOR CONTROL

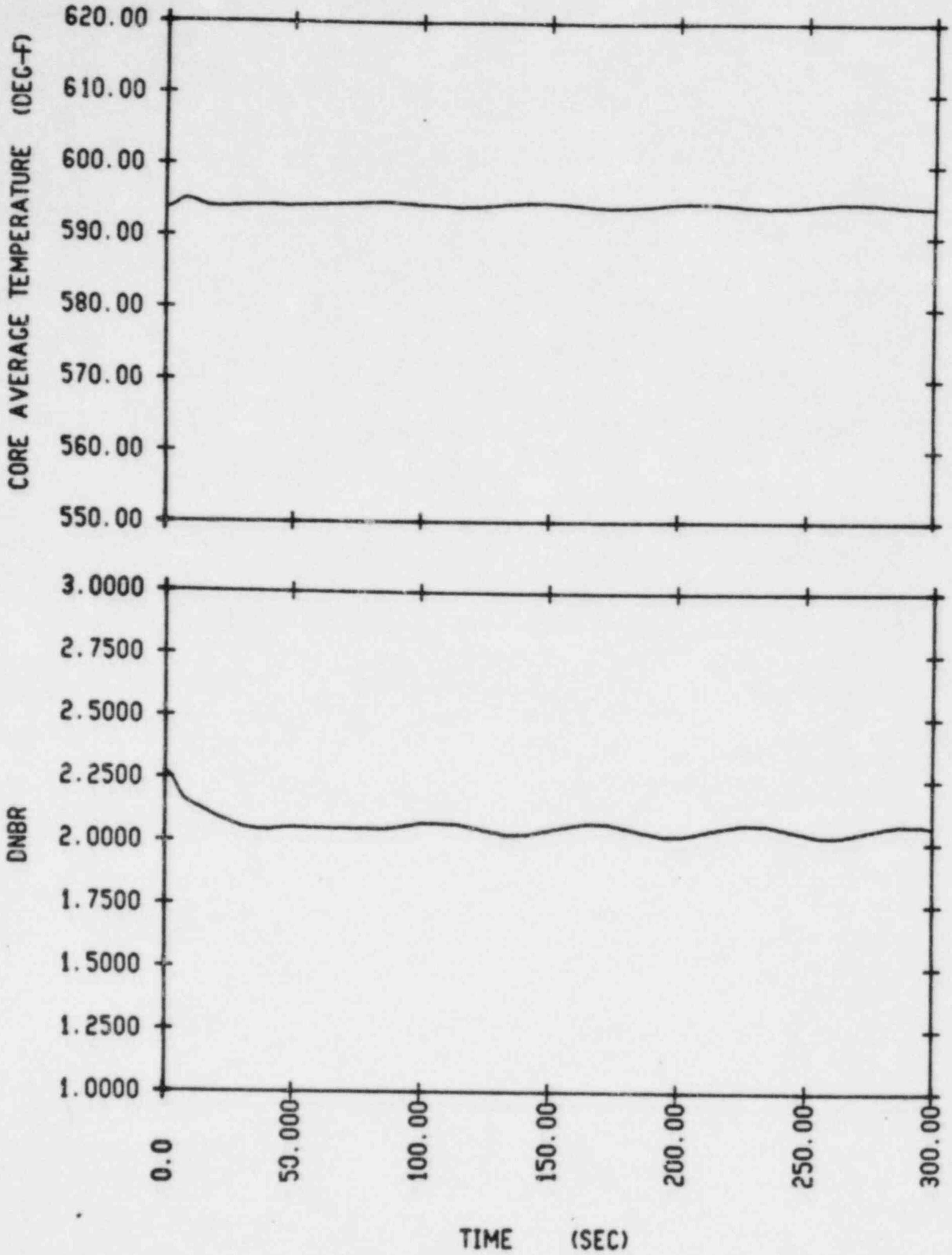

FIGURE 15.1-8 TEN PERCENT STEP LOAD INCREASE, MINIMUM MODERATOR FEEDBACK, AUTOMATIC REACTOR CONTROL

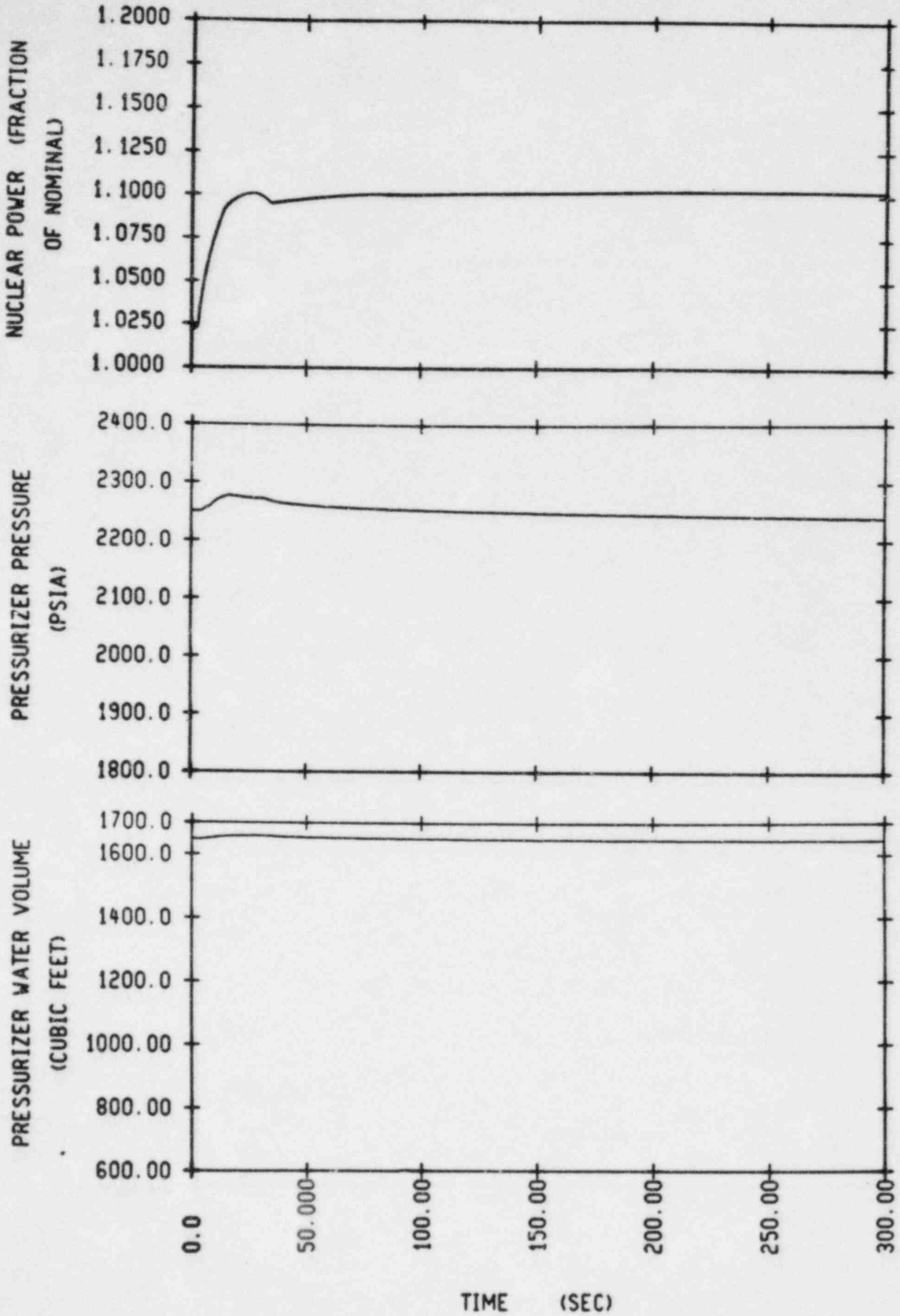

FIGURE 15.1-9 TEN PERCENT STEP LOAD INCREASE, MAXIMUM MODERATOR FEEDBACK, AUTOMATIC REACTOR CONTROL

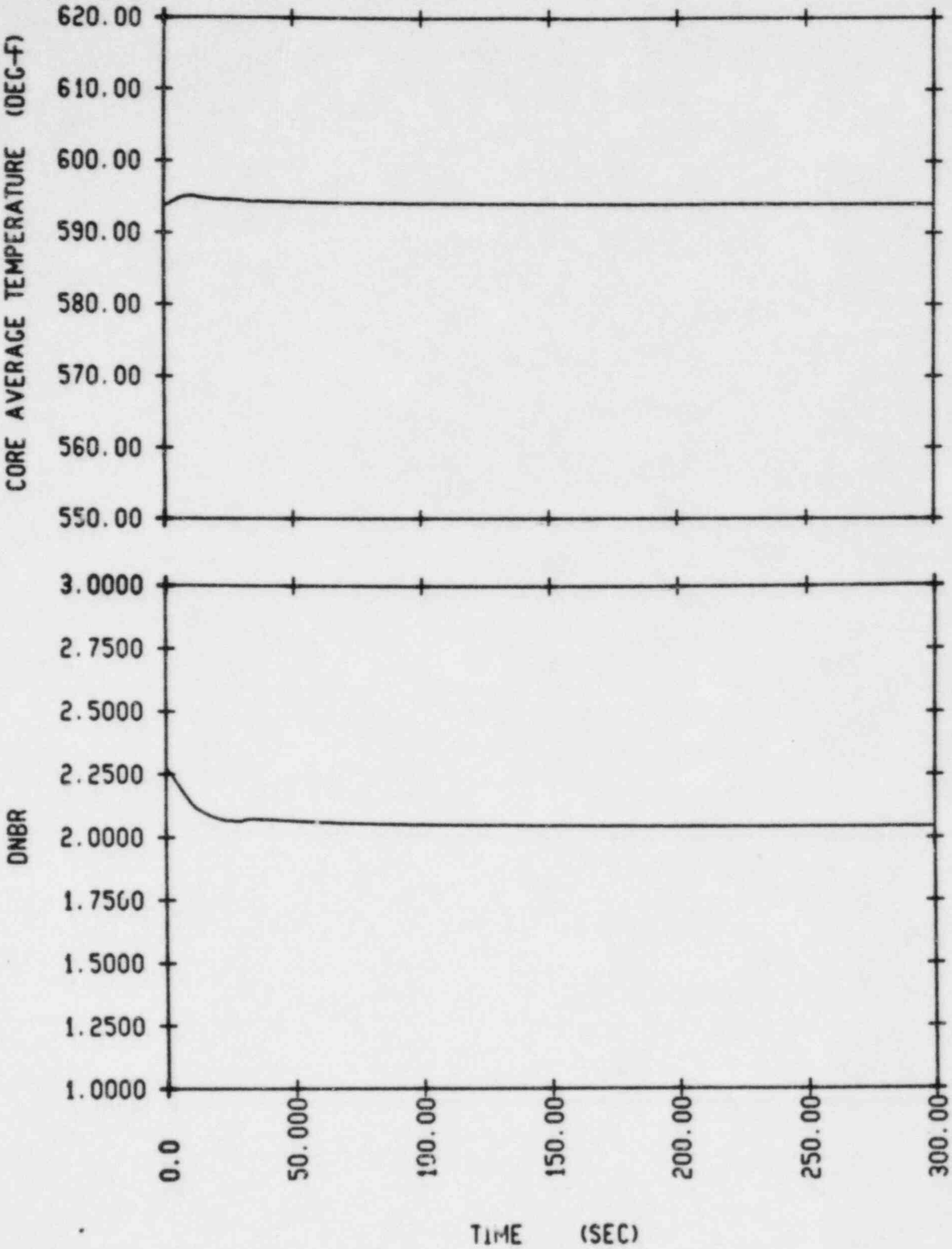

FIGURE 15.1-10 TEN PERCENT STEP LOAD INCREASE, MAXIMUM MODERATOR FEEDBACK, AUTOMATIC REACTOR CONTROL

Figure 15.1-12 This Figure is Not Applicable

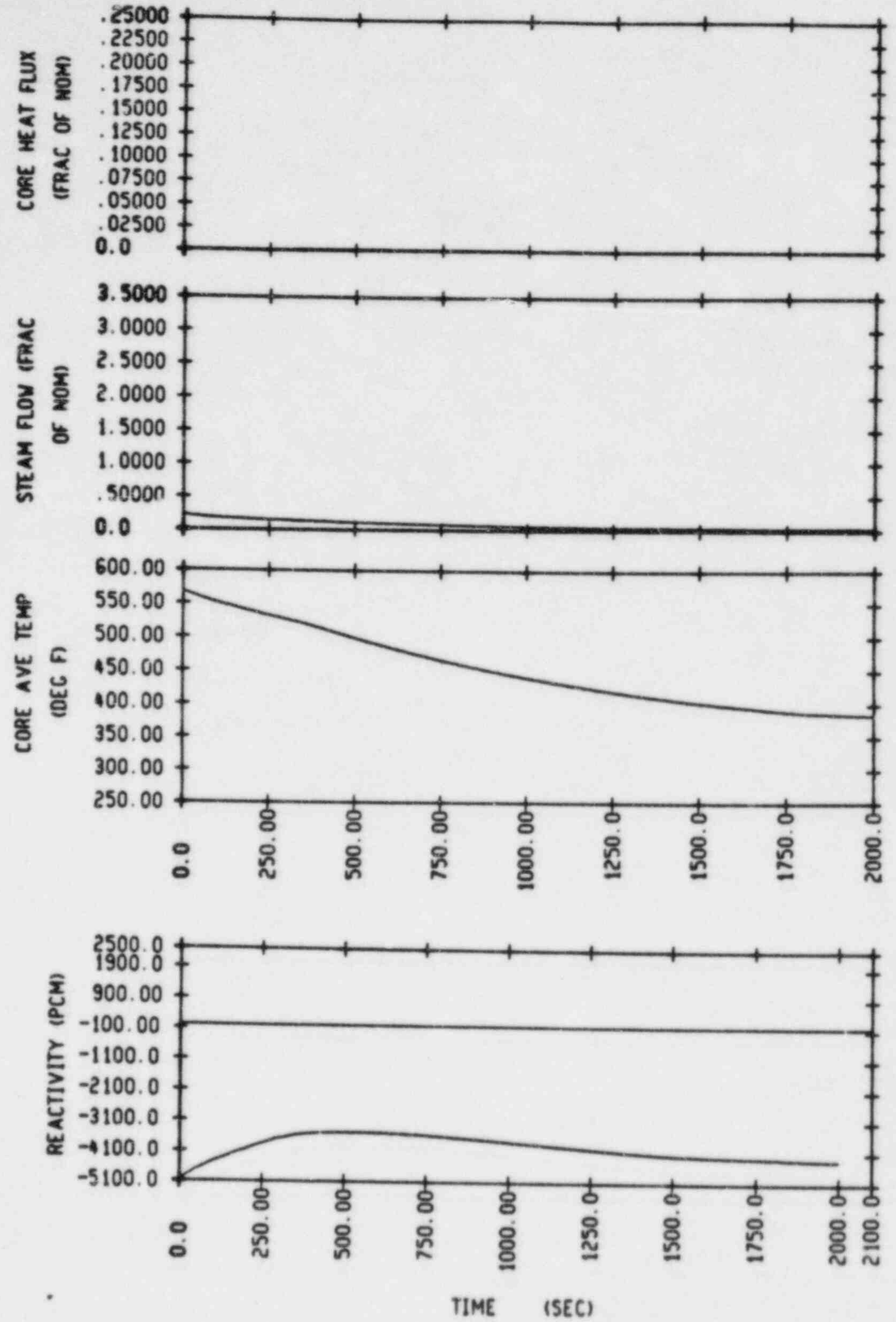

FIGURE 15.1-13 FAILURE OF A STEAM GENERATOR SAFETY OR DUMP VALVE - HEAT FLUX VS TIME, STEAM FLOW VS TIME, AVERAGE TEMPERATURE VS TIME, REACTIVITY VS TIME

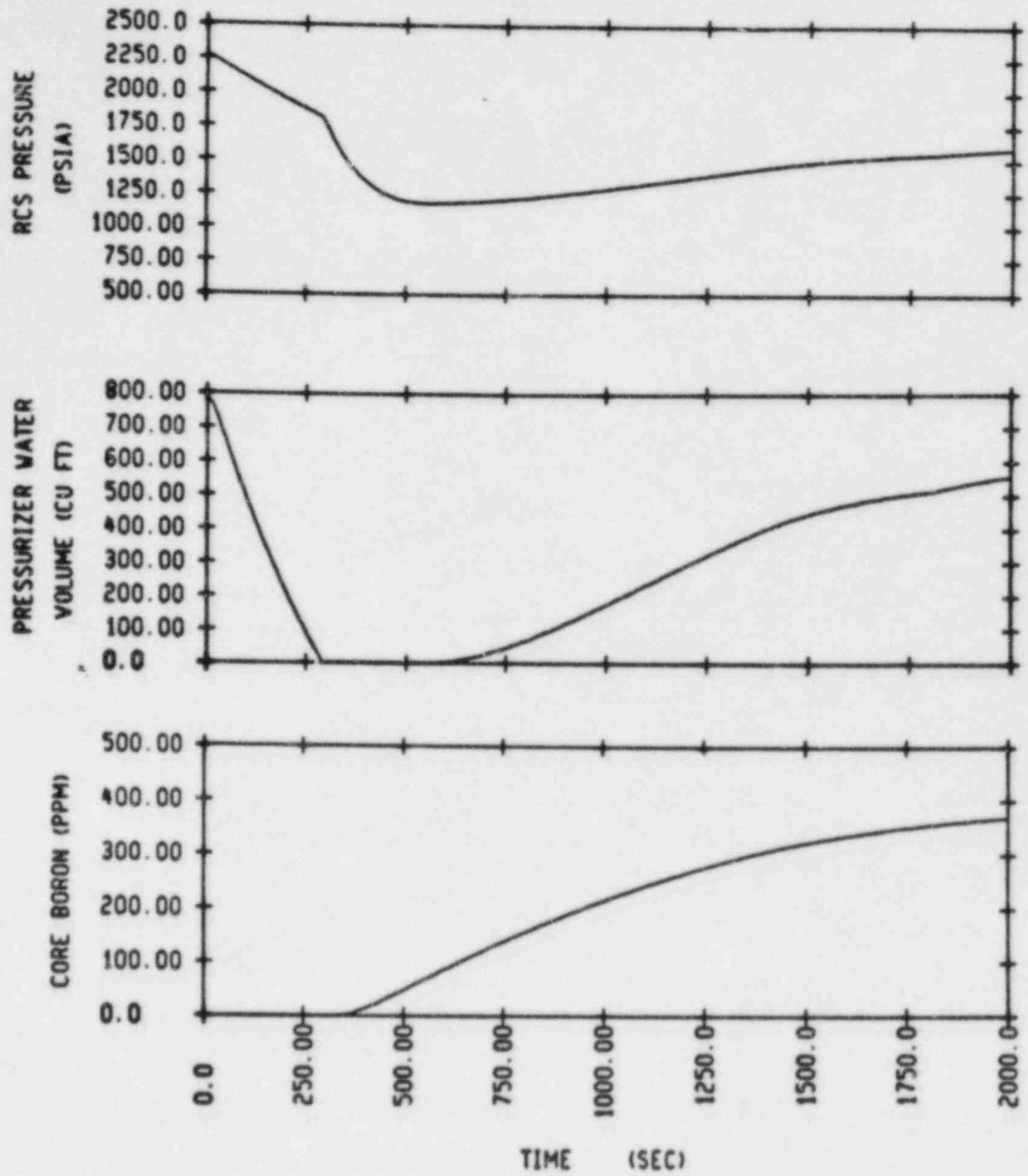

FIGURE 15.1-14 FAILURE OF A STEAM GENERATOR SAFETY OR DUMP VALVE - RCS PRESSURE VS TIME, PRESSURIZER WATER VOLUME VS TIME, BORON CONCENTRATION VS TIME

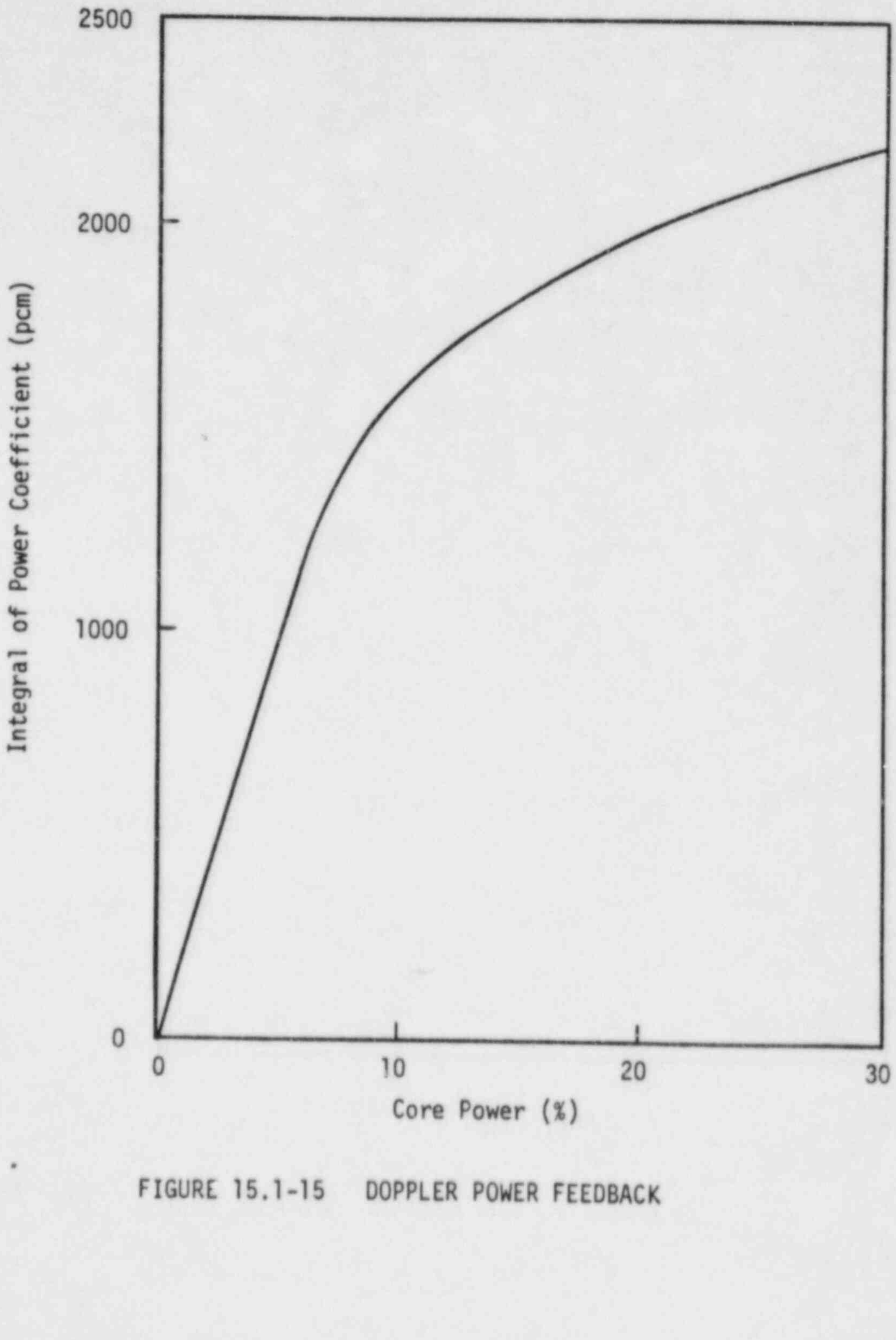

FIGURE 15.1-15 DOPPLER POWER FEEDBACK

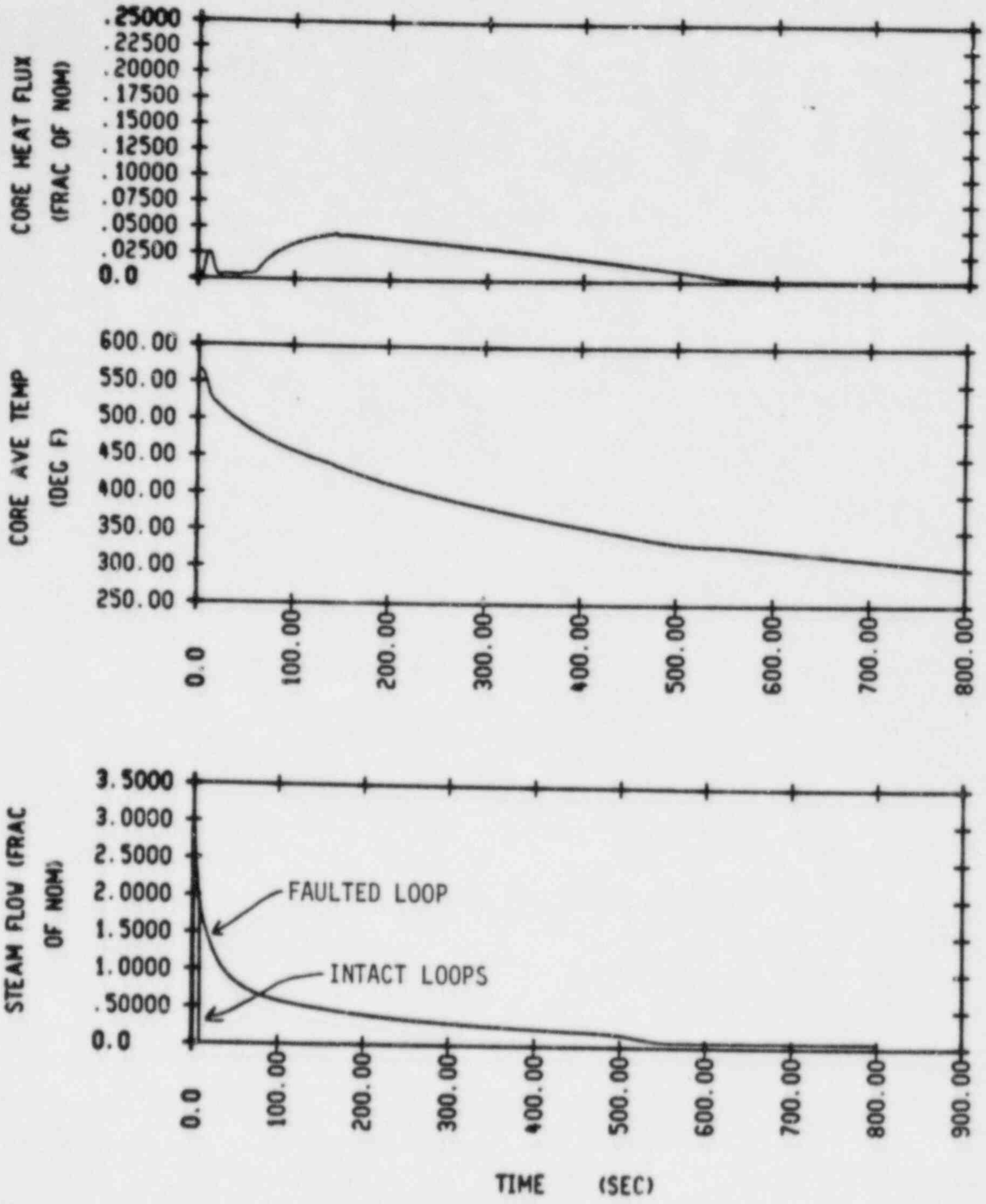

FIGURE 15.1-16 1.4 FT^2 STEAMLINE RUPTURE, OFFSITE POWER AVAILABLE - HEAT FLUX VS TIME, AVERAGE TEMPERATURE VS TIME, STEAM FLOW PER LOOP VS TIME

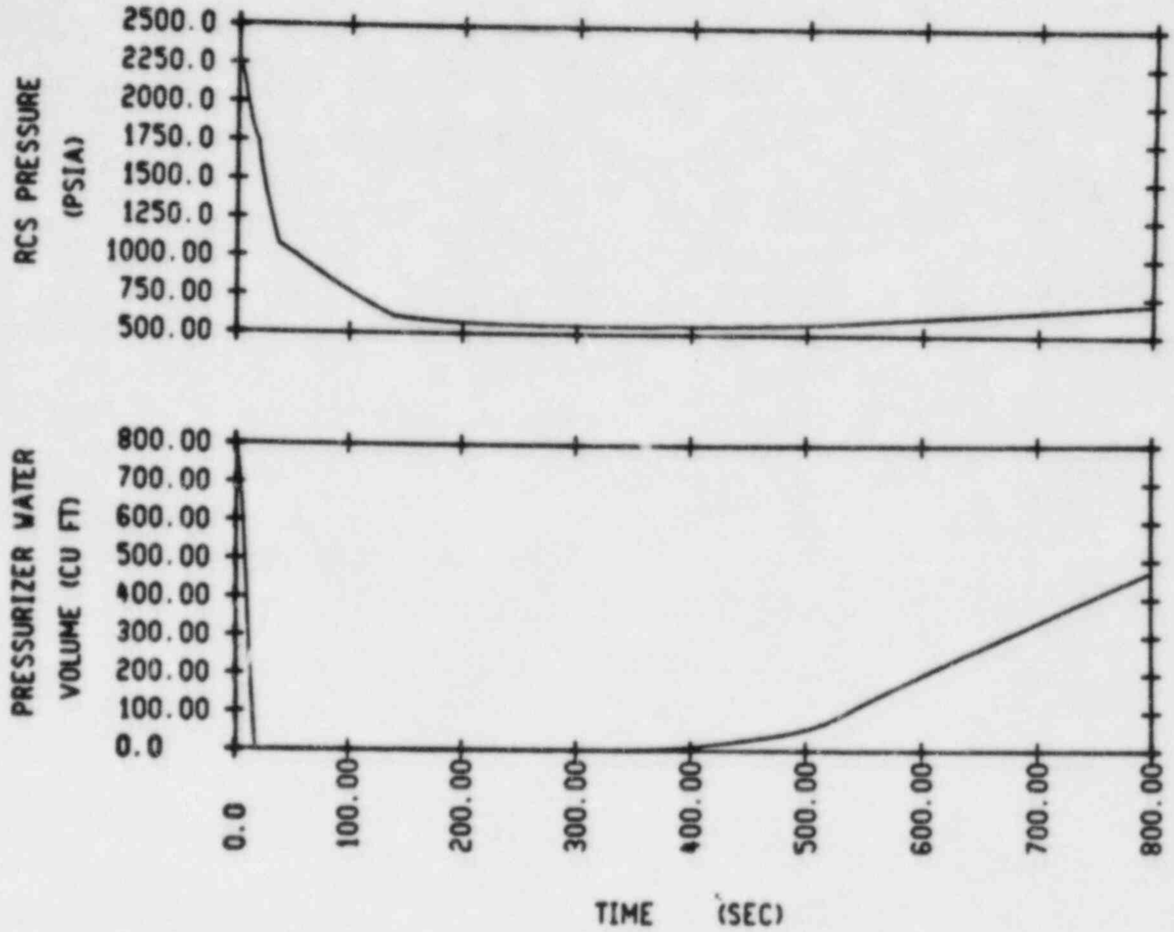

FIGURE 15.1-17 1.4 FT^2 STEAMLINE RUPTURE, OFFSITE POWER AVAILABLE, RCS PRESSURE VS TIME, PRESSURIZER WATER VOLUME VS TIME

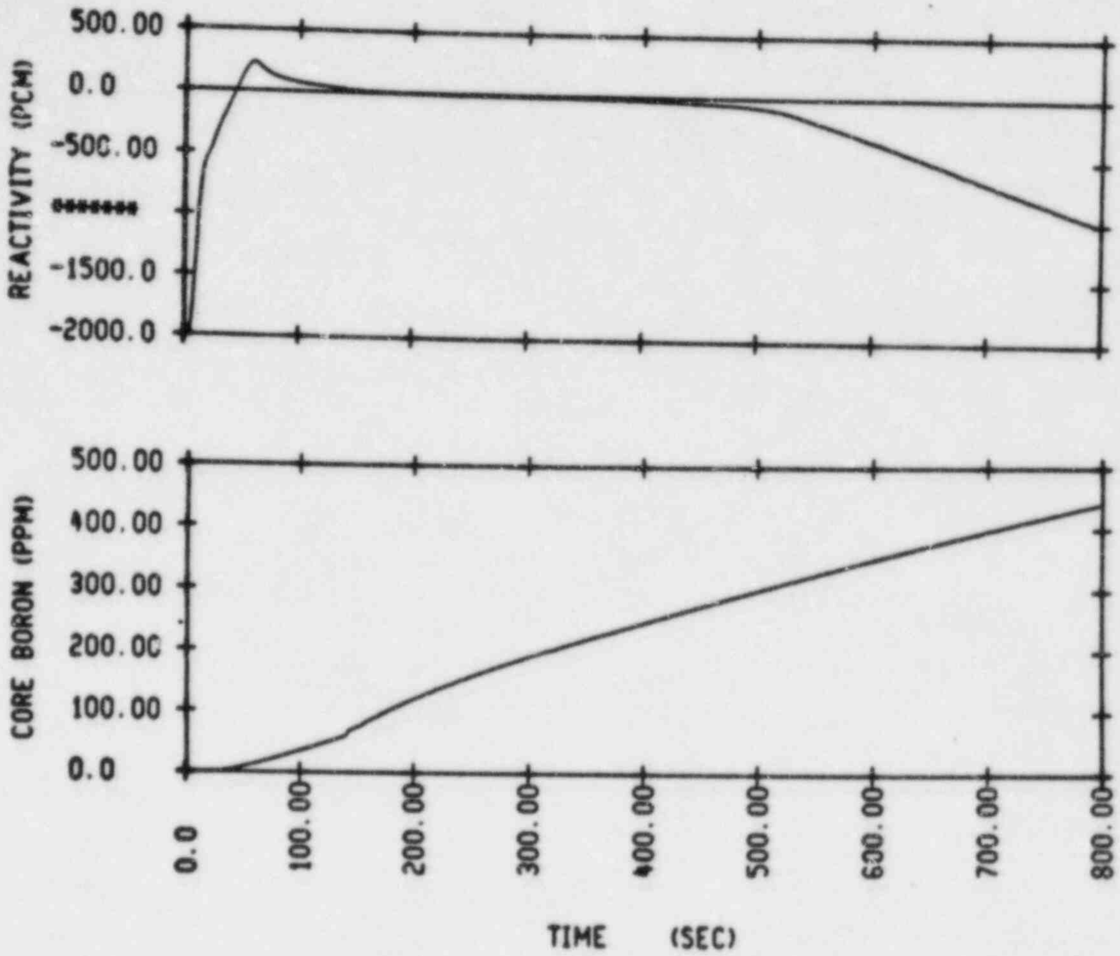

FIGURE 15.1-18 1.4 FT^2 STEAMLINE RUPTURE, OFFSITE POWER AVAILABLE, BORON CONCENTRATION VS TIME, REACTIVITY VS TIME

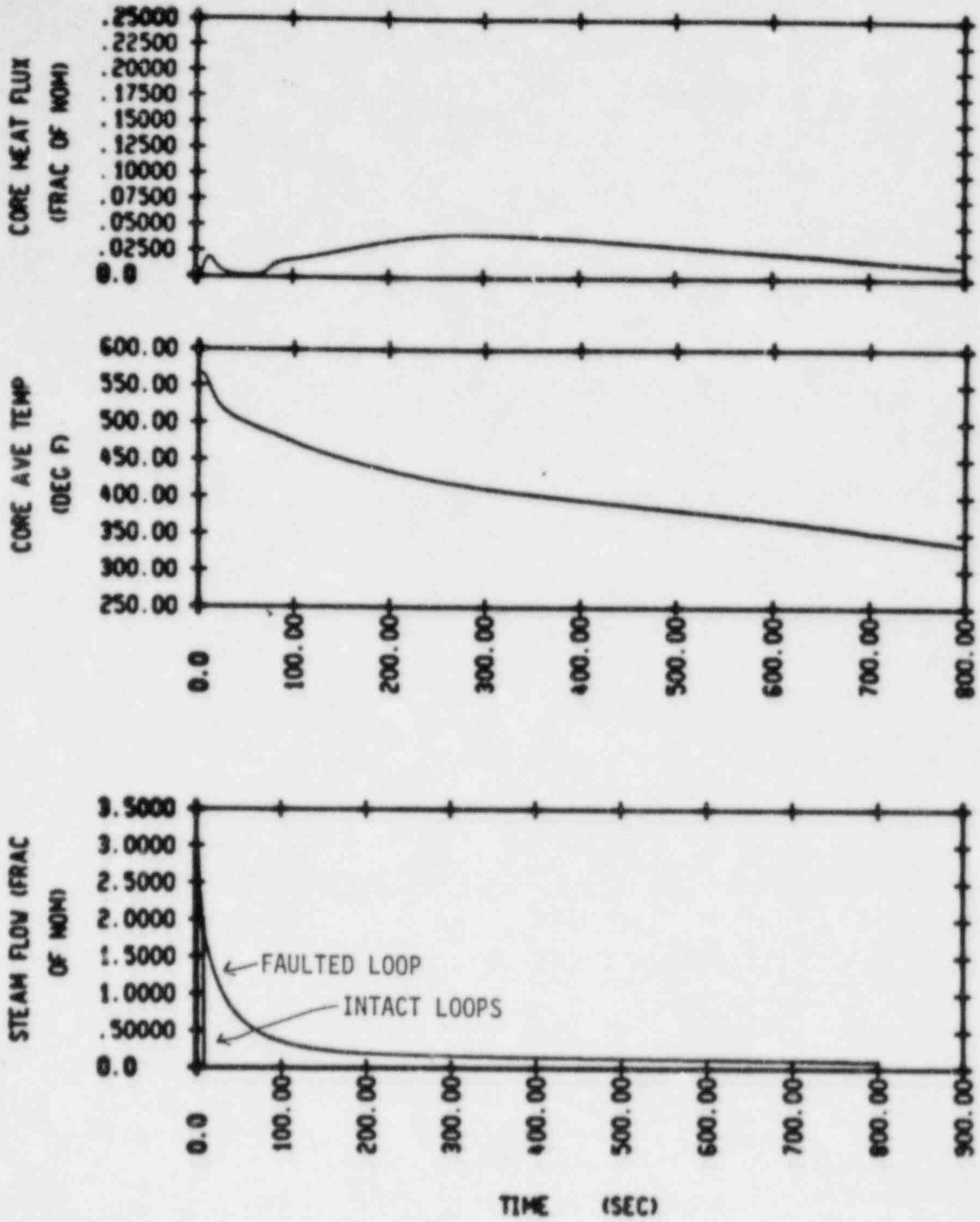

FIGURE 15.1-19 1.4 FT^2 STEAMLINE RUPTURE, OFFSITE POWER NOT AVAILABLE - HEAT FLUX VS TIME, AVERAGE TEMP VS TIME, STEAM FLOW PER LOOP VS TIME

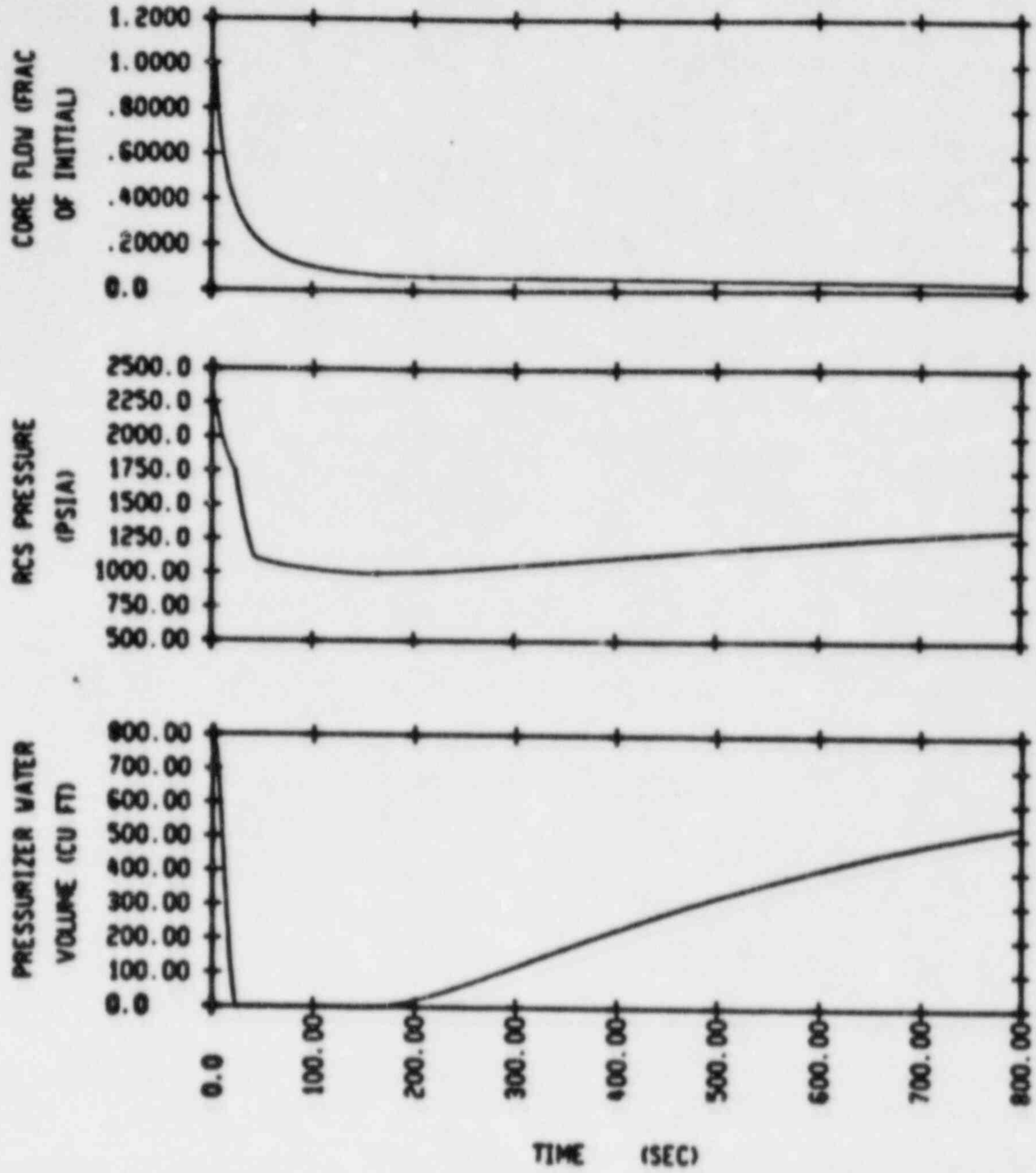

FIGURE 15.1-20 1.4 FT^2 STEAMLINE RUPTURE, OFFSITE POWER NOT AVAILABLE - CORE FLOW VS TIME, RCS PRESSURE VS TIME, PRESSURIZER WATER VOLUME VS TIME

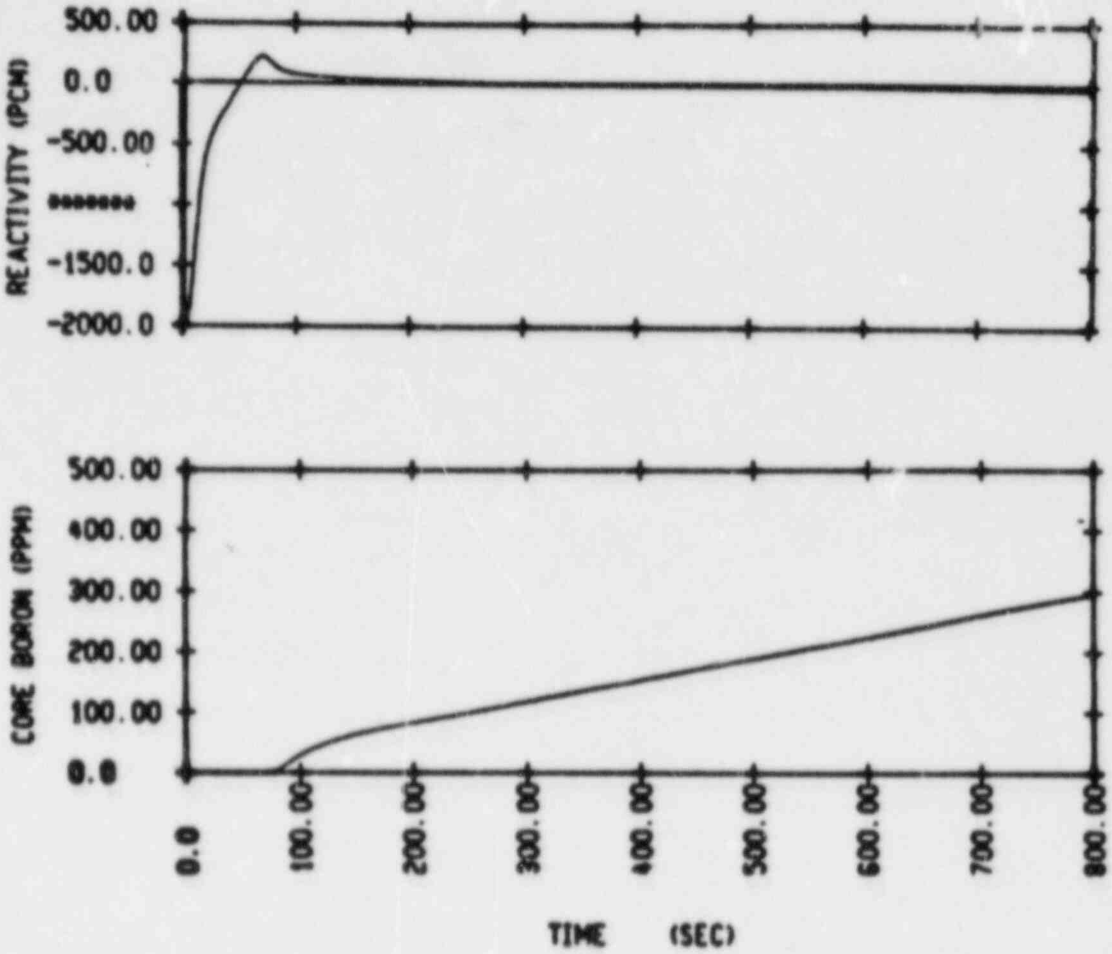

FIGURE 15.1-21 1.4 FT^2 STEAMLINE RUPTURE, OFFSITE POWER NOT AVAILABLE - REACTIVITY VS TIME, BORON CONCENTRATION VS TIME

15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

A number of transients and accidents have been postulated which could result in a reduction of the capacity of the secondary system to remove heat generated in the reactor coolant system (RCS). Detailed analyses are presented in this section for several such events which have been identified as more limiting than the others.

Discussions of the following RCS coolant heatup events are presented in this section:

1. Steam pressure regulator malfunction or failure that results in decreasing steam flow (not applicable).

2. Loss of external electrical load (Subsection 15.2.2).

3. Turbine trip (Subsection 15.2.3).

4. Inadvertent closure of main steam isolation valves (Subsection 15.2.4).

5. Loss of condenser vacuum and other events causing a turbine trip (Subsection 15.2.5).

6. Loss of nonemergency AC power to the plant auxiliaries (Subsection 15.2.6)

7. Loss of normal feedwater flow (Subsection 15.2.7).

8. Feedwater system pipe break (Subsection 15.2.8).

The above items are considered to be American Nuclear Society (ANS) Condition II events,, with the exception of a feedwater system pipe break, which is considered to be an ANS Condition IV event.

15.2.1 Steam Pressure Regulator Malfunction or Failure That Results In Decreasing Steam Flow

There are no pressure regulators whose malfunction or failure could cause a steam flow transient.

15.2.2 Loss of External Electrical Load

15.2.2.1 Identification of Causes and Accident Description

A loss of external electrical load may occur due to some electrical system disturbance. Offsite AC power remains available to operate plant components such as the reactor coolant pumps; as a result, the onsite standby diesel-generators are not required to function for this event. Following the loss of generator load, an immediate fast closure of the turbine control valves will occur. This will cause a sudden reduction in steam flow, resulting in an increase in pressure and temperature in the steam generator shell. As a result, the heat transfer rate in the steam generator is reduced, causing the reactor coolant temperature to rise, which in turn causes coolant expansion, pressurizer insurge, and RCS pressure rise.

For a loss of external electrical load without subsequent turbine trip, no direct reactor trip signal would be generated. With full load rejection capability, plant operation would be expected to continue without a reactor trip. The plant would be expected to trip from the reactor protection system if a safety limit were approached. A continued steam load of approximately 5 percent would exist after total loss of external electrical load because of the steam demand of plant auxiliaries.

In the event that a safety limit is approached, protection would be provided by the high pressurizer pressure and low DNBR trips. Following a complete loss of load, the maximum turbine would be approximately 8 to 9 percent, resulting in an overfrequency of less than 6 Hz. Any increased frequency to the reactor coolant pump motors will result in a corresponding increase in flowrate and subsequent additional margin to safety limits. For postulated loss of load and subsequent turbine generator overspeed, any overfrequency condition does not effect other safety-related pump motors, reactor protection system equipment, or other safeguard loads. Safeguard loads are supplied from offsite power or, alternatively, from standby diesels. Reactor protection system equipment is supplied from the 118 volt AC instrument power supply system, which in turn is supplied from the inverters; the inverters are supplied from a d-c bus energized from batteries or by a rectified AC voltage from safeguard buses.

In the event the steam dump valves fail to open following a large loss of load, the steam generator safety valves may lift and the reactor may be tripped by the high pressurizer pressure signal, the high pressurizer water level signal, or the low DNBR signal. The steam generator shell side pressure and reactor coolant temperatures will increase rapidly. The pressurizer safety valves and steam generator safety valves are, however, sized to protect the RCS and steam generator against over-pressure for all load losses without assuming the operation of the steam dump system, pressurizer spray, pressurizer power operated relief valves, automatic rod control or direct reactor trip on turbine trip.

The steam generator safety valve capacity is sized to relieve 105 percent of steam flow at rated power from the steam generator without exceeding 110 percent of the steam system design pressure. The pressurizer safety valve capacity is sized based on a complete loss of load with the plant initially operating at the maximum calculated turbine load. The pressurizer safety valves and steam generator safety valves are able to relieve sufficient steam to maintain the RCS pressure within 110 percent of the RCS design pressure.

A more complete discussion of overpressure protection can be found in Reference 1.

A loss of external load is classified as an ANS Condition II event, fault of moderate frequency. See Subsection 15.0.2.

A loss of external load event results in an NSSS transient that is less severe than a turbine trip event (see Subsection 15.2.3). Therefore, a detailed transient analysis is not presented for the loss of external load. The primary-side transient is caused by a decrease in the heat transfer capability from primary to secondary due to a rapid termination of steam flow to the turbine, accompanied by an automatic reduction of feedwater flow (should feed flow not be reduced, a larger heat sink should be available and the transient would be less severe). Termination of steam flow to the turbine following a loss of external load occurs due to automatic fast closure of the turbine control valves. Following a turbine trip event, termination of steam flow occurs via turbine stop valve closure. Therefore, the transient in primary pressure, temperature, and water volume will be less severe for the loss of external load than for the turbine trip due to a slightly slower loss of heat transfer capability.

The protection available to mitigate the consequences of a loss of external load is the same as that for a turbine trip, as listed in Table 15.0-6.

15.2.2.2 Analysis of Effects and Consequences

Method of Analysis

Refer to Subsection 15.2.3.2 for the method used to analyze the limiting transient (turbine trip) in this grouping of events. The results of the turbine trip event analysis are more severe than those expected for the loss of external load as discussed in Subsection 15.2.2.1.

Normal reactor control systems and engineered safety feature systems are not required to function. The emergency feedwater system may, however, be automatically actuated following a loss of main feedwater; this will further mitigate the effects of the transient.

The reactor protection system may be required to function following a complete loss of external load to terminate core heat input and prevent departure from nucleate boiling (DNB). Depending on the magnitude of the load loss, pressurizer safety valves and/or steam generator safety valves may be required to open to maintain system pressure below allowable limits. No single active failure will prevent operation of any system required to function. Refer to Reference 2 for a discussion of ATWT considerations.

15.2.2.3 Radiological Consequences

Loss of external load from full power would result in the operation of the steam dump system. This system keeps the main turbine generator operating to supply auxiliary electrical loads. Operation of the steam dump system results in bypassing steam to the condenser. If steam dumps are not available, steam generator safety and relief valves relieve to the atmosphere. Since no fuel damage is postulated for this transient the radiological releases will be less severe than those for the steamline break accident analyzed in Subsection 15.1.5.3.

15.2.2.4 Conclusions

Based on results obtained for the turbine trip event (Subsection 15.2.3) and considerations described in Subsection 15.2.2.1, the applicable acceptance criteria for a loss of external load event are met. The radiological consequences of this event are not limiting.

15.2.3 Turbine Trip

15.2.3.1 Identification of Causes and Accident Description

For a turbine trip event, the reactor would be tripped directly (unless below 10 percent power) by a signal derived from the turbine auto-stop oil pressure (Westinghouse turbine) and turbine stop valves. The turbine stop valves close rapidly on loss of trip-fluid pressure actuated by one of a number of turbine trip signals. Turbine trip initiation signals include:

1. low condenser vacuum,

2. low bearing oil pressure,

3. turbine thrust bearing failure,

4. turbine overspeed,

5. DE-H DC power failure, and

6. manual trip.

Upon initiation of stop valve closure, steam flow to the turbine stops abruptly. Sensors on the stop valves detect the turbine trip and enable the steam dump system and, if above 10 percent power, trip the reactor. The loss of steam flow results in a rapid rise in secondary system temperature and presssure with a resultant primary system transient as described in Subsection 15.2.2.1 for the loss of external load event. The turbine trip event is analyzed because it results in the most rapid reduction in steam flow.

The automatic steam dump system would normally accommodate the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly. If the turbine condenser is not available, the excess steam generation would be dumped to the atmosphere. Feedwater flow would be maintained by the emergency feedwater system to ensure adequate residual and decay heat removal capability. Should the steam dump system fail to operate, the steam generator safety valves may lift to provide pressure control. See Subsection 15.2.2.1 for a further discussion of the transient.

A turbine trip is classified as an ANS Condition II event, fault of moderate frequency (see Subsection 15.0.2).

The plant systems and equipment available to mitigate the consequences of a turbine trip are discussed in Subsection 15.0.9, and listed in Table 15.0-6.

15.2.3.2 Analysis of Effects and Consequences

Method of Analysis

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from full power without direct reactor trip primarily to show the adequacy of the pressure relieving devices and also to demonstrate core protection margins; that is, the turbine is assumed to trip without actuating all the sensors for reactor trip on the turbine stop valves and autostop oil pressure. The assumption delays reactor trip until conditions in the RCS result in a trip due to other signals. Thus, the analysis assumes a worst case transient. In addition, no credit is taken for steam dump. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for auxiliary feedwater to mitigate the consequences of the transient.

The turbine trip transients are analyzed by employing the detailed digital computer program LOFTRAN (Reference 3). The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level.

Major assumptions are summarized below:

1. Initial Operating Conditions

Initial reactor power, pressure, and RCS temperatures are assumed to be at their maximum values consistent with steady state full power operation including allowances for calibration and instrument errors.

2. Moderator and Doppler Coefficients of Reactivity

The turbine trip is analyzed with both a least negative moderator temperature coefficient and a large negative moderator temperature coefficient. The most negative Doppler power coefficient is used for all cases (see Figure 15.0-2).

3. Reactor Control

From the standpoint of the maximum pressures attained it is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.

4. Steam Release

No credit is taken for the operation of the steam dump system or steam generator power operated relief valves. The steam generator pressure rises to the safety valve setpoint where steam release through safety valves limits secondary steam pressure at the setpoint value.

5. Pressurizer Spray and Power-Operated Relief Valves

Two cases for both the minimum and maximum moderator feedback cases are analyzed:

a. Full credit is taken for the effect of pressurizer spray and power operated relief valves in reducing or limiting the coolant pressure. Safety valves are also available.

b. No credit is taken for the effect of pressurizer spray and power operated relief valves in reducing or limiting the coolant pressure. Safety valves are operable.

6. Feedwater Flow

Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for emergency feedwater flow since a stabilized plant condition will be reached before emergency feedwater initiation is normally assumed to occur. The emergency feedwater flow would remove core decay heat following plant stabilization.

7. Reactor Trip

Reactor trip is actuated by the first reactor protection system trip setpoint reached with no credit taken for the direct reactor trip on the turbine trip. Trip signals are expected due to high pressurizer pressure, low DNBR and high pressurizer water level.

Except as discussed above, normal reactor control system and engineered safety systems are not required to function. Several cases are presented in which pressurizer spray and power operated relief valves are assumed to operate, but the more limiting cases where these functions are not assumed are also presented.

The reactor protection system may be required to function following a turbine trip. Pressurizer safety valves and/or steam generator safety valves may be required to open to maintain system pressures below allowable limits. No single active failure will prevent operation of any system required to function. A discussion of ATWT considerations is presented in Reference 2.

<u>Results</u>

The transient responses for a turbine trip from full power operation are shown for four cases: two cases for minimum moderator feedback and two cases for maximum moderator feedback (Figures 15.2-1 through 15.2-8). For the minimum moderator feedback cases, the core has the least negative moderator coefficient of reactivity. For the maximum moderator feedback cases, the moderator temperature coefficient has its highest absolute value. The calculated sequence of events for the accident is shown in Table 15.2-1.

Figures 15.2-1 and 15.2-2 show the transient responses for the turbine trip with minimum moderator feedback, assuming full credit for the pressurizer spray and pressurizer power operated relief valves. No credit is taken for the steam dump. The reactor is tripped by the high pressurizer pressure trip signal. The minimum DNBR remains well above the limit value. The pressurizer safety valves are actuated, and maintain primary system pressure below 110 percent of the design value. The steam generator safety valves limit the secondary steam conditions to saturation of the safety valve setpoint.

Figures 15.2-3 and 15.2-4 show the responses for the total loss of steam load with maximum moderator feedback. All other plant parameters are the same as the above. The DNBR increases throughout the transient and never drops below its initial value. Pressurizer relief valves and steam generator safety valves prevent overpressurization in the primary and secondary systems, respectively. The pressurizer safety valves are not actuated for this case.

In the event that feedwater flow is not terminated at the time of turbine trip for this case, flow would continue under automatic control with the reactor at a reduced power. The operator would take action to terminate the transient and bring the plant to a stabilized condition.

If no action were taken by the operator, the reduced power operation would continue until the condenser hotwell was emptied. A low-low steam generator water level reactor trip would be generated along with emergency feedwater initiation signals. Emergency feedwater would then be used to remove decay heat.

The results would be less severe than those presented in Subsection 15.2.7, loss of normal feedwater flow.

The turbine trip accident was also studied assuming the plant to be initially operating at full power with no credit taken for the pressurizer spray, pressurizer power-operated relief valves, or steam dump. The reactor is tripped on the high pressurizer pressure signal. Figures 15.2-5 and 15.2-6 show the transients with minimum moderator feedback. The neutron flux remains essentially constant at full power until the reactor is tripped. The DNBR increases throughout the transient. In this case the pressurizer safety valves are actuated, and maintain system presssure below 110 percent of the design value.

Figures 15.2-7 and 15.2-8 are the transients with maximum moderator feedback with the other assumptions being the same as in the preceding case. Again, the DNBR increases throughout the transient and the pressurizer safety valves are actuated to limit primary pressure.

Following the reactor trip, the plant will approach a stabilized condition at hot standby; normal plant operating procedures may then be followed. The operating procedures would call for operator action to control RCS boron concentration and pressurizer level using the CVCS, and to maintain steam generator level through control of the main or auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of thirty minutes following reactor trip.

Reference 1 presents additional results of analysis for a complete loss of heat sink including loss of main feedwater. This analysis shows the overpressure protection that is afforded by the pressurizer and steam generator safety valves.

15.2.3.3 Radiological Consequences

The radiological consequences resulting from atmospheric steam dump will be less severe than the steamline break event analyzed in Subsection 15.1.5.3 since no fuel damage is postulated to occur.

15.2.3.4 Conclusions

Results of the analyses, including those in Reference 1, show that the plant design is such that a turbine trip, without a direct or immediate reactor trip, presents no hazard to the integrity of the RCS or the main steam system. Pressure relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits.

The DNBR remains above the limit value for all cases analyzed; thus, the DNB design basis as described in Section 4.4 is met. The above analysis demonstrates the ability of the NSSS to safely withstand a full load rejection. The radiological consequences of this event will be less than the steamline break event analyzed in Subsection 15.1.5.3.

15.2.4 Inadvertent Closure of Main Steam Isolation Valves

The inadvertent closure of the main steam isolation valves would result in a turbine trip and other consequences as discussed in Subsection 15.2.5.

15.2.5 Loss of Condenser Vacuum And Other Events Resulting In Turbine Trip

Loss of condenser vacuum is one of the events that can cause a turbine trip. Turbine trip initiating events are described in Subsection 15.2.3. A loss of condenser vacuum would preclude the use of steam dump to the condenser; however, since steam dump is assumed not to be available in the turbine trip analysis, no additional adverse effects would result if the turbine trip were caused by loss of condenser vacuum. Therefore, the analysis results and conclusions contained in Subsection 15.2.3 apply to loss of condenser vacuum. In addition, analyses for the other possible causes of a turbine trip, as listed in Subsection 15.2.3.1, are covered by Subsection 15.2.3. Possible overfrequency effects due to a turbine overspeed condition are discussed in Subsection 15.2.2.1 and are not a concern for this type of event.

15.2.6 Loss of Non Emergency AC Power To The Plant Auxiliaries (Loss of Offsite Power)

15.2.6.1 Identification of Causes and Accident Description

A complete loss of nonemergency AC power may result in the loss of all power to the station auxiliaries, i.e., the reactor coolant pumps, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the plant, or by a loss of the onsite AC distribution system.

This transient is more severe than the turbine trip event analyzed in Subsection 15.2.3 because for this case the decrease in heat removal by the secondary is accompanied by a flow coastdown which further reduces the capacity of the primary coolant to remove heat from the core. The reactor will trip due to: 1) turbine trip; 2) upon reaching one of the trip setpoints in the primary and secondary systems as a result of the flow coastdown and decrease in secondary heat removal; or 3) due to loss of power to the control rod drive mechanisms as a result of the loss of power to the plant.

Following a loss of AC power with turbine and reactor trips, the sequence described below will occur:

1. Plant vital instruments are supplied from emergency DC power sources.

2. As the steam system pressure rises following the trip, the steam generator power-operated relief valves may be automatically opened to the atmosphere. Steam dump to the condenser is assumed not to be available. If the steam flow rate through the power relief valves is not available, the steam generator self actuated safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.

3. As the no-load temperature is approached, the steam generator power operated relief valves (or the safety valves, if the power-operated relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot shutdown condition.

4. The emergency diesel generators, started on loss of voltage on the plant emergency buses, begin to supply plant vital loads.

The start-up feedwater system (SFWS) is started automatically on any of the following:

1. Low level in any steam generator.

2. Low feedwater flow to any steam generator.

The SFWS is a control grade system and is used primarily for low power, plant heatups, plant cooldown, and start-up operations. The SFWS consisting of a single pump, designed to provide an even flow split to the steam generators and is designed to minimize the actuation of the EFWS.

The emergency feedwater system (EFWS) is started automatically on any of the following:

1. Any safety injection signal.

2. Low level in two or more steam generators in coincidence with low start-up feedwater flow.

3. Low-low level in any steam generator.

The EFWS consists of 4 emergency feedwater pumps. For diversity, a motor driven and turbine driven pump are headered together to feed each pair of steam generators.

The motor driven emergency feedwater pumps are supplied by power from the ESF buses. The turbine-driven emergency feedwater pumps are driven by steam from the secondary system and exhaust to the atmosphere. Both types of pumps are designed to start and supply rated flow within one minute of the initiating signal. The emergency feedwater pumps take suction from the emergency feedwater storage tank for delivery to the steam generators.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops.

A loss of non-emergency AC power to the station auxiliaries is classified as an ANS Condition II event, fault of moderate frequency. See Subsection 15.0.2 for a discussion of Condition II events.

A loss of non-emergency AC power event is a more limiting event than the turbine-trip-initiated decrease in secondary heat removal without loss of AC power, which was analyzed in Subsection 15.2.3. However, a loss of AC power to the plant auxiliaries as postulated above could also result in a loss of normal feedwater if the condensate pumps lose the power supply.

Following the reactor coolant pump coastdown caused by the loss of AC power, the natural circulation capability of the RCS will remove residual and decay heat from the core, aided by emergency feedwater in the secondary system. An analysis is presented here to show that the natural circulation flow in the RCS following a loss of AC power event is sufficient to remove residual heat from the core.

The plant systems and equipment available to mitigate the consequences of a loss of AC power event are discussed in Subsection 15.0.9, and listed in Table 15.0-6.

15.2.6.2 Analysis of Effects and Consequences

Method of Analysis

A detailed analysis using the LOFTRAN Code (Reference 3) is performed to obtain the plant transient following a station blackout. The simulation describes the plant thermal kinetics, RCS including the natural circulation, pressurizer, steam generators and feedwater system. The digital program computes pertinent variables including the steam generator level, pressurizer water level, and reactor coolant average temperature.

The assumptions used in the analysis are as follows:

1. The plant is initially operating at 102 percent of the nominal NSSS power rating.

2. A conservative core residual heat generation based upon long term operation at the initial power level preceding the trip.

3. A heat transfer coefficient in the steam generator associated with RCS natural circulation following the reactor coolant pump coast-down is assumed.

4. Reactor trip occurs on steam generator low level. No credit is taken for immediate release of the control rod drive mechanisms caused by a loss of offsite power.

5. Emergency feedwater is delivered to all four steam generators.

6. The worst single failure in the EFWS is assumed.

7. No credit is taken for the SFWS in mitigating the transient.

8. Secondary system steam relief is achieved through the steam generator safety valves.

9. The initial reactor coolant average temperature is 4°F higher than the nominal value.

The assumptions used in the analysis are similar to the loss of normal feedwater flow incident (Subsection 15.2.7) except that power is assumed to be lost to the reactor coolant pumps at the time of reactor trip.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.4.

Results

The transient response of the RCS following a loss of AC power is shown in Figures 15.2-9 and 15.2-10.

The first few seconds after the loss of power to the reactor coolant pumps will closely resemble a simulation of the complete loss of flow incident (see Subsection 15.3.2), i.e., core damage due to rapidly increasing core temperatures is prevented by promptly tripping the reactor. After the reactor trip, stored and residual decay heat must be removed to prevent damage to either the RCS or the core.

The LOFTRAN results show that the natural circulation flow available is sufficient to provide adequate core decay heat removal following reactor trip and RCP coastdown.

The calculated sequence of events for this accident are listed in Table 15.2-1.

15.2.6.3 Radiological Consequences

A loss of nonessential AC power to plant auxiliaries would result in a turbine and reactor trip and loss of condenser vacuum. Heat removal from the secondary system would occur through the steam generator power relief valves or safety valves. The parameters to be used in calculation of the radiological consequences of the loss of AC power analysis are summarized in Table 15.2-2. Since no fuel damage is postulated to occur from this transient, the radiological consequences will be less severe than the steamline break event analyzed in Subsection 15.1.5.3.

15.2.6.4 Conclusions

Analysis of the natural circulation capability of the RCS has demonstrated that sufficient heat removal capability exists following reactor coolant pump coastdown to prevent fuel or clad damage. The radiological consequences of this event are given in Table 15.2-2.

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Identification of Causes and Accident Description

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite AC power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If an alternative supply of feedwater were not supplied to the plant, core residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer would occur, resulting in a substantial loss of water from the RCS. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables never approach a DNB condition.

The following occur upon loss of normal feedwater (assuming main feedwater pump failures or valve malfunctions):

1. As the steam system pressure rises following the trip, the steam generator power-operated relief valves are automatically opened to the atmosphere. Steam dump to the condenser is assumed not to be available.

If the steam flow through the power relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.

2. As the no-load temperature is approached, the steam generator power-operated relief valves (or the safety valves, if the power operated relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot shutdown condition.

A loss of normal feedwater is classified as an ANS Condition II event, fault of moderate frequency. See Subsection 15.0.2 for a discussion of Condition II events.

Reactor trip on low water level in any steam generator provides protection for a loss of normal feedwater.

The startup feedwater system (SFWS) and emergency feedwater system (EFWS) are started automatically as discussed in Subsection 15.2.6.1. The motor-driven emergency feedwater pumps are supplied by power from the ESF buses. The turbine-driven emergency feedwater pumps are driven by steam from the secondary system and exhaust to the atmosphere. The pumps take suction directly from the emergency feedwater storage tank for delivery to the steam generators.

An analysis of the system transient is presented below to show that following a loss of normal feedwater, the EFWS is capable of removing the stored and residual heat, thus preventing either overpressurization of the RCS or loss of water from the reactor core, and returning the plant to a safe condition.

15.2.7.2 Analysis of Effects and Consequences

Method of Analysis

A detailed analysis using the LOFTRAN Code, Reference 3, is performed in order to obtain the plant transient following a loss of normal feedwater. The simulation describes the plant thermal kinetics, RCS, pressurizer, steam generator and feedwater system. The digital program computes pertinent variables including the steam generator level, pressurizer water level, and reactor coolant average temperature.

Assumptions made in the analysis are:

1. The plant is initially operating at 102 percent of the nominal NSSS power rating.

2. A conservative core residual heat generation based upon long term operation at the initial power level preceding the trip.

3. Reactor trip occurs on steam generator low level.

4. The worst single failure in the EFWS occurs.

5. Emergency feedwater is delivered to all four steam generators.

6. Secondary system steam relief is achieved through the steam generator safety valves.

7. The initial reactor coolant average temperature is 4°F higher than the nominal value.

8. No credit is taken for the SFWS in mitigating the transient.

The loss of normal feedwater analysis is performed to demonstrate the adequacy of the reactor protection and engineered safeguards system (e.g., the EFWS) in removing long term decay heat and preventing excessive heatup of the RCS with possible resultant RCS overpressurization or loss of RCS water.

As such, the assumptions used in this analysis are designed to minimize the energy removal capability of the system and to maximize the possibility of water relief from the coolant system by maximizing the coolant system expansion, as noted in the assumptions listed above.

For the loss of normal feedwater transient, the reactor coolant volumetric flow remains at its normal value, and the reactor trips via the low steam generator level trip. The reactor coolant pumps may be manually tripped at some later time to reduce heat addition to the RCS.

The assumptions used in the analysis are similar to the loss of AC power incident (Subsection 15.2.6) except that the reactor coolant pumps are assumed to continue to operate.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.4.

Plant systems and equipment which are available to mitigate effects of a loss of normal feedwater accident are discussed in Subsection 15.0.9 and listed in Table 15.0.6. Normal reactor control systems are not required to function. The reactor protection system is required to function following a loss of normal feedwater as analyzed here. The EFWS is required to deliver a minimum feedwater flow rate. No single active failure will prevent operation of any system required to function. A discussion of ATWT considerations is presented in Reference 2.

Results

Figures 15.2-11 and 15.2-12 show the significant plant parameters following a loss of normal feedwater.

Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to the reduction of steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. One minute following the initiation of the EFWS, the emergency feedwater pumps are automatically started and delivering full flow reducing the rate of water level decrease.

The capacity of the emergency feedwater pumps are such that the water level in the steam generators being fed does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the RCS relief or safety valves. Figures 15.2-11 and 15.2-12 show that at no time is there water relief from the pressurizer.

The calculated sequence of events for this accident is listed in Table 15.2-1.

As shown in Figures 15.2-11 and 15.2-12, the plant will slowly approach a stabilized condition at hot standby with emergency feedwater removing decay heat. The plant may be maintained at hot standby or further cooled through manual control of the auxiliary feed flow. The operating procedures would also call for operator action to control RCS boron concentration and pressurizer level using the CVCS and to maintain steam generator level through control of the emergency feedwater system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of thirty minutes following reactor trip.

15.2.7.3 Radiological Consequences

If steam dump to the condenser is assumed to be lost, heat removal from the secondary system would occur through the steam generator power relief valves or safety valves. Since no fuel damage is postulated to occur, radiological consequences resulting from this transient would be less severe than the steamline break accident analyzed in Subsection 15.1.5.3.

15.2.7.4 Conclusions

Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the RCS, or the steam system since the emergency feedwater capacity is such that reactor coolant system does not overpressurize and water is not relieved from the pressurizer relief or safety valves. The radiological consequences of this event would be less severe than the steamline break accident analyzed in Subsection 15.1.5.3.

15.2.8 Feedwater System Pipe Break

15.2.8.1 Identification of Causes and Accident Description

A major feedwater line rupture is defined as a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell side fluid inventory in the steam generators. If the break is postulated in a feedline between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. (A break upstream of the feedline check valve would affect the Nuclear Steam Supply System only as a loss of feedwater. This case is covered by the evaluation in Subsection 15.2.7).

Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either a RCS cooldown (by excessive energy discharge through the break) or RCS heatup.

Potential RCS cooldown resulting from a secondary pipe rupture is evaluated in Subsection 15.1.5. Therefore, only the RCS heatup effects are evaluated for a feedwater line rupture.

A feedwater line rupture reduces the ability to remove heat generated by the core from the RCS for the following reasons:

1. Feedwater flow to the steam generators is reduced. Since feedwater is subcooled, its loss may cause reactor coolant temperatures to increase prior to reactor trip.

2. Fluid in the steam generator may be discharged through the break, and would then not be available for decay heat removal after trip.

3. The break may be large enough to prevent the addition of any main feedwater after trip.

An emergency feedwater system (EFWS) is provided to assure that adequate feedwater will be available such that:

1. No substantial overpressurization of the RCS shall occur.

2. Sufficient liquid in the RCS shall be maintained in order to provide adequate decay heat removal.

A major feedwater line rupture is classified as an ANS Condition IV event. See Subsection 15.0.2 for a discussion of Condition IV events.

The severity of the feedwater line rupture transient depends on a number of system parameters including break size, initial reactor power, and credit taken for the functioning of various control and safety systems. A number of cases of feedwater line break have been analyzed. Based on these analyses, it has been shown that the most limiting feedwater line ruptures is a double ended rupture of the largest feedwater line.

The following provides the necessary protection for a main feedwater rupture:

1. A reactor trip on any of the following conditions:

    a. High pressurizer pressure,
    b. Low steam generator water level in any steam generator,
    c. Safety injection signals from any of the following:
        1) low steamline pressure in any loop,
        2) high containment pressure (Hi-1).

2. An emergency feedwater system to provide an assured source of feedwater to the steam generators for decay heat removal.

15.2.8.2 Analysis of Effects and Consequences

Method of Analysis

A detailed analysis using the LOFTRAN Code (Reference 3) is performed in order to determine the plant transient following a feedwater line rupture. The code describes the plant thermal kinetics, RCS including natural circulation, pressurizer, steam generators, and feedwater system, and computes pertinent variables including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

The cases analyzed assume a double ended rupture of the largest feedwater pipe at full power. Major assumptions made in the analyses are as follows:

1. The plant is initially operating at 102 percent of the nominal NSSS power rating.

2. Initial reactor coolant average temperature is 4.0°F above the nominal value, and the initial pressurizer pressure is 30 psi above its nominal value.

3. Initial pressurizer level is at the nominal programmed value plus 5 percent (error); initial steam generator water level is at the nominal value plus 5% in the faulted steam generator and at the nominal value minus 5% in the intact steam generators.

4. No credit is taken for the high pressurizer pressure reactor trip.

5. Main feedwater flow to all steam generators is assumed to be lost at the time the break occurs (all main feedwater spills out through the break).

6. The full double-ended break area is assumed.

7. A conservative feedwater line break discharge quality is assumed prior to the time the reactor trip occurs, thereby maximizing the time the trip setpoint is reached. After the trip occurs, a saturated liquid discharge is assumed until all the water inventory is discharged from the affected steam generator. This minimizes the heat removal capability of the affected steam generator.

8. Reactor trip is assumed to be actuated when the low level trip setpoint in the ruptured steam generator is reached.

9. The EFWS is actuated by the low steam generator water level signal. The EFWS is assumed to supply a total of 750 gpm to two intact steam generators, including allowance for spillage through the main feedwater line break. A 60-second delay was assumed following the low-low level signal to allow time for startup of the standby diesel generators and the emergency feed pumps. An additional 240 seconds was assumed before the feedwater lines were purged and the relatively cold (120°F) emergency feedwater entered the unaffected steam generators.

10. No credit is taken for heat energy deposited in reactor coolant system metal during the reactor coolant system heatup.

13. No credit is taken for charging or letdown.

14. Steam generator heat transfer area is assumed to decrease as the shell-side liquid inventory decreases.

15. Conservative core residual heat generation is assumed based upon long-term operation at the initial power level preceding the trip.

16. No credit is taken for the following potential protection logic signals to mitigate the consequences of the accident:

    a. High pressurizer pressure.
    b. High pressurizer level.
    c. High containment pressure.

Receipt of a low-low steam generator water level in at least one steam generator initiates start up of the EFWS. Similiarly, receipt of a low steam line pressure signal in at least one steam line initiates a steam line isolation signal which closes the main steam line isolation valves in all steam lines.

Emergency operating procedures following a main feed line rupture require the following actions to be taken:

1. Isolate feedwater flow spilling out the break from the ruptured feedwater line and align system so level in intact steam generators recovers.

2. Stop high head safety injection pumps if reactor coolant pressure is stable or increasing, pressurizer level is greater than fifty percent of span, and steam generator narrow range indication exists in at least one steam generator.

Isolating feedwater flow through the break allows additional emergency feedwater flow to be diverted to the intact steam generators.

Subsequent to recovery of level in the intact steam generators, the plant operating procedures will be followed in cooling the plant to hot shutdown conditions.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.4.

The reactor protection system is required to function following a feedwater line rupture as analyzed here. No single active failure will prevent operation of this system.

The engineered safety systems assumed to function are the emergency feedwater system and the safety injection system. One headered turbine and motor driven pump pair is assumed unavailable due to the loss of steam supply to the turbine driven pump and failure of the motor driven pump. The remaining EFW pumps deliver 750 gpm to the two intact steam generators. Only one train of safety injection has been assumed to be available which is consistent with the single failure criterion and minimum safeguards analysis. (The SI pumps have a 1800 psi cutoff head).

Following the trip of the reactor coolant pumps, there will be a flow coastdown until flow in the loops reaches the natural circulation value. The natural circulation capability of the RCS has been shown in Subsection 15.2.6, for the loss of AC power transient, to be sufficient to remove core decay heat following reactor trip. Pump coastdown characteristics are demonstrated in Subsections 15.3.1 and 15.3.2 for single and multiple reactor coolant pump trips, respectively.

A detailed description and analysis of the safety injection system is provided in Section 6.3 of RESAR-SP/90 PDA module 7, "Structural/Equipment Design". The EFWS is described in Section 6.6 of RESAR-SP/90 PDA module 1, "Primary Sides Safeguards".

Results

Calculated plant parameters following a major feedwater line rupture are shown in Figures 15.2-13 through 15.2-24. Results for the case with offsite power available are presented in Figures 15.2-13 through 15.2-18. Results for the case where offsite power is lost are presented in Figures 15.2-19 through 15.2-24. The calculated sequence of events for both cases analyzed are listed in Table 15.2-1.

The system response following the feedwater line rupture is similar for both cases analyzed. Results presented in Figures 15.2-14 and 15.2-17 (with offsite power available) and Figures 15.2-20 and 15.2-23 (without offsite power) show that pressures in the reactor coolant system and main steam system remain below 110 percent of the respective design pressures. Pressurizer pressure increases until reactor trip occurs on low steam generator water level. Pressure then decreases, due to the loss of heat input, until the time at which the mass inventory in the intact steam generators is not sufficient to remove the core decay heat, and until steamline isolation and safety injection actuation occur. The pressurizer relief valves open to maintain RCS pressure at an acceptable value. Addition of the safety injection flow aids in cooling down the primary and helps to ensure that sufficient fluid exists to keep the core covered with water.

Reactor coolant system pressure will be maintained at the safety valve setpoint until safety injection flow is terminated. The reactor core remains covered with water throughout the transient, as water relief due to thermal expansion is limited by the heat removal capability of the emergency feedwater system.

The major difference between the two cases analyzed can be seen in the plots of hot and cold-leg temperatures, Figures 15.2-15 and 15.2-16 (with offsite power available) and Figures 15.2-21 and 15.2-22 (without offsite power). It is apparent that for the initial transient (300 seconds), the case without offsite power results in higher temperatures in the hot leg. For longer times, however, the case with offsite power results in a more severe rise in temperature due to the addition of pump heat.

The pressurizer fills more rapidly for the case with power due to the increased coolant expansion resulting from the pump heat addition; hence, more water is relieved for the cases with power. As previously stated, however, the core remains covered with water for all cases.

15.2.8.3 Radiological Consequences

The feedwater line break with the most significant consequences would be one that occurred inside the containment between a steam generator and the feedwater check valve. In this case, the contents of the steam generator would be released to the containment. Since no fuel failures are postulated, the radioactivity released would be less than that for the steamline break, as analyzed in Subsection 15.1.5.3. Furthermore, automatic isolation of the containment would further reduce any radiological consequences from this postulated accident.

15.2.8.4 Conclusions

Results of the analyses show that for the postulated feedwater line rupture, the assumed EFWS capacity is adequate to remove decay heat, to prevent overpressurizing the RCS, and to prevent uncovering the reactor core. Radiological doses from the postulated feedwater line rupture would be less than those previously presented for the postulated steam line break.

15.2.9 References

1. M. A. Mangan, "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, October 1971.

2. "Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August 1974.

3. T. W. T. Burnett et al., "LOFTRAN Code Description," WCAP-7907, June 1972. Also supplementary information in letter from T. M. Anderson, NS-TMS-1802, May 26, 1978 and NS-TMA-1824, June 16, 1978.

4. M. S. Balwin et al., "An Evaluation of Loss of Flow Accidents caused by Power System Frequency Transients in Westinghouse PWR's," WCAP-8424, Revision 1, June 1975.

TABLE 15.2-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

ACCIDENT	EVENT	TIME (sec) N Loop
Turbine Trip		
1. With pressurizer control (minimum moderator feedback)	Turbine trip, loss of main feedwater flow	0.0
	High pressurizer pressure reactor trip point reached	14.0
	Initiation of steam release from steam generator safety valves	10.0
	Rods begin to drop	16.0
	Minimum DNBR occurs	17.0
	Peak pressurizer pressure occurs	15.5

*DNBR does not decrease below its initial value.

TABLE 15.2-1 (Continued)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

ACCIDENT	EVENT	TIME (sec) N Loop
2. With pressurizer control (maximum moderator feedback)	Turbine trip, loss of main feed flow	0.0
	Initiation of steam release from steam generator safety valves	10.0
	Minimum DNBR occurs	*
	Peak pressurizer pressure occurs	9.0
3. Without pressurizer control (minimum moderator feedback)	Turbine trip, loss of main feed flow	0.0

*DNBR does not decrease below its initial value.

TABLE 15.2-1 (Continued)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

ACCIDENT	EVENT	TIME (sec) N Loop
	High pressurizer pressure reactor trip point reached	6.5
	Rods begin to drop	8.5
	Initiation of steam release from steam generator safety valves	10.0
	Minimum DNBR occurs	*
	Peak pressurizer pressure occurs	11.0
4. Without pressurizer control (maximum moderator feedback)	Turbine trip, loss of main feed flow	0.0
	High pressurizer pressure reactor trip point reached	6.4

*DNBR does not decrease below its initial value.

TABLE 15.2-1 (Continued)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

ACCIDENT	EVENT	TIME (sec) N Loop
	Rods begin to drop	8.4
	Initiation of steam release from steam generator safety valves	10.0
	Minimum DNBR occurs	*
	Peak pressurizer pressure occurs	10.5
Loss of Non-Emergency AC Power	Main feedwater flow stops	0.0
	Low steam generator water level trip	84.0
	Rods begin to drop	86.0
	Reactor coolant pumps begin to coastdown	86.0

*DNBR does not decrease below its initial value.

TABLE 15.2-1 (Continued)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

ACCIDENT	EVENT	TIME (sec) N Loop
	Two steam generators begin to receive emergency feedwater (EFW) from one emergency feedwater pump	184.0
	Main feedwater lines purged, delivery of cold EFW	664.0
	Core decay heat decreases to emergency feedwater heat removal capacity	1000
Loss of Normal Feed-water flow	Main feedwater flow stops	0.0
	Low steam generator water level trip	84.0
	Rods begin to drop	86.0
	Two steam generators begin to receive emergency feedwater (EFW) (EFW) from two emergency pumps	184.0

TABLE 15.2-1 (Continued)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

ACCIDENT	EVENT	TIME (sec) N Loop
	Main feedwater lines purged; delivery of cold EFW	664.0
	Core decay heat decreases to emergency feedwater heat removal capacity	<1800
Feedwater System Pipe Break		
1. With Offsite Power Available	Main feedline rupture occurs	10
	Low steam generator level reactor trip setpoint reached in ruptured steam generator	18.5
	Rods begin to drip	20.5
	Emergency feedwater (EFW) is delivered to intact steam generators	118.5

TABLE 15.2-1 (Continued)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

ACCIDENT	EVENT	TIME (sec) N Loop
	Low steamline pressure setpoint reached in ruptured steam generator	286.1
	All main steamline isolation valves close	293.1
	Steam generator safety valve setpoint reached intact steam generators	1550
	Core decay heat plus pump heat decreases to emergency feedwater heat removal capacity	<1800
2. Without Offsite Power	Main feedline rupture occurs	10
	Low steam generator level reactor trip setpoint reached in ruptured steam generator	18.50

TABLE 15.2-1 (Continued)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

ACCIDENT	EVENT	TIME (sec) N Loop
	Rods begin to drop, power lost to the reactor coolant pumps	20.5
	Low steamline pressure setpoint reached in ruptured steam generator	174.0
	All main steamline isolocation valves close	181.0
	Emergency feedwater (EFW) is delivered to intact steam generators	118.5
	Steam generator safety valve setpoint reached in intact steam generators	2984
	Core decay heat decreases to emergency feedwater heat removal capacity	<4000

TABLE 15.2-2

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LOSS OF NONEMERGENCY AC POWER
(Sheet 1 of 3)

I. Source Data	
a. Core power level, MWt	3876
b. Total steam generator tube leakage, gpm	1
c. Reactor coolant iodine activity:	
1. Accident Initiated Spike	Initial activity equal to the D.E. of 1.0 μ Ci/gm of I-131 with an iodine spike that increases the rate of iodine release into the reactor coolant by a factor of 500. See Tables 15A-5 and 15A-6.
2. Pre-Accident Spike	An assumed pre-accident iodine spike, which has resulted in the dose equivalent of 60 μCi/gm of I-131 in the reactor coolant. See Table 15A.5.
d. Reactor coolant noble gas activity	Based on one percent defective fuel. See Table 15A-7.
e. Secondary system initial activity	Dose equivalent of 0.1 μCi/gm of I-131.
f. Secondary coolant mass (4 generators), grams	2.0×10^8
g. Reactor coolant mass, grams	3.3×10^8
h. Offsite power	Lost after Trip
i. Primary-to-secondary leakage duration	8 hours
j. Species of iodine	100 percent elemental
II. Atmospheric Dispersion Factors	See Table 15A-2

TABLE 15.2-2

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LOSS OF NONEMERGENCY AC POWER
(Sheet 2 of 3)

III. Activity Release Data

a. Primary to secondary leak rate (gpm)*	1.0
b. Steam Released	
0 - 2 hr (lb)	675,000
2 - 8 hr	1,500,000
c. Iodine partition factor	100

IV. Activity Released to the Environment

a. Accident Initiated Spike

Isotope	0-2 h (Ci)	2-8 h (Ci)
I-131	0.22	1.64
I-132	0.25	2.2
I-133	0.34	2.9
I-134	0.07	0.3
I-135	0.19	1.8

b. Pre-Accident Spike

I-131	0.28	1.58
I-132	0.21	0.36
I-133	0.41	2.0
I-134	0.03	0.01
I-135	0.21	0.77

c. Noble gases - both cases

Xe-131m	0.5	2.1
Xe-133m	5.5	16.0
Xe-133	87.0	254.0
Xe-135m	0.03	1.0×10^{-4}
Xe-135	2.0	5.0
Xe-138	0.04	3×10^{-4}
Kr-85m	0.4	0.9
Kr-85	1.8	7.0
Kr-87	0.2	0.1
Kr-88	0.6	1.1

*Based on water at 590°F, 2250 psia

TABLE 15.2-2

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LOSS OF NONEMERGENCY AC POWER
(Sheet 3 of 3)

RADIOLOGICAL CONSEQUENCES

Case 1. Accident Initiated Iodine Spike	
Exclusion area boundary (0-2 h) Thyroid (rem)	0.027
Low-population zone outer boundary (8 h) Thyroid (rem)	0.05
Case 2. Pre-Accident Iodine Spike	
Exclusion area boundary (0-2 h) Thyroid (rem)	0.034
Low-population zone outer boundary (8 h) Thyroid (rem)	0.047
Both Cases. Whole Body Gamma (Rem)	
Exclusion area boundary (0-2 h)	2.7×10^{-4}
Low-population zone outer boundary (8 h)	2.2×10^{-4}

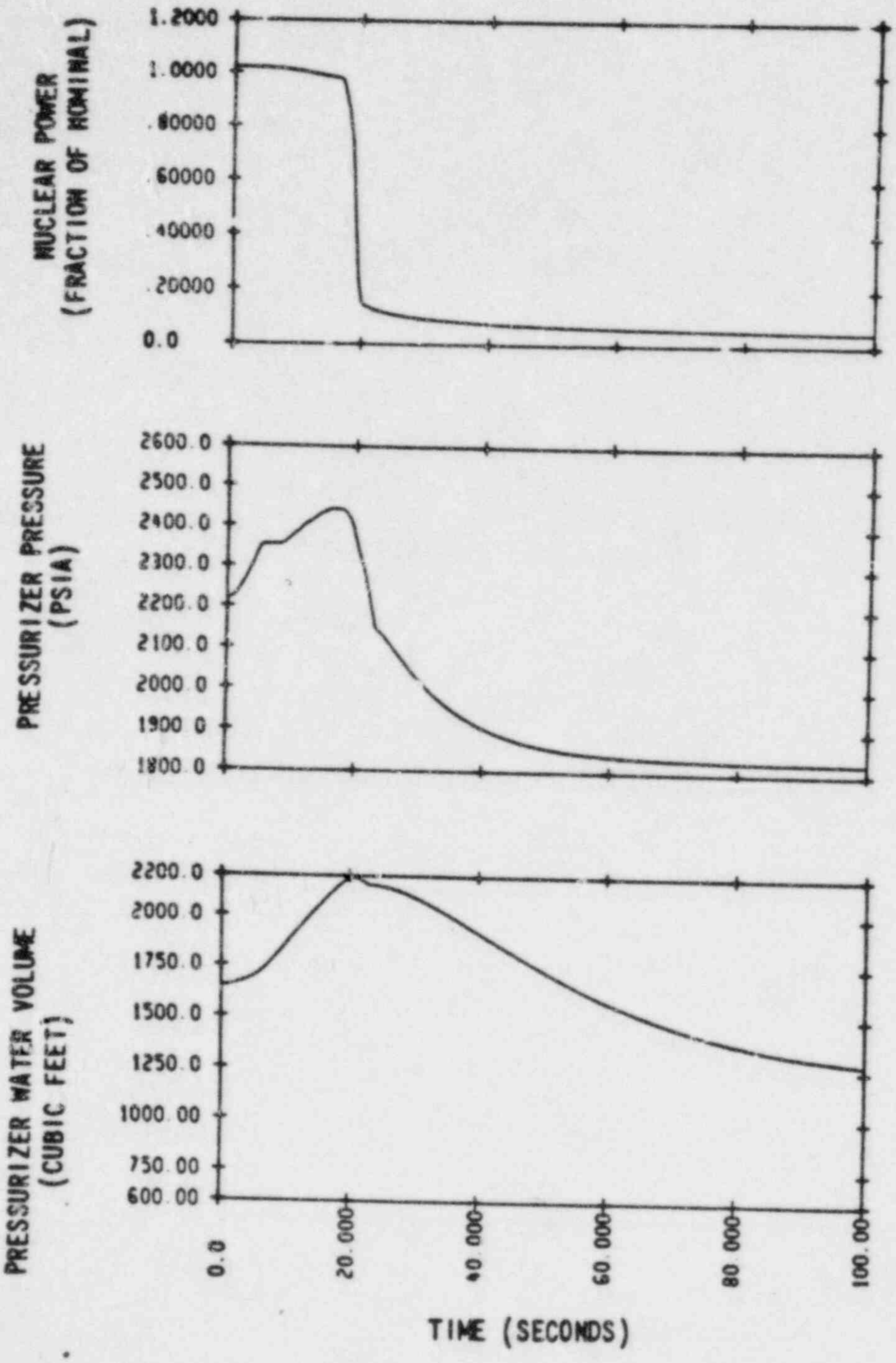

Figure 15.2-1.

Turbine Trip Accident With Pressurizer Spray and Power-Operated Relief Valves, Minimum Moderator Feedback

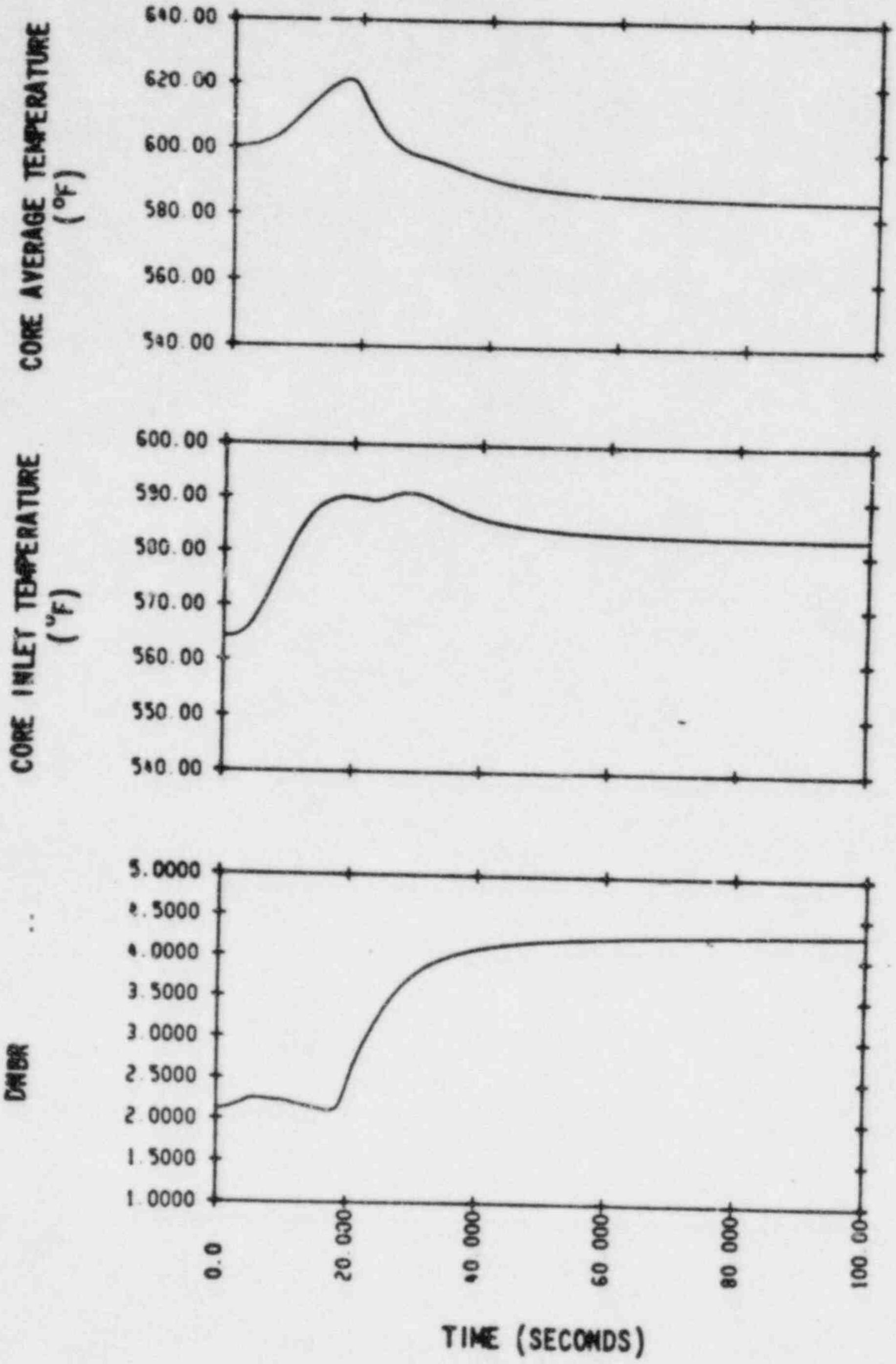

Figure 15.2-2.

Turbine Trip Accident With Pressurizer Spray and Power-Operated Relief Valves, Minimum Moderator Feedback

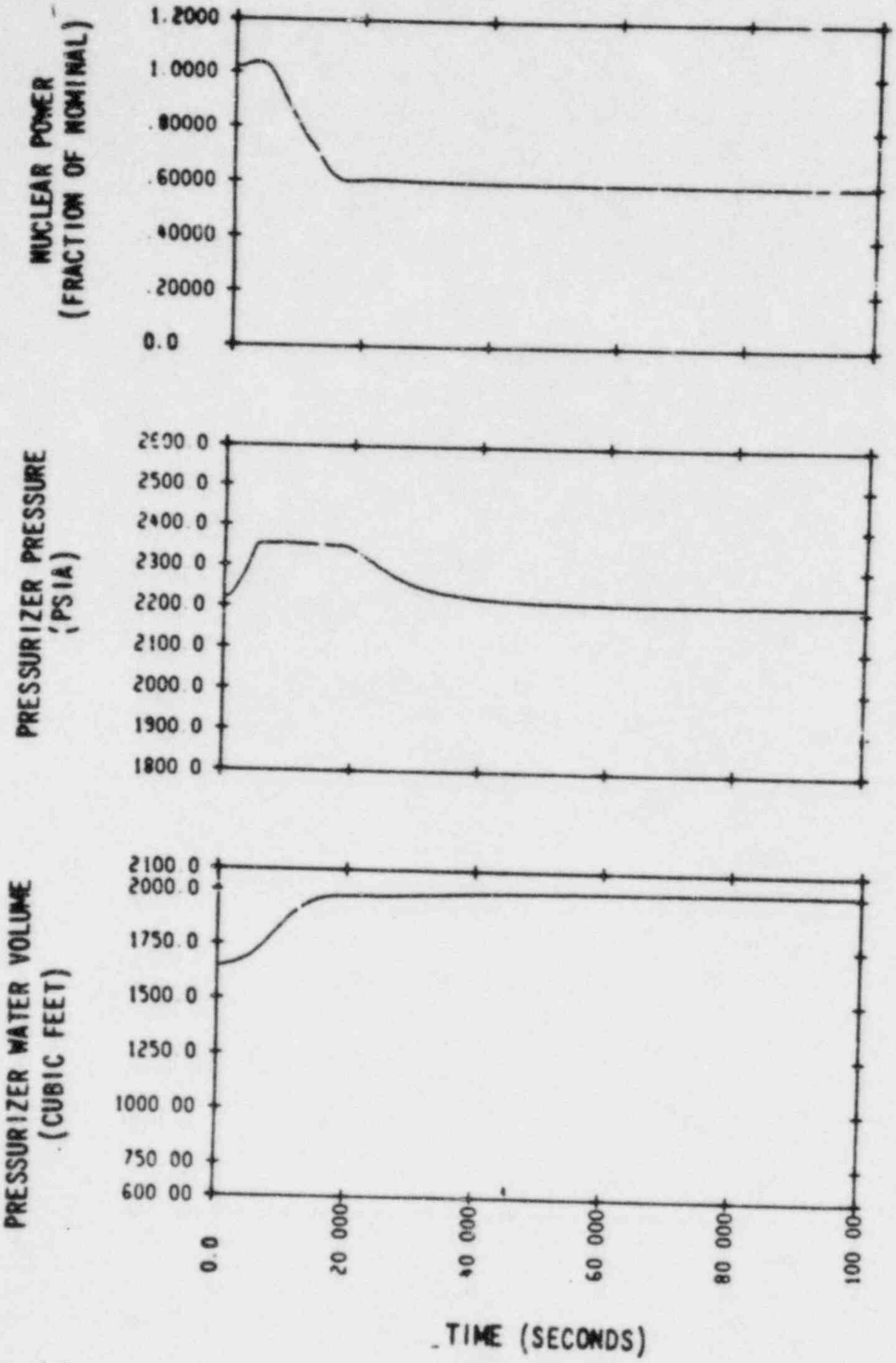

Figure 15.2-3.

Turbine Trip Accident With Pressurizer Spray and Power-Operated Relief Valves, Maximum Moderator Feedback

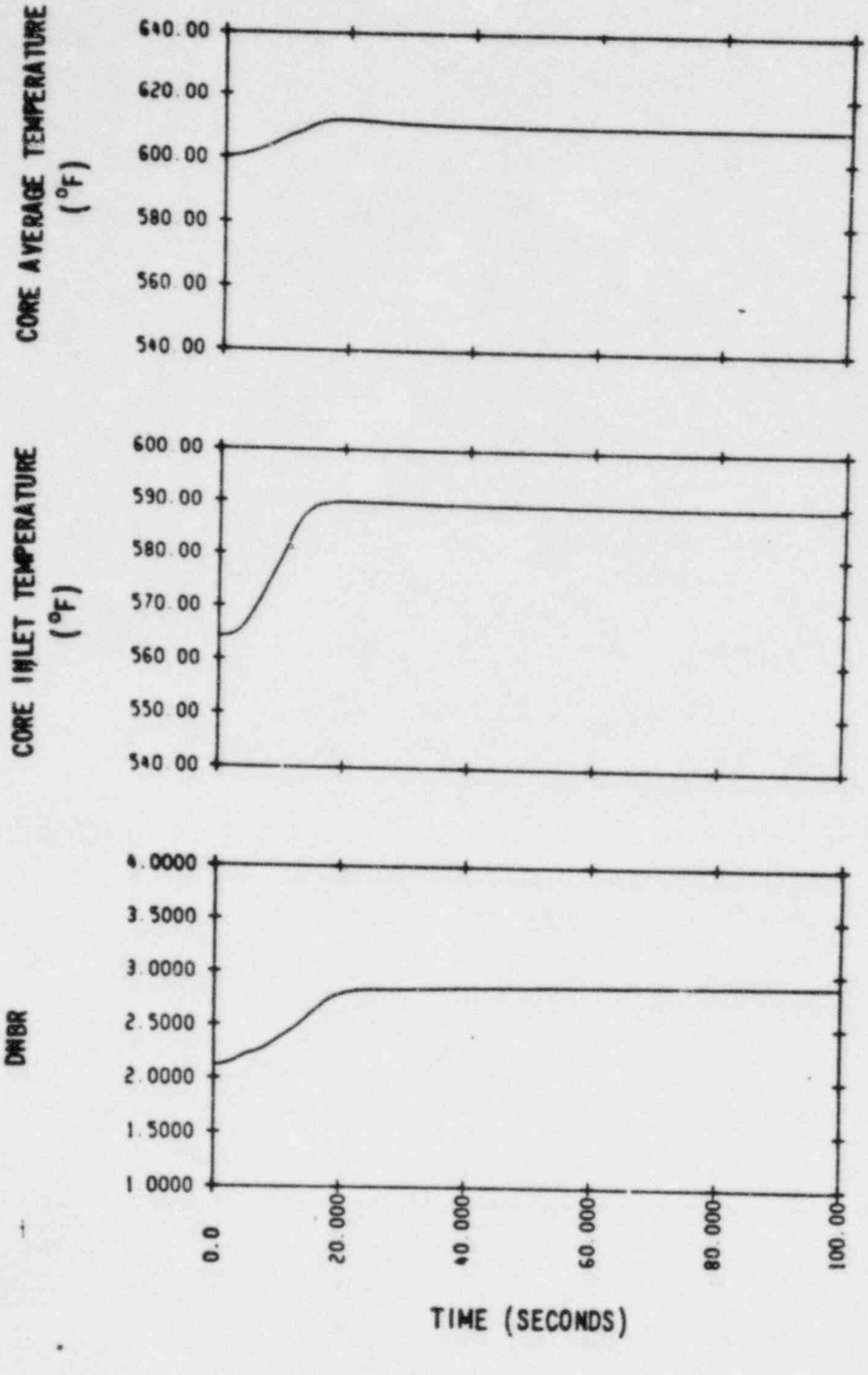

Figure 15.2-4.

Turbine Trip Accident With Pressurizer Spray and Power-Operated Relief Valves, Maximum Moderator Feedback

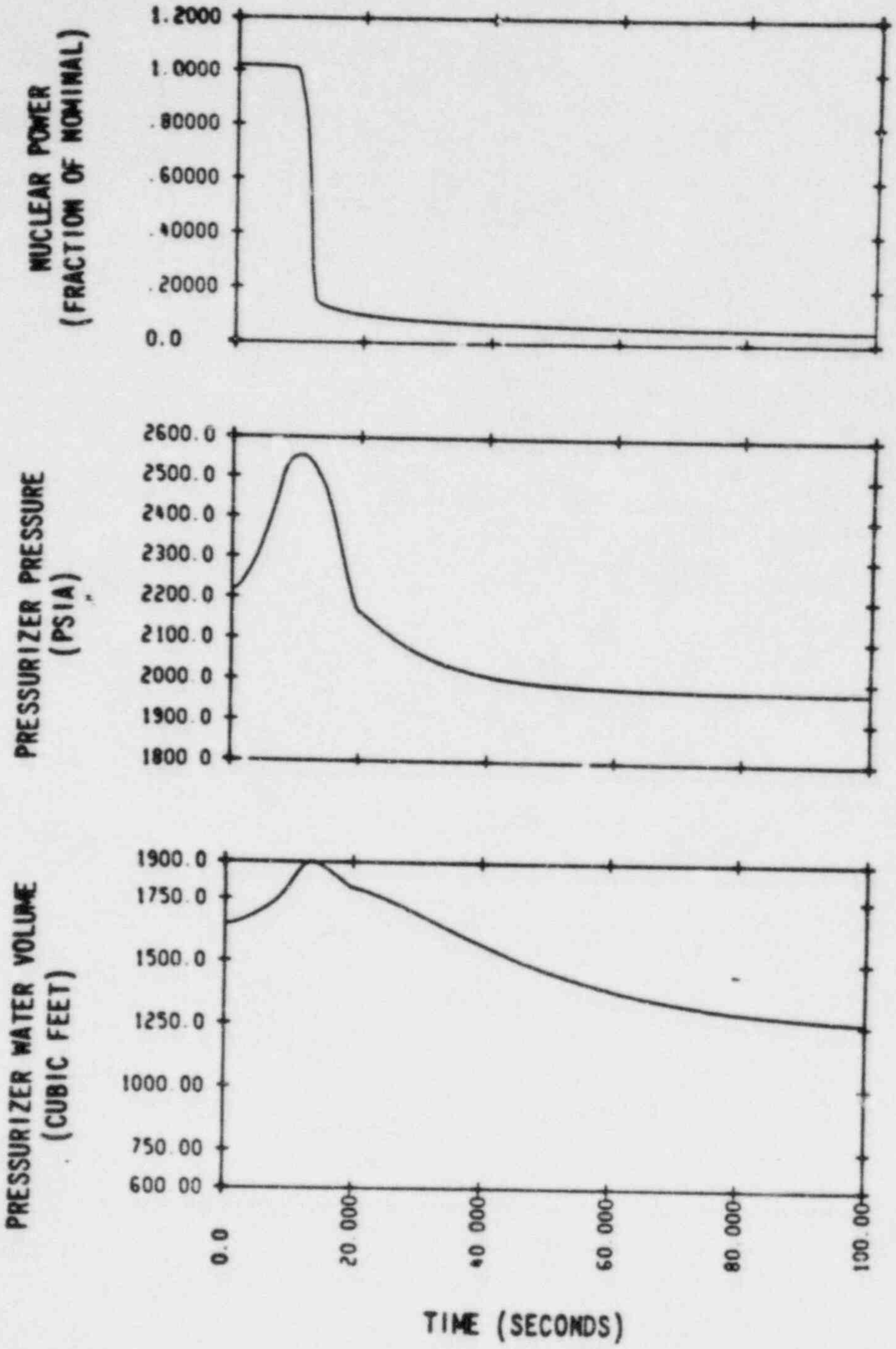

Figure 15.2-5.

Turbine Trip Accident Without Pressurizer Spray and Power-Operated Relief Valves, Minimum Moderator Feedback

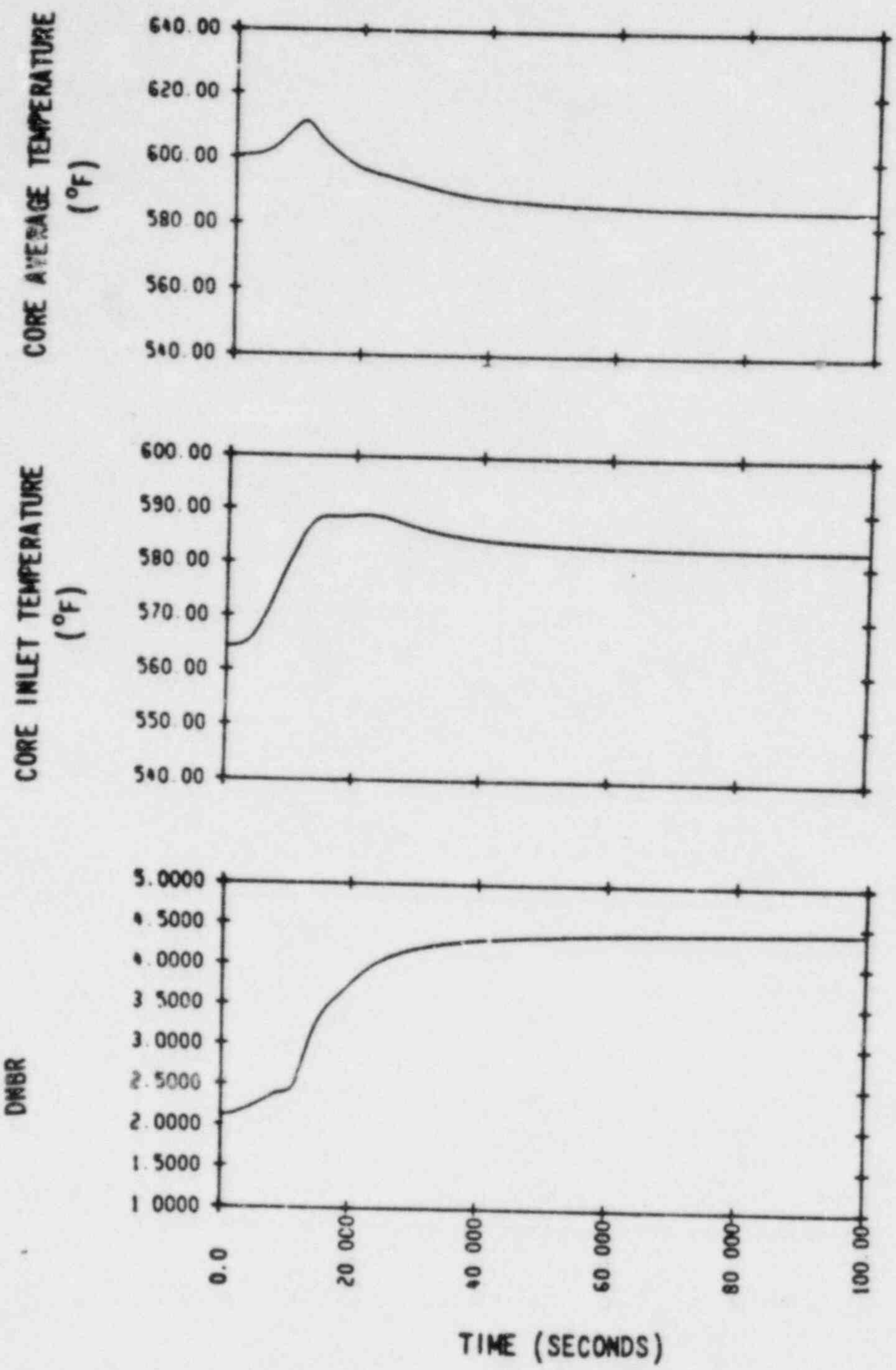

Figure 15.2-6.

Turbine Trip Accident Without Pressurizer Spray
and Power-Operated Relief Valves,
Minimum Moderator Feedback

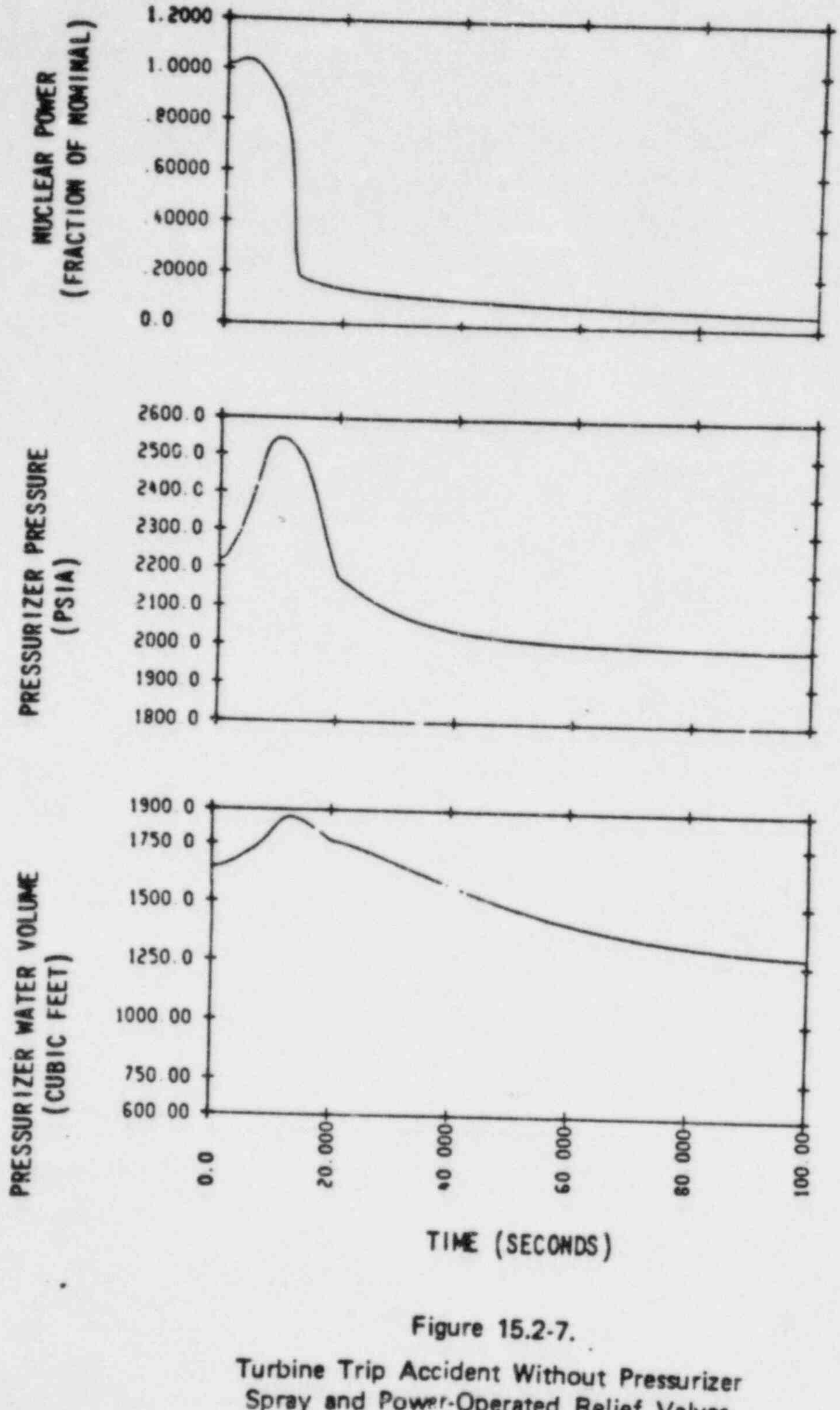

Figure 15.2-7.

Turbine Trip Accident Without Pressurizer Spray and Power-Operated Relief Valves, Maximum Moderator Feedback

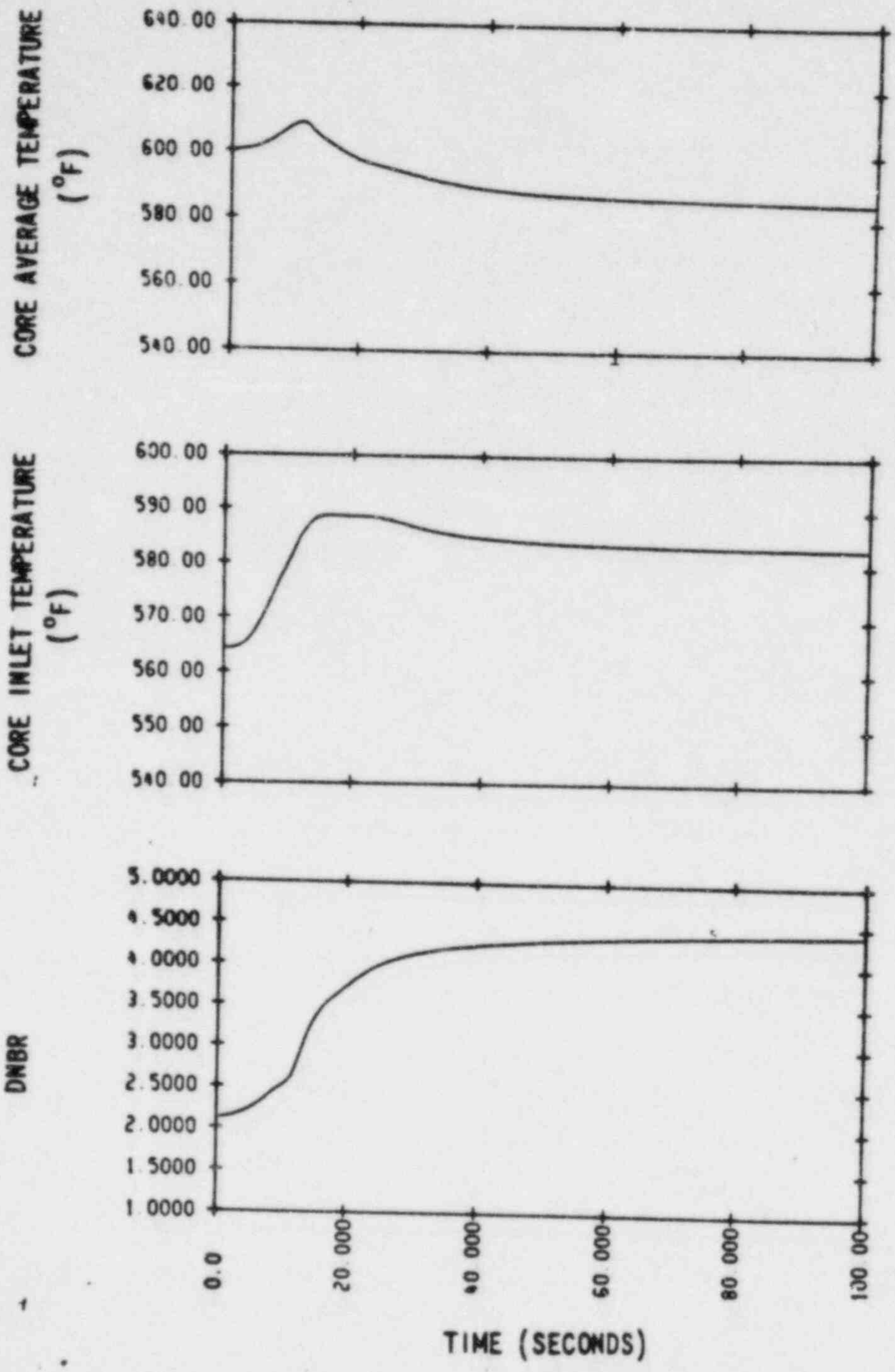

Figure 15.2-8.

Turbine Trip Accident Without Pressurizer Spray and Power-Operated Relief Valves, Maximum Moderator Feedback

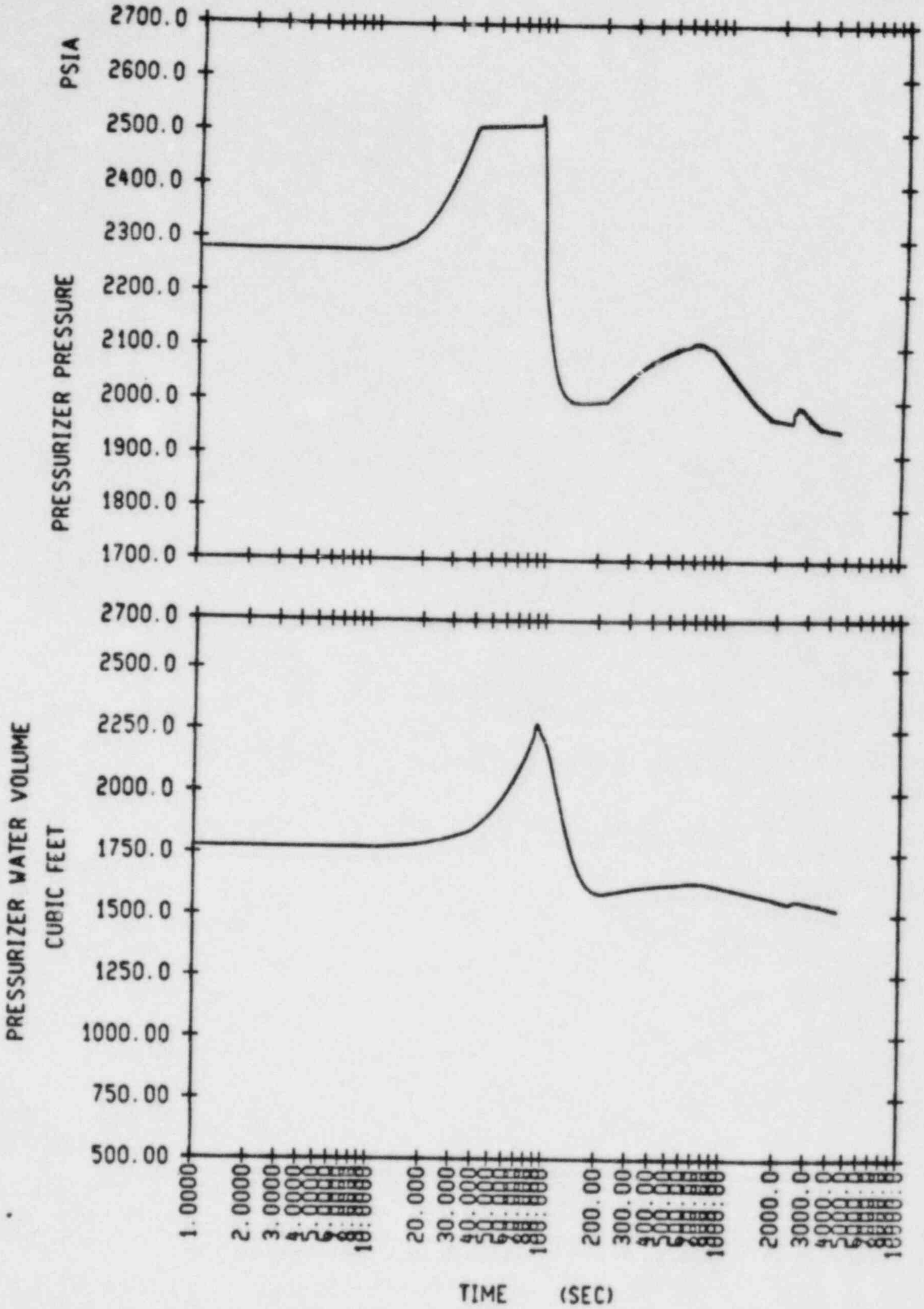

Figure 15.2-9 Pressurizer Pressure and Water Volume Transients for Loss of AC Power

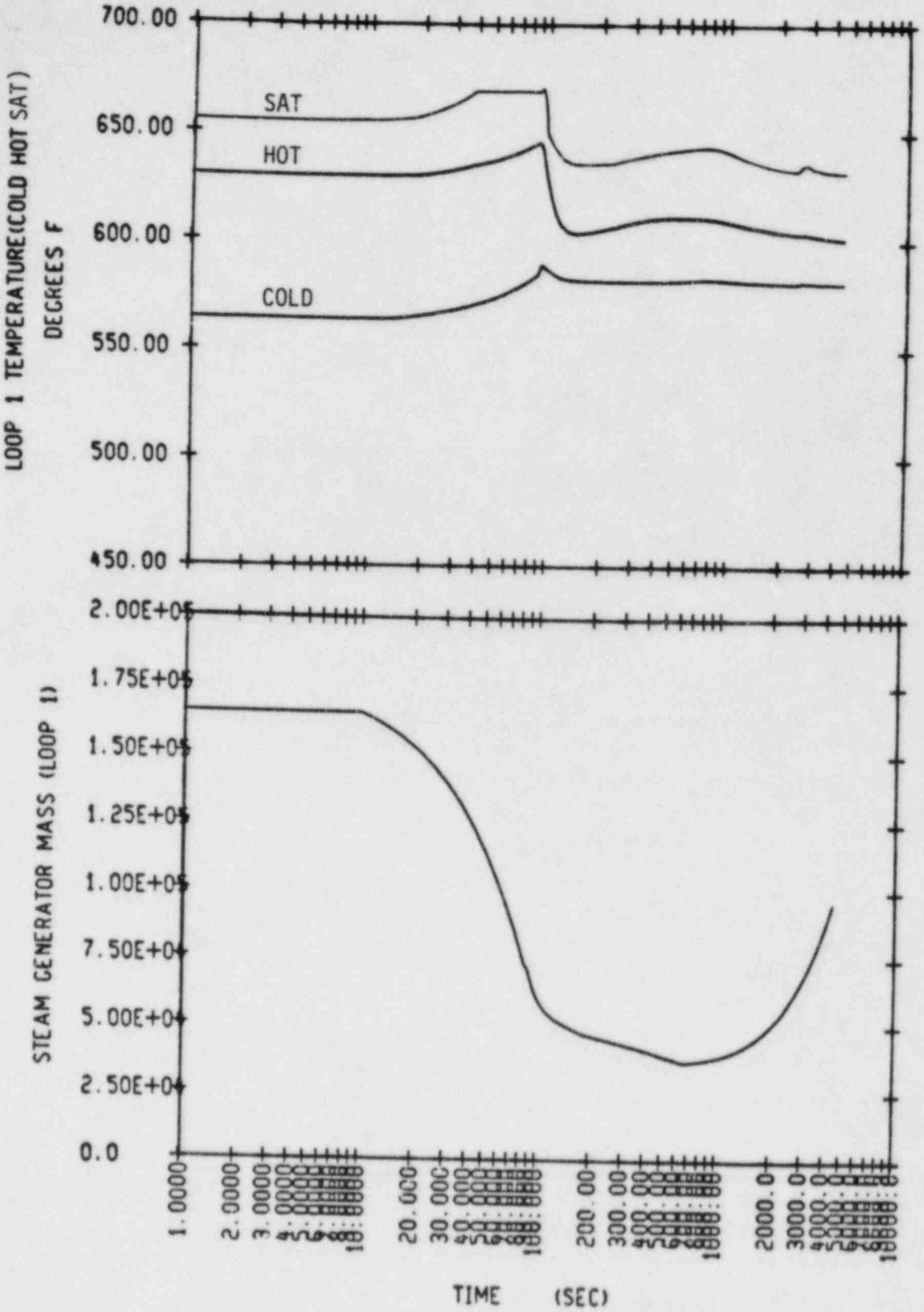

Figure 15.2-10 Loop Temperatures and Steam Generator Mass Transients for Loss of AC Power

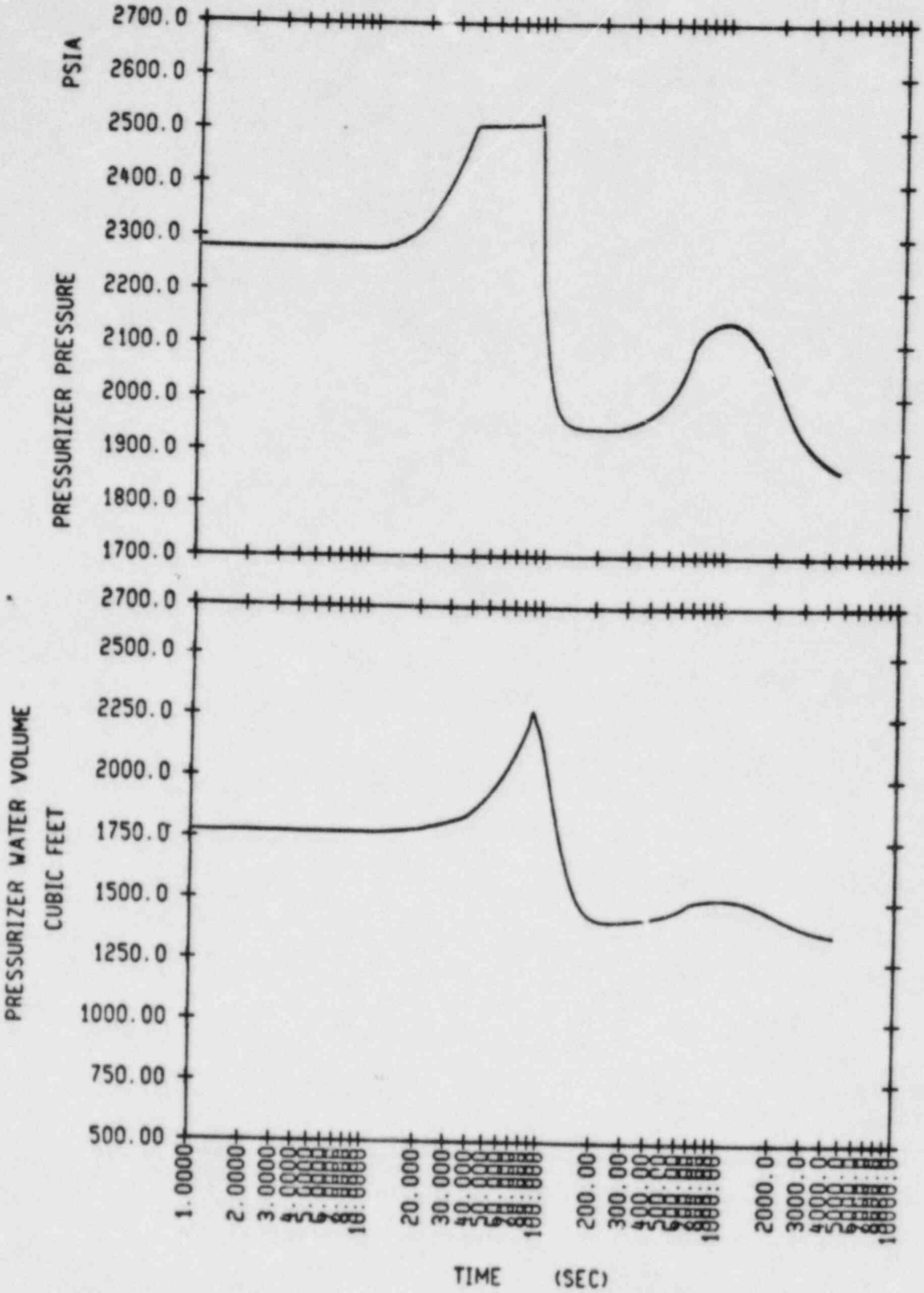

Figure 15.2-11 Pressurizer Pressure and Water Volume Transients for Loss of Normal Feedwater

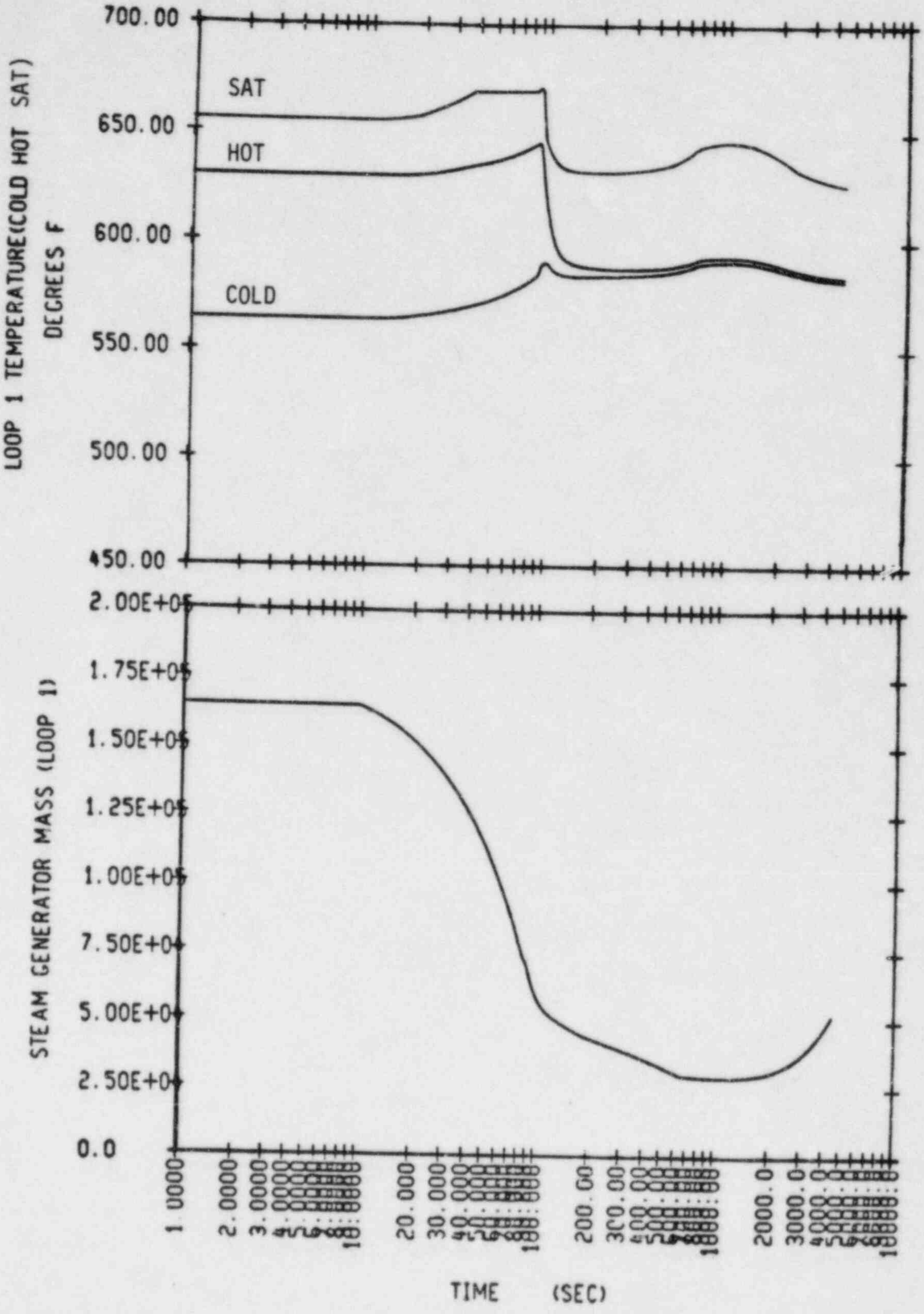

Figure 15.2-12 Loop Temperatures and Steam Generator Mass Transients for Loss of Normal Feedwater

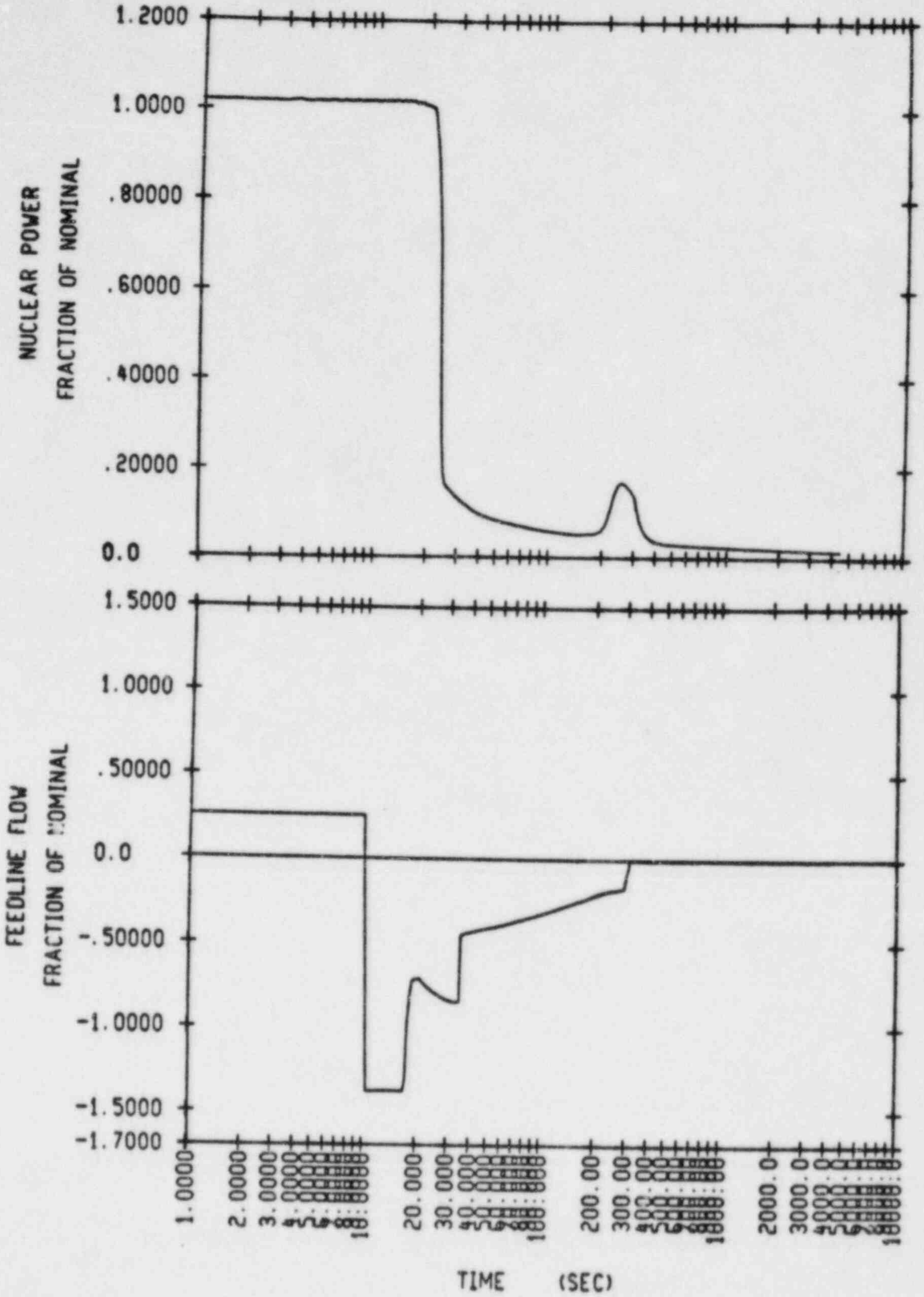

Figure 15.2-13 Nuclear Power Transient and Feedline Break Transient for Main Feedline Rupture With Offsite Power Available

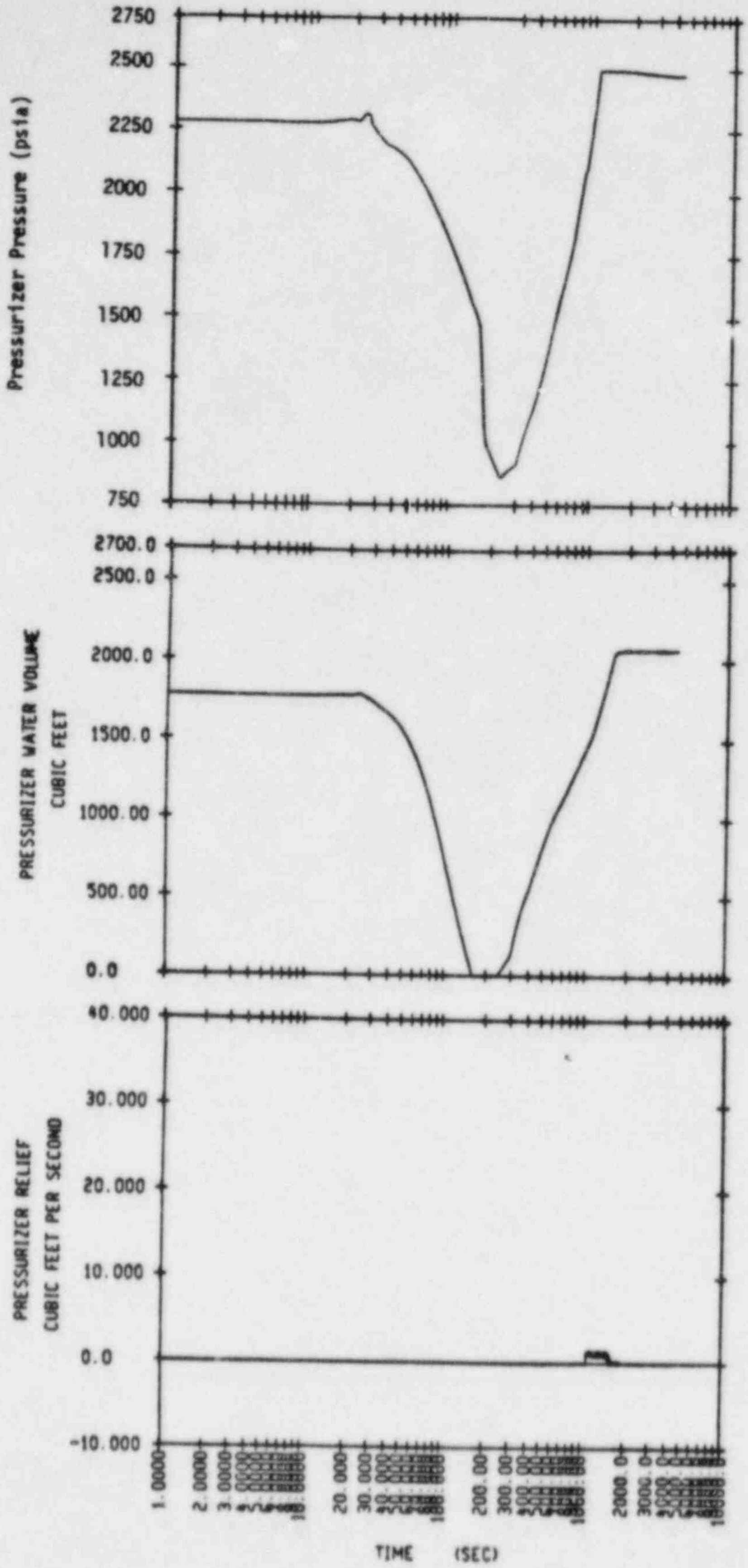

Figure 15.2-14 Pressurizer Pressure, Water Volume, and Relief Transients for Main Feedline Rupture With Offsite Power Available

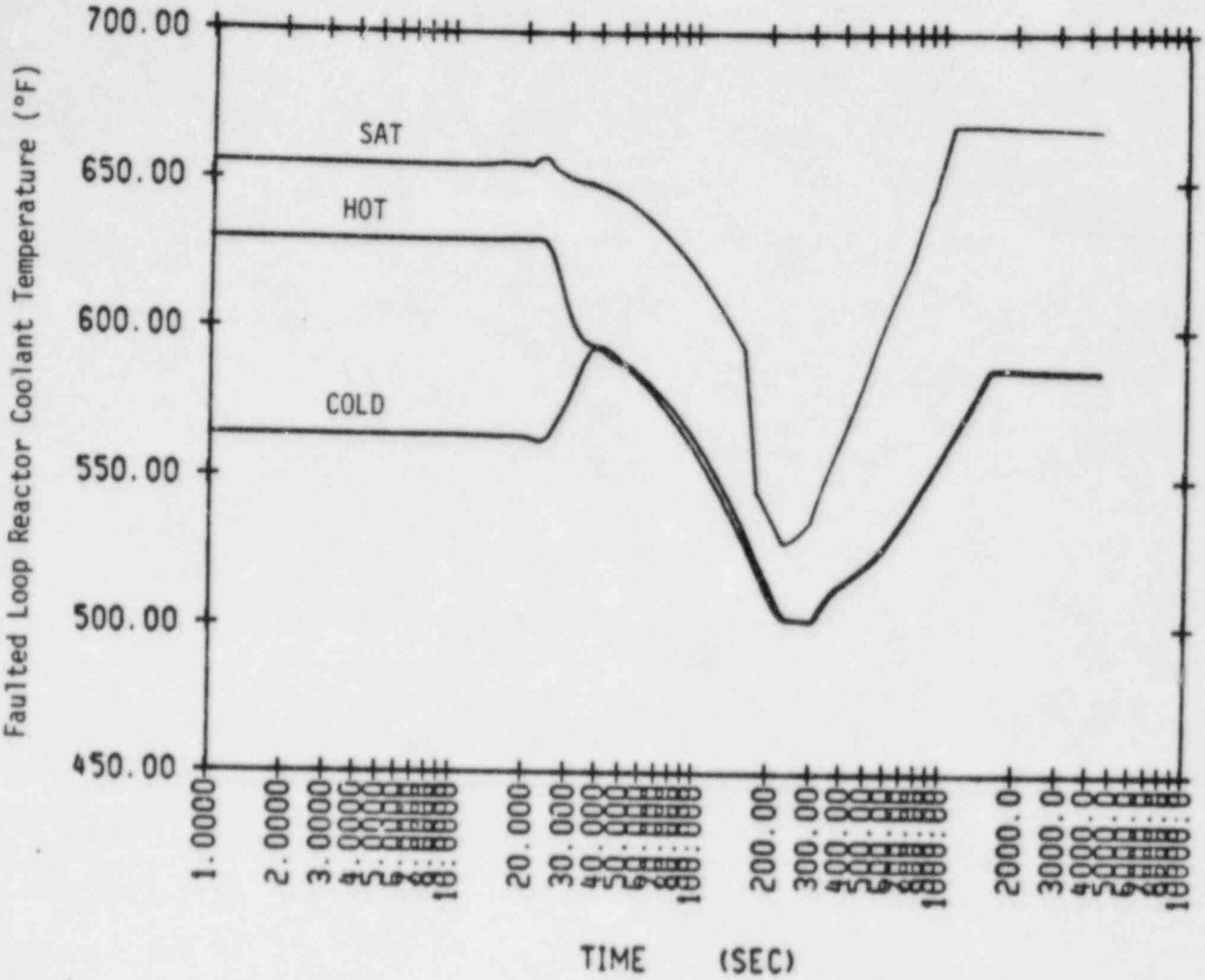

Figure 15.2-15 Reactor Coolant Temperature Transients for the Faulted Loop for Main Feedline Rupture With Offsite Power Available

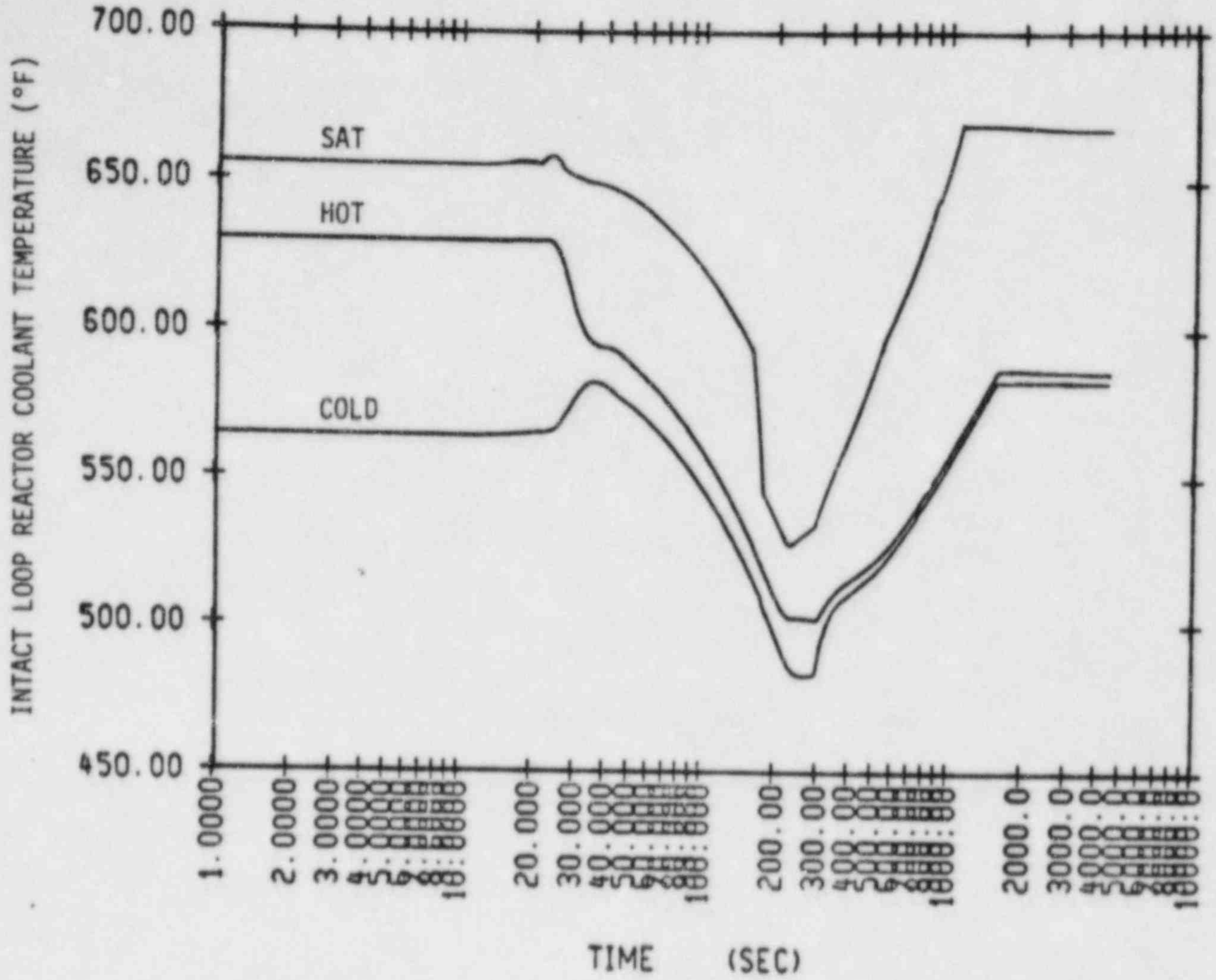

Figure 15.2-16 Reactor Coolant Temperature Transients for an Intact Loop for Main Feedline Rupture With Offsite Power Available

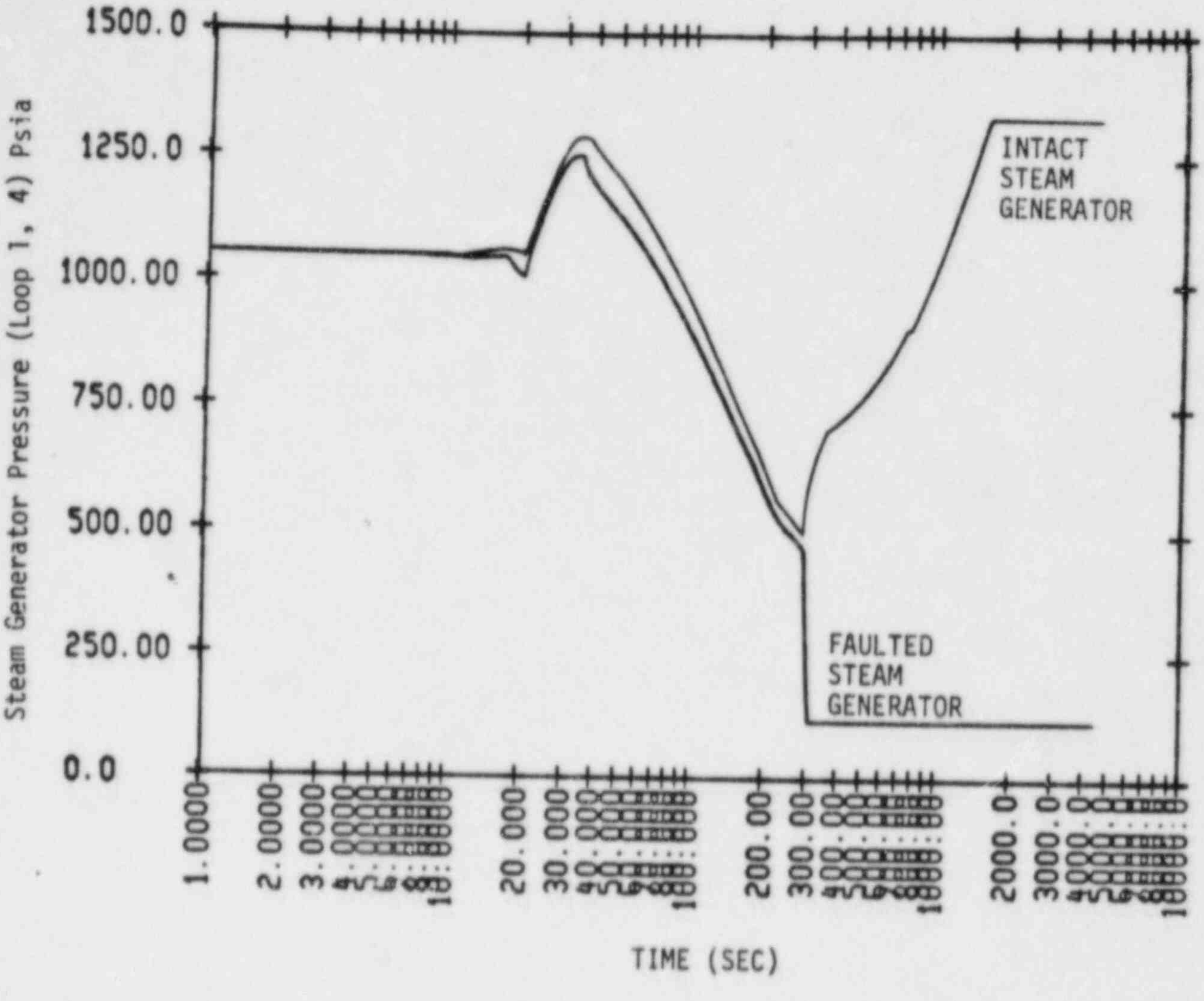

Figure 15.2-17 Steam Generator Pressure Transients for Main Feedline Rupture With Offsite Power Available

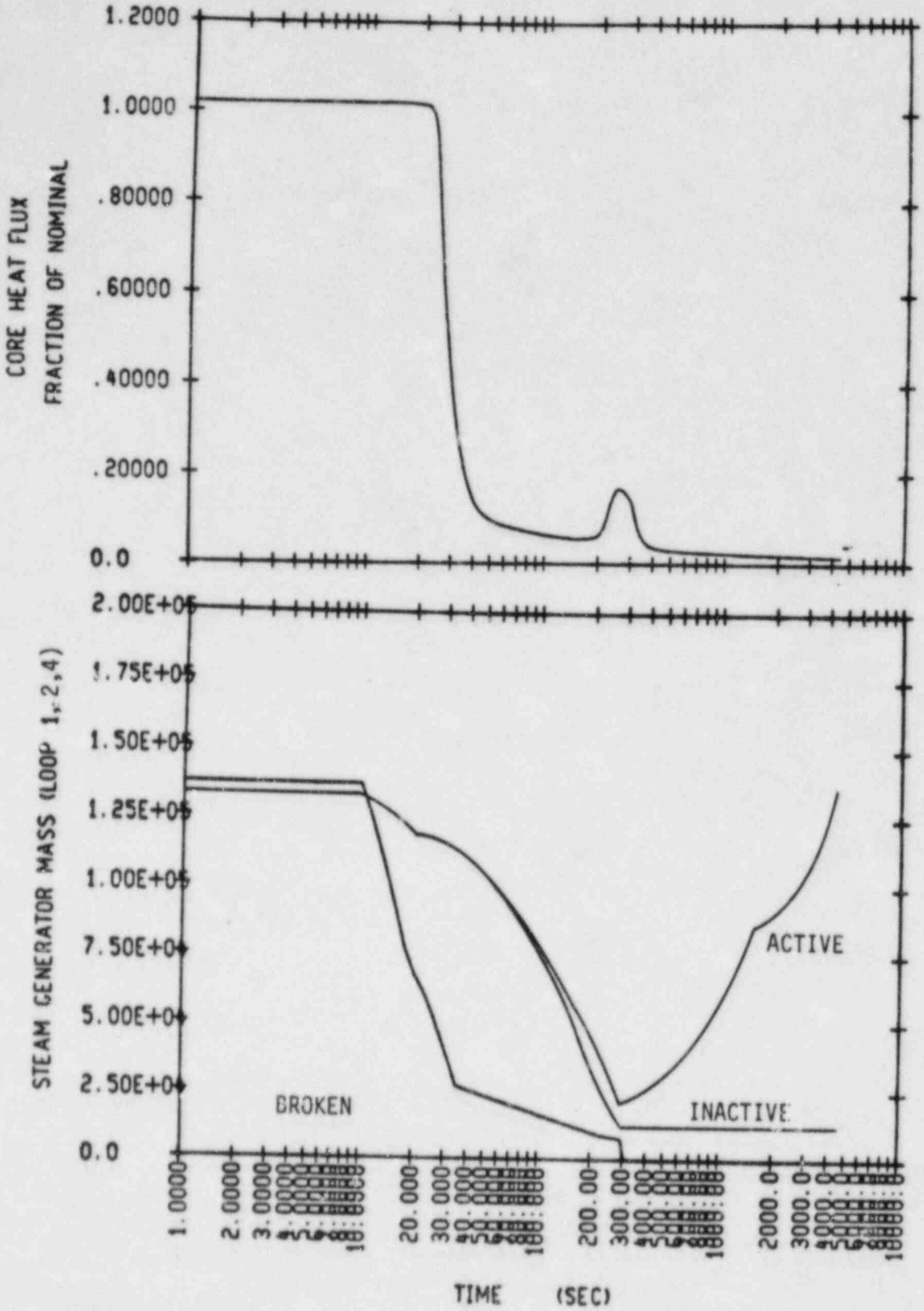

Figure 15.2-18 Core Heat Flux Transient and Steam Generator Mass Transients for Main Feedline Rupture With Offsite Power Available

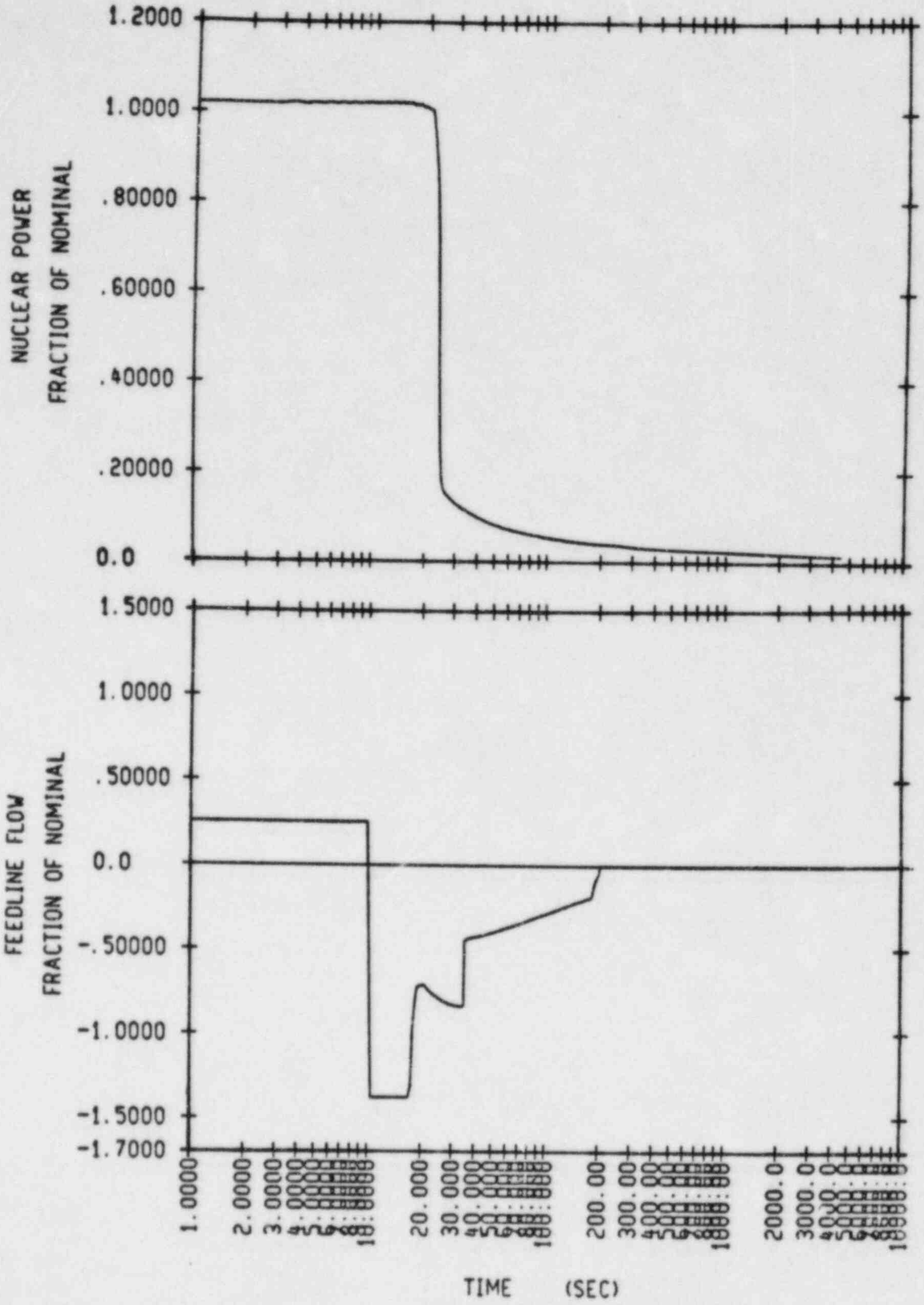

Figure 15.2-19 Nuclear Power Transient, and Feedline Break Flow Transient for Main Feedline Rupture Without Offsite Power

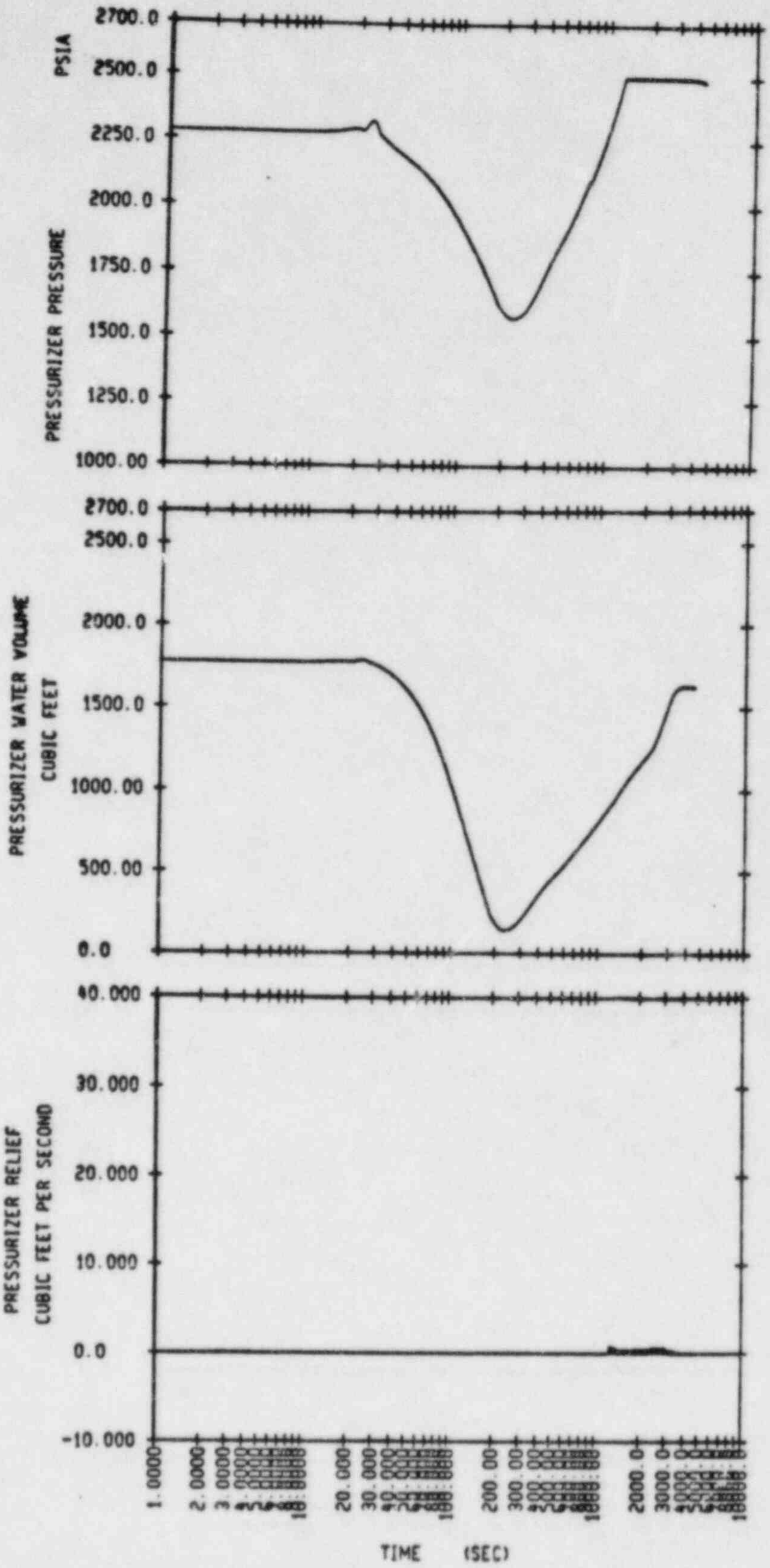

Figure 15.2-20 Pressurizer Pressure, Water Volume, and Relief Rate for Main Feedline Rupture Without Offsite Power

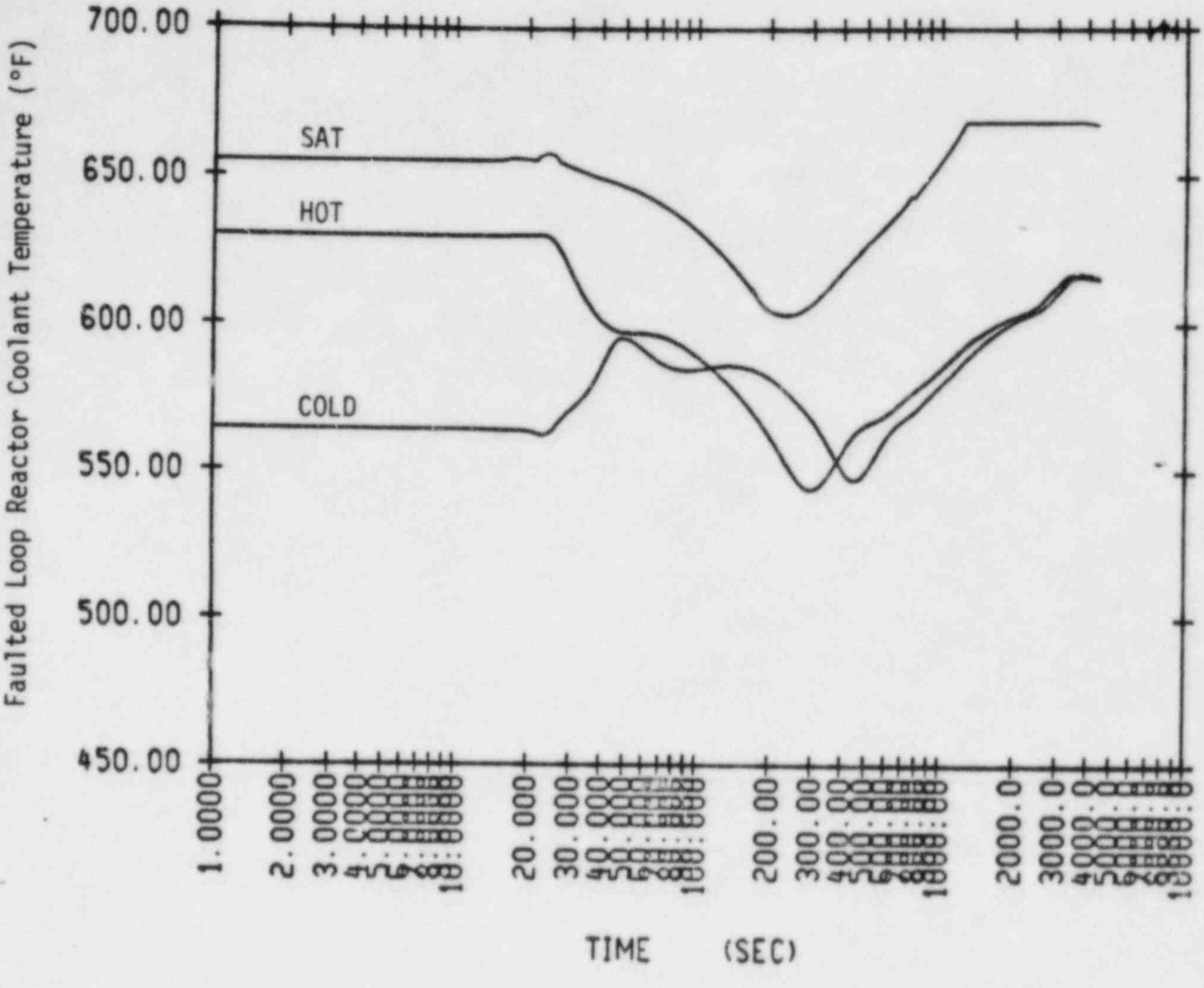

Figure 15.2-21 Reactor Coolant Temperature Transients for the Faulted Loop for Main Feedline Rupture Without Offsite Power

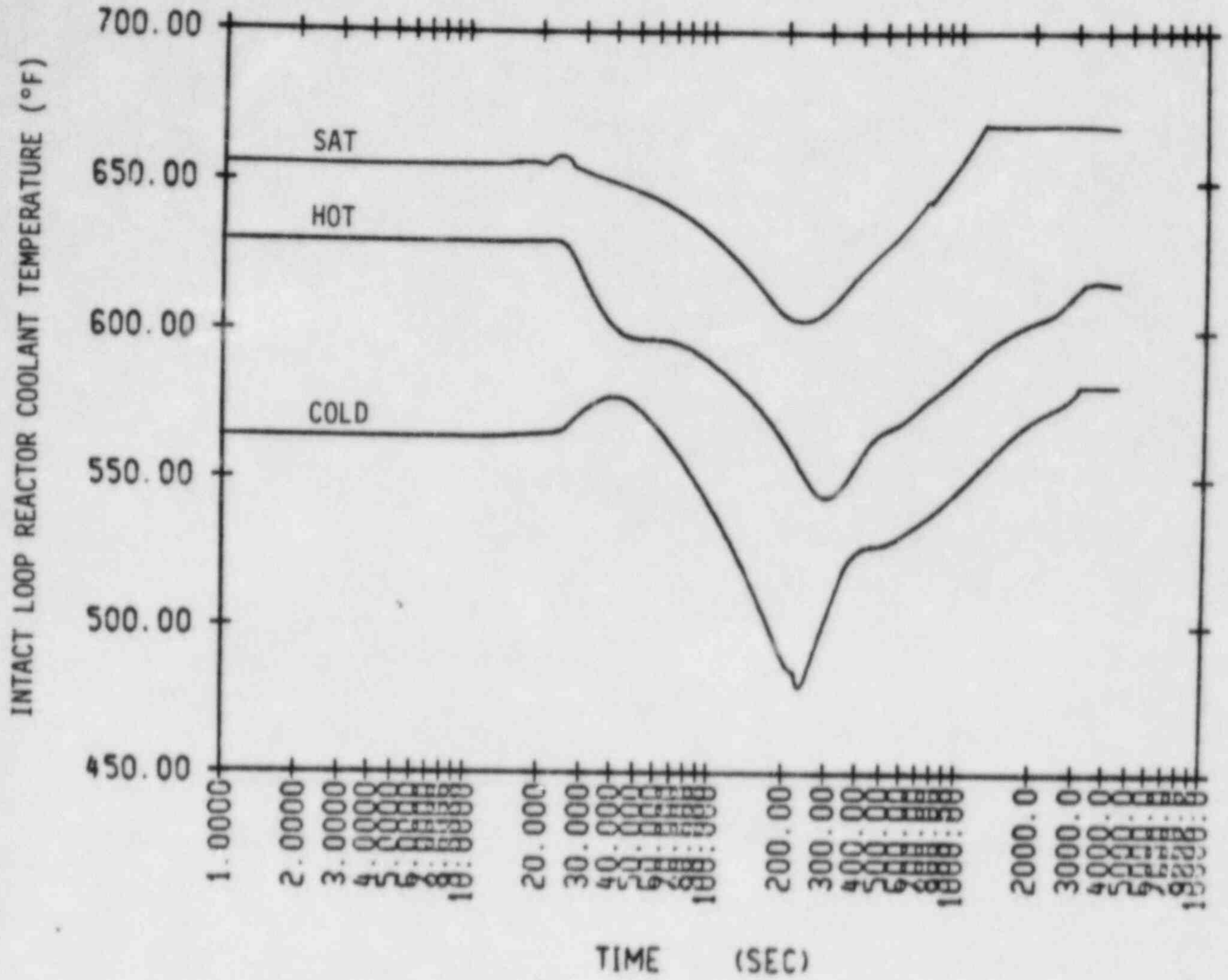

Figure 15.2-22 Reactor Coolant Temperature Transients for an Intact Loop for Main Feedline Rupture Without Offsite Power

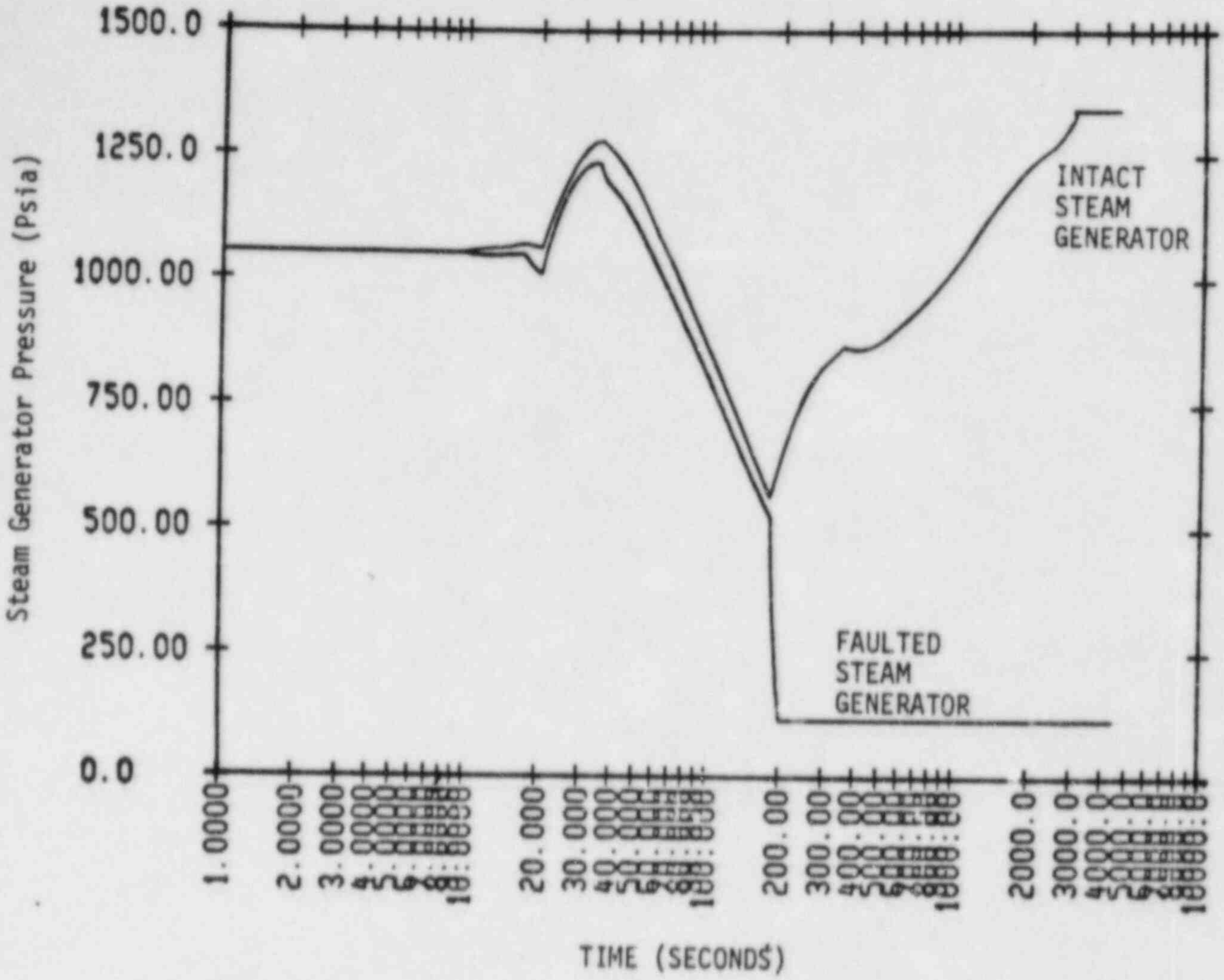

Figure 15.2-23 Steam Generator Pressure Transients for Main Feedline Rupture Without Offsite Power

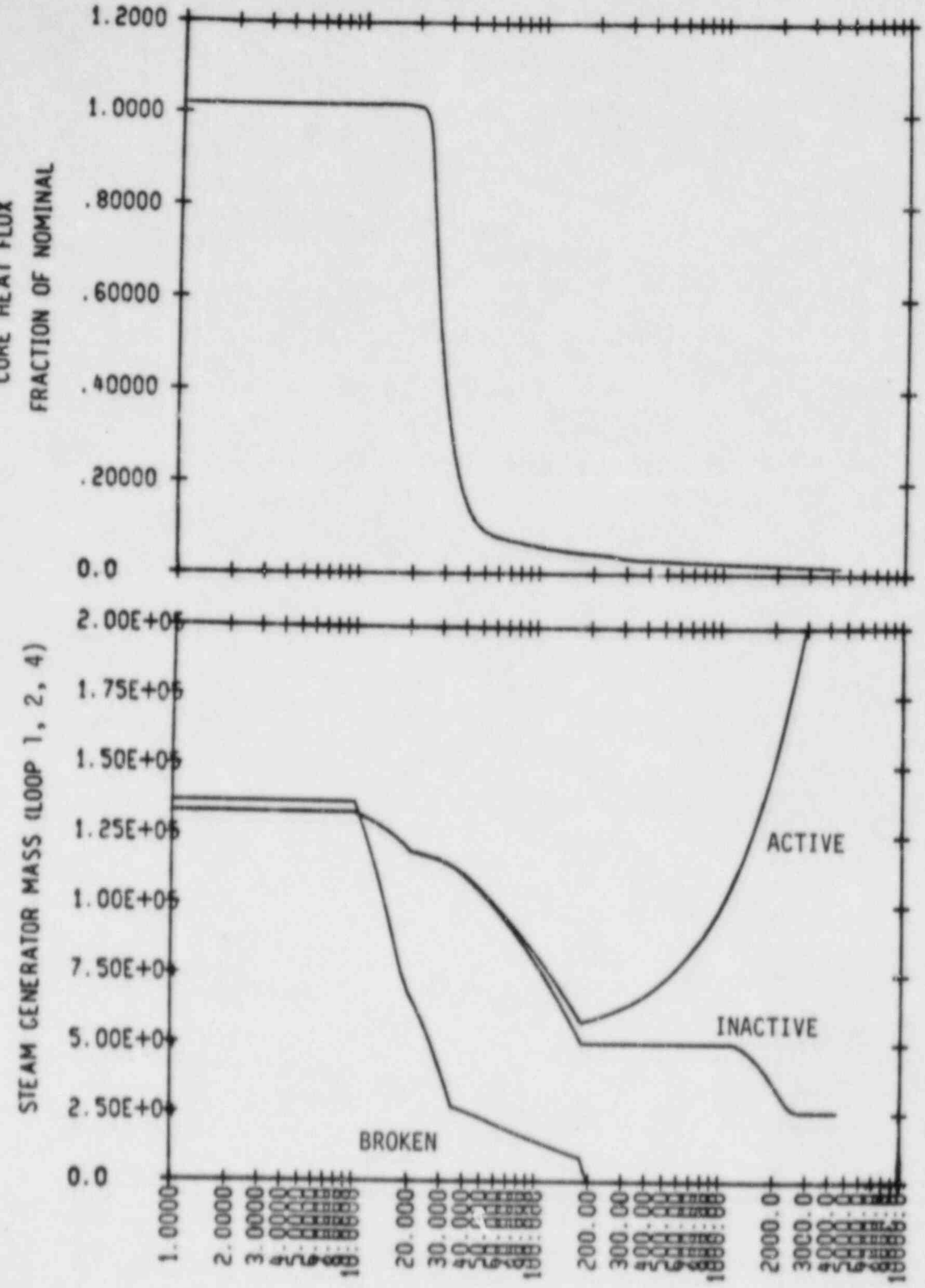

Figure 15.2-24 Core Average Heat Flux Transient and Steam Generator Mass Transients for Main Feedline Rupture Without Offsite Power

15.6 DECREASE IN REACTOR COOLANT INVENTORY

15.6.3 Steam Generator Tube Rupture

15.6.3.1 Identification of Cause and Accident Description

The accident examined is the complete severance of a single steam generator tube. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with the design basis reactor coolant activity (see RESAR-SP/90 PDA Module 12, "Waste Management"). The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the RCS. In the event of a coincident loss of offsite power, or failure of the steam dump system discharge to the condenser, discharge of radioactivity to the atmosphere takes place via the steam generator safety and/or power-operated relief valves.

Due to a series of alarms as described below, the operator will readily determine that a steam generator tube rupture has occurred, and identify and isolate the faulty steam generator. Sufficient indications and controls are provided to enable the operator to carry out these functions satisfactorily.

Assuming normal operation of the various plant control systems, the following sequence of events is initiated by a tube rupture:

a) Pressurizer low pressure and low level alarms are actuated and charging pump flow increases in an attempt to maintain pressurizer level. On the secondary side there is a steam flow/feedwater flow mismatch before trip as feedwater flow to the affected steam generator is reduced due to the additional break flow which is now being supplied to that Unit. A high steamline radiation alarm is actuated for the affected steamline.

b) The condenser vacuum pump discharge radiation monitor and steam generator blowdown sample radiation monitor will alarm, further indicating a sharp increase in radioactivity in the secondary system.

c) Continued loss of reactor coolant inventory leads to a reactor trip signal generated by low pressurizer pressure or by a low DNBR signal. Resultant plant cooldown following reactor trip leads to a rapid change of pressurizer level, and the safety injection signal, initiated by low pressurizer pressure, follows soon after the reactor trip. The safety injection signal automatically terminates normal feedwater supply and initiates the secondary side safeguards system.

d) The reactor trip automatically trips the turbine and, if offsite power is available, the steam dump valves open permitting steam dump to the condenser. In the event of coincident station blackout, the steam dump valves would automatically close to protect the condenser. The steam generator pressure would rapidly increase resulting in steam discharge to the atmosphere through the steam generator safety and/or power-operated relief valves.

e) Following reactor trip, the continued action of secondary side safeguards system and borated safety injection flow (supplied from the emergency water storage tank) provide a heat sink which absorbs some of the decay heat. Thus, steam bypass to the condenser from the faulted steam generator, or, in the case of loss of steam dump or condenser, steam relief to the atmosphere, is attenuated while the recovery procedure leading to isolation is being carried out. In the analysis presented, emergency feedwater flow is throttled early, maximizing steam release.

f) Water level in the intact steam generators rises gradually due to the imbalance between emergency feedwater flow and steam relief. In the affected unit, the break flow causes the level to rise more rapidly. If the operator does not stop feed flow to that steam generator and equalize the RCS and steam generator pressures, the level continues to rise until one of two parallel overflow valves is opened by two out of four high water level signals. This action releases water from the steam generator to mix with water in the emergency water storage tank inside the reactor containment building. The flow is great enough that the water level soon drops to a lower level, above the overflow nozzle, where the water level logic closes the valve. This fill and relief cycle, a few minutes long, would be repeated until the feed and break flow were stopped, thus preventing the steam generator from ever filling with water to the top of the level indicator span.

g) Following reactor trip, the first recovery action is to identify the faulted steam generator and isolate all feedwater to it.

h) The operator's subsequent actions are directed toward equalizing the RCS pressure and the affected steam generator pressure. This requires isolating the affected-loop steamline from the main steam header; cooling down the RCS by increasing steam relief from the intact steam generators to the condenser or to the atmosphere, until the circulating reactor coolant is subcooled relative to the affected steam generator pressure; and depressurizing the RCS using pressurizer power-operated relief valves. Safety injection is then stopped, allowing pressure to gradually equalize across the break, stopping break flow.

i) By this time, steam flow from the affected steam generator to the condenser and/or atmosphere has also stopped. The plant is stabilized, with the intact steam generators maintaining the desired RCS temperature with automatically or manually controlled steam relief and feed flow.

j) The plant is cooled down in a controlled manner by relieving steam from the intact steam generators through power-operated relief valves to the atmosphere, or through the dump system to the condenser if they are available. Depressurization of the faulted steam generator is accomplished by the opearator's opening of the overfill control valves, relieving steam to the emergency water storage tank. RCS pressure and pressurizer water level are controlled with pressurizer PORV's and with makeup by safety injection or by charging, if it is available. Cooldown is completed using RHR.

15.6.3.2 Analysis of Effects and Consequences

Method of Analysis - In estimating the mass transfer from the RCS through the broken tube and from the steam generators to the atmosphere the following assumptions are made:

a) Reactor trip occurs automatically as a result of low DNBR.

b) Following reactor trip, only safety grade equipment is available to mitigate the consequences of the event.

c) To maximize calculated steam release from the effected steam generator, a loss of offsite power on turbine trip is assumed. Furthermore, the operator is assumed to throttle emergency feedwater flow to all steam generators following EFW actuation. For the faulted steam generator feedwater flow is thus stopped early in the transient. This also increases subsequent steam release from the faulted steam generator.

d) After this initial action, the operator is assumed to wait until the faulted steam generator water level is clearly within the narrow-range level indicator span as positive indication of the affected steam generator before proceding with the cooldown and pressure equalization. This delays further recovery actions beyond 30 minutes since EFW flow is assumed terminated earlier to the faulty steam generator. In the expected case, EFW flow would not be throttled until level has returned into the narrow range. Although the operator actions assumed in this analysis are not completely consistent with the expected recovery actions, they have been conservatively chosen to maximize steam releases and to demonstrate additional safety margin.

e) The PORV on one intact steam generator is assumed to fail to open, slowing the subsequent cooldown.

To illustrate the bounding water volume increase in the faulted steam generator, a second analysis was performed assuming maximum initial secondary water mass, maximum emergency feedwater flow, with no emergency feedwater control until 30 minutes had elapsed. The 30 minute delay was chosen to demonstrate the effectiveness of this WAPWR design feature, even though the operator would be expected to initiate recovery actions much earlier.

Results

Figures 15.6-1 through 15.6-5 illustrate the time varying values of the key parameters for the steam generator tube rupture event. The radiological consequences are discussed in the next section. Figures 15.6-5 through 15.6-8 illustrate the time varying values assuming maximum water supply capability.

15.6.3.3 Radiological Consequences

The evaluation of the radiological consequences of a steam generator tube rupture (SGTR), assumes that the reactor has been operating with a small percent of defective fuel and leaking generator tubes for sufficient time to establish equilibrium concentrations of radionuclides in the reactor coolant and in the secondary coolant. Radionuclides from the primary coolant enter the steam generator, via the ruptured tube, and are released to the atmoshpere through the turbine condenser air ejector exhaust, the emergency feedwater drive turbine and the steam generator safety or power operated relief valves.

The radioactivity released to the environment, due to a SGTR, depends upon primary and secondary coolant activity, iodine spiking effects, primary to secondary break flow, time dependent break flow, flashing fractions, time dependent scrubbing of flashed activity, partitioning of activity between the steam generator liquid and steam and the mass of fluid discharged to the environment. All of these parameters were conservatively evaluated for the design basis double ended rupture of a single tube.

15.6.3.3.1 Analytical Assumptions

The major assumptions and parameters used in the analysis are itemized in Table 15.6-1. The following is a more detailed discussion of the source term.

15.6.3.3.1.1 Source Term Calculations

The concentrations of nuclides in the primary and secondary system, prior to and following the SGTR are determined as follows:

A. The iodine concentrations in the reactor coolant will be based upon preaccident and accident initiated iodine spikes.

   1. Accident Initiated Spike - The reactor trip or primary system depressurization associated with the SGTR creates an iodine spike in the primary system which increases the iodine release rate from the fuel to the primary coolant to a value 500 times greater than the release rate corresponding to the maximum equilibrium primary system iodine concentration of 1 μCi/gram of D.E. I-131. The duration of the spike, 3.3 hours, is sufficient to increase the initial RCS I-131 inventory by a factor of 100.

   2. Preaccident Spike - A reactor transient has occurred prior to the SGTR and has raised the primary coolant iodine concentration from 1 μCi/gram to 60 μCi/gram of Dose Equivalent (D.E.) I-131.

B. The noble gas concentrations in the reactor coolant are based on 1 percent defective fuel.

C. The secondary coolant activity is based on the D.E. of 0.1 μCi/gram of I-131.

15.6.3.3.1.2 Mathematical Models Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

A. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A.

B. The atmospheric dispersion factors used in the analysis were calculated based on typical onsite meteorological measurement programs.

C. The thyroid inhalation dose and total-body gamma immersion doses to a receptor at the exclusion area boundary and outer boundary of the low-population zone were analyzed, using the models described in Appendix 15A.

15.6.3.3.1.3 Identification of Leakage Pathways and Resultant Leakage Activity

Radionuclides carried by the rupture flow to the faulted generator are released to the environment via the steam line safety or power operated relief valves. For the release of iodines, a fraction of the rupture flow is assumed to flash to steam and immediately release radioiodines to the environment. The fraction that does not flash is assumed to mix with the bulk generator water and partition between the generator liquid and steam before release to the environment. Noble gases are also assumed to be directly released.

15.6.3.3.2 Identification of Uncertainties and Conservatisms in the Analysis

A. Reactor coolant activities are based on the Technical Specification limit of 1.0 µCi/g of D.E. I-131 with extremely large iodine spike values result in equivalent concentrations many times greater than the reactor coolant activities based on 0.12-percent defective fuel associated with normal operating conditions.

B. The noble gas activities are based on one percent defective fuel and cannot exist simultaneously with 1.0 µCi/g I-131. For iodines, one percent defects would be approximately three times the technical specification limit.

C. A 1-gal/min steam generator primary-to-secondary leakage is assumed, which is significantly greater than that anticipated during normal operation. Furthermore, it was conservatively assumed that 0.35 gpm leakage goes to the faulted steam generator.

D. The rupture site is conservatively assumed to be at the top of the tube bundle to minimize any scrubbing of the flashed coolant as the steam bubbles rise from the rupture site to the surface of the generator water.

E. The rupture is conservatively assumed to be on the hot leg side of the tube sheet to maximize the flashing fraction.

F. The iodine partition factors assumed for the steam generator and for the condenser are at least a factor of 10 less than expected.

G. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that the assumed meteorological conditions would be present during the course of the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meteorological conditions assumed, are conservative.

15.6.3.3.3 Conclusions

15.6.3.3.3.1 Filter Loadings

The only ESF filtration system considered in the analysis which limits the consequences of the SGTR is the control room filtration system.

Integrated activity on the control room filters has been evaluated for the more limiting loss-of-coolant accident (LOCA) analysis, as discussed in Subsection 15.6.5 of RESAR-SP/90 PDA Module 1, "Primary Side Safeguards System". Since the control room filters are capable of accommodating the potential design basis LOCA fission product iodine loadings, there will be sufficient capacity to accommodate any fission product loading due to a postulated SGTR.

15.6.3.3.3.2 Dose to Receptor at the Exclusion Area Boundary and Low-Population Zone Outer Boundary

The potential radiological consequences resulting from the occurrence of a postulated SGTR have been conservatively analyzed using assumptions and models described. The total-body gamma dose due to immersion from direct radiation and the thyroid dose due to inhalation have been analyzed for the 0 to 2-hour dose at the exclusion area boundary and for the duration of the accident (0 to 8 hours) at the low-population zone outer boundary. The resultant doses listed in Table 15.6-2 are small fractions of the guideline values of 10 CFR 100.

TABLE 15.6-1
PARAMETERS USED IN EVALUATING
THE RADIOLOGICAL CONSEQUENCES OF A
STEAM GENERATOR TUBE RUPTURE

I. Source Data

a. Core power level, MWt	3876
b. Total steam generator tube leakage, gpm	1
c. Reactor coolant iodine activity:	
1. Accident Initiated Spike	Initial activity equal to the dose equivalent of 1.0 µCi/gm of I-131 with an assumed iodine spike that increases the rate of iodine release into the reactor coolant by a factor of 500. See Tables 15A-5 and 15A-6.
2. Pre-Accident Spike	An assumed pre-accident iodine spike, which has resulted in the dose equivalent of 60 µCi/gm of I-131 in the reactor coolant. See Table 15A-5.
d. Reactor coolant noble gas activity (both cases)	Based on one percent defective fuel. See Table 15A-7.
e. Secondary system initial activity	Dose equivalent of 0.1 µCi/gm of I-131.
f. Reactor coolant mass, grams	3.3×10^8
g. Secondary coolant mass (4 generators), grams	2.0×10^8
h. Offsite power	Lost after Trip
i. Primary-to-secondary leakage duration	8 hours
j. Species of iodine	100 percent elemental

TABLE 15.6-1 (Sheet 2)

II.	Atmospheric Dispersion Factors (See Table 15A-2)	
III.	Activity Released Data for the Faulted Generator	
	a. Steam released	
	0-690 sec (lb), to turbine condenser	847,600
	690 - 1.8 hr, to atmosphere	106,100
	> 2 hr(1)	0
	b. Rupture flow	
	0-690 sec (lb)	30,600
	690 sec - 1.8 hr	200,400
	> 1.8 hr	0
	c. Rupture flow flashing fraction	
	0-690 sec	0.12
	690-1800 sec	0.05
	1800-2400 sec	0.04
	2400 - 1.2 hr	0.03
	> 1.2 hr	0
	d. Elemental iodine attenuation factor for flashed coolant	1.0
	e. Elemental iodine partiation factor	100
IV.	Activity Release Data for the Intact Steam Generators	
	a. Steam released	
	0-690 sec (lb), turbine condenser	2,523,400
	690 sec - 2 hr, to atmosphere	432,900
	2-8 hr, to atmosphere	1,345,000
	b. Primary to secondary leak rate (gpm)(2)	0.65
	c. Elemental iodine partiation factor	100
V.	Condenser	
	a. Elemental iodine partition factor	100

(1) After 2 hours, discharge from the faulted SG is diverted to the in-containment EWST.

(2) Based on water at 590°F, 2250 psia.

TABLE 15.6-1 (Sheet 3)

VI. Activity Released to the Environment

a. Accident Initiated Spike

Isotope	0-2 h (Ci)	2-8 h (Ci)
I-131	3.9×10^{1}	6.9×10^{-1}
I-132	1.8×10^{2}	1.3×10^{0}
I-133	7.9×10^{1}	1.9×10^{0}
I-134	8.6×10^{1}	1.6×10^{-1}
I-135	7.3×10^{1}	1.2×10^{0}

b. Pre-Accident Spike

I-131	1.1×10^{2}	1.9×10^{-1}
I-132	8.4×10^{1}	2.7×10^{-1}
I-133	1.6×10^{2}	1.5×10^{0}
I-134	1.6×10^{1}	1.0×10^{-2}
I-135	8.4×10^{1}	5.7×10^{-1}

c. Noble gases - both cases

Xe-131m	2×10^{2}	1.0×10^{0}
Xe-133m	1.5×10^{3}	7.4×10^{0}
Xe-133	2.4×10^{4}	1.2×10^{2}
Xe-135m	1.3×10^{1}	8.1×10^{-5}
Xe-135	6.0×10^{2}	2.2×10^{0}
Xe-138	1.4×10^{1}	1.4×10^{-4}
Kr-85m	1.6×10^{2}	4.3×10^{-1}
Kr-85	6.6×10^{2}	3.4×10^{0}
Kr-87	7.7×10^{1}	5.9×10^{-2}
Kr-88	2.7×10^{2}	5.3×10^{-1}

TABLE 15.6-2

RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE

Case 1.	Accident Initiated Iodine Spike	
	Exclusion area boundary (0-2 h) Thyroid (rem)	5.3
	Low-population outer boundary (8 h) Thyroid (rem)	2.2
Case 2.	Pre-Accident Iodine Spike	
	Exclusion area boundary (0-2 h) Thyroid (rem)	13.3
	Low-population zone outer boundary (8h) Thyroid (rem)	5.4
Both Cases.	Whole Body Gamma (rem)	
	Exclusion area boundary (0-2 h)	0.24
	Low-population zone outer boundary (8 h)	0.1

TABLE 15.6-2

RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE

Case 1.	Accident Initiated Iodine Spike	
	Exclusion area boundary (0-2 h) Thyroid (rem)	5.3
	Low-population outer boundary (8 h) Thyroid (rem)	2.2
Case 2.	Pre-Accident Iodine Spike	
	Exclusion area boundary (0-2 h) Thyroid (rem)	13.3
	Low-population zone outer boundary (8h) Thyroid (rem)	5.4
Both Cases.	Whole Body Gamma (rem)	
	Exclusion area boundary (0-2 h)	0.24
	Low-population zone outer boundary (8 h)	0.1

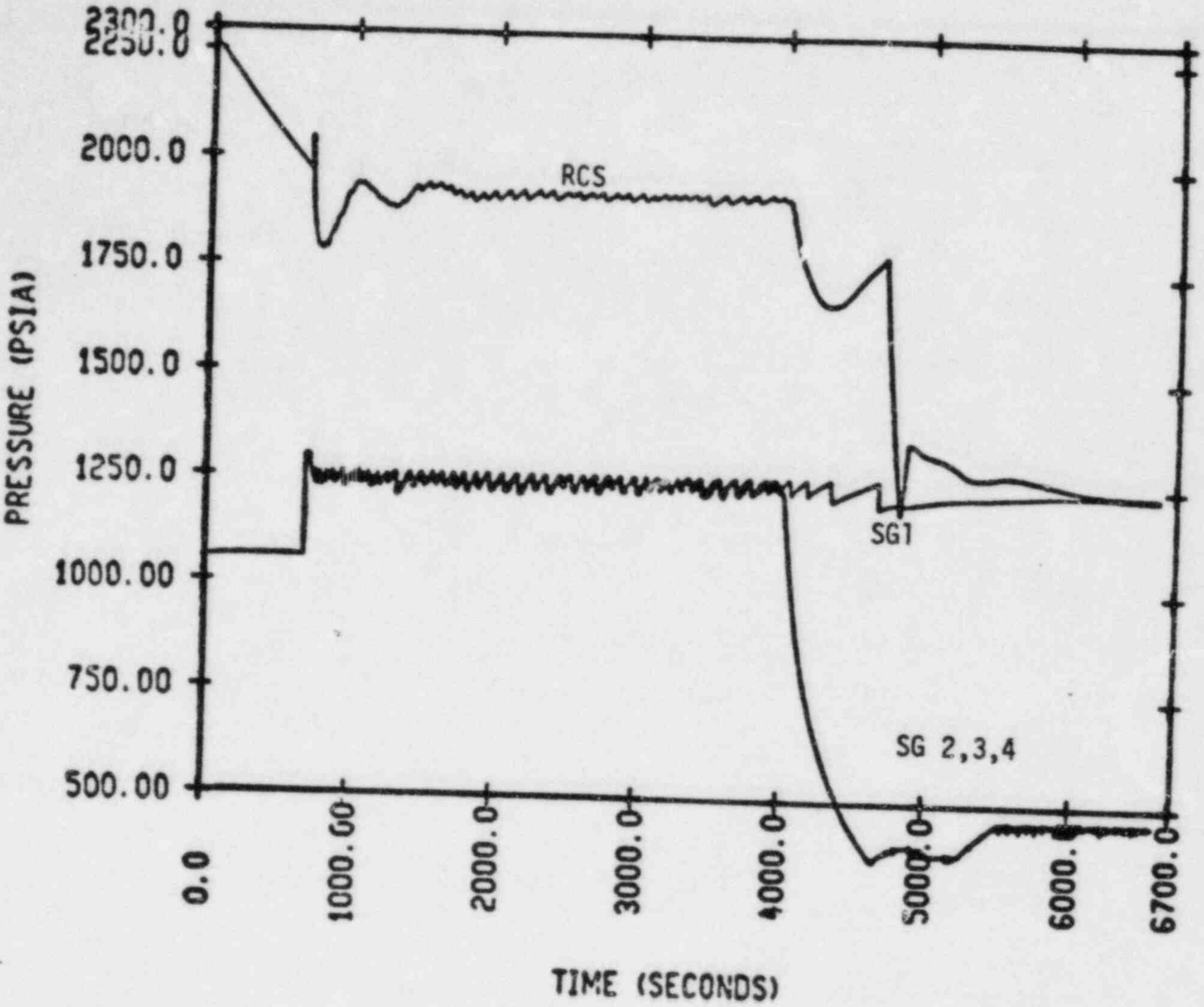

FIGURE 15.6-1 STEAM GENERATOR TUBE RUPTURE PRESSURES VS TIME

IN REACTOR COOLANT SYSTEM (RCS), FAULTED STEAM GENERATOR (SG 1) AND ACTIVE STEAM GENERATORS (SG 2,3,4)

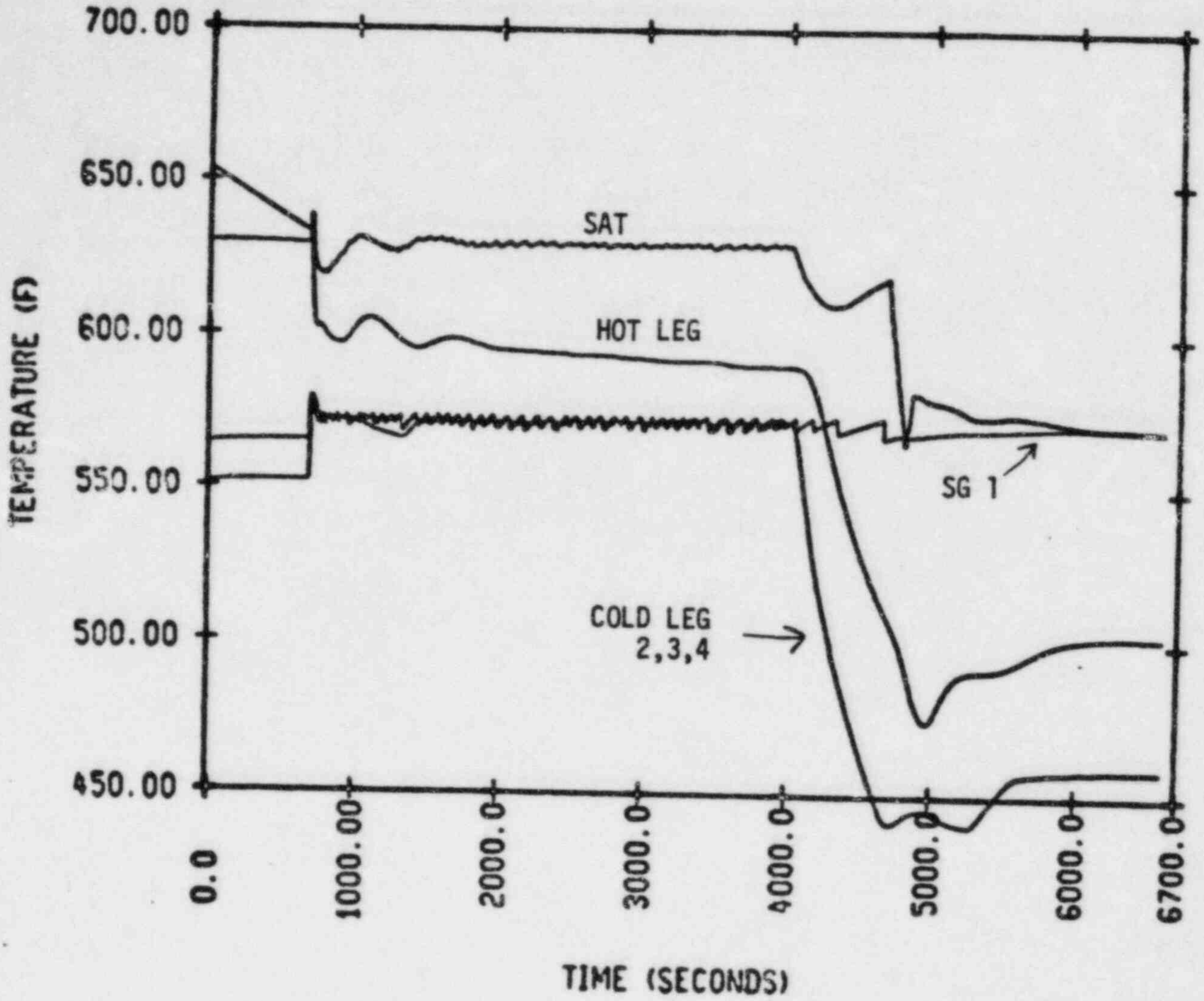

FIGURE 15.6-2 STEAM GENERATOR TUBE RUPTURE TEMPERATURES VS TIME

PRESSURIZER SATURATION TEMPERATURE (SAT), HOT LEG INLET TEMPERATURE, FAULTED STEAM GENERATOR STEAM TEMPERATURE (SG1), AND INTACT COLD LEG TEMPERATURES

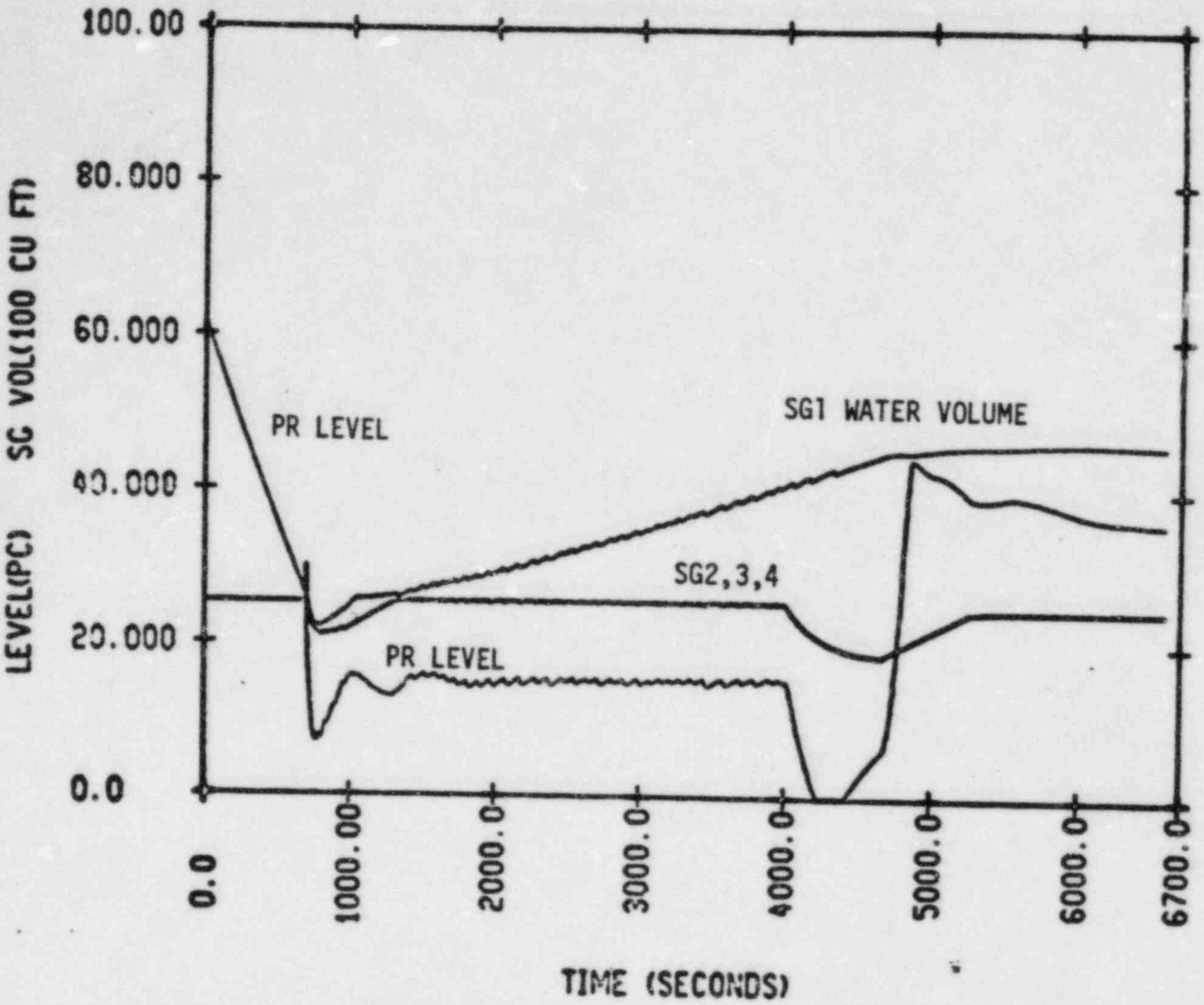

FIGURE 15.6-3 STEAM GENERATOR TUBE RUPTURE WATER VOLUME AND LEVELS VS TIME

WATER VOLUME (CUBIC FT) IN FAULTED STEAM GENERATOR (SG1), INTACT STEAM GENERATORS (SG2,3,4). INDICATED WATER LEVEL IN PRESSURIZER (PR LEVEL), PERCENT OF SPAN.

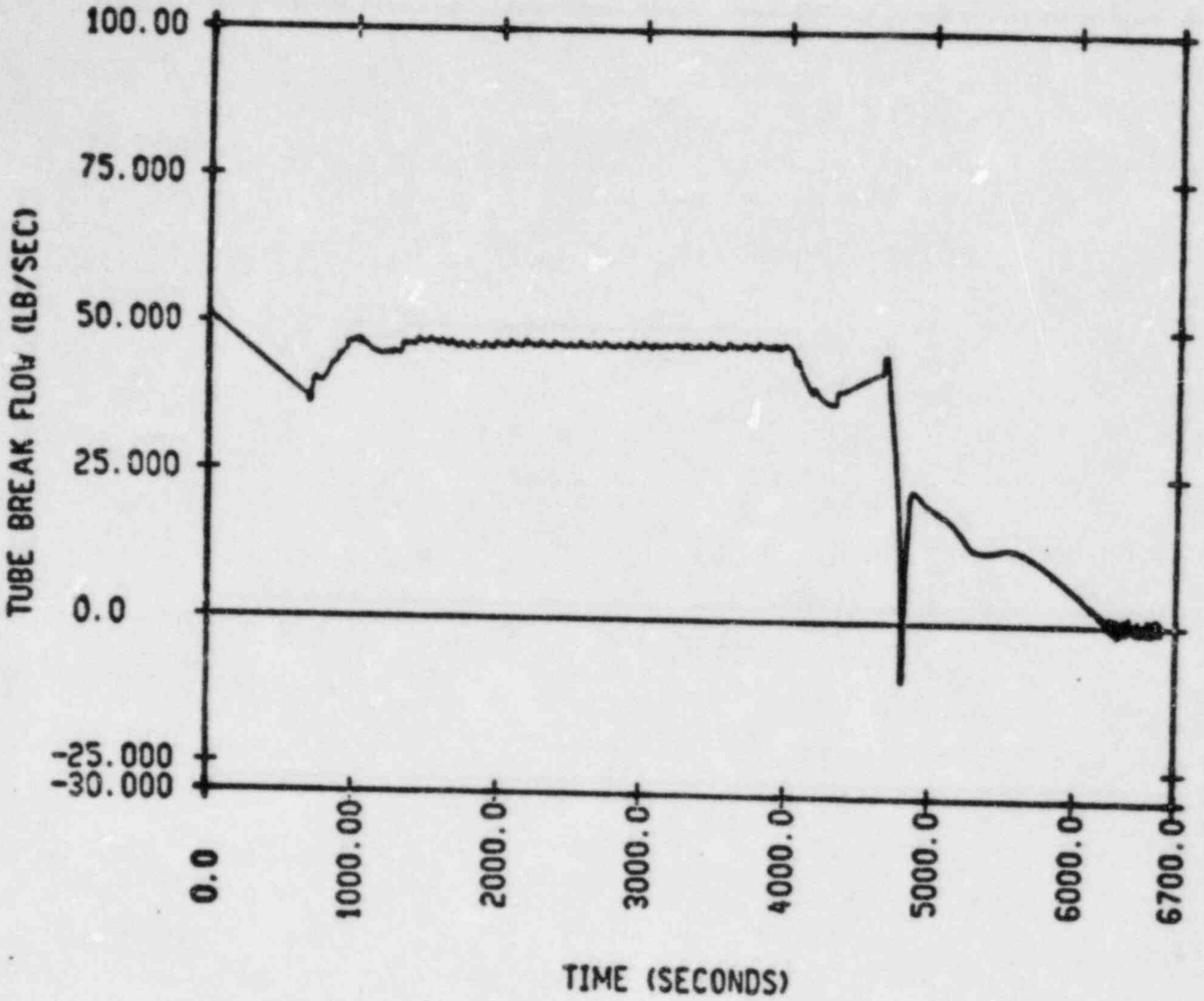

FIGURE 15.6-4 STEAM GENERATOR TUBE RUPTURE BREAK FLOW RATE VS TIME

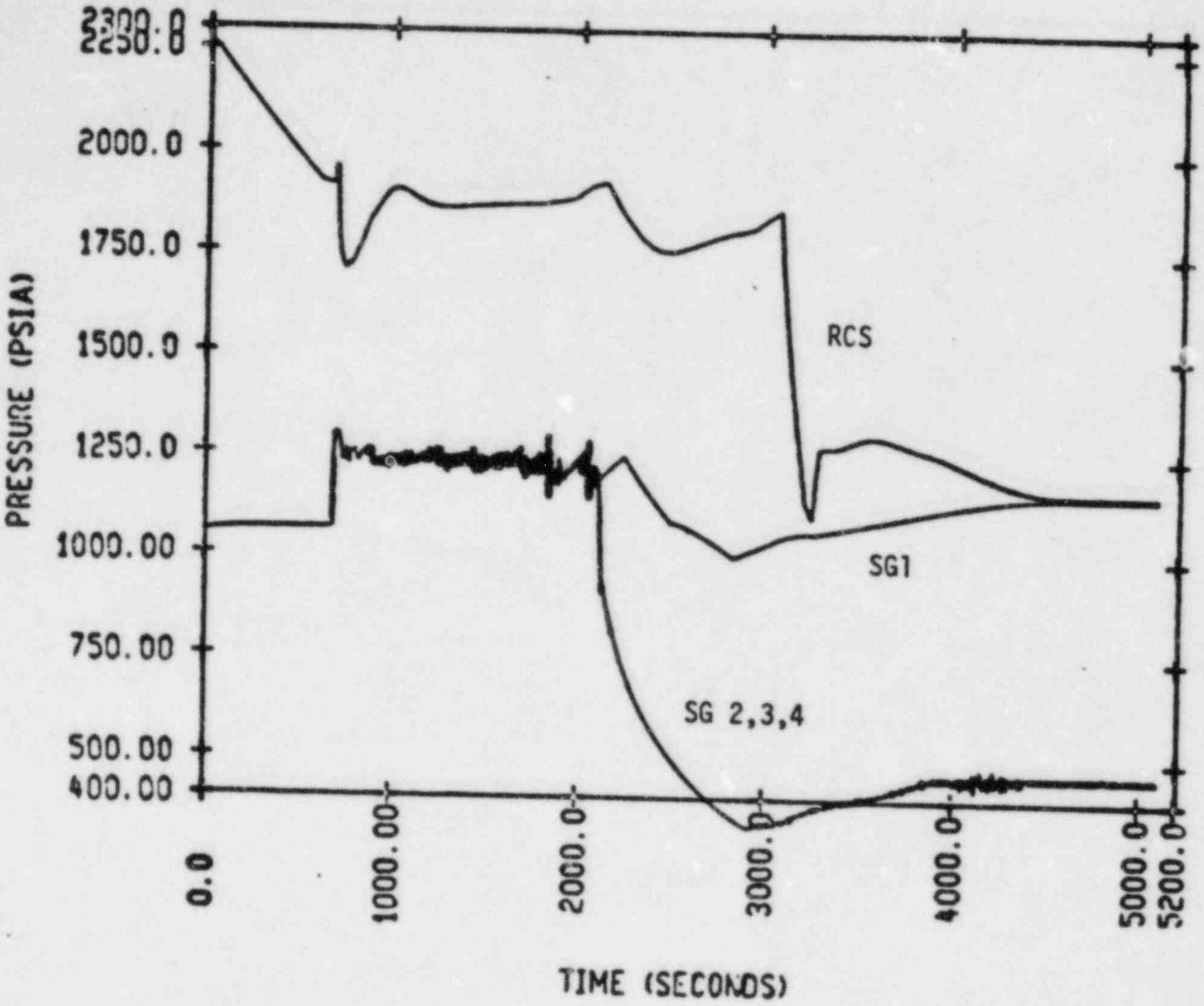

FIGURE 15.6-5 STEAM CENERATOR TUBE RUPTURE, HIGH WATER LEVEL, PRESSURE RESPONSE VS TIME IN REACTOR COOLANT SYSTEM (RCS), FAULTED STEAM GENERATOR (SG1) AND INTACT STEAM GENERATORS (SG 2,3,4)

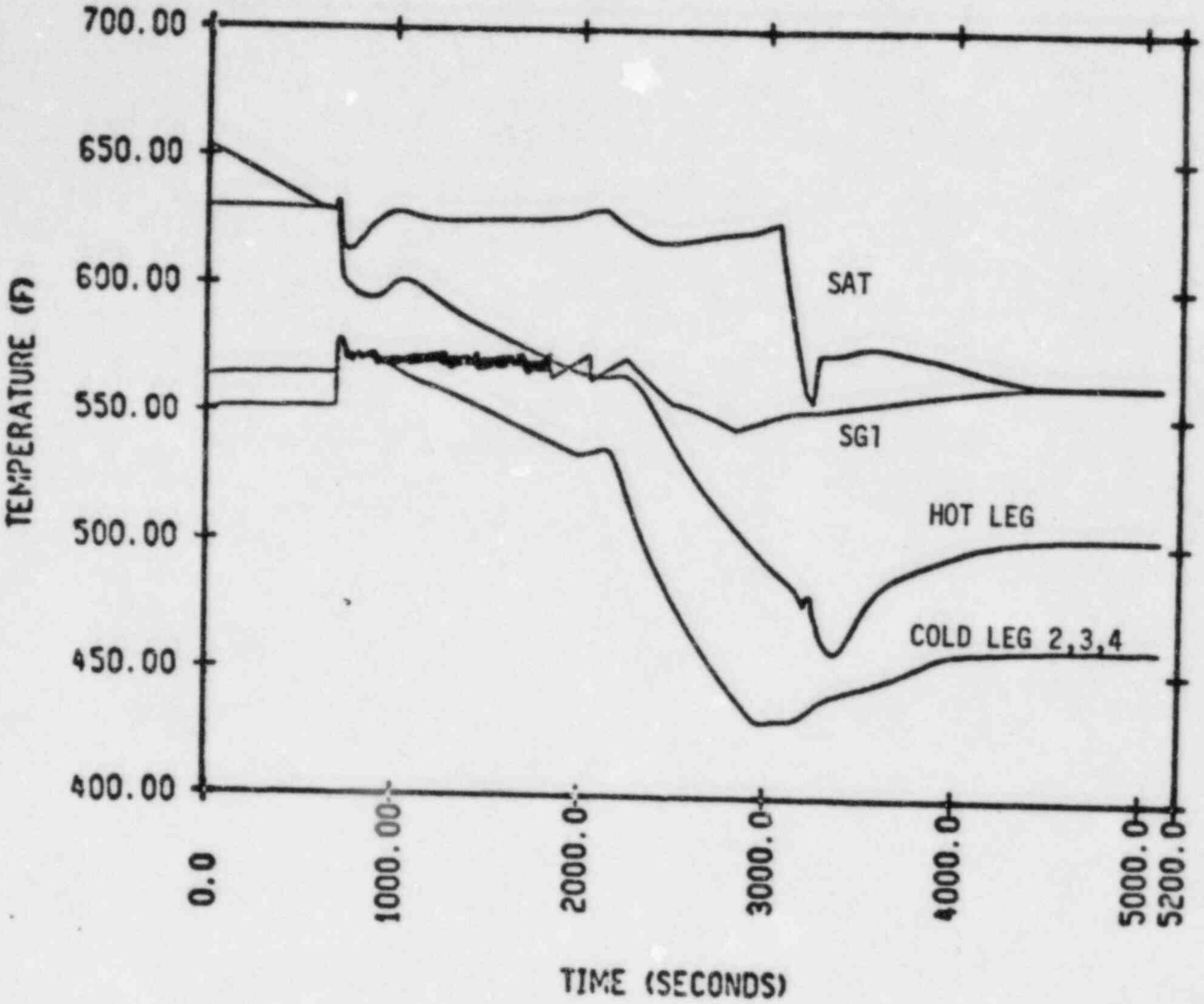

FIGURE 15.6-6 STEAM GENERATOR TUBE RUPTURE, HIGH WATER LEVEL, TEMPERATURE RESPONSE VS TIME, PRESSURIZER SATURATION TEMPERATURE (SAT), HOT LEG INLET TEMPERATURE, FAULTED STEAM GENERATOR STEAM TEMPERATURE (SG1), AND INTACT LOOP COLD LEG TEMPERATURES

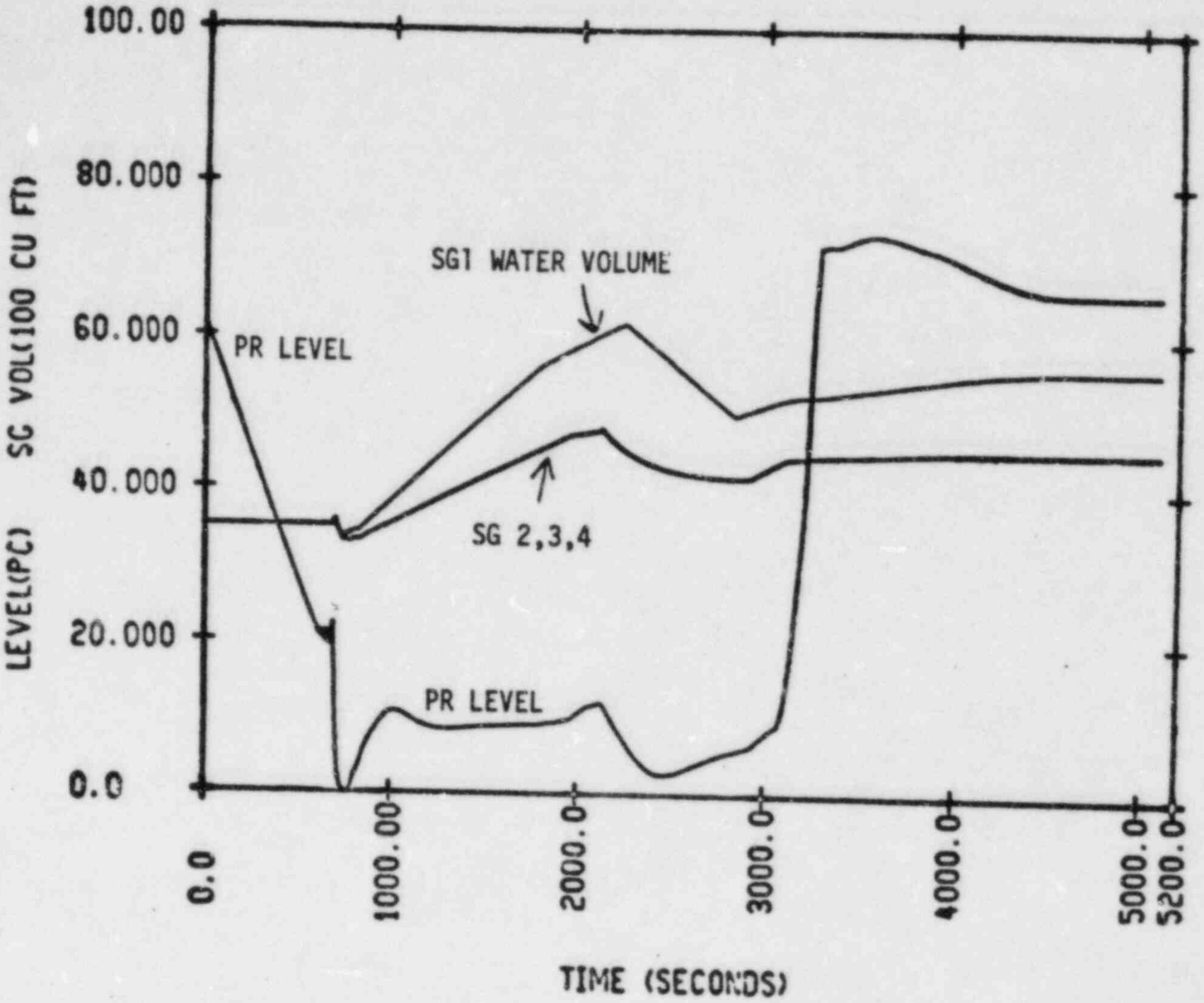

FIGURE 15.6-7 STEAM GENERATOR TUBE RUPTURE, HIGH WATER LEVEL, WATER VOLUME AND LEVELS VS TIME

WATER VOLUME (CUBIC FT) IN FAULTED STEAM GENERATOR (SG1) AND INTACT STEAM GENERATORS (SG 2,3,4). INDICATED WATER LEVEL IN PRESSURIZER (PR LEVEL), PERCENT OF SPAN.

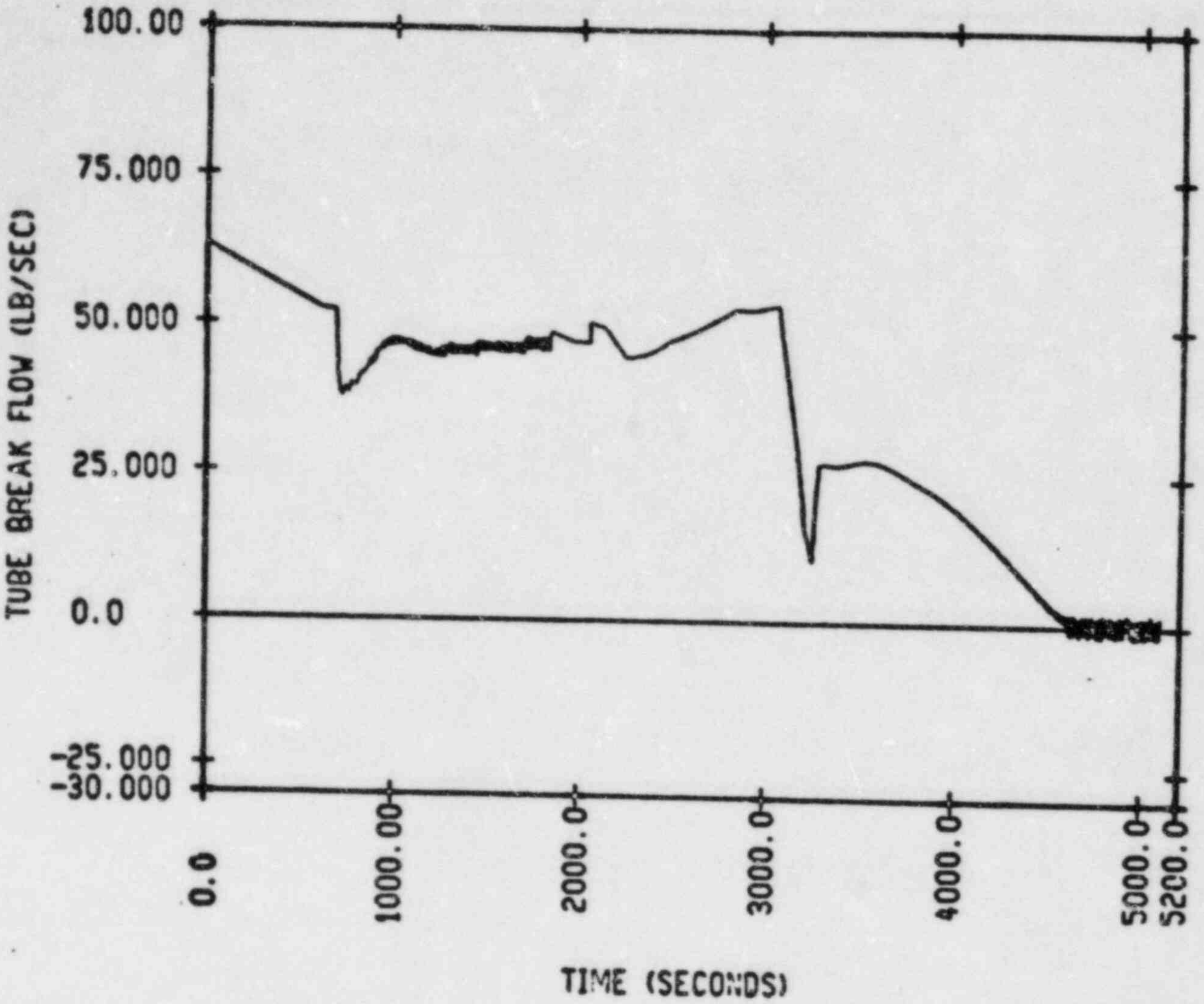

FIGURE 15.6-8 STEAM GENERATOR TUBE RUPTURE, HIGH WATER LEVEL, BREAK FLOW RATE RESPONSE VS TIME

APPENDIX 15A[1]

ACCIDENT ANALYSIS RADIOLOGICAL CONSEQUENCES EVALUATION MODELS AND PARAMETERS

15A.1 GENERAL ACCIDENT PARAMETERS

This appendix contains the parameters used in analyzing the radiological consequences of postulated accidents. Table 15A-1 contains the general parameters used in all the accident analyses. For parameters specific only to particular accidents, refer to that accident parameter section. The site specific, ground-level release, short-term dispersion factors (For accidents, ground-level releases are assumed.) are based on Regulatory Guide 1.145 (reference 1) methodology and represent the 0.5-percent worst-sector meteorology and these are given in Table 15A-2. The thyroid (via inhalation pathway), beta-skin, and gamma body (via immersion pathway) dose factors based on reference 2 are given in Table 15A-3.

15A.2 OFFSITE RADIOLOGICAL CONSEQUENCES CALCULATIONAL MODELS

This section presents the models and equations used for calculating the integrated activity released to the environment, the accident flowpaths, and the equations for dose calculations. Two major release models are considered:

1. A single holdup system with no internal cleanup.

2. A holdup system wherein a two-region spray model is used for internal cleanup.

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(1) This Appendix is included in all RESAR-SP/90 Modules for which there are transients with potential radiological consequences.

15A.2.1 ACCIDENT RELEASE PATHWAYS

The release pathways for the major accidents are given in Figure 15A-1. The accident and their pathways are as follows:

A. Loss-of-Coolant Accident (LOCA)

Immediately following a postulated LOCA, the release of radioactivity from the containment in to the environment with the Integrated Safeguards System (ISS) in full operation. The release in this case is calculated using equations 6a and 6b which take into account a two-region spray model within the containment.

B. Control Assembly Ejection (CAE)

Radioactivity release to the environment due to the CAE accident is direct and unfiltered. The releases from the primary system are calculated using equation 5 which considers holdup in the single-region primary system (the spray removal is not assumed); the secondary (steam) releases via the relief valves are calculated without any holdup. The pathways for these releases are A-B and A'-B.

15A.2.2 SINGLE-REGION RELEASE MODEL

It is assumed that any activity released to the holdup system instantaneously diffuses to uniformly occupy the system volume.

The following equations are used to calculate the integrated activity released from postulated accidents.

$$A_1(t) = A_1(0)e^{-\lambda t} \qquad (1)$$

where $A_1(0)$ = initial source activity at time t_o, Ci

$A_1(t)$ = source activity at time t, Ci

λ_1 = total removal constant from primary holdup system, S^{-1}

$$\lambda_1 = \lambda_d + \lambda_{1\ell} + \lambda_r \qquad (2)$$

where

λ_d = decay removal constant, S^{-1}

$\lambda_{1\ell}$ = primary holdup leak or release rate, S^{-1}

λ_r = internal removal constant, i.e., sprays, plateout, etc.; S^{-1}

Thus, the direct release rate to the atmosphere from the primary holdup system

$$R_U(t) = \lambda_{1\ell}\,[A_1(t)] \qquad (3)$$

where:

$R_u(t)$ = unfiltered release rate (Ci/s)

The integrated activity release is the integral of the above equation.

$$IAR\ (t) = \int_o^t R_u(t)\,dt = \int_o^t \lambda_{1\ell}\,A_1(o)e^{-\lambda_1 t}\,dt \qquad (4)$$

This yields:

$$IAR(t) = (\lambda_{1\ell}\,A_1(o)/\lambda_1)\,(1 - e^{-\lambda_1 t}) \qquad (5)$$

15A.2.3 TWO-REGION SPRAY MODEL IN CONTAINMENT (LOCA)

A two-region spray model is used to calculate the integrated activity released to the environment. The model consists of sprayed and unsprayed regions in containment and a constant mixing rate between them.

As it is assumed that there are no sources after initial release of the fission products, the remaining processes are removal and transfer so that the multivolume containment is described by a system of coupled first-order differential equations.

For a two-region model, the above system reduces to

$$\frac{dA_1}{dt} = - \sum_{j=1}^{K_1} \lambda_{1j} A_1 - Q_{12} \frac{A_1}{V_1} + Q_{21} \frac{A_2}{V_2} \qquad (6a)$$

$$\frac{dA_2}{dt} = - \sum_{j=1}^{K_2} \lambda_{2j} A_2 - Q_{21} \frac{A_2}{V_2} + Q_{12} \frac{A_1}{V_1} \qquad (6b)$$

where

A_i = fission product activity in volume i, Ci

$Q_{i\ell}$ = transfer rate from volume i to volume ℓ, cc/s

V_i = volume of the ith compartment, cc

λ_{ij} = removal rate of the jth removal process in volume i, s^{-1}

K_i = total number of removal processes in the volume i

To calculate the integrated activity released to the atmosphere, the release rate of activity is first calculated. This is found from

$$R(t) = \sum_{i=1}^{2} \lambda_{1\ell} \; A(t) \qquad (7)$$

The integrated activity released from time $t_0 - t_1$ is then

$$IAR = \int_{t_o}^{t_1} R\,(t)\,dt$$

15A.2.4 OFFSITE THYROID DOSE CALCULATION MODEL

Offsite thyroid doses are calculated using the equation:

$$D_{TH} = \sum_i \; DCF_{THi} \; \sum_j \; (IAR)_{ij} \; (BR)_j \; (\chi/Q)_j \qquad (8)$$

where

$(IAR)_{ij}$ = integrated activity of isotope i released[a] during the time interval j, Ci

$(BR)_j$ = breathing rate during time interval j, m^3/s

$(\chi/Q)_j$ = offsite atmospheric dispersion factor during time interval j, s/m^3

DCF_{THi} = thyroid dose conversion factor via inhalation for isotope i rem/Ci

D_{TH} = thyroid dose via inhalation, rems

15A.2.5 OFFSITE BETA-SKIN DOSE CALCULATIONAL MODEL

Assuming a semi-infinite cloud of beta emitters, off-site beta-skin doses are calculated using the equation:

$$D_{BS} = \sum_i DCF_{Bi} \sum_j (IAR)_{ij} (\chi/Q)_j$$

where

D_{BS} = beta-skin dose in rem

DCF_{Bi} = beta-skin dose conversion factor for the ith isotope in rem-m^3/Ci-s

and $(IAR)_{ij}$ and $(\chi/Q)_j$ are defined in Subsection 15A.2.4.

15A.2.6 OFFSITE GAMMA-BODY DOSE CALCULATIONAL MODEL

Assuming a semi-infinite cloud of gamma emitters, offsite gamma-body doses are calculated using the equation:

---

a. No credit is taken for cloud depletion by ground deposition and radioactive decay during transport to the exclusion area boundary or the outer boundary of the low-population zone.

$$D_{\gamma B} = \sum_i DCF_{\gamma i} \sum_j (IAR)_{ij} (\chi/Q)_j$$

where

$(IAR)_{ij}$ and $(\chi Q)_j$ are defined in Section 15A.2.4.

and

$DCF_{\gamma i}$ = gamma-body dose conversion factor for the ith isotope in rem-m^3/Ci-s

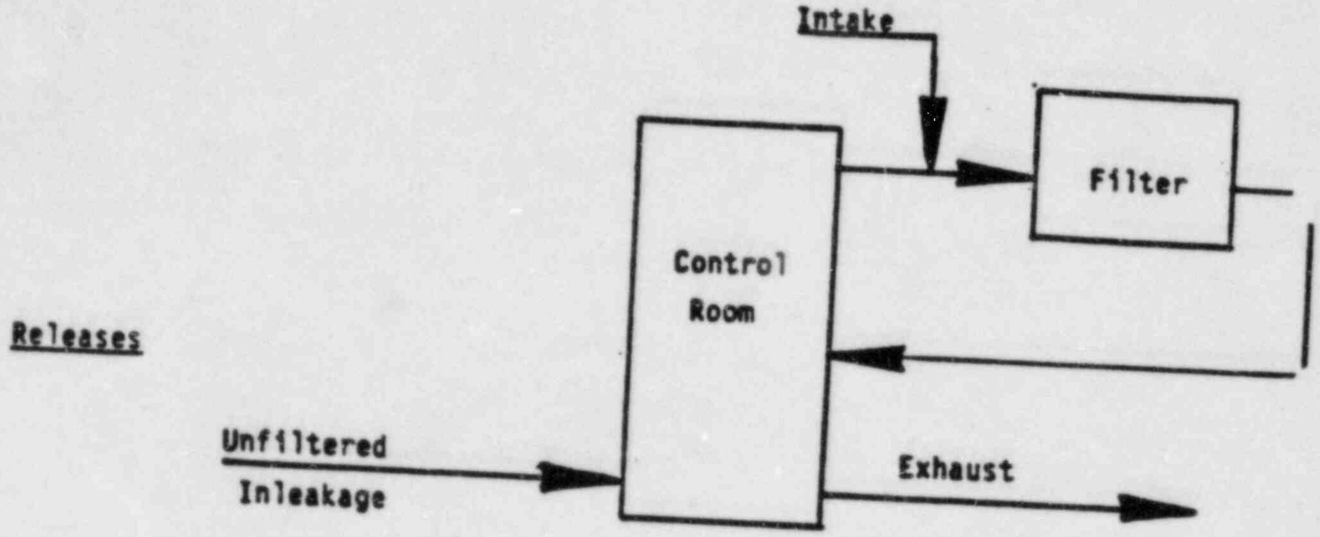
$D_{\gamma B}$ = gamma-body dose in rem

15A.3 CONTROL ROOM RADIOLOGICAL CONSEQUENCES CALCULATIONAL MODELS

Radiation doses to a control room operator as a result of a postulated LOCA are presented in this chapter. (A study of the radiological consequences in the control room due to various postulated accidents indicate that the LOCA is the limiting case.)

15A.3.1 INTEGRATED ACTIVITY IN CONTROL ROOM

The integrated activity in the control room during each time interval is found by multiplying the release by the appropriate χ/Q to give the concentration at the control room intake. This activity is brought into the control room through the filtered intake and by unfiltered inleakage. The control room ventilation system recirculates control room air through charcoal filters and exhausts a portion to the atmosphere.



From this we can calculate the total integrated activity in the control room during any time interval.

The activity in the control room can be calculated by the same method used to calculate activity in the containment.

15A.3.2 INTEGRATED ACTIVITY CONCENTRATION IN CONTROL ROOM FROM SINGLE-REGION SYSTEM

To calculate the integrated activity concentration in the control room we must first calculate the activity in the control room at any time t, and then integrate again to find the integrated activity.

$$\frac{dA_{CR}(t)}{dt} = [F_2 R_{FIN} + R_{UIN} \; \frac{\chi}{Q} \; R(t) - \lambda_3 A_{CR}(t)$$

where:

$A_{CR}(t)$ = activity in the control room at any time t, Ci

F_2 = filter nonremoval fraction on intake

R_{FIN} = filtered intake rate in m^3/s

R_{UIN} = unfiltered intake rate in m^3/s

R(t) = activity of release in Ci/s given in equation 3 of subsection 15A.2.2

$\lambda_3 = \lambda_{3\ell} + \lambda_d + \lambda_r$

where

λ_3 = total removal rate from control room in s^{-1}

$\lambda_{3\ell}$ = exhaust rate from control room in s^{-1}

λ_d = isotopic decay constant in s^{-1}

λ_r = recirculation removal rate in s^{-1}

The integrated activity in the control room (IA_{CR}) is determined by the expression

$$IA_{CR}(t) = \frac{1}{V_{CR}} \int_0^t A_{CR}(t)\ dt$$

Where: V_{CR} = control room volume

This $IA_{CR}(t)$ is used to calculate the doses to the operator in the control room. This activity is multiplied by an occupancy factor which accounts for the time fraction the operator is in the control room.

15A.3.3 CONTROL ROOM THYROID DOSE CALCULATIONAL MODEL

Control room thyroid doses via inhalation pathway are calculated using the following equation:

$$D_{TH-CR} = BR \sum_{i} DCF_{THi} \sum_{j} (IA_{CRij})\,(O_j)$$

where

D_{TH-CR} = control room thyroid dose in rem

BR = breathing rate assumed to be always 3.47×10^{-4} m^3/s

DCF_{THi} = thyroid dose conversion factor for adult via inhalation in rem/Ci for isotope i

IA_{CRij} = integrated activity concentration in control room, $Ci\text{-}s/m^3$ for isotope i during time interval j

o_j = control room occupancy fraction during time interval j

15A.3.4 CONTROL ROOM BETA-SKIN DOSE CALCULATIONAL MODEL

The beta-skin doses to a control room operator are calculated using the following equation:

$$D_{\beta-CR} = {}_{i}\; DCF_{\beta i}\; {}_{j}\; (IA_{CRij}) \times o_j$$

$D_{\beta-CR}$ = beta skin dose in the control room (rem).

$DCF_{\beta i}$ = beta skin dose conversion factor for isotope i ($rem\text{-}m^3/Ci\text{-}s$)

IA_{CRij} = integrated activity concentration in the control room, Ci-s for isotope i during time interval jm^3.

o_j = control room occupancy fraction during time interval j.

15A.3.5 CONTROL ROOM GAMMA-BODY DOSE CALCULATION

Due to the finite structure of the control room, the gamma-body doses to a control room operator will be substantially less than wha. tney would be due to immersion in an infinite cloud of gamma emitters. The finite cloud gamma doses are calculated using Murphy's method (reference 3) which models the control room at a hemisphere. The following equation is used:

$$D_{B-CR} = \frac{1}{GF} \sum_i DCF_i \sum_j (IA_{CRij}) (O_j)$$

where

GF = dose reduction due to control room geometry factor

GF = $1173/V_1^{0.338}$

V_1 = volume of the control room, ft^3

DCF_i = gamma-body dose conversion factor for isotope i, rem-m^3/Ci-s

D_{B-CR} = gamma-body dose in the control room, rem

Other symbols have been defined in Subsections 15A.2.5 and 15A.3.3.

15A.3.5.1 Model for Radiological Consequences Due to Radioactive Cloud External to the Control Room

This dose is calculated based on the semi-infinite cloud model which is modified using the protection factors described in Subsection 7.5.4 of reference 4 to account for the control room walls.

15A.4 REFERENCES

1. USNRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," August 1979.

2. USNRC Regulatory Guide 1.109, Rev. 1, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50 Appendix I," October 1977.

3. Murphy, K. G., and Campe, K. M., "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," Paper presented at the 13th AEC Air Cleaning Conference.

4. "Meteorology and Atomic Energy 1968," D. H. Slade (ed.), USAEC Report, TID 24190, 1968.

TABLE 15A-1

PARAMETERS USED IN ACCIDENT ANALYSIS

General	',c)
Core power level, MWt	
Full-power operation, effective full-power days (EFPD)	
Maximum radial peaking factor	
Steam generator tube leak rate, gal/min	
Sources	
Activity Release Parameters	
Free volume of containment, ft^3	
Containment leak rate	
0-24 h, percent per day	
After 24 h, percent per day	
Control room	
Free volume, ft^3	
Unfiltered infiltration rate, ft^3/min	
Filtered intake rate, ft^3/min	
Internal recirculation rate through filters, ft^3/min	
Iodine removal efficiency for recirculation filters (all forms of iodine), percent	95
Iodine removal efficiency for intake filters (all forms of iodine), percent	95
High efficiency particulate air filter efficiency, percent	99
Miscellaneous	
Atmospheric dispersion factors (χ/Q), s/m^3	Table 15A-2
Dose conversion factors	
Gamma-body and beta skin, rem-m^3/Ci-s	Table 15A-4
Thyroid, rem/Ci	Table 15A-4

TABLE 15A-2

LIMITING SHORT-TERM ATMOSPHERIC DISPERSION FACTORS FOR ACCIDENT ANALYSIS (s/m^3)*

Location Type/ Time Interval (h)	(χ/Q)
Site boundary	
0-2	2.0E-4
Low-population zone	
0-2	7.0E-5
2-8	3.5E-5
8-24	2.0E-5
24-96	9.0E-6
96-720	3.0E-6
Control room	
0-2	4.0E-3
2-8	3.0E-3
8-24	2.8E-3
24-96	2.0E-3
96-720	1.5E-3

* For the A. W. Vogtle Site.

TABLE 15A-3
DOSE CONVERSION FACTORS USED IN ACCIDENT ANALYSIS

Nuclide	Gamma-Body $\frac{Rem\text{-}m^3}{Ci\text{-}s}$	Beta-Skin $\frac{Rem\text{-}m^3}{Ci\text{-}s}$	Thyroid (Rem/Ci)
I-131	NA	NA	1.49E+6
I-132	NA	NA	1.43E+4
I-133	NA	NA	2.69E+5
I-134	NA	NA	3.73E+3
I-135	NA	NA	5.60E+4
Kr-85m	3.71E-2	4.63E-2	NA
Kr-85	5.11E-4	4.25E-2	NA
Kr-87	1.88E-1	3.09E-1	NA
Kr-88	4.67E-1	7.52E-2	NA
Xe-131m	2.91E-3	1.51E-2	NA
Xe-133m	7.97E-3	3.15E-2	NA
Xe-133	9.33E-3	9.70E-3	NA
Xe-135m	9.91E-2	2.25E-2	NA
Xe-135	5.75E-2	5.90E-2	NA
Xe-138	2.80E-1	1.31E-1	NA

TABLE 15A-4

DOSE CONVERSION FACTORS USED IN ACCIDENT ANALYSIS

Nuclide	Gamma-Body Rem-m^3/Ci-s	Beta-Skin Rem-m^3/Ci-s	Thyroid (Rem/Ci)
I-131	NA	NA	1.49E+6
I-132	NA	NA	1.43E+4
I-133	NA	NA	2.69E+5
I-134	NA	NA	3.73E+3
I-135	NA	NA	5.60E+4
Kr-85m	3.71E-2	4.63E-2	NA
Kr-85	5.11E-4	4.25E-2	NA
Kr-87	1.88E-1	3.09E-1	NA
Kr-88	4.67E-1	7.52E-2	NA
Xe-131m	2.91E-3	1.51E-2	NA
Xe-133m	7.97E-3	3.15E-2	NA
Xe-133	9.33E-3	9.70E-3	NA
Xe-135m	9.91E-2	2.25E-2	NA
Xe-135	5.75E-2	5.90E-2	NA
Xe-138	2.80E-1	1.31E-1	NA

TABLE 15A-5

REACTOR COOLANT IODINE CONCENTRATIONS FOR 1 μCi/GRAM AND 60 μCI/GRAM OF DOSE EQUIVALENT I-131

Nuclide	Reactor Coolant Concentration (Ci/gm)	
	1 μCi/gm D.E. I-131	60 μCi/gm D.E. I-131
I-131	0.76	45.6
I-132	0.76	45.6
I-133	1.14	68.4
I-134	0.195	11.7
I-135	0.63	37.8

TABLE 15A-6

IODINE APPEARANCE RATES IN THE REACTOR COOLANT (Curies/sec)

	*Equilibrium Appearance Rates due to Fuel Defects	**Appearance Rates Due to an Accident Initiated Iodine Spike
I-131	4.6×10^{-3}	2.3
I-132	2.5×10^{-2}	12.6
I-133	9.9×10^{-3}	5.0
I-134	1.5×10^{-2}	7.6
I-135	9.5×10^{-3}	4.8

* Based on RCS concentration of 1 μ Ci/gm of dose equivalent I-131

** 500 x equilibrium appearance rate

TABLE 15A-7

REACTOR COOLANT NOBLE GAS SPECIFIC ACTIVITY BASED ON ONE PERCENT DEFECTIVE FUEL

Nuclide	Activity (μc/gram)
Kr-85m	2.0
Kr-85	7.3
Kr-87	1.3
Kr-88	3.6
Xe-131m	2.2
Xe-133m	1.7×10^{1}
Xe-133	2.7×10^{2}
Xe-135m	4.8×10^{-1}
Xe-135	7.2
Xe-138	6.4×10^{-1}

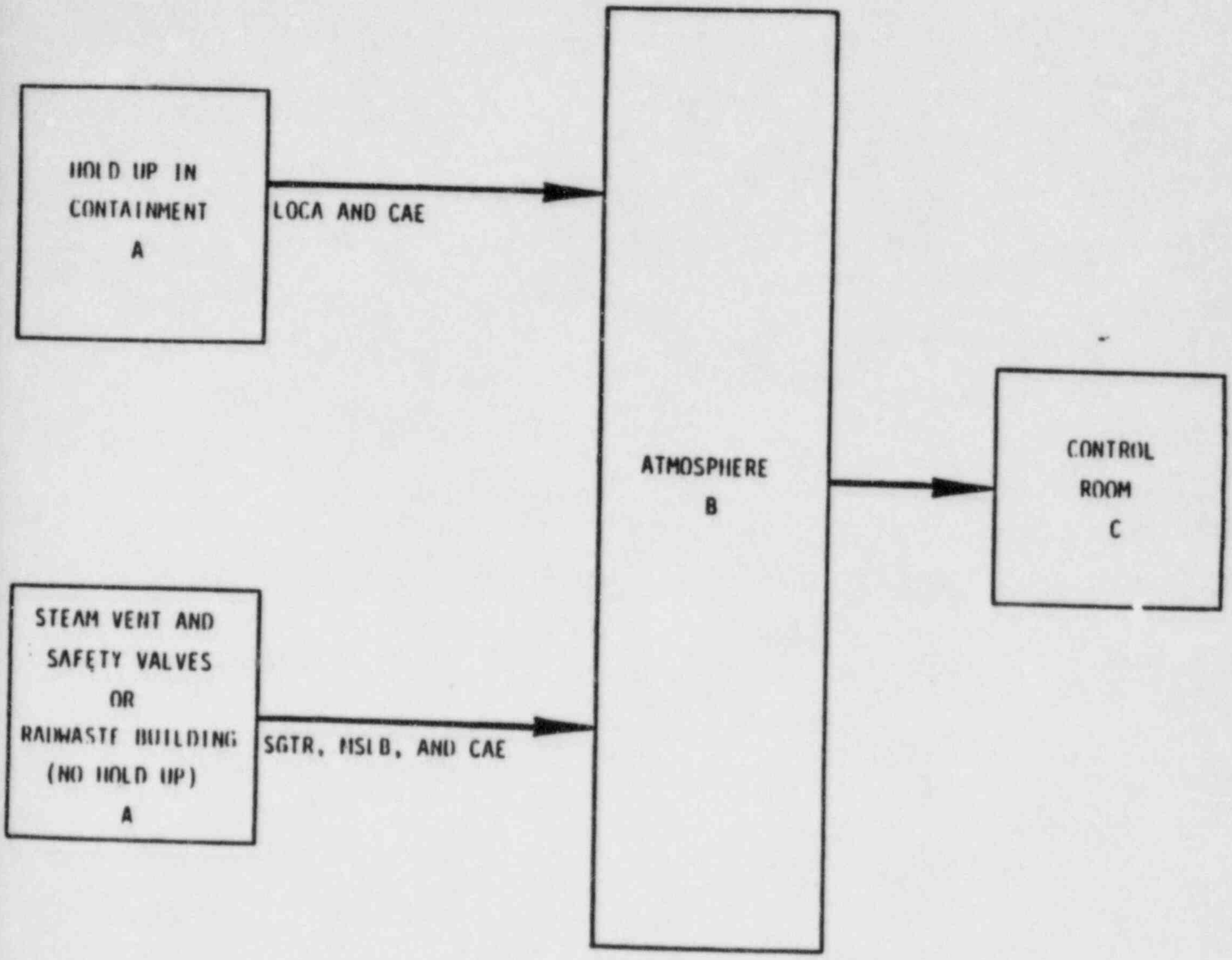


Figure 15.A-1 Release Pathways

16.0 TECHNICAL SPECIFICATIONS

No portion of this chapter is pertinent to the RESAR-SP/90 "Secondary Side Safeguards System/Steam and Power Conversion System" module.

17.0 QUALITY ASSURANCE

17.1 QUALITY ASSURANCE DURING DESIGN AND CONSTRUCTION

The Westinghouse Water Reactor Divisions Quality Assurance Program is described in Reference 1.

See the WAPWR integrated PDA submittal for a description of the complete WAPWR Quality Assurance Program, including modifications to reflect the expanded design and construction scope of the WAPWR Nuclear Power Block.

17.1.1 References

1. "Westinghouse Water Reactor Divisions Quality Assurance Plan," WCAP-8370, Revisions 9A, Amendment 1, February, 1981.