TABLE OF CONTENTS

CHAPTER 15.0

ACCIDENT ANALYSIS

Section	Title	Page
15.0	ACCIDENT ANALYSIS	15.0-1
15.0.1	CLASSIFICATION OF PLANT CONDITIONS	15.0-1
15.0.1.1	Condition I - Normal Operation and Operational Transients	15.0-1
15.0.1.2	Condition II - Faults of Moderate Frequency	15.0-3
15.0.1.3	Condition III - Infrequent Faults	15.0-4
15.0.1.4	Condition IV - Limiting Faults	15.0-5
15.0.2	OPTIMIZATION OF CONTROL SYSTEMS	15.0-6
15.0.3	PLANT CHARACTERISTICS AND INITIAL CONDITIONS ASSUMED IN THE ACCIDENT ANALYSES	15.0-6
15.0.3.1	Design Plant Conditions	15.0-6
15.0.3.2	Initial Conditions	15.0-7
15.0.3.3	Power Distribution	15.0-8
15.0.4	REACTIVITY COEFFICIENTS ASSUMED IN THE ACCIDENT ANALYSES	15.0-9
15.0.5	ROD CLUSTER CONTROL ASSEMBLY INSERTION CHARACTERISTICS	15.0-9
15.0.6	TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES	15.0-10
15.0.7	INSTRUMENTATION DRIFT AND CALORIMETRIC ERRORS - POWER RANGE NEUTRON FLUX	15.0-11
15.0.8	PLANT SYSTEMS AND COMPONENTS AVAILABLE FOR MITIGATION OF ACCIDENT EFFECTS	15.0-11
15.0.9	FISSION PRODUCT INVENTORIES	15.0-12
15.0.9.1	Activities in the Core	15.0-12
15.0.9.2	Activities in the Fuel Pellet Clad Gap	15.0-12
15.0.10	RESIDUAL DECAY HEAT	15.0-13
15.0.10.1	Total Residual Heat	15.0-13
15.0.10.2	Distribution of Decay Heat Following Loss-of-Coolant Accident	15.0-13
15.0.11	COMPUTER CODES UTILIZED	15.0-13
15.0.11.1	FACTRAN	15.0-14
15.0.11.2	LOFTRAN	15.0-14
15.0.11.3	PHOENIX-P	15.0-15

TABLE OF CONTENTS (CONTINUED)

Section	Title	Page
15.0.11.4	ANC	15.0-15
15.0.11.5	TWINKLE	15.0-15
15.0.11.6	APOLLO	15.0-16
15.0.11.7	RETRAN-02	15.0-16
15.0.11.8	VIPRE-01	15.0-16
15.0.12	LIMITING SINGLE FAILURES	15.0-16
15.0.13	OPERATOR ACTIONS	15.0-17
15.0.14	REFERENCES	15.0-19
15.1	INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM	15.1-1
15.1.1	FEEDWATER SYSTEM MALFUNCTIONS THAT RESULT IN A DECREASE IN FEEDWATER TEMPERATURE	15.1-1
15.1.1.1	Identification of Causes and Accident Description	15.1-1
15.1.1.2	Analysis of Effects and Consequences	15.1-2
15.1.1.3	Conclusions	15.1-3
15.1.2	FEEDWATER SYSTEM MALFUNCTIONS THAT RESULT IN AN INCREASE IN FEEDWATER FLOW	15.1-3
15.1.2.1	Identification of Causes and Accident Description	15.1-3
15.1.2.2	Analysis of Effects and Consequences	15.1-4
15.1.2.3	Conclusions	15.1-6
15.1.3	EXCESSIVE INCREASE IN SECONDARY STEAM FLOW	15.1-7
15.1.3.1	Identification of Causes and Accident Description	15.1-7
15.1.3.2	Analysis of Effects and Consequences	15.1-7
15.1.3.3	Conclusions	15.1-9
15.1.4	INADVERTENT OPENING OF A STEAM GENERATOR ATMOSPHERIC RELIEF OR SAFETY VALVE	15.1-10
15.1.4.1	Identification of Causes and Accident Description	15.1-10
15.1.4.2	Analysis of Effects and Consequences	15.1-11
15.1.4.3	Conclusions	15.1-13

TABLE OF CONTENTS (CONTINUED)

Section	Title	Page
15.1.5	STEAM SYSTEM PIPING FAILURE	15.1-13
15.1.5.1	Identification of Causes and Accident Description	15.1-13
15.1.5.2	Analysis of Effects and Consequences	15.1-16
15.1.5.3	Radiological Consequences	15.1-21
15.1.5.4	Conclusions	15.1-25
15.1.5.5	Notes	15.1-25
15.1.6	REFERENCES	15.1-26
15.2	DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM	15.2-1
15.2.1	STEAM PRESSURE REGULATOR MALFUNCTION OR FAILURE THAT RESULTS IN DECREASING STEAM FLOW	15.2-1
15.2.2	LOSS OF EXTERNAL ELECTRICAL LOAD	15.2-2
15.2.2.1	Identification of Causes and Accident Description	15.2-2
15.2.2.2	Analysis of Effects and Consequences	15.2-4
15.2.2.3	Conclusions	15.2-4
15.2.3	TURBINE TRIP	15.2-4
15.2.3.1	Identification of Causes and Accident Description	15.2-4
15.2.3.2	Analysis of Effects and Consequences	15.2-5
15.2.3.3	Conclusions	15.2-9
15.2.4	INADVERTENT CLOSURE OF MAIN STEAM ISOLATION VALVES	15.2-10
15.2.5	LOSS OF CONDENSER VACUUM AND OTHER EVENTS RESULTING IN TURBINE TRIP	15.2-10
15.2.6	LOSS OF NONEMERGENCY AC POWER TO THE STATION AUXILIARIES	15.2-10
15.2.6.1	Identification of Causes and Accident Description	15.2-10
15.2.6.2	Analysis of Effects and Consequences	15.2-12
15.2.6.3	Radiological Consequences	15.2-14
15.2.6.4	Conclusions	15.2-16
15.2.7	LOSS OF NORMAL FEEDWATER FLOW	15.2-17
15.2.7.1	Identification of Causes and Accident Description	15.2-17
15.2.7.2	Analysis of Effects and Consequences	15.2-18
15.2.7.3	Conclusions	15.2-20

TABLE OF CONTENTS (CONTINUED)

Section	Title	Page
15.2.8	FEEDWATER SYSTEM PIPE BREAK	15.2-20
15.2.8.1	Identification of Causes and Accident Description	15.2-20
15.2.8.2	Analysis of Effects and Consequences	15.2-22
15.2.8.3	Conclusions	15.2-26
15.2.9	REFERENCES	15.2-26
15.3	DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE	15.3-1
15.3.1	PARTIAL LOSS OF FORCED REACTOR COOLANT FLOW	15.3-1
15.3.1.1	Identification of Causes and Accident Description	15.3-1
15.3.1.2	Analysis of Effects and Consequences	15.3-2
15.3.1.3	Conclusions	15.3-4
15.3.2	COMPLETE LOSS OF FORCED REACTOR COOLANT FLOW	15.3-4
15.3.2.1	Identification of Causes and Accident Description	15.3-4
15.3.2.2	Analysis of Effects and Consequences	15.3-5
15.3.2.3	Conclusions	15.3-6
15.3.3	REACTOR COOLANT PUMP SHAFT SEIZURE (LOCKED ROTOR)	15.3-6
15.3.3.1	Identification of Causes and Accident Description	15.3-6
15.3.3.2	Analysis of Effects and Consequences	15.3-7
15.3.3.3	Radiological Consequences	15.3-11
15.3.3.4	Conclusions	15.3-15
15.3.4	REACTOR COOLANT PUMP SHAFT BREAK	15.3-15
15.3.4.1	Identification of Causes and Accident Description	15.3-15
15.3.4.2	Conclusions	15.3-16
15.3.5	REFERENCES	15.3-16
15.4	REACTIVITY AND POWER DISTRIBUTION ANOMALIES	15.4-1
15.4.1	UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER STARTUP CONDITION	15.4-2
15.4.1.1	Identification of Causes and Accident Description	15.4-2
15.4.1.2	Analysis of Effects and Consequences	15.4-4
15.4.1.3	Conclusions	15.4-6
15.4.2	UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL AT POWER	15.4-7
15.4.2.1	Identification of Causes and Accident Description	15.4-7

TABLE OF CONTENTS (CONTINUED)

Section	Title	Page
15.4.2.2	Analysis of Effects and Consequences	15.4-8
15.4.2.3	Conclusions	15.4-12
15.4.3	ROD CLUSTER CONTROL ASSEMBLY MISOPERATION	15.4-12
15.4.3.1	Identification of Causes and Accident Description	15.4-12
15.4.3.2	Analysis of Effects and Consequences	15.4-14
15.4.3.3	Conclusions	15.4-17
15.4.4	STARTUP OF AN INACTIVE REACTOR COOLANT PUMP AT AN INCORRECT TEMPERATURE	15.4-17
15.4.4.1	Identification of Causes and Accident Description	15.4-17
15.4.4.2	Analysis of Effects and Consequences	15.4-18
15.4.4.3	Conclusions	15.4-19
15.4.5	A MALFUNCTION OR FAILURE OF THE FLOW CONTROLLER IN A BWR LOOP THAT RESULTS IN AN INCREASED REACTOR COOLANT FLOW RATE	15.4-19
15.4.6	CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT RESULTS IN A DECREASE IN THE BORON CONCENTRATION IN THE REACTOR COOLANT	15.4-19
15.4.6.1	Identification of Causes and Accident Description	15.4-19
15.4.6.2	Analysis of Effects and Consequences	15.4-21
15.4.6.3	Conclusions	15.4-22
15.4.7	INADVERTENT LOADING AND OPERATION OF A FUEL ASSEMBLY IN IMPROPER POSITION	15.4-24
15.4.7.1	Identification of Causes and Accident Description	15.4-24
15.4.7.2	Analysis of Effects and Consequences	15.4-25
15.4.7.3	Conclusions	15.4-26
15.4.8	SPECTRUM OF ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENTS	15.4-26
15.4.8.1	Identification of Causes and Accident Description	15.4-26
15.4.8.2	Analysis of Effects and Consequences	15.4-30
15.4.8.3	Radiological Consequences	15.4-36
15.4.8.4	Conclusions	15.4-41
15.4.9	REFERENCES	15.4-41

TABLE OF CONTENTS (CONTINUED)

Section	Title	Page
15.5	INCREASE IN REACTOR COOLANT INVENTORY	15.5-1
15.5.1	INADVERTENT OPERATION OF THE EMERGENCY CORE COOLING SYSTEM DURING POWER OPERATION	15.5-1
15.5.1.1	Identification of Causes and Accident Description	15.5-1
15.5.1.2	Analysis of Effects and Consequences	15.5-2
15.5.1.3	Conclusions	15.5-4
15.5.2	CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT INCREASES REACTOR COOLANT INVENTORY	15.5-4
15.5.2.1	Identification of Causes and Accident Description	15.5-4
15.5.2.2	Analysis of Effects and Consequences	15.5-5
15.5.2.3	Conclusions	15.5-7
15.5.3	A NUMBER OF BWR TRANSIENTS	15.5-7
15.5.4	REFERENCES	15.5-7
15.6	DECREASE IN REACTOR COOLANT INVENTORY	15.6-1
15.6.1	INADVERTENT OPENING OF A PRESSURIZER SAFETY OR RELIEF VALVE	15.6-1
15.6.1.1	Identification of Cause and Accident Description	15.6-1
15.6.1.2	Analysis of Effects and Consequences	15.6-2
15.6.1.3	Conclusions	15.6-3
15.6.2	BREAK IN INSTRUMENT LINE OR OTHER LINES FROM REACTOR COOLANT PRESSURE BOUNDARY THAT PENETRATE CONTAINMENT	15.6-3
15.6.2.1	Radiological Consequences	15.6-4
15.6.3	STEAM GENERATOR TUBE RUPTURE (SGTR)	15.6-7
15.6.3.1	Steam Generator Tube Rupture with Failure of Faulted Steam Generator AFW Control Valve	15.6-7
15.6.3.1.1	Identification of Causes and Accident Description	15.6-7
15.6.3.1.2	Analysis of Effects and Consequences	15.6-9
15.6.3.2	Steam Generator Tube Rupture with Postulated Stuck-Open Atmospheric Relief Valve	15.6-12
15.6.3.2.1	Identification of Causes and Accident Description	15.6-12
15.6.3.2.2	Analysis of Effects and Consequences	15.6-12

TABLE OF CONTENTS (CONTINUED)

Section	Title	Page
15.6.3.3	Radiological Consequences	15.6-14
15.6.3.4	Conclusions	15.6-17
15.6.3.5	References	15.6-18
15.6.4	SPECTRUM OF BWR STEAM SYSTEM PIPING FAILURES OUTSIDE OF CONTAINMENT	15.6-18
15.6.5	LOSS-OF-COOLANT ACCIDENTS RESULTING FROM A SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY	15.6-18
15.6.5.1	Identification of Causes and Frequency Classification	15.6-18
15.6.5.2	Sequence of Events and Systems Operations	15.6-19
15.6.5.3	Core and System Performance	15.6-21
15.6.5.4	Radiological Consequences	15.6-28
15.6.6	A NUMBER OF BWR TRANSIENTS	15.6-33
15.6.7	REFERENCES	15.6-33
15.7	RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT	15.7-1
15.7.1	RADIOACTIVE WASTE GAS DECAY TANK FAILURE	15.7-1
15.7.1.1	Identification of Causes	15.7-1
15.7.1.2	Sequence of Events and System Operations	15.7-1
15.7.1.3	Core and System Performance	15 7-2
15.7.1.4	Barrier Performance	15.7-2
15.7.1.5	Radiological Consequences	15.7-2
15.7.2	RADIOACTIVE LIQUID WASTE SYSTEM LEAK OR FAILURE	15.7-5
15.7.2.1	Identification of Causes	15.7-5
15.7.2.2	Sequence of Events and System Operation	15.7-6
15.7.2.3	Core and System Performance	15.7-6
15.7.2.4	Barrier Performance	15.7-6
15.7.2.5	Radiological Consequences	15.7-6
15.7.3	POSTULATED RADIOACTIVE RELEASE DUE TO LIQUID TANK FAILURES	15.7-9
15.7.4	FUEL HANDLING ACCIDENTS	15.7-9
15.7.4.1	Identification of Causes and Accident Description	15.7-9
15.7.4.2	Sequence of Events and Systems Operations	15.7-9
15.7.4.3	Core and System Performance	15.7-9
15.7.4.4	Barrier Performance	15.7-9
15.7.4.5	Radiological Consequences	15.7-10

TABLE OF CONTENTS (CONTINUED)

Section	Title	Page
15.7.5	SPENT FUEL CASK DROP ACCIDENTS	15.7-17
15.8	ANTICIPATED TRANSIENTS WITHOUT SCRAM	15.8-1
App. 15A	ACCIDENT ANALYSIS RADIOLOGICAL CONSEQUENCES EVALUATION MODELS AND PARAMETERS	15A-1
15A.1	GENERAL ACCIDENT PARAMETERS	15A-1
15A.2	OFFSITE RADIOLOGICAL CONSEQUENCES CALCULATIONAL MODELS	15A-1
15A.2.1	Accident Release Pathways	15A-1
15A.2.2	Single-Region Release Model	15A-2
15A.2.3	Two-Region Spray Model in Containment (LOCA)	15A-3
15A.2.4	Offsite Thyroid Dose Calculational Model	15A-7
15A.2.5	Offsite Total-Body Dose Calculational Model	15A-8
15A.3	CONTROL ROOM RADIOLOGICAL CONSEQUENCES CALCULATIONAL MODELS	15A-8
15A.3.1	Integrated Activity in Control Room	15A-8
15A.3.2	Control Room Thyroid Dose Calculational Model	15A-11
15A.3.3	Control Room Beta-Skin Dose Calculational Model	15A-12
15A.3.4	Control Room Total-Body Dose Calculation	15A-12
15A.3.4.1	Model for Radiological Consequences Due to Radioactive Cloud External to the Control Room	15A-13
15A.4	REFERENCES	15A-13

TABLE OF CONTENTS (CONTINUED)

LIST OF TABLES

Table No.	Title
15.0-1	Nuclear Steam Supply System Power Ratings
15.0-2	Summary of Initial Conditions and Computer Codes Used
15.0-3	Nominal Values of Pertinent Plant Parameters Utilized in the Accident Analyses
15.0-4	Trip Points and Time Delays to Trip Assumed in Accident Analyses
15.0-5	Determination of Maximum Overpower Trip Point - Power Range Neutron Flux Channel - Based on Nominal Setpoint Considering Inherent Instrumentation Errors
15.0-6	Major Plant System and Equipment Available for Mitigation of Transient and Accident Conditions
15.0-7	Single Failures Assumed in Accident Analyses
15.0-8	Operator Actions Required for Small and Large LOCAs
15.0-9	Short Term Operator Actions Required for Steam Generator Tube Rupture
15.1-1	Time Sequence of Events for Incidents that Result in an Increase in Heat Removal by the Secondary System
15.1-2	Equipment Required Following a Rupture of a Main Steam Line
15.1-3	Parameters Used in Evaluating the Radiological Consequences of a Main Steam Line Break
15.1-4	Radiological Consequences of a Main Steam Line Break
15.2-1	Time Sequence of Events for Incidents Which Result in a Decrease in Heat Removal by the Secondary System
15.2-2	Parameters Used in Evaluating Radiological Consequences of Loss of Nonemergency AC Power
15.2-3	Radiological Consequences of Loss of Non-Emergency AC Power
15.2-4	Natural Circulation Flow

TABLE OF CONTENTS (CONTINUED)

LIST OF TABLES

Table No.	Title
15.3-1	Time Sequence of Events for Incidents Which Result in a Decrease in Reactor Coolant System Flow Rate
15.3-2	Summary of Results for Locked Rotor Transients
15.3-3	Parameters Used in Evaluating the Radiological Consequences of a Locked Rotor Accident
15.3-4	Radiological Consequences of a Locked Rotor Accident
15.4-1	Time Sequence of Events for Incidents Which Result in Reactivity and Power Distribution Anomalies
15.4-2	Parameters Used in the Analysis of the Rod Cluster Control Assembly Ejection Accident
15.4-3	Parameters Used in Evaluating the RCCA Ejection Accident
15.4-4	Radiological Consequences of a Rod-Ejection Accident
15.5-1	Time Sequence of Events for Incidents Which Result in an Increase in Reactor Coolant Inventory
15.6-1	Time Sequence of Events for Incident Which Results in a Decrease in Reactor Coolant Inventory
15.6-2	Parameters Used in Evaluating the Radiological Consequence of the CVCS Letdown Line Rupture Outside of Containment
15.6-3	Radiological Consequences of a CVCS Letdown Line Break Outside of Containment
15.6-4	Parameters Used in Evaluating the Radiological Consequences of a Steam Generator Tube Rupture (SGTR) With Forced Overfill
15.6-5	Radiological Consequences of a Steam Generator Tube Rupture

TABLE OF CONTENTS (CONTINUED)

LIST OF TABLES

Table No.	Title
15.6-6	Parameters Used in Evaluating the Radiological Consequences of a Loss-of-Coolant-Accident
15.6-7	Design Comparison to the Regulatory Positions of Regulatory Guide 1.4 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," Revision 2, June 1974
15.6-8	Radiological Consequences of a Loss-of-Coolant Accident
15.6-9	Input Parameters Used in the ECCS Analysis
15.6-10	Time Sequence of Events for Loss-of-Coolant Accidents
15.6-11	Large Break LOCAs Results Fuel Cladding Data
15.6-12	Small Break LOCAs Results Fuel Cladding Data
15.6-13	Safety Injection Pumped Flow Assumed for Large Break LOCAs
15.6-14	Safety Injection Pumped Flow Assumed for Small Break LOCAs
15.7-1	Design Comparison to the Regulatory Positions of Regulatory Guide 1.24 "Assumptions Used for Evaluating the Potential Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure" Revision 0, Dated March 23, 1972
15.7-2	Design Comparison to the Regulatory Positions of Regulatory Guide 1.25 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors" Revision 0, Dated March 23, 1972
15.7-3	Parameters Used in Evaluating the Radiological Consequences of a Waste Gas Decay Tank Rupture
15.7-4	Radiological Consequences of a Waste Gas Decay Tank Rupture
15.7-5	Parameters Used in Evaluating the Radiological Consequences of a Liquid Radwaste Tank Failure
15.7-6	Radiological Consequences of a Liquid Radwaste Tank Failure

TABLE OF CONTENTS (CONTINUED)

LIST OF TABLES

Table No.	Title
15.7-7	Parameters Used in Evaluating the Radiological Consequences of a Fuel Handling Accident
15.7-8	Radiological Consequences of a Fuel Handling Accident
15A-1	Parameters Used in Accident Analysis
15A-2	Limiting Short-term Atmospheric Dispersion Factors (X/Qs) for Accident Analysis
15A-3	Fuel and Rod Gap Inventories - Core
15A-4	Dose Conversion Factors Used in Accident Analysis

CHAPTER 15 - LIST OF FIGURES

*Refer to Section 1.6 and Table 1.6-3. Controlled drawings were removed from the USAR at Revision 17 and are considered incorporated by reference.

Figure #	Sheet	Title	Drawing #*
15.0-1	0	Illustration of Overtemperature and Overpower T Protection	
15.0-2	0	Doppler Power Coefficient Used in Accident Analysis	
15.0-3	0	RCCA Position Versus Time to Dashpot	
15.0-4	0	Normalized Rod Worth Versus Percent Inserted	
15.0-5	0	Normalized RCCA Bank Reactivity Worth Versus Normalized Drop Time	
15.0-6	0	Minimum Moderator Density Coefficient Used in Analysis	
15.0-7	0	Abbreviations and Symbols Used in Sequence Diagrams	
15.0-8	0	Excessive Heat Removal Due to Feedwater Systems Malfunction	
15.0-9	0	Excessive Load Increase	
15.0-10	0	Depressurization of Main Steam System	
15.0-11	0	Loss of External Load	
15.0-12	0	Loss of Non-Emergency AC Power to the Station Auxiliaries	
15.0-13	0	Loss of Normal Feedwater	
15.0-14	0	Major Rupture of a Main Feedwater Line	
15.0-15	0	Loss of Forced Reactor Coolant Flow	
15.0-16	0	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal	
15.0-17	0	Dropped Rod Cluster Control Assembly	
15.0-18	0	Single Rod Cluster Control Assembly Withdrawal at Full Power	
15.0-19	0	Startup of an Inactive Reactor Coolant Loop	
15.0-20	0	Boron Dilution	
15.0-21	0	Rupture of Control Rod Drive Mechanism Housing	
15.0-22	0	Inadvertent ECCS Operation at Power	
15.0-23	0	Accidental Depressurization of Reactor Coolant System	
15.0-24	0	Steam Generator Tube Rupture	
15.0-25	0	Loss of Coolant Accident	
15.0-26	0	CVCS Letdown Line Rupture	
15.0-27	0	GWPS Gas Decay Tank Rupture	
15.0-28	0	Floor Drain Tank Failure	
15.0-29	0	Fuel Handling Accident in Fuel Building	
15.0-30	0	Fuel Handling Accident Inside Containment	
15.0-31	0	Spent Fuel Cask Drop Accident	

CHAPTER 15 - LIST OF FIGURES

*Refer to Section 1.6 and Table 1.6-3. Controlled drawings were removed from the USAR at Revision 17 and are considered incorporated by reference.

Figure #	Sheet	Title	Drawing #*
15.1-1	0	Nuclear Power, and Pressurizer Pressure Transients for Feedwater Control Valve Malfunction	
15.1-2	0	Reactor Coolant Loop ÷T, and Core Heat Flux Transient for Feedwater Control Valve Malfunction	
15.1-2A	0	Core Average Temperature Transient and Minimum DNBR for Feedwater Control Valve Malfunction	
15.1-3	0	Nuclear Power, Pressurizer Pressure and Water Volume Transients for Ten Percent Step Load Increase, Minimum Reactivity Feedback, Manual Reactor Control	
15.1-4	0	DNBR and Vessel Average Temperature Transients for Ten Percent Step Load Increase, Minimum Reactivity Feedback, Manual Reactor Control	
15.1-5	0	Nuclear Power, Pressurizer Pressure and Water Volume Transients for Ten Percent Step Load Increase, Maximum Reactivity Feedback, Manual Reactor Control	
15.1-6	0	DNBR and Vessel Average Temperature Transients for Ten Percent Step Load Increase, Minimum Reactivity Feedback, Manual Reactor Control	
15.1-7	0	Nuclear Power, Pressurizer Pressure and Water Volume Transients for Ten Percent Step Load Increase Minimum Reactivity Feedback, Automatic Reactor Control	
15.1-8	0	DNBR and Vessel Average Temperature Transients for Ten Percent Step Load Increase, Minimum Reactivity Feedback, Automatic Reactor Control	
15.1-9	0	Nuclear Power, Pressurizer Pressure and Water Volume Transients for Ten Percent Step Load Increase, Maximum Reactivity Feedback, Automatic Reactor Control	
15.1-10	0	DNBR and Vessel Average Temperature Transients for Ten Percent Step Load Increase, Maximum Reactivity Feedback, Automatic Reactor Control	
15.1-11	0	K_{eff} Versus Temperature	
15.1-12	0	Failure of a Steam Generator Safety or Atmospheric Relief Valve	
15.1-13	0	Failure of a Steam Generator Safety or Atmospheric Relief Valve	
15.1-14	0	Doppler Power Feedback	
15.1-15	0	Normalized Core Power and Normalized Core Heat Flux SLB with Offsite Power	
15.1-16	0	Pressurizer Pressure and Pressurizer Liquid Volume SLB with Offsite Power	

**The acronym fon, which is utilized on select Chapter 15 Figures, is defined as fraction of nominal.

CHAPTER 15 - LIST OF FIGURES

*Refer to Section 1.6 and Table 1.6-3. Controlled drawings were removed from the USAR at Revision 17 and are considered incorporated by reference.

Figure #	Sheet	Title	Drawing #*
15.1-17	0	Core Inlet Temperature and Core Average Temperature SLB with Offsite Power	
15.1-18	0	Total Core Reactivity and Core Boron Concentration SLB with Offsite Power	
15.1-19	0	Normalized Feedwater Flow and Break Flow SLB with Offsite Power	
15.1-20	0	Steamline Pressure and Core Inlet Flow SLB with Offsite Power	
15.1-21	0	Normalized Core Power and Normalized Core Heat Flux SLB Without Offsite Power	
15.1-22	0	Pressurizer Pressure and Pressurizer liquid Volume SLB Without Offsite Power	
15.1-23	0	Core Inlet Temperature and Core Average Temperature SLB Without Offsite Power	
15.1-24	0	Total core Reactivity and core Boron Concentration SLB Without Offsite Power	
15.1-25	0	Normalized Feedwater Flow and Break Flow SLB Without Offsite Power	
15.1-26	0	Steamline Pressure and Core Inlet Flow SLB Without Offsite Power	
15.2-1	0	Nuclear Power and Pressure for Turbine Trip Event with Minimum Reactivity Feedback Overpressure Evaluation	
15.2-2	0	Pressurizer Liquid Volume and RCS Temperature for Turbine Trip Event with Minimum Reactivity Feedback Overpressure Evaluation	
15.2-3	0	Nuclear Power and Pressure for Turbine Trip Event with Maximum Reactivity Feedback Overpressure Evaluation	
15.2-4	0	Pressurizer Liquid Volume and RCS Temperature for Turbine Trip Event with Maximum Reactivity Feedback Overpressure Evaluation	
15.2-5	0	Nuclear Power and Pressure for Turbine Trip Event with Minimum Reactivity Feedback DNB Evaluation	
15.2-6	0	Pressurizer Liquid Volume and RCS Temperature for Turbine Trip Event with Minimum Reactivity Feedback DNB Evaluation	
15.2-7	0	Nuclear Power and Pressure for Turbine Trip Event With Maximum Reactivity Feedback DNB Evaluation	
15.2-8	0	Pressurizer Liquid Volume and RCS Temperature for Turbine Trip Event With Maximum Reactivity Feedback DNB Evaluation	

**The acronym fon, which is utilized on select Chapter 15 Figures, is defined as fraction of nominal.

CHAPTER 15 - LIST OF FIGURES

*Refer to Section 1.6 and Table 1.6-3. Controlled drawings were removed from the USAR at Revision 17 and are considered incorporated by reference.

Figure #	Sheet	Title	Drawing #*
15.2-9	0	Nuclear Power, Reactor Coolant Mass Flow Rate and Steam Generator Pressure Transients for Loss of AC Power	
15.2-10	0	Reactor Coolant Temperature, Maximum RCS Pressure and Pressurizer Liquid Level for Loss of AC Power	
15.2-11	0	Nuclear Power and Steam Generator Pressure Transients for Loss of Normal Feedwater	
15.2-12	0	Core Average Temperature, Maximum RCS Pressure and Pressurizer Liquid Level Transients for Loss of Normal Feedwater	
15.2-13	0	Nuclear Power, Core Heat Flux and Total Core Reactivity Transients for Main Feedwater Line Rupture with Offsite Power Available	
15.2-14	0	Pressurizer and Maximum System Pressure and Pressurizer Water Volume for Main Feedwater Line Rupture with Offsite Power Available	
15.2-15	0	Reactor Coolant Flow and Feedwater Line Break Flow for Main Feedwater Line Rupture with Offsite Power Available	
15.2-16	0	Reactor Coolant Temperature (Faulted Loop) and Reactor Coolant Temperature (Intact Loop) for Main Feedwater Line Rupture with Offsite Power Available	
15.2-17	0	Steam Generator Pressure for Main Feedwater Line Rupture with Offsite Power Available	
15.2-18	0	Nuclear Power, Core Heat Flux and Total Core Reactivity for Main Feedwater Line Rupture Without Offsite Power	
15.2-19	0	Pressurizer and Maximum System Pressure and Pressurizer Water Volume for Main Feedwater Line Rupture Without Offsite Power	
15.2-20	0	Reactor Coolant Flow and Feedwater Line Break Flow for Main Feedwater Line Rupture Without Offsite Power	
15.2-21	0	Faulted Loop and Intact Loop Reactor Coolant Temperature for Main Feedwater Line Rupture Without Offsite Power	
15.2-22	0	Steam Generator Pressure for Main Feedwater Line Rupture Without Offsite Power	
15.3-1	0	Pressurizer Pressure Transient for Four Loops in Operation, Two Pumps Coasting Down	
15.3-2	0	Total Core Flow Transient for Four Loops in Operation, Two Pumps Coasting Down	
15.3-3	0	Faulted Loop Flow Transient for Four Loops in Operation, Two Pumps Coasting Down	

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CHAPTER 15 - LIST OF FIGURES

*Refer to Section 1.6 and Table 1.6-3. Controlled drawings were removed from the USAR at Revision 17 and are considered incorporated by reference.

Figure #	Sheet	Title	Drawing #*
15.3-4	0	Nuclear Power Transient for Four Loops in Operation, Two Pumps Coasting Down	
15.3-5	0	Core Heat Flux and Core Average Temperature for Four Loops in Operation, Two Pumps Coasting Down	
15.3-6	0	DNBR Transient for Four Loops in Operation, Two Pumps Coasting Down	
15.3-7	0	Total Core Flow Transient for Four Loops in Operation, Four Pumps Coasting Down	
15.3-8	0	Pressurizer Pressure Transient for Four Loops in Operation, Four Pumps Coasting Down	
15.3-9	0	Normalized Core Power Transient for Four Loops in Operation, Four Pumps Coasting Down	
15.3-10	0	Core Heat Flux Transient for Four Loops in Operation, Four Pumps Coasting Down	
15.3-11	0	Core Average Temperature for Four Loops in Operation, Four Pumps Coasting Down	
15.3-12	0	DNBR Transient for Four Loops in Operation, Four Pumps Coasting Down	
15.3-13	0	Peak RCS Pressure Transient, Locked Rotor With and Without Offsite Power	
15.3-14	0	Total Core Flow Transient, Locked Rotor With and Without Offsite Power	
15.3-15	0	Faulted Loop Flow Transient, Locked Rotor With and Without Offsite Power	
15.3-16	0	Normalized Power Transient, Locked Rotor With and Without Offsite Power	
15.3-17	0	Core Heat Flux Transient, Locked Rotor With and Without Offsite Power	
15.3-18	0	Core Average Temperature, Locked Rotor With and Without Offsite Power	
15.3-19	0	Outer Clad Temperature for Locked Rotor without Offsite Power	
15.4-1	0	Core Average Neutron Flux Transient for Uncontrolled Rod Withdrawal from a Subcritical Condition	
15.4-2	0	Thermal Flux Transient for Uncontrolled Rod Withdrawal from a Subcritical Condition	
15.4-3	0	Fuel and Clad Temperature Transient for Uncontrolled Rod Withdrawal from a Subcritical Condition	
15.4-4	0	Uncontrolled RCCA Bank Withdrawal from Full Power with Minimum Reactivity Feedback (5 pcm/sec Withdrawal Rate)	

CHAPTER 15 - LIST OF FIGURES

*Refer to Section 1.6 and Table 1.6-3. Controlled drawings were removed from the USAR at Revision 17 and are considered incorporated by reference.

Figure #	Sheet	Title	Drawing #*
15.4-5	0	Uncontrolled RCCA Bank Withdrawal from Full Power with Minimum Reactivity Feedback (5 pcm/sec Withdrawal Rate)	
15.4-6	0	Withdrawal from Full Power with Beginning-of-Life Reactivity, (5 pcm/sec Withdrawal Rate)	
15.4-7	0	Minimum DNBR Versus Reactivity Insertion Rate, Rod Withdrawal from 100-Percent Power	
15.4-8	0	Minimum DNBR Versus Reactivity Insertion Rate, Rod Withdrawal from 60-Percent Power	
15.4-9	0	Minimum DNBR Versus Reactivity Insertion Rate, Rod Withdrawal from 10-Percent Power	
15.4-10	0	Nuclear Power Transient and Core Heat Flux Transient for Dropped Rod Cluster Control Assembly	
15.4-11a	0	Pressurizer Pressure Transient and Core Average Temperature Transient for Dropped Rod Cluster Control Assembly	
15.4-11b	0	Deleted	
15.4-12	0	Nuclear Power Transient for Startup of an Inactive Reactor Coolant Loop	
15.4-13	0	Core Flow Transient for Startup of an Inactive Reactor Coolant Loop	
15.4-14	0	Pressurizer Pressure Transient for Startup of an Inactive Reactor Coolant Loop	
15.4-15	0	Core Average Temperature and Pressurizer Pressure Transient for Startup of an Inactive Reactor Coolant Loop	
15.4-16	0	DNBR Transient for Startup of an Inactive Reactor Coolant Loop	
15.4-17	0	Representative Percent Change in Local Assembly Average Power for an Interchange Between a Region 1 and a Region 3 Assembly	
15.4-18	0	Representative Percent Change in Local Assembly Average Power for an Interchange Between a Region 1 and a Region 2 Assembly with the Burnable Poison Rods Being Retained by the Region 2 Assembly	
15.4-19	0	Representative Percent Change in Local Assembly Average Power for an Interchange Between a Region 1 and a Region 2 Assembly with the Burnable Poison Rods Being Transferred to the Region 1 Assembly	
15.4-20	0	Representative Percent Change in Local Assembly Average Power for an Enrichment Error (A Region 2 Assembly Loaded into the Core Central Position)	

CHAPTER 15 - LIST OF FIGURES

*Refer to Section 1.6 and Table 1.6-3. Controlled drawings were removed from the USAR at Revision 17 and are considered incorporated by reference.

Figure #	Sheet	Title	Drawing #*
15.4-21	0	Representative Percent Change in Local Assembly Average Power for Loading a Region 2 Assembly into a Region 1 Position Near the Core Periphery	
15.4-22	0	Nuclear Power Transient, BOL, HFP, Rod Ejection Accident	
15.4-23	0	Fuel Centerline, Fuel Average, and Cladding Surface Temperature Transients, BOL, HFP, Rod Ejection Accident	
15.4-24	0	Nuclear Power Transient, EOL, HZP, Rod Ejection Accident	
15.4-25	0	Fuel Centerline, Fuel Average, and Cladding Surface Temperature Transients, EOL, HZP, Rod Ejection Accident	
15.4-26	0	Steam Generator Secondary Pressure Transient for Rod Ejection Analysis	
15.4-27	0	RCS Pressure Transient for Rod Ejection Analysis	
15.4-28	0	Reactor Coolant System Integrated Break Flow Following a Rod Ejection Accident	
15.5-1	0	Inadvertent Operation of ECCS During Power Operation	
15.5-2	0	Inadvertent Operation of ECCS During Power Operation	
15.5-3	0	Inadvertent Operation of ECCS During Power Operation	
15.5-4	0	Chemical and Volume Control System Malfunction Maximum Reactivity Feedback with Pressurizer Spray	
15.5-5	0	Chemical and Volume Control System Malfunction Maximum Reactivity Feedback with Pressurizer Spray	
15.5-6	0	Chemical and Volume Control System Malfunction Maximum Reactivity Feedback without Pressurizer Spray	
15.5-7	0	Chemical and Volume Control System Malfunction Maximum Reactivity Feedback without Pressurizer Spray	
15.5-8	0	Chemical and Volume Control System Malfunction Minimum Reactivity Feedback with Pressurizer Spray	
15.5-9	0	Chemical and Volume Control System Malfunction Minimum Reactivity Feedback with Pressurizer Spray	
15.5-10	0	Chemical and Volume Control System Malfunction Minimum Reactivity Feedback without Pressurizer Spray	
15.5-11	0	Chemical and Volume Control System Malfunction Minimum Reactivity Feedback without Pressurizer Spray	
15.6-1	0	Nuclear Power, Pressurizer Pressure and Reactor Coolant Average Temperature for Inadvertent Opening of a Pressurizer Safety Valve	
15.6-2	0	DNBR Transient for Inadvertent Opening of a Pressurizer Safety Valve	

CHAPTER 15 - LIST OF FIGURES

*Refer to Section 1.6 and Table 1.6-3. Controlled drawings were removed from the USAR at Revision 17 and are considered incorporated by reference.

Figure #	Sheet	Title	Drawing #*
15.6-3A	0	Pressurizer Pressure Transient SGTR Forced with Stuck Open Steam Generator Safety Valve	
15.6-3B	0	Faulted Loop RCS Temperature Transients SGTR Forced Overfill with Stuck Open Steam Generator Safety Valve	
15.6-3C	0	Intact Loop RCS Temperature Transients SGTR Forced Overfill with Stuck Open Steam Generator Safety Valve	
15.6-3D	0	Steam Generator Pressure Transients SGTR Forced Overfill with Stuck Open Steam Generator Safety Valve	
15.6-3E	0	Steam Generator Temperature Transients SGTR Forced Overfill with Stuck Open Steam Generator Safety Valve	
15.6-3F	0	Pressurizer Water Level Transient SGTR Forced Overfill with Stuck Open Steam Generator Safety Valve	
15.6-3G	0	Faulted Steam Generator Steam Flow Transient SGTR Forced Overfill with Stuck Open Steam Generator Safety Valve	
15.6-3H	0	Faulted Steam Generator AFW Flow and Narrow Range Level Transient SGTR Forced Overfill with Stuck Open Steam Generator Safety Valve	
15.6-3I	0	Faulted Steam Generator Break Flow Transient SGTR Forced Overfill with Stuck Open Steam Generator Safety Valve	
15.6-3J	0	Faulted Steam Generator Mixture Volume Transient SGTR Forced Overfill with Stuck Open Steam Generator Safety Valve	
15.6-3K	0	Pressurizer Pressure Transient SGTR W/Stuck-Open ARV	
15.6-3L	0	Faulted Loop RCS Temperature SGTR W/Stuck-Open ARV	
15.6-3M	0	Intact Loop RCS Temperature Transient SGTR W/Stuck-Open ARV	
15.6-3N	0	Steam Generator Pressure (Stuck-Open ARV)	
15.6-3O	0	Steam Generator Temperature Transient SGTR W/Stuck-Open ARV	
15.6-3P	0	Pressurizer Water Level Transient SGTR W/Stuck-Open ARV	
15.6-3Q	0	Faulted SG Steam Flow Transient SGTR (Stuck-Open ARV)	
15.6-3R	0	Faulted SG-AFW Flow and Indicated Narrow Range Level Transient SGTR (Stuck-Open ARV)	
15.6-3S	0	Faulted SG-Total Break Flow Transient SGTR (Stuck-Open ARV)	
15.6-3T	0	Faulted SG-Mixture Volume Transient SGTR (Stuck-Open ARV)	

CHAPTER 15 - LIST OF FIGURES

*Refer to Section 1.6 and Table 1.6-3. Controlled drawings were removed from the USAR at Revision 17 and are considered incorporated by reference.

Figure #	Sheet	Title	Drawing #*
15.6-4	0	Sequence of Events for Large Break LOCA Analysis	
15.6-5	0	Code Interface Description for Large Break Model	
15.6-6	0	Code Interface Description for Small Break Model	
15.6-7	0	Clad Temperature Transient Hot Spot, Reduced T_{AVG} - DECLG (C_D = 0.4), Min Safeguards	
15.6-8	0	Core Pressure, Reduced T_{AVG} - DECLG (C_D = 0.4), Min Safeguards	
15.6-9	0	Reflood Mixture Levels, Reduced T_{AVG} - DECLG (C_D = 0.4), Min Safeguards	
15.6-10	0	Heat Transfer Coefficient Hot Spot, Reduced T_{AVG} - DECLG (C_D = 0.4), Min Safeguards	
15.6-11	0	Fluid Temperature @ Hot Spot, Reduced T_{AVG} - DECLG (C_D = 0.4), Min Safeguards	
15.6-12a	0	Fluid Quality @ Hot Spot, Reduced T_{AVG} - DECLG (C_D = 0.4), Min Safeguards	
15.6-12b	0	Core Inlet Flow Velocity, Reduced T_{AVG} - DECLG (C_D = 0.4), Min Safeguards	
15.6-12c	0	Core Power Transient, Reduced T_{AVG} - DECLG (C_D = 0.4), Min Safeguards	
15.6-12d	0	Core Flowrate, Reduced T_{AVG} - DECLG (C_D = 0.4), Min Safeguards	
15.6-12e	0	Break Mass Flow Rerate, Reduced T_{AVG} - DECLG (C_D = 0.4), Min Safeguards	
15.6-12f	0	Break Energy Flow Rate, Reduced T_{AVG} - DECLG (C_D = 0.4), Min Safeguards	
15.6-12g	0	Accumulator Injection, Reduced T_{AVG} - DECLG (C_D = 0.4), Min Safeguards	
15.6-12h	0	Accumulator Injection, Reduced T_{AVG} - DECLG (C_D = 0.4), Min Safeguards	
15.6-12i	0	Fluid Velocity @ Hot Spot, Reduced T_{AVG} - DECLG (C_D = 0.4), Min Safeguards	
15.6-13	0	Clad Temperature Transient Hot Spot, Reduced T_{AVG} - DECLG (C_D = 0.4), Min Safeguards	
15.6-14	0	Core Pressure, Reduced T_{AVG} - DECLG (C_D = 0.6)	
15.6-15	0	Reflood Mixture Levels, Reduced T_{AVG} - DECLG (C_D = 0.6), Min Safeguards	

CHAPTER 15 - LIST OF FIGURES

*Refer to Section 1.6 and Table 1.6-3. Controlled drawings were removed from the USAR at Revision 17 and are considered incorporated by reference.

Figure #	Sheet	Title	Drawing #*
15.6-16	0	Heat Transfer Coefficient Hot Spot, Reduced T_{AVG} - DECLG (C_D - 0.6), Min Safeguards	
15.6-17	0	Fluid Temperature @ Hot Spot, Reduced T_{AVG} - DECLG (C_D = 0.6), Min Safeguards	
15.6-18	0	Fluid Quality @ Hot Spot, Reduced T_{AVG} - DECLG (C_D = 0.6), Min Safeguards	
15.6-19	0	Clad Temperature Transient Hot Spot, Reduced T_{AVG} - DECLG (C_D = 0.6), Max Safeguards	
15.6-20	0	Reflood Mixture Levels, Reduced T_{AVG} - DECLG (C_D = 0.6), Max Safeguards	
15.6-21	0	Heat Transfer Coefficient, Hot Spot, Reduced T_{AVG} - DECLG (C_D = 0.6), Max Safeguards	
15.6-22	0	Fluid Temperature @ Hot Spot, Reduced T_{AVG} - DECLG (C_D = 0.6), Max Safeguards	
15.6-23	0	Fluid Quality @ Hot Spot, Reduced T_{AVG} - DECLG (C_D = 0.6), Max Safeguards	
15.6-24a	0	Pumped Safety Injection, Reduced T_{AVG} - DECLG (C_D = 0.6), Min Safeguards	
15.6-24b	0	Pumped Safety Injection, Reduced T_{AVG} - DECLG (C_D = 0.6), Max Safeguards	
15.6-25	0	Clad Temperature Transient Hot Spot, Reduced T_{AVG} - DECLG (C_D = 0.8) Min Safeguards	
15.6-26	0	Core Pressure, Reduced T_{AVG} - DECLG (C_D = 0.8) Min Safeguards	
15.6-27	0	Reflood Mixture Levels, Reduced T_{AVG} - DECLG (C_D = 0.8) Min Safeguards	
15.6-28	0	Heat Transfer Coefficient Hot Spot, Reduced T_{AVG} - DECLG (C_D = 0.8) Min Safeguards	
15.6-29	0	Fluid Temperature @ Hot Spot, Reduced T_{AVG} - DECLG (C_D = 0.8) Min Safeguards	
15.6-30	0	Fluid Quality @ Hot Spot, Reduced T_{AVG} - DECLG (C_D = 0.8) Min Safeguards	
15.6-31	0	Reactor Coolant System Depressurization Transient (3 inch break)	
15.6-32	0	Core Mixture Level (3 inch break)	
15.6-33	0	Clad Temperature Transient Hot Spot (3 inch break)	
15.6-34	0	Steam Flow (3 inch break)	

CHAPTER 15 - LIST OF FIGURES

*Refer to Section 1.6 and Table 1.6-3. Controlled drawings were removed from the USAR at Revision 17 and are considered incorporated by reference.

Figure #	Sheet	Title	Drawing #*
15.6-35	0	Rod Film Heat Transfer Coefficient (3 inch break)	
15.6-36	0	Hot Spot Fluid Temperature (3 inch break)	
15.6-37	0	Reactor Coolant System Depressurization Transient (2 inch break)	
15.6-38	0	Reactor Coolant System Depressurization Transient (4 inch break)	
15.6-39	0	Reactor Coolant System Depressurization Transient (6 inch break)	
15.6-40	0	Core Mixture Level (2 inch break)	
15.6-41	0	Core Mixture Level (4 inch break)	
15.6-42	0	Core Mixture Level (6 inch break)	
15.6-43	0	Clad Temperature Transient Hot Rod (2 inch break)	
15.6-44	0	Clad Temperature Transient Hot Rod (4 inch break)	
15.6-45	0	Clad Temperature Transient Hot Rod (6 inch break)	
15.6-46	0	Safety Injection Flowrate	
15.6-47	0	Small Break Power Distribution	
15.6-48	0	Core Power(including Residual Fission) After Reactor Trip (Applies to All Small Breaks)	
15.A-1	0	Release Pathways	

15.0 ACCIDENT ANALYSIS

15.0.1 CLASSIFICATION OF PLANT CONDITIONS

Since 1970, the American Nuclear Society (ANS) classification of plant conditions divides 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:

a. Condition I*: Normal operation and operational transients
b. Condition II*: Faults of moderate frequency
c. Condition III*: Infrequent faults
d. 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 and engineered safeguards system functioning is assumed to the extent allowed by considerations, such as the single failure criterion, in fulfilling this principle, (i.e. only seismic Category I, Class IE, and IEEE qualified equipment, instrumentation, and components are used in the ultimate mitigation of the consequences of faulted conditions Condition II, III and IV events). Step-by-step sequence-of-events diagrams are provided for each transient in Figures 15.0-8 through 15.0-31. Figure 15.0-7 provides the legend used in these diagrams. The accident analysis radiological consequences evaluation models and parameters are discussed in Appendix 15A.

15.0.1.1 Condition I - Normal Operation and Operational Transients

Condition I occurrences are those which are expected frequently in the course of power operation, refueling, maintenance, or maneuvering of the plant. 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 events occur

* For the definition of Conditions I, II, III, and IV events, refer to ANSI-N18.2, "Nuclear Safety Criteria for the Design of Stationary PWR Plants," Section 5, 1973.

frequently, they must be considered from the point of view of their effect on 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 that can occur during Condition I operation.

A typical list of Condition I events is given below:

a. Steady state and shutdown operations

   1. Power operation

   2. Startup

   3. Hot standby

   4. Hot shutdown

   5. Cold shutdown

   6. Refueling

b. Operation with permissible deviations

   Various deviations from normal operation but specifically allowed by the Technical Specifications which may occur during continued operation are considered in conjunction with other operational modes. These include:

   1. Operation with components or systems out of service (such as an inoperable RCCA)

   2. Leakage from fuel with limited clad defects

   3. Excessive radioactivity in the reactor coolant

      (a) Fission products
      (b) Corrosion products
      (c) Tritium

   4. Operation with steam generator leaks

   5. Testing

c. Operational transients

1. Plant heatup and cooldown

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

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

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

15.0.1.2 Condition II - Faults of Moderate Frequency

These faults, at worst, result in a reactor trip with the plant being capable of returning to operation. 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 failures or reactor coolant system or secondary system overpressurization. The following faults are included in this category:

a. Feedwater system malfunctions that result in a decrease in feedwater temperature (Section 15.1.1).

b. Feedwater system malfunctions that result in an increase in feedwater flow (Section 15.1.2).

c. Excessive increase in secondary steam flow (Section 15.1.3).

d. Inadvertent opening of a steam generator atmospheric relief or safety valve (Section 15.1.4).

e. Loss of external electrical load (Section 15.2.2).

f. Turbine trip (Section 15.2.3).

g. Inadvertent closure of main steam isolation valves (Section 15.2.4).

h. Loss of condenser vacuum and other events resulting in turbine trip (Section 15.2.5).

i. Loss of nonemergency ac power to the station auxiliaries (Section 15.2.6).

j. Loss of normal feedwater flow (Section 15.2.7).

k. Partial loss of forced reactor coolant flow (Section 15.3.1).

l. Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition (Section 15.4.1).

m. Uncontrolled rod cluster control assembly bank withdrawal at power (Section 15.4.2).

n. Rod cluster control assembly misalignment (dropped RCCA, dropped RCCA bank, or statically misaligned RCCA) (Section 15.4.3).

o. Startup of an inactive reactor coolant pump at an incorrect temperature (Section 15.4.4).

p. Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant (Section 15.4.6).

q. Inadvertent operation of the emergency core cooling system during power operation (Section 15.5.1).

r. Chemical and volume control system malfunction that increases reactor coolant inventory (Section 15.5.2).

s. Inadvertent opening of a pressurizer safety or relief valve (Section 15.6.1).

t. Break in instrument line or other lines from reactor coolant pressure boundary that penetrate the containment (Section 15.6.2).

15.0.1.3 Condition III - Infrequent Faults

Condition III events are faults which may occur very infrequently during the life of the plant. They may result in the failure of only a small fraction of the fuel rods. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion area boundary, per the guidelines of 10 CFR 100. A Condition III event will not, by itself, generate a Condition IV event or result in a consequential loss of function of the reactor coolant system or containment barriers. The following faults are included in this category:

a. Steam system piping failure (minor) (Section 15.1.5).

b. Complete loss of forced reactor coolant flow (Section 15.3.2).

c. Rod cluster control assembly misalignment (single rod cluster control assembly withdrawal at full power) (Section 15.4.3).

d. Inadvertent loading and operation of a fuel assembly in an improper position (Section 15.4.7).

e. Loss-of-coolant accidents resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (small break) (Section 15.6.5).

f. Radioactive gas waste system leak or failure (Section 15.7.1).

g. Radioactive liquid waste system leak or failure (Section 15.7.2).

h. Postulated radioactive releases due to liquid tank failures (Section 15.7.3).

i. Spent fuel cask drop accidents (Section 15.7.5).

15.0.1.4 Condition IV - Limiting Faults

Condition IV events are faults which are not expected to take place, but are postulated because their consequences would include the potential of the release of significant amounts of radioactive material. They are the most drastic which must be designed against and represent limiting design cases. Condition IV events are not to cause a fission product release to the environment resulting in doses in excess of guideline values of 10 CFR 100. A single Condition IV event is not to cause a consequential loss of required functions of systems needed to cope with the fault, including those of the emergency core cooling system and the containment. The following faults have been classified in this category:

a. Steam system piping failure (major) (Section 15.1.5).

b. Feedwater system pipe break (Section 15.2.8).

c. Reactor coolant pump shaft seizure (locked rotor) (Section 15.3.3).

d. Reactor coolant pump shaft break (Section 15.3.4).

e. Spectrum of rod cluster control assembly ejection accidents (Section 15.4.8).

f. Steam generator tube failure (Section 15.6.3).

g. Loss-of-coolant accidents, resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary (large break) (Section 15.6.5).

h. Design basis fuel handling accidents (Section 15.7.4).

15.0.2 OPTIMIZATION OF CONTROL SYSTEMS

A control system setpoint study is performed prior to operation to simulate performance of the reactor control and protection systems. In this study, emphasis is placed on the development of a control and protection system which will automatically maintain prescribed conditions in the plant even under the most adverse set of anticipated plant operating transients with respect to both system stability and equipment performance. For each mode of plant operation, a group of optimum controller setpoints is determined. A consistent set of control system parameters is derived from this study that satisfies plant operational requirements throughout the core life and for various levels of power operation.

The study is comprised of an analysis of the following control systems: rod cluster control assembly, steam dump, steam generator level, pressurizer pressure, and pressurizer level.

15.0.3 PLANT CHARACTERISTICS AND INITIAL CONDITIONS ASSUMED IN THE ACCIDENT ANALYSES

15.0.3.1 Design Plant Conditions

Table 15.0-1 lists the principal power rating values which are assumed for the transients analyzed. Two ratings are given:

a. The guaranteed NSSS thermal power output. This power output includes the thermal power generated by the reactor coolant pumps.

b. The rated reactor core thermal power output is 3565 MWt.

Allowances for errors in the determination of the steady-state power level are made as described in Section 15.0.3.2. The core thermal power used for each transient is given in Table 15.0-2.

The values of other pertinent plant parameters utilized in the accident analyses are given in Table 15.0-3.

15.0.3.2 Initial Conditions

For transients in which DNB is of concern, the WCGS Stastitical Core Design (SCD) methodology (Ref 14) is used. In application of this methodology, the following steady-state errors are considered:

a.	Core power	Nominal
b.	Average reactor coolant system temperature	±1.65°F to allow for steam generator fouling
c.	Pressurizer pressure	Nominal

For transients in which RCS overpressurization or pressurizer overfill is of concern the following steady-state errors are considered:

a.	Core power	+2 percent allowance for calorimetric error
b.	Average reactor coolant system temperature	±4.85°F allowance for controller deadband and measurement error plus an additional 1.65°F allowance for steam generator fouling
c.	Pressurizer pressure	±30 psi allowance for steady state fluctuations and measurement errors

Table 15.0-2 summarizes the initial conditions and computer codes used in the accident analyses.

15.0.3.3 Power Distribution

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of fuel assemblies and control rods. Power distribution may be characterized by the nuclear enthalpy rise hot channel factor (F_{DH}) and the total peaking factor (F_q). Unless specifically noted otherwise, the peaking factors used in the accident analyses are those presented in Chapter 4.0.

For transients which may be DNB limited, the radial peaking factor is of importance. The radial peaking factor increases with decreasing power level due to rod insertion. This increase in F_{DH} is included in the core limits illustrated in Figure 15.0-1. All transients that may be DNB limited are assumed to begin with a F_{DH} consistent with the initial power level defined in the Technical Specifications.

The axial power shape used in the DNB calculation is the 1.55 chopped cosine, as discussed in Chapter 4 for transients analyzed at full power and the most limiting power shape calculated or allowed for accidents initiated at non-full power or asymmetric RCCA conditions.

The radial and axial power distributions described above are input to the VIPRE code, as described in Chapter 4.

For transients which may be overpower limited, the total peaking factor (F_q) is of importance. All transients that may be over-power limited are assumed to begin with plant and reactor operating conditions consistent with the restrictions defined in the Technical Specifications.

For overpower transients that are slow with respect to the fuel rod thermal time constant, for example the chemical and volume control system malfunction that results in a slow decrease in the boron concentration in the reactor coolant system as well as an excessive increase in secondary steam flow, that may reach equilibrium without causing a reactor trip, the fuel rod thermal evaluations are performed as discussed in Chapter 4. For overpower transients that are fast with respect to the fuel rod thermal time constant, for example, the uncontrolled RCCA bank withdrawal from subcritical or low power startup and RCCA ejection incident, both of which result in a large power rise over a few seconds, a detailed fuel transient heat transfer calculation must be performed.

15.0.4 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 are discussed in detail in Section 4.3.2.3 and shown on Figure 15.0-6.

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 on reactivity feedback effects. The values used are given in Table 15.0-2. 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 versus small reactivity coefficient values are treated on an event-by-event basis. In some cases, conservative combinations of parameters are used to bound the effects of core life. For example, in a load increase transient it is conservative to use a small Doppler defect and a small moderator coefficient.

15.0.5 ROD CLUSTER CONTROL ASSEMBLY INSERTION CHARACTERISTICS

All accident analysis results contained herein are applicable to both Ag-In-Cd and hafnium control rods.

The negative reactivity insertion following a reactor trip is a function of the acceleration of the RCCAs as a function of time and the variation in rod worth as a function of rod position. For accident analyses, the critical parameter is the time of insertion up to the dashpot entry, or approximately 85 percent of the rod cluster travel. The RCCA position versus time assumed in accident analyses is shown in Figure 15.0-3. The RCCA insertion time to dashpot entry is taken as 2.7 seconds unless otherwise noted in the discussion. The use of such a long insertion time provides conservative results for all accidents and is intended to be applicable to all types of RCCAs, which may be used throughout plant life. Drop time testing requirements are specified in the plant Technical Specifications.

Figure 15.0-4 shows the fraction of total negative reactivity insertion versus 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 versus 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, significantly more negative reactivity would have been inserted than that shown in the curve, due to the more favorable axial distribution existing prior to trip.

The normalized RCCA negative reactivity insertion versus 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-percent Δk 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 Table 4.3-3. For Figures 15.0-3 and 15.0-4, the RCCA drop time is normalized to 2.7 seconds, unless otherwise noted for a particular event, in order to provide a bounding analysis for all RCCAs to be used in the WCGS core.

The normalized RCCA negative reactivity insertion versus time curve for an axial power distribution skewed to the bottom (Figure 15.0-5) is used in those transient analyses for which a point kinetics core model is used. Where special analyses require 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 RCCA position versus time of Figure 15.0-3 is used as code input.

15.0.6 TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

A reactor trip signal acts to open two trip breakers connected in series feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the RCCAs, 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 mechanisms. 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. The safety analysis 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 that table to overtemperature and overpower ΔT trip shown in Figure 15.0-1. Figure 15.0-1 shows the core thermal limits over the range of allowable operating pressures assuming a maximum loop Thermal Design Flow (TDF) of 90,324 gpm. The DNB limit lines correspond to a safety analysis limit DNBR value as determined by the WRB-2 CHF Correlation (see Section 4.4.1.1)(Reference 16). The figure demonstrates the adequacy of the Overtemperature and Overpower Delta T trip functions in conjunction with the steam generator safety valve lift settings to protect the core thermal limits.

The difference between the safety analysis trip setpoint assumed for the analysis and the nominal trip setpoint represents an allowance for instrumentation channel error. Nominal trip setpoints are specified in the plant Technical Specifications Bases. During plant startup tests, it was 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.

15.0.7 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 top and bottom sections) is calibrated (set equal) to this measured power on a periodic basis.

Two options are available for indication of secondary power. These are (1) a feedwater calorimetric or (2) a steam flow calorimetric. The feedwater calorimetric uses feedwater flow (feedwater venturi differential pressure), feedwater inlet temperature to the steam generators, and steam pressure (from which feedwater pressure is inferred). The steam flow calorimetric uses the same inputs but uses the pressure drop across the steam generator orifice instead of the feedwater flow. This pressure drop value is normalized to a precision feedwater flow measurement at the beginning of the cycle. High accuracy instruments provided measurements with accuracy tolerances much tighter than those which are required to normally control feedwater flow.

15.0.8 MAJOR PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR MITIGATION OF TRANSIENT AND ACCIDENT CONDIITONS

The plant is designed to afford protection against the possible effects of natural phenomena, postulated environmental conditions, and dynamic effects of the postulated accidents. In addition, the design incorporates features which minimize the probability and effects of fires and explosions. Chapter 17.0 discusses the quality assurance program which has been implemented to assure that the NSSS will satisfactorily perform its assigned safety functions. The incorporation of these features in the plant, coupled with the reliability of the design, ensures that the normally operating systems and equipment listed in Table 15.0-6 are available for mitigation of the events discussed in Chapter 15.0. 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 Standard 379-1972 and Regulatory Guide 1.53, in the application of the single failure criterion.

In the analysis of the Chapter 15.0 events, control system action is considered only if that action results in more severe accident results. 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 15.0 events.

15.0.9 FISSION PRODUCT INVENTORIES

15.0.9.1 Activities in the Core

The calculation of the core iodine fission product inventory is consistent with the inventories given in TID-14844 (Ref. 1) and is based on a core power level of 3565 MWt. The fission product inventories for other isotopes that are important from a health hazards point of view are consistent with the data from APED-5398 (Ref. 2). These inventories are given in Table 15A-3. The isotopes included in Table 15A-3 (Appendix 15A) are the isotopes controlling from considerations of inhalation dose (iodines) and from direct dose due to immersion (noble gases).

The isotopic yields used in the calculations are from the data of APED-5398, utilizing the isotopic yield data for thermal fissioning of uranium-235 as the sole fissioning source. The change in fission product inventory resulting from the fissioning of other fissionable atoms has been reviewed. The results of this review indicated that inclusion of all fission source data would result in small (less than 10 percent) change in the isotopic inventories.

15.0.9.2 Activities in the Fuel Pellet Clad Gap

The fuel-clad gap activities are determined, using the model given in Regulatory Guide 1.77. Thus, the amount of activity accumulated in the fuel clad gap is assumed to be 10 percent of the core activity for all isotopes except for Kr-85. For Kr-85 it is assumed to be 30 percent of the core activity. The gap activities are given in Table 15A-3.

15.0.10 RESIDUAL DECAY HEAT

15.0.10.1 Total Residual Heat

Residual heat in a subcritical core is calculated for the LOCA per the requirements of Appendix K of 10 CFR 50.46, as described in References 3 and 4. 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.

15.0.10.2 Distribution of Decay Heat Following Loss-of-Coolant Accident

During a LOCA, the core is rapidly shut down by void formation or RCCA insertion, or both, and a large fraction of the heat generation to be considered comes from fission product decay gamma rays. This heat is not distributed in the same manner as steady state fission power. Local peaking effects which are important for the neutron dependent part of the heat generation do not apply to the gamma ray contribution. The steady state factor of 97.4 percent, which represents the fraction of heat generated within the clad and pellet, drops to 95 percent for the hot rod in a LOCA.

For example, consider the transient resulting from the postulated double ended break of the largest reactor coolant system pipe; 1/2 second after the rupture about 30 percent of the heat generated in the fuel rods is from gamma ray absorption. The gamma power shape is less peaked than the steady state fission power shape, reducing the energy deposited in the hot rod at the expense of adjacent colder rods. A conservative estimate of this effect is a reduction of 10 percent of the gamma ray contribution or 3 percent of the total. Since the water density is considerably reduced at this time, an average of 98 percent of the available heat is deposited in the fuel rods, the remaining 2 percent being absorbed by water, thimbles, sleeves, and grids. The net effect is a factor of 0.95, rather than 0.974, to be applied to the heat production in the hot rod.

15.0.11 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 (see Section 15.6), are summarized in their respective accident analyses sections. The codes used in the analyses of each transient have been listed in Table 15.0-2.

15.0.11.1 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 clad, using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, and density). The code uses a fuel model which simultaneously exhibits the following features:

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

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

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

FACTRAN is further discussed in Reference 5.

15.0.11.2 LOFTRAN

The LOFTRAN program is used for studies of transient response of a PWR system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by a model containing reactor vessel, hot and cold leg piping, steam generator (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 high neutron flux, overtemperature ΔT, overpower ΔT, 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. The emergency core cooling system, 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 on Figure 15.0-1. The core limits represent the minimum value of DNBR, as calculated for typical or thimble cell.

LOFTRAN is further discussed in Reference 6.

15.0.11.3 PHOENIX-P

PHOENIX-P is a two dimensional, multi-group transport theory computer code. The nuclear cross-section library used by PHOENIX-P contains cross-section data based on a 42-energy group structure derived from ENDF/B-V files. PHOENIX-P performs a 2D 42-group nodal flux calculation which couples the individual subcell regions as well as surrounding rods via a collision probability technique. This 42-group solution is normalized by a coarse energy group flux solution derived from a discrete ordinates calculation. PHOENIX-P is capable of modeling all cell types needed for PWR core design applications.

PHOENIX-P is further described in Reference 7.

15.0.11.4 ANC

ANC is a multidimensional nodal analysis program used to predict nuclear reactor core reactivity and assembly and rod distributions for normal and off-normal conditions. The code allows for the treatment of enthalpy, xenon, and Doppler feedback. A high degree of automation has been incorporated into the code to address fuel depletion, reactivity coefficients, control rod worths for nonuniform inlet temperature distribution.

ANC is further described in Reference 8.

15.0.11.5 TWINKLE

The TWINKLE program is a multidimensional 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, and three dimensions. The code uses six delayed neutron groups, and contains a detailed multiregion fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2,000 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, and others. Various edits are provided, e.g., channel-wise power, axial offset, enthalpy, volumetric surge, point-wise 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.

15.0.11.6 APOLLO

APOLLO is a two-group, one dimensional neutron diffusion code designed to characterize the axial behavior of the core during plant operation in situations where three dimensional analyses would be time consuming and cumbersome. Pertinent uses include evaluation of limiting power shapes and scoping the adequacy of reactor protection systems in order to maintain margin to overpower and DNB limits.

APOLLO is further described in Reference 10.

15.0.11.7 RETRAN-02

RETRAN-02 is a thermal-hydraulic systems analysis code employing a one-dimensional, homogeneous equilibrium mixture thermal-hydraulic model for the RCS, a point kinetics model for the reactor core, special component and auxiliary models (ex., pumps, temperature transport, non-equilibrium pressurizer) and control system models. The code computes pertinent plant transient information including core power level, RCS pressure and temperature.

RETRAN-02 is further described in Reference 12.

15.0.11.8 VIPRE-01

The core thermal-hydraulic analysis code VIPRE-01 is an open channel code designed to evaluate DNBR and coolant state for steady state operations and transients using subchannel analysis.

VIPRE-01 is further described in Reference 13 and Section 4.4.4.5.

15.0.12 LIMITING SINGLE FAILURES

The most limiting single failure as described in Section 3.1 of safety-related equipment, where one exists, is identified in each analysis description, and the consequences of this failure are described therein. In some instances, because of redundancy in protection equipment, no single failure which could adversely affect the consequences of the transient has been identified. The failure assumed in each analysis is listed in Table 15.0-7.

15.0.13 OPERATOR ACTIONS

For most of the events analyzed in Chapter 15.0 the plant will be in a safe and stable hot standby condition following the automatic actuation of reactor trip. This condition will, in fact, be similar to plant conditions following any normal, orderly shutdown of the reactor. At this point, the actions taken by the operator would be no different than normal operating procedures. The exact actions taken, and the time at which these actions would occur, will depend on what systems are available (e.g., turbine bypass system, main feedwater system, etc.) and the plans for further plant operation. As a minimum, to maintain the hot stabilized condition, decay heat must be removed via the steam generators. The main feedwater system and the steam dump or atmospheric relief system could be used for this purpose. Alternatively, the auxiliary feedwater system and the steam generator safety valves may be used, both of which are safety grade systems. Although the auxiliary feed system may be started manually, it will be automatically actuated, if needed, by one of the signals shown on Figure 7.2-1, such as low-low steam generator water level. Also, if the hot standby condition is maintained for an extended period of time (greater than approximately 18 hours), operator action may be required to add boric acid via the CVCS to compensate for xenon decay and maintain shutdown margin.

Where a stabilized condition is reached automatically following a reactor trip and only actions typical of normal operation are required, this has been stated in the text of the Chapter 15.0 events. For several events involving breaks in the reactor coolant system or secondary system piping, additional requirements for operator action are identified.

Following the postulated MSLB, a steamline isolation signal will be generated almost immediately, causing the steamline isolation valves to close within a few seconds. If the break is downstream of the isolation valves, all of which subsequently close, the break will be isolated. If the break is upstream of the isolation valves, or if one valve fails to close, the break will be isolated to three steam generators while the affected steam generator will continue to blow down. Only the case in which one steam generator continues to blow down is discussed here, since the break followed by isolation of all steam generators will terminate the transient.

Steam pressure from the steam generators is relieved by the turbine bypass system, secondary system atmospheric relief valves, or secondary system safety valves. The operator is instructed to terminate auxiliary feedwater flow to the affected steam generator, as soon as he determines which steam generator is affected. As soon as an indicated water level returns to the pressurizer and pressure is no longer decreasing, the operator is instructed to terminate the charging pump flow to limit system repressurization.

For long-term cooling following a steamline break, the operator is instructed to use the intact steam generators for the purpose of removing decay heat and plant stored energy. This is done by feeding the steam generators with auxiliary feed-water to maintain an indicated water level in the steam generator narrow-range span.

A safety injection signal (generated a few seconds after the break on low steam line pressure) will cause main feedwater isolation to occur. A steam line break protection signal (Section 7.3.8.3.2) will also cause main feedwater isolation to occur. The only source of water available to the affected generator is then the auxiliary feedwater system. Following steamline isolation, steam pressure in the steamline with the affected steam generator will continue to fall rapidly, while the pressure stabilizes in the remaining three steam lines. The indication of the different steam pressures will be available to the operator, within a few seconds of steamline isolation. This will provide the information necessary to identify the affected steam generator so that auxiliary feedwater to it can be isolated. Manual controls are provided in the control room for start and stop of the auxiliary feedwater pumps and for the control valves associated with the auxiliary feedwater system. The means for detecting the affected steam generator and isolating auxiliary feedwater to it requires only the use of safety grade equipment available following the break. The removal of decay heat in the long term (following the initial cooldown), using the remaining steam generators, requires only the auxiliary feedwater system as a water source and the secondary system safety valves to relieve steam.

The operator has available, in the control room, an indication of pressurizer water level from the reactor protection system instrumentation. Indicated water level returns to the pressurizer in approximately 5 to 7 minutes following the steamline break. To maintain the indicated water level, the operator can start and stop the charging pumps as necessary. The pressurizer level instrumentation and manual controls for the operation of the charging pumps meet the required standards for safety systems.

As indicated, the information for terminating auxiliary feedwater to the affected steam generator is available to the operator within 1 minute of the break, while the information required for terminating the charging/SI flow becomes available within 5 to 7 minutes following the break. The requirements to terminate auxiliary feedwater flow to the affected steam generator can be met by switch actions by the operators, i.e., closing auxiliary feed discharge valve. Thus, the required actions to limit the cooldown can be recognized, planned, and performed within 10 minutes. After it is determined that the pressurizer level is restored and SI flow is no longer required, normal charging flow is established and the SI flow is eventually terminated to prevent repressurization of the RCS. For decay heat removal and plant cooldown, the operator has a considerably longer time period in which to respond because of the large initial cooldown associated with a steamline break transient.

For a feedwater line break, the required operator actions and times are discussed in Section 15.2.8 and Table 15.2-1. Auxiliary feedwater flow is initiated automatically, as is safety injection. As in the steamline break, the operator terminates auxiliary feedwater flow to the affected steam generator as soon as he determines which unit is affected, using safety grade equipment. Where possible, the operator should also increase auxiliary feedwater flow to the intact steam generators in order to shorten the time until primary temperatures begin to decrease. The analysis presented in Section 15.2.8 assumes a 30-minute delay until these actions occur.

As soon as primary temperature begins to decrease, the operator can use the steam dump system or the steam generator atmospheric relief valves to begin a controlled cooldown. In addition, if the U-tubes of the intact steam generators are covered with water as indicated by post accident monitoring system (PAMS) steam generator water level instrumentation (see Chapter 7.0), the operator can modulate the high-head charging pumps, so that the primary pressure decreases while ensuring that voiding does not occur within the RCS. The primary pressure-temperature relationship can be monitored by the operator via the PAMS widerange RCS pressure and temperature instruments.

Using the above-mentioned PAMS indications, the operator can maintain the plant in a hot shutdown condition for an extended period of time, or can proceed to a cold shutdown condition as desired.

The safety-related indicators for steamline pressure and pressurizer water level noted above are further discussed in Section 7.5.

Tables 15.0-8 and 15.0-9 list the short term operator actions required to bring the plant to a stable condition for the LOCA and steam generator tube rupture (SGTR). Further information (including alarms which alert the operator) on operator action for these two accidents are given in Section 6.3.2.8 for the LOCA and Section 15.6.3 for the SGTR.

Process information available to the operator in the control room following either of these accidents (LOCA or SGTR) is given in Section 7.5.

Instrumentation and controls provided to allow the operator to complete required manual actions are classified as Class IE. Electrical components are also classified as Class IE. Mechanical components are classified as Safety Class 1, 2, or 3.

Safety systems required for accident mitigation are designed to function after the occurrence of the worst postulated single failure. There are no adverse impacts as a result of these actions.

15.0.14 REFERENCES

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TABLE 15.0-1

NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

NSSS thermal power output, MWt	3,579
Nominal thermal power generated by the reactor coolant pumps, MWt	14
Rated reactor core thermal power output, MWt	3,565

TABLE 15.0-2

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

	Faults	Computer Codes Utilized	Reactivity Coefficients (a) Assumed: Moderator Density /Temp Coeff	Reactivity Coefficients (a) Assumed: Doppler	Initial NSSS Thermal Power Output Assumed (b) (MWt)
15.1	Increase in heat removal by the secondary system				
	Feedwater system malfunctions that result in an increase in feedwater flow	RETRAN, VIPRE	0.43 Δk/gm/cc	Lower (a)	0 and 3,565
	Excessive increase in secondary steam flow	LOFTRAN, VIPRE	Figure 15.0-6 and 0.43	Lower (a)	3,565
	Inadvertent opening of a steam generator atmospheric relief or safety valve	RETRAN, VIPRE	Function of moderator density (see Section 15.1.4, Figure 15.1-11)	See Section 15.1.5	0 (Subcritical)
	Steam system piping failure	RETRAN, VIPRE	Function of moderator density (see Section 15.1.5, Figure 15.1-11)	See Section 15.1.5	0 (Subcritical)
15.2	Decrease in heat removal by the secondary system				
	Loss of external electrical load and/or turbine trip	RETRAN, VIPRE	+7 pcm/°F	Lower and upper (a)	3,565

TABLE 15.0-2 (Sheet 2)

	Loss of nonemergency ac power to the station auxiliaries	RETRAN, VIPRE	+7 pcm/°F	Upper (a)	3,565
	Loss of normal feedwater flow	RETRAN, VIPRE	+7 pcm/°F	Upper (a)	3,565
	Feedwater system pipe break	RETRAN, VIPRE	+7 pcm/°F	Lower (a)	3,565
15.3	Decrease in reactor coolant system flow rate				
	Partial and complete loss of forced reactor coolant flow	RETRAN, VIPRE	+7.0 pcm/°F	Upper (a)	3,565
	Reactor coolant pump shaft seizure (locked rotor)	RETRAN, VIPRE FACTRAN	+7.0 pcm/°F	Upper (a)	3,565
15.4	Reactivity and power distribution anomalies				
	Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition	TWINKLE, FACTRAN, VIPRE	+7.0 pcm/°F	Consistent with lower limit shown on Figure 15.0-2	0
	Uncontrolled rod cluster control assembly bank withdrawal at power	RETRAN, VIPRE	+7.0 pcm/°F and 0.43 Δk/gm/cc	Lower and upper (a)	3,565
	Rod cluster control assembly misalignment	PHOENIX, ANC,	NA	NA	3,565
	Startup of an inactive reactor coolant loop at an incorrect temperature	LOFTRAN, VIPRE	0.43 Δk/gm/cc	Lower (a)	2,495
	Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant	NA	NA	NA	0 and 3,565

TABLE 15.0-2 (Sheet 3)

	Inadvertent loading and operation of a fuel asembly in an improper position	ANC, PHOENIX	NA	NA	3,565
	Spectrum of rod cluster control assembly ejection accidents	TWINKLE, FACTRAN, ANC	Refer to Section 15.4.8 (c)	Consistent with lower limit shown on Figure 15.0-2	0 and 3,565
15.5	Increase in reactor coolant inventory				
	Inadvertent operation of the ECCS during power operation	RETRAN, VIPRE	+7.0 pcm/°F	Lower (a)	3,565
	Chemical and volume control system malfunction that increases reactor coolant inventory	LOFTRAN	0.43 Δk/gm/cc and +7.0 pcm/°F	Lower and Upper (a)	3,565
15.6	Decrease in reactor coolant inventory				
	Inadvertent opening of a pressurizer safety or relief valve	RETRAN, VIPRE	0.43 Δk/gm/cc and +7.0 pcm/°F	Upper (a)	3,565
	Steam generator tube rupture	RETRAN	NA	NA	3,565
	Loss-of-coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	SB LOCA: NOTRUMP, LOCTA-IV. LB LOCA: SATAN-VI, WREFLOOD COCO, BASH, LOCBART	See Section 15.6.5 references (c)	See Section 15.6.5, references	3,565

(a) See Figure 15.0-2.
(b) A minimum of 2 percent margin is applied to the values shown for analysis purposes.
(c) The moderator density coefficients for these accidents were translated into moderator temperature coefficients.

TABLE 15.0-3

NOMINAL VALUES OF PERTINENT PLANT PARAMETERS
UTILIZED IN THE ACCIDENT ANALYSES *

Thermal output of NSSS, MWt	See Table 15.0-2
Core inlet temperature, °F	555.8
Vessel average temperature, °F	588.4
Reactor coolant system pressure, psia	2,250
Reactor coolant flow per loop, gpm	90,324
Steam flow from NSSS, Mlb/hr	15.92
Steam pressure at steam generator outlet, psia	944
Maximum steam moisture content, %	0.25
Assumed feedwater temperature at steam generator inlet, °F	446
Average core heat flux, Btu/hr-ft^2	198340
Steam generator tube plugging, %	10.0

* Steady-state errors discussed in Section 15.0.3.2 are added to these values to obtain initial conditions for transient analyses except where discussed otherwise.

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TABLE 15.0-4

TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Point Assumed In Analyses	Time Delays (Seconds)
Power range high neutron flux, high setting	118%	0.5
Power range high neutron flux, low setting	35%	0.5
High neutron flux, P-8	84%	0.5
Overtemperature ΔT	Variable see Figure 15.0-1	6.0*
Overpower ΔT	Variable see Figure 15.0-1	6 0*
High pressurizer pressure	2,460 psia	2.0
Low pressurizer pressure	1,900 psia	2.0
Low reactor coolant flow (from loop flow detectors)	87 % loop flow	1.0
Undervoltage trip	68% nominal	1.5
Turbine trip	Not applicable	2.0
Low-low steam generator level	0% of narrow range level span	2.0
High steam generator level trip of the feedwater pumps and closure of feedwater system valves, and turbine trip	100% of narrow range level span	2.0

* Total time delay (including scoop delay and thermal lag, combined RTD/thermowell response, and trip circuit channel electronics delay) from the time the temperature difference in the coolant loop 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 INSTRUMENTATION ERRORS

Nominal setpoint (% of rated power) 109

Calorimetric errors in the measurement of secondary system thermal power:

Variable	Accuracy of Measurement of Variable (% error)	Estimated Effect on Thermal Power Determination (% error)
Assumed calorimetric error (% of rated power for both feedwater calorimetric and steam flow calorimetric measurements)		2 (a)*
Axial power distribution effects on total ion chamber current		
Estimated error (% of rated power)	3	
Assumed error (% of rated power)		5 (b)*
Instrumentation channel drift and setpoint reproducibility		
Estimated error (% of rated power)	1	

TABLE 15.0-5 (Sheet 2)

Variable	Accuracy of Measurement of Variable (% error)	Estimated Effect on Thermal Power Determination (% error)
Assumed error (% of rated power)		2 (c)*
* Total assumed error in setpoint (% of rated power) (a) + (b) + (c)		±9
Maximum overpower trip point, assuming all individual errors are simultaneously in the most adverse direction (% of rated power)		118

TABLE 15.0-6

MAJOR PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR MITIGATION OF TRANSIENT AND ACCIDENT CONDITIONS

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
15.1	Increase in heat removal by the secondary system	-	-	-	-
	Feedwater system malfunctions that result in an increase in feedwater flow	Source range high flux, intermediate range high flux, power range high flux, low-low steam generator level, manual overpower ΔT, overtemperature ΔT, turbine trip	High-high SG level produced feedwater isolation and turbine trip	Feedwater isolation valves	-
	Excessive increase in secondary steam flow	Power range high flux, overtemperature ΔT, overpower ΔT, manual	-	Pressurizer safety valves, steam generator safety valves	-
	Inadvertent opening of a steam generator atmospheric relief or safety valve	Manual, SIS, power range high flux, overpower ΔT	Low pressurizer pressure, low compensated steam line pressure high negative steam pressure rate	Feedwater isolation valves, steam line isolation valves	Auxiliary feedwater system, safety injection system
	Steam system piping failure	Power range high flux, pressure, manual, SIS overpower ΔT	Low pressurizer pressure, low compensated steam line pressure, hi-1 containment pressure, hi-2 containment pressure, high negative steam pressure rate, SIS, high-high SG level, low-low SG level, manual	Feedwater isolation valves, steam line isolation valves	Auxiliary feedwater system, safety injection system
15.2	Decrease in heat removal by the secondary system	-	-	-	-
	Loss of external load/turbine trip	Low-low SG level, turbine trip, high pressurizer pressure, overtemperature ΔT, manual high pressurizer water level	Low-low SG level, manual	Pressurizer safety valves, steam generator safety valves	-

TABLE 15.0-6 (Sheet 2)

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
	Loss of non-emergency ac power to the station auxiliaries	Manual, low-low SG level, low flow	Low-low SG level, loss of offsite power, manual	Steam generator safety valves, pressurizer safety valves	Auxiliary feedwater system
	Loss of normal feedwater flow	Low-low SG level, manual, overtemperature ΔT	Low-low SG level, loss of offsite power, manual	Steam generator safety valves, pressurizer safety valves	Auxiliary feedwater system
	Feedwater system pipe break	Low-low SG level, high pressurizer pressure, SIS, manual, overtemperature ΔT	Hi-1 containment pressure, low-low SG level, low compensated steam line pressure SIS, loss of offsite power, manual	Steam line isolation valves, feedline isolation, pressurizer safety valves, steam generator safety valves	Auxiliary 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, undervoltage underfrequency, manual	-	Pressurizer safety valves	-
	Reactor coolant pump shaft seizure (locked rotor)	Low flow, high pressurizer pressure, manual	-	Pressurizer safety valves, steam generator safety valves	-
15.4	Reactivity and power distribution anomalies	-	-	-	-
	Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition	Source range high flux, intermediate range high flux, power range high flux (high setpoint), power range high flux (low setpoint), manual, power range high flux rate	-	-	-

TABLE 15.0-6 (Sheet 3)

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
	Uncontrolled rod cluster control assembly bank withdrawal at power	Power range high flux, overtemperature ΔT, high pressurizer pressure, manual, high pressurizer water level, overpower ΔT, low-low SG level	-	Pressurizer safety valves,steam generator safety valves	-
	Rod cluster control assebmly mis-alignment	Power range negative flux rate, overtemperature ΔT, manual	-	-	-
	Startup of an inactive reactor coolant loop at an incorrect temperature	Manual, power range high flux above P8 setpoint with low flow	-	Low insertion limit annun-ciators for boration	-
	Chemical and volume control, system malfunction that results in a decrease in boron concentrattion in the reactor coolant	Power range high flux, overtemperature ΔT, manual, overpower ΔT high pressurizer pressure, high pressurizer water level	-	Pressurizer safety valves, steam generator safety valves rod insertion limit alarms, position step counter, RT alarm, audible count rate from BF3 detector source range high flux alarm	-
	Spectrum of rod cluster control assembly (RCCA) ejection accidents	Source range high flux, intermediate range high flux, power range high flux, high positive flux rate, manual	Low pressurizer pressure, Hi-1 containment pressure, Hi-3 containment pressure, manual.	Pressurizer safety valves	ESFAS
15.5	Increase in reactor coolant inventory	-	-	-	-
	Inadvertent operation of the ECCS during power operation	Low pressurizer pressure, manual, safety injection trip	-	-	Safety injection system

TABLE 15.0-6 (Sheet 4)

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	Low pressurizer pressure, overtemperature ΔT, manual	Low pressurizer pressure, manual	-	Safety injection system
Steam generator tube rupture	Low pressurizer pressure, SIS, manual, overtemperaturer ΔT	Low pressurizer pressure manual	Essential service water system, component cooling water system, steam generator shell side fluid operating system, steam generator safety and/or atmospheric relief valves, steam line isolation valves	Emergency core cooling system, auxiliary feed-water system, emergency power system containment isolation
Loss of coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	Pressurizer low pressure, SIS, manual	Hi-1 containment pressure, low pressurizer pressure Hi-3 containment pressure, manual AFAS on loss of offsite power	Essential service water system, component cooling water system, steam gen-erator safety and/or atmospheric relief valves	Emergency core cooling system, auxiliary feedwater system, containment heat removal system, emergency power system containment isolation

TABLE 15.0-7

SINGLE FAILURES ASSUMED IN ACCIDENT ANALYSES

Event Description	Worst Failure Assumed
Feedwater temperature reduction	(1)
Excessive feedwater flow	One protection train
Excessive steam flow	(1)
Inadvertent secondary depressurization	One safety injection train
Steam system piping failure	One safety injection train
Steam pressure regulator malfunction	(2)
Loss of external load	One protection train
Turbine trip	One protection train
Inadvertent closure of MSIV	One protection train
Loss of condenser vacuum	One protection train
Loss of ac power	One auxiliary feedwater pump
Loss of normal feedwater	One auxiliary feedwater pump
Feedwater system pipe break	One protection train
Partial loss of forced reactor coolant flow	One protection train
Complete loss of forced reactor coolant flow	One protection train
RCP locked rotor	One protection train
RCP shaft break	One protection train
RCCA bank withdrawal from subcritical	One protection train
RCCA bank withdrawal at power	One protection train
Dropped RCCA, dropped RCCA bank	One nuclear instrumentation system channel
Statically misaligned RCCA	(3)
Single RCCA withdrawal	One protection train
Inactive RC pump startup	One protection train
Flow controller malfunction	(2)
Uncontrolled boron dilution	Standby charging pump is operating (4)
Improper fuel loading	(3)
RCCA ejection	One protection train
Inadvertent ECCS operation at power	One protection train
Increase in RCS inventory	One protection train
BWR transients	(2)
Inadvertent RCS depressurization	One protection train
Failure of small lines carrying primary coolant outside containment	(3)
Steam Generator Tube Rupture	AFW flow control valve to ruptured SG fails open
Fuel Handling Accident	(3)
BWR piping failures	(2)
Spectrum of LOCA	
Small breaks	One safety injection train
Large breaks	One RHR pump

NOTES:
(1) No protection action required
(2) Not applicable to WCGS
(3) No transient analysis involved
(4) Applies to power and startup operations only. A single failure in the VCT level controller which results in the continuation of the dilution flow, even after the “stop auto makeup” setpoint is exceeded, is assumed for Mode 3, 4 and 5.

TABLE 15.0-8

OPERATOR ACTIONS [1] REQUIRED FOR SMALL AND LARGE LOCAS

Time	Operator Action
Reactor trip signal is actuated [2]	None
Safety injection signal is actuated [2]	None
Prior to generation of RWST low level signal	Reset safeguards actuation signal. Check sump water level indicator.
Generation of RWST low level signal properly	Verify completion of automatic switch-over to ensure components have been realigned. Perform the additional valve alignments required for switchover to recirculation [3].
Switchover to cold leg injection plus 10 hours	Perform operations necessary to switch to simultaneous hot and cold leg recirculation.
To final stabilized condition with	Monitor system pressure and temperature. Control pressurizer water level safety injection system.

(1) Actions associated with primary system protection.
(2) These times can be found in the sequence of events tables in Section 15.6.5.
(3) See Section 6.3.2.8 for the manual actions required for completion of switchover. Operator actions associated with containment protection is discussed in Section 6.2.

TABLE 15.0-9

SHORT TERM OPERATOR ACTIONS REQUIRED FOR STEAM GENERATOR TUBE RUPTURE

Time	System	Operator Action
Reactor trip signal actuated	Reactor trip system	None
Safety injection signal actuated	Safety injection system	None
Post SI signal generation to plant stabilization	Safety injection system, auxiliary feedwater system, steam dump system	Perform operator actions necessary to isolate the ruptured steam generator. [Isolate steam flow from and feed flow to the ruptured steam generator]
		Reduce RCS temperature to increase RCS subcooling. Depressurize RCS to equilibrate RCS and ruptured steam generator pressure. Terminate SI, establish normal charging/letdown flow.
		Proceed with normal plant cooldown, while monitoring reactor coolant system pressure, temperature, and boron concentration.

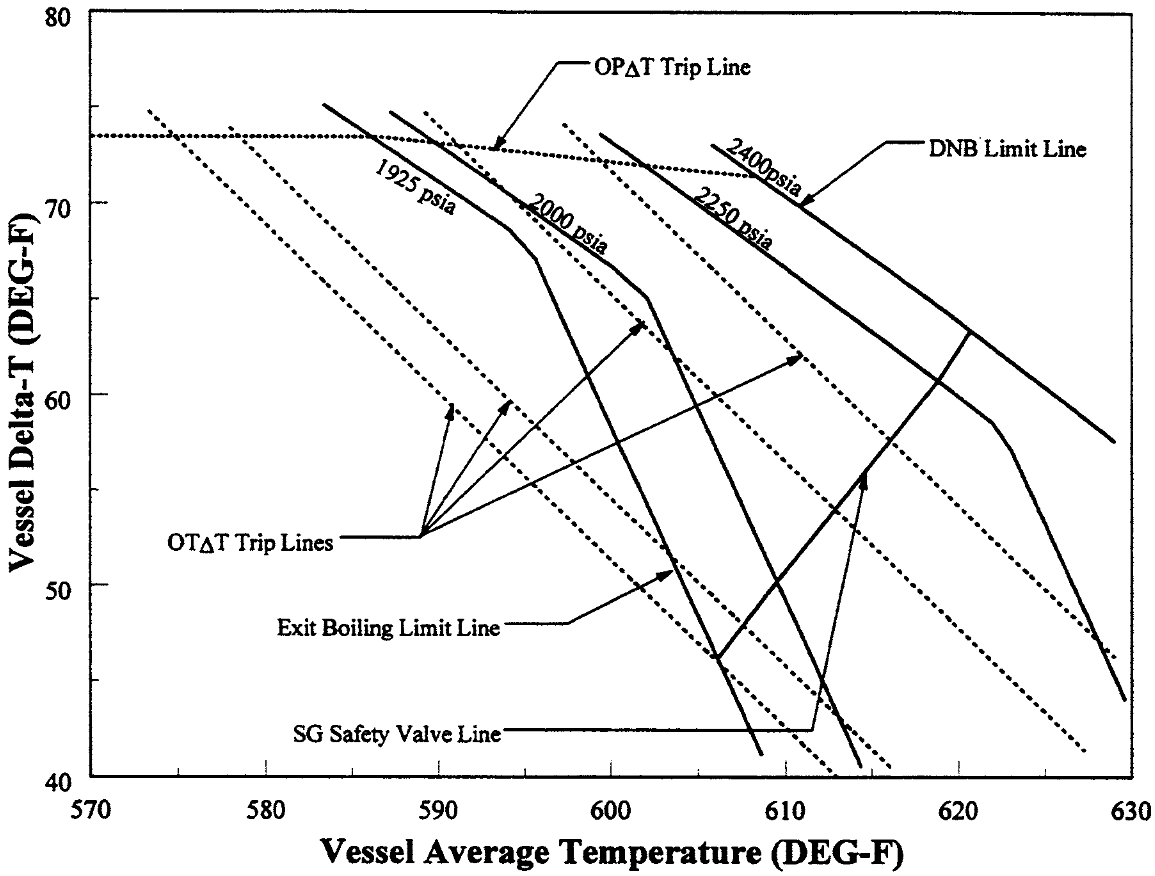

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FIGURE 15.0-1
Illustration of Overtemperature and Overpower ΔT Protection

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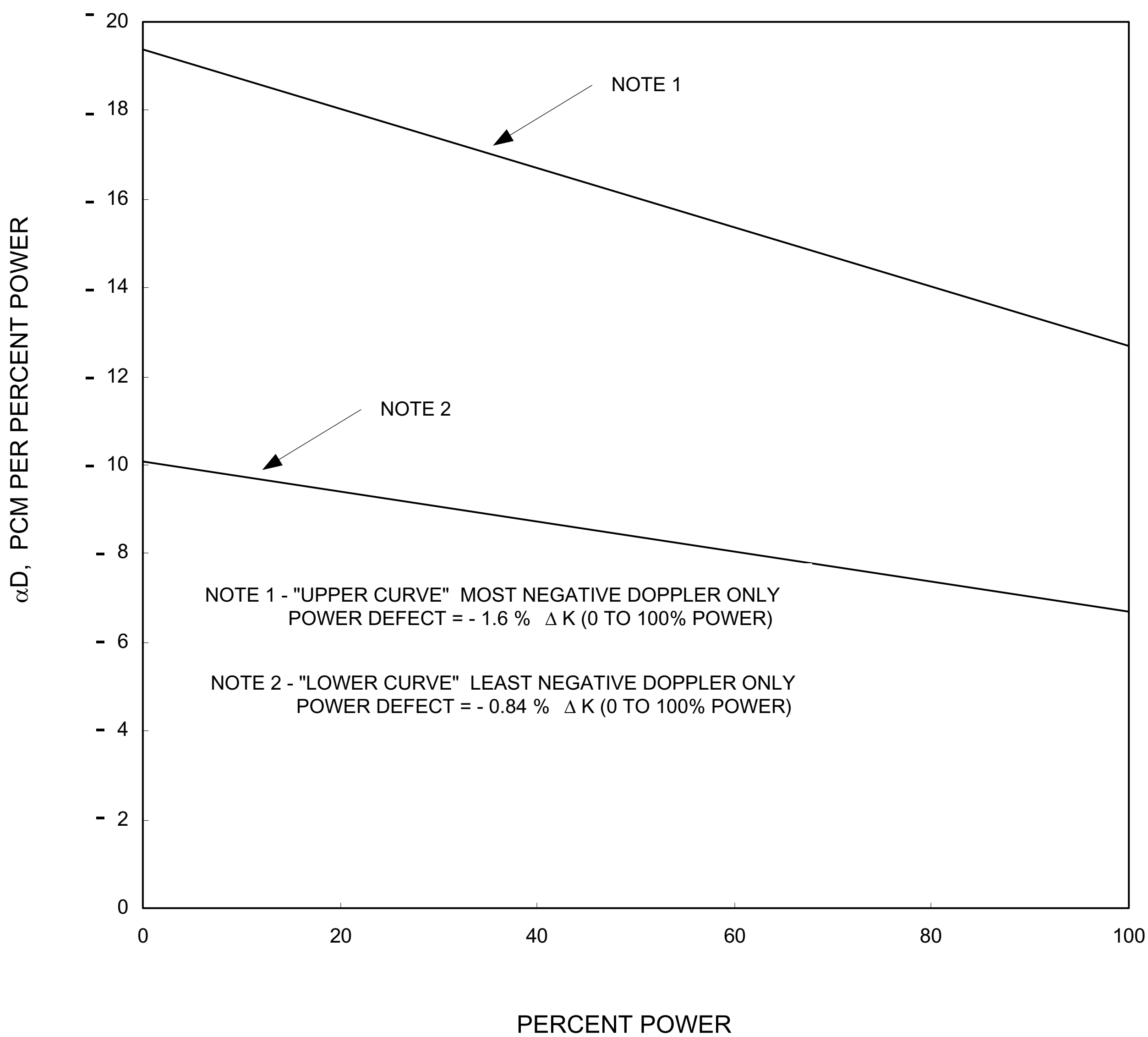

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FIGURE 15.0-2

DOPPLER POWER COEFFICIENT
USED IN ACCIDENT ANALYSIS

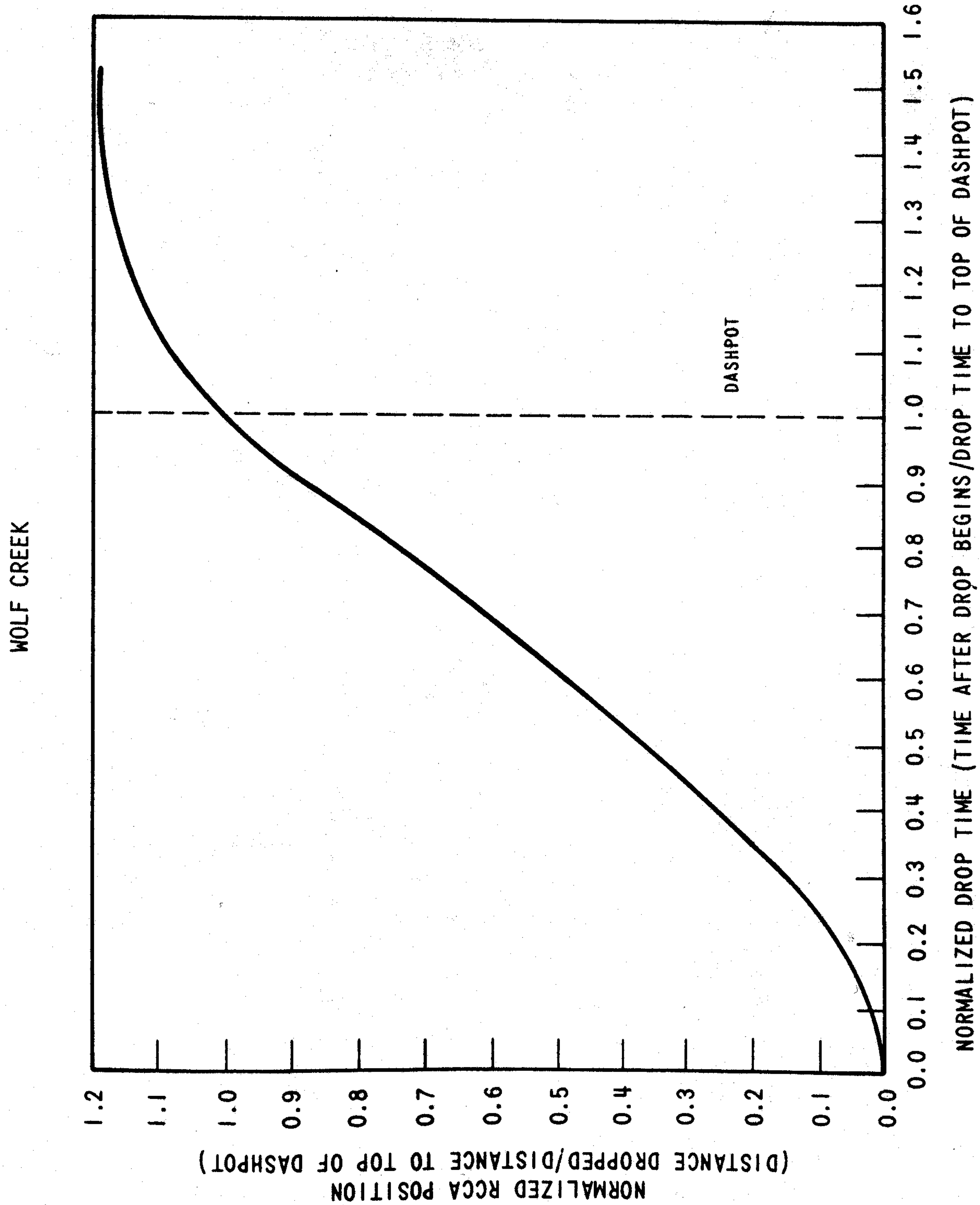

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FIGURE 15.0-3

RCCA POSITION VERSUS TIME TO DASHPOT

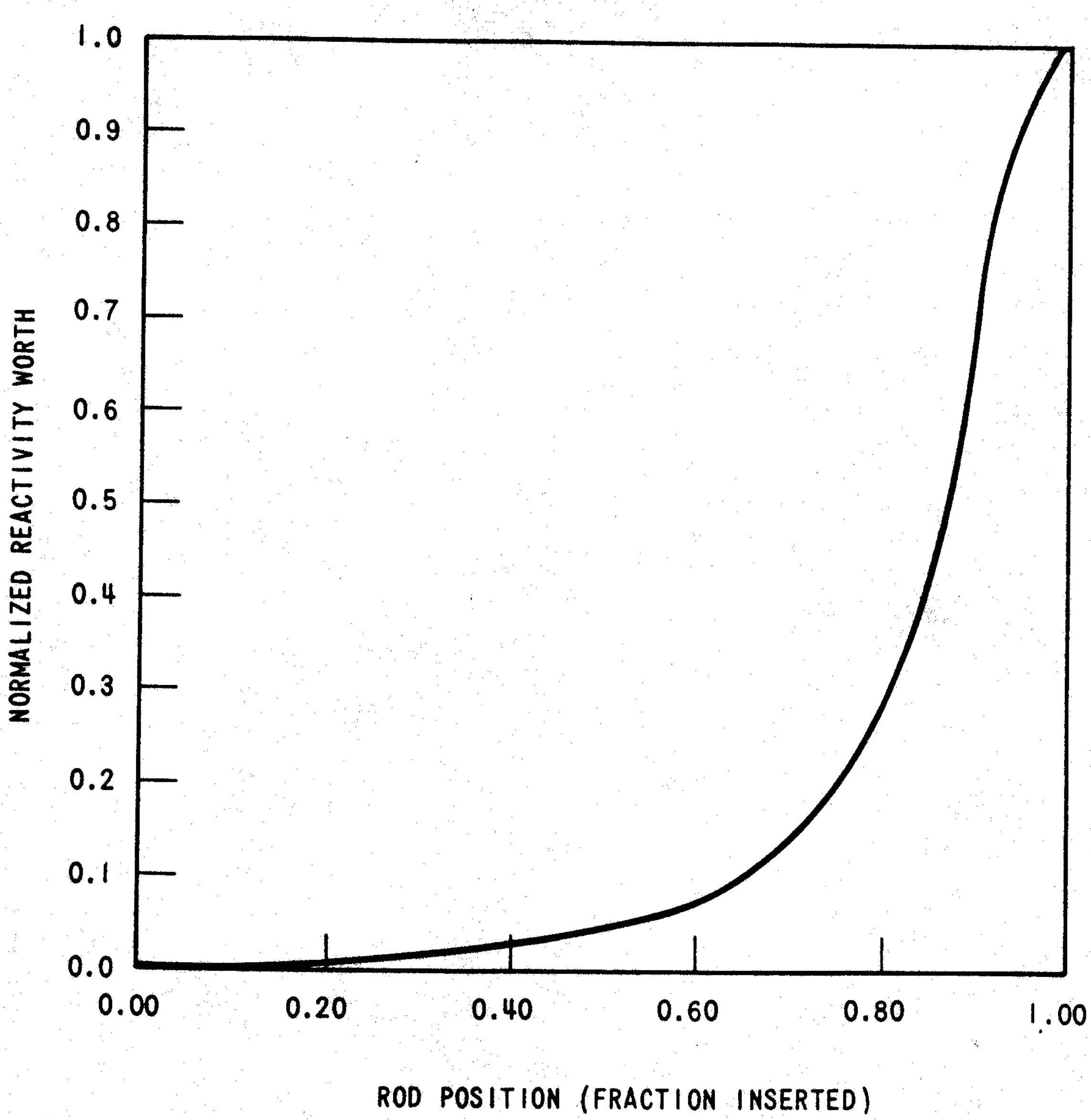

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FIGURE 15.0-4

NORMALIZED ROD WORTH VERSUS PERCENT INSERTED

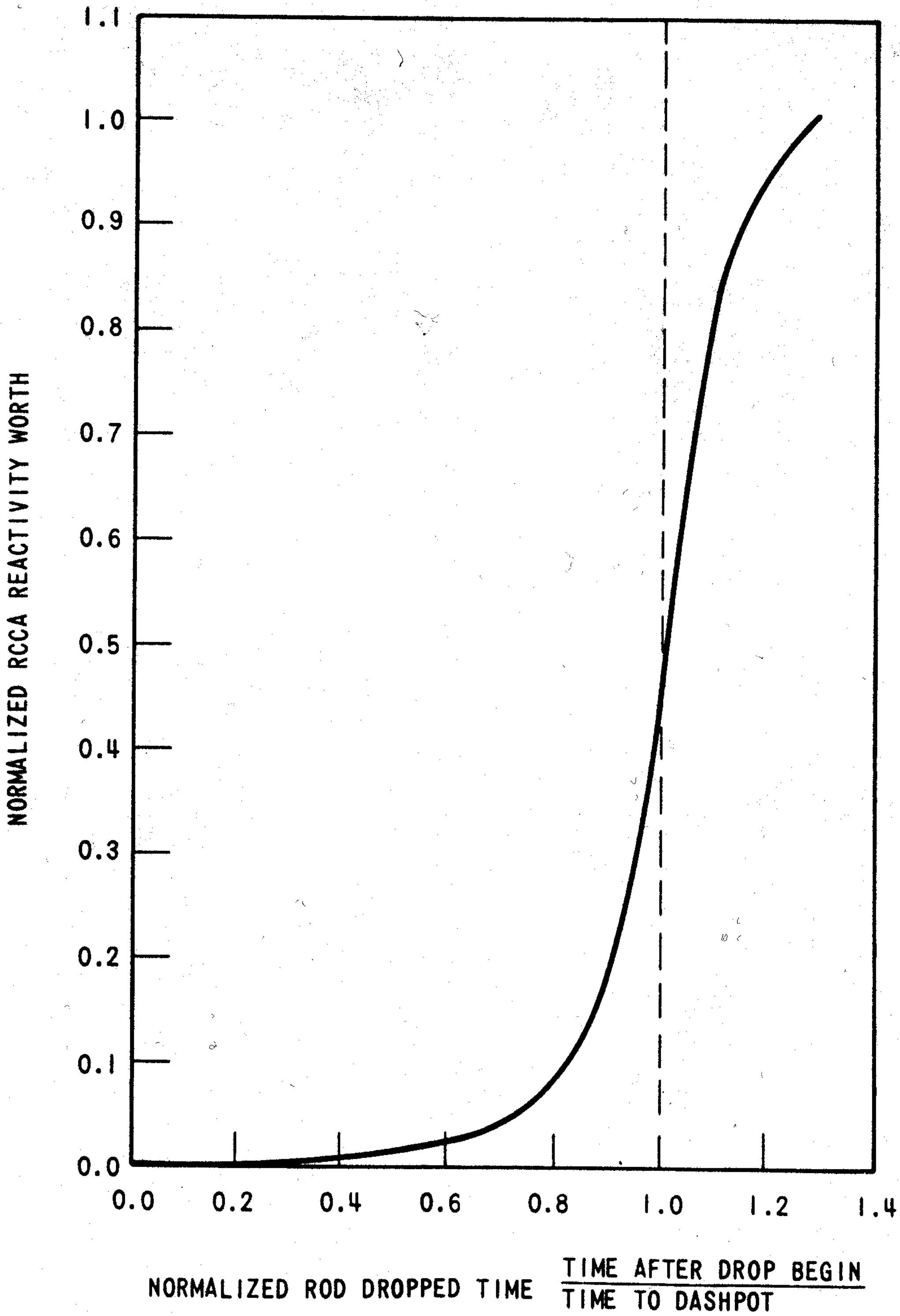

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FIGURE 15.0-5

NORMALIZED RCCA BANK REACTIVITY WORTH VERSUS NORMALIZED DROP TIME

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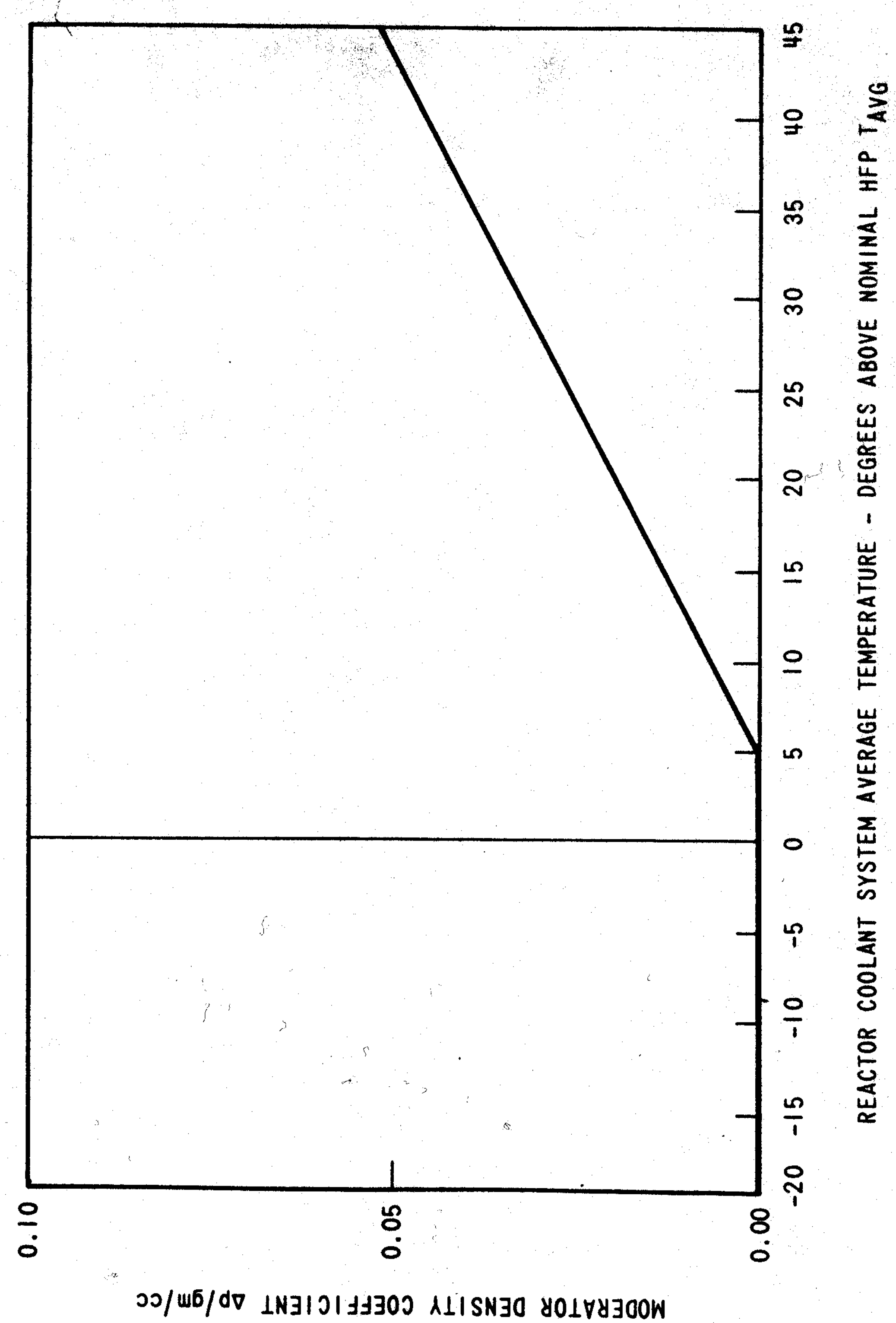

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FIGURE 15.0-6

MINIMUM MODERATOR DENSITY COEFFICIENT USED IN ANALYSIS

ABBREVIATIONS USED:

AFWS	-	AUXILIARY FEEDWATER SYSTEM
CVCS	-	CHEMICAL AND VOLUME CONTROL SYSTEM
ESFAS	-	ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
FW	-	FEEDWATER
RTS	-	REACTOR TRIP SYSTEM
SIS	-	SAFETY INJECTION SYSTEM
SI	-	SAFETY INJECTION
RT	-	REACTOR TRIP
CS	-	CONTIANMENT SPRAY
ESF	-	ENGINEERED SAFETY FEATURE
ECCS	-	EMERGENCY CORE COOLING SYSTEM
HL	-	HOT LEG
CL	-	COLD LEG
CCWS	-	COMPONENT COOLING WATER SYSTEM
RCS	-	REACTOR COOLANT SYSTEM
ESWS	-	ESSENTIAL SERVICE WATER SYSTEM
HPI	-	HIGH PRESSURE INJECTION
LPI	-	LOW PRESSURE INJECTION
CI	-	CONTAINMENT ISOLATION
SG	-	STEAM GENERATOR
GWPS	-	GASEOUS WASTE PROCESSING SYSTEM

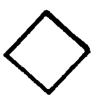
NOTES;

1. FOR TRIP INITIATION AND SAFETY SYSTEM ACTUATION. MULTIPLE SIGNALS ARE SHOWN BUT ONLY A SINGLE SIGNAL IS REQUIRED. THE OTHER SIGNALS ARE BACKUPS.

2. NO TIMING SEQUENCE IS IMPLIED BY POSITION OF VARIOUS BRANCHES. REFER TO EVENT TIMING SEQUENCES PRESENTED IN TABULAR FORM IN PERTINENT ACCIDENT ANALYSIS SECTION OF CHAPTER 15.0 OF THE FSAR.

DIAGRAM SYMBOLS:


- EVENT TITLE


- BRANCH POINT FOR DIFFERENT PLANT CONDITIONS


- SAFETY SYSTEM


- SAFETY ACTION

- SYSTEM REQUIRED TO MEET SINGLE-FAILURE CRITERIA

- MANUAL ACTION REQUIRED DURING SYSTEM OPERATION

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FIGURE 15.0 7

ABBREVIATIONS AND SYMBOLS USED IN SEQUENCE DIAGRAMS

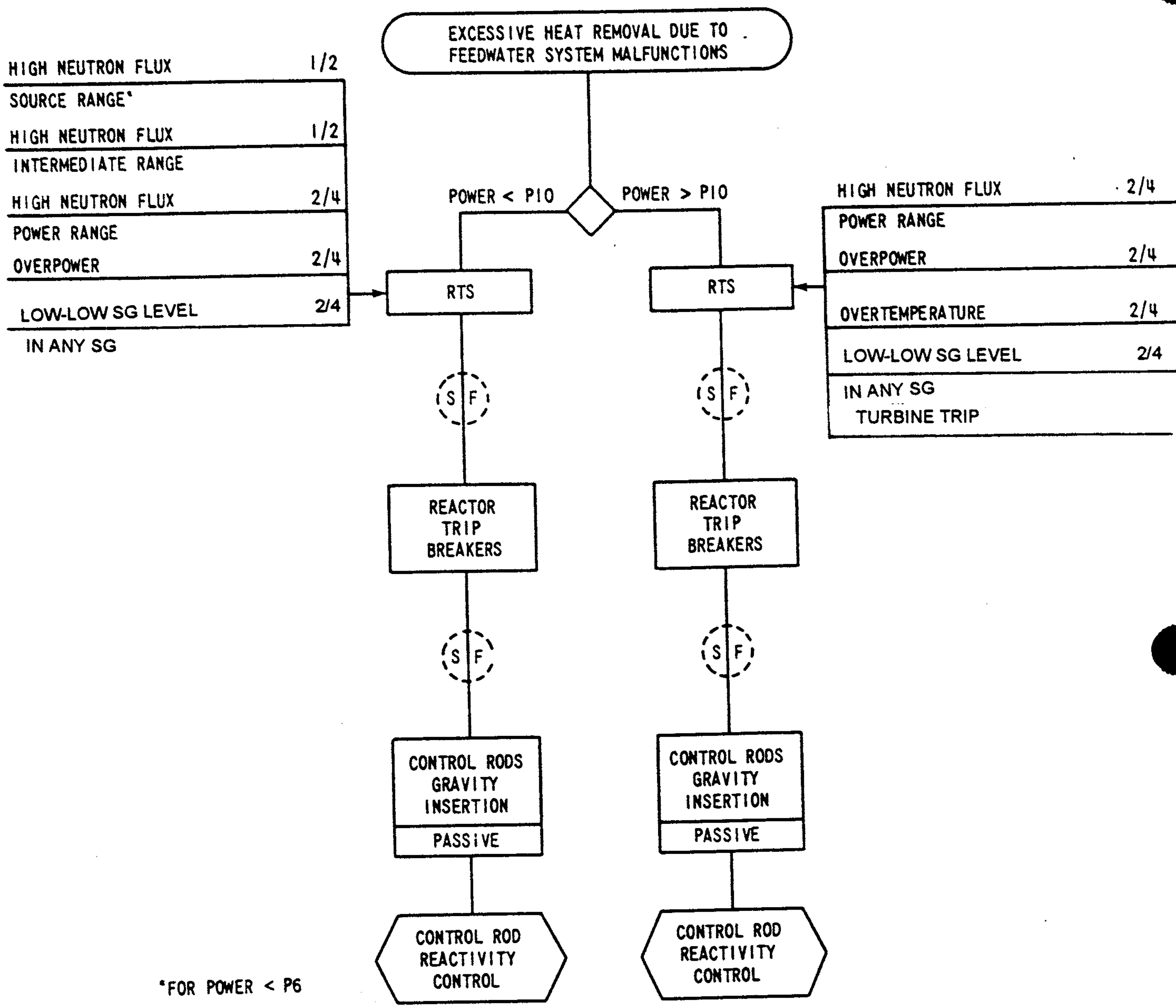

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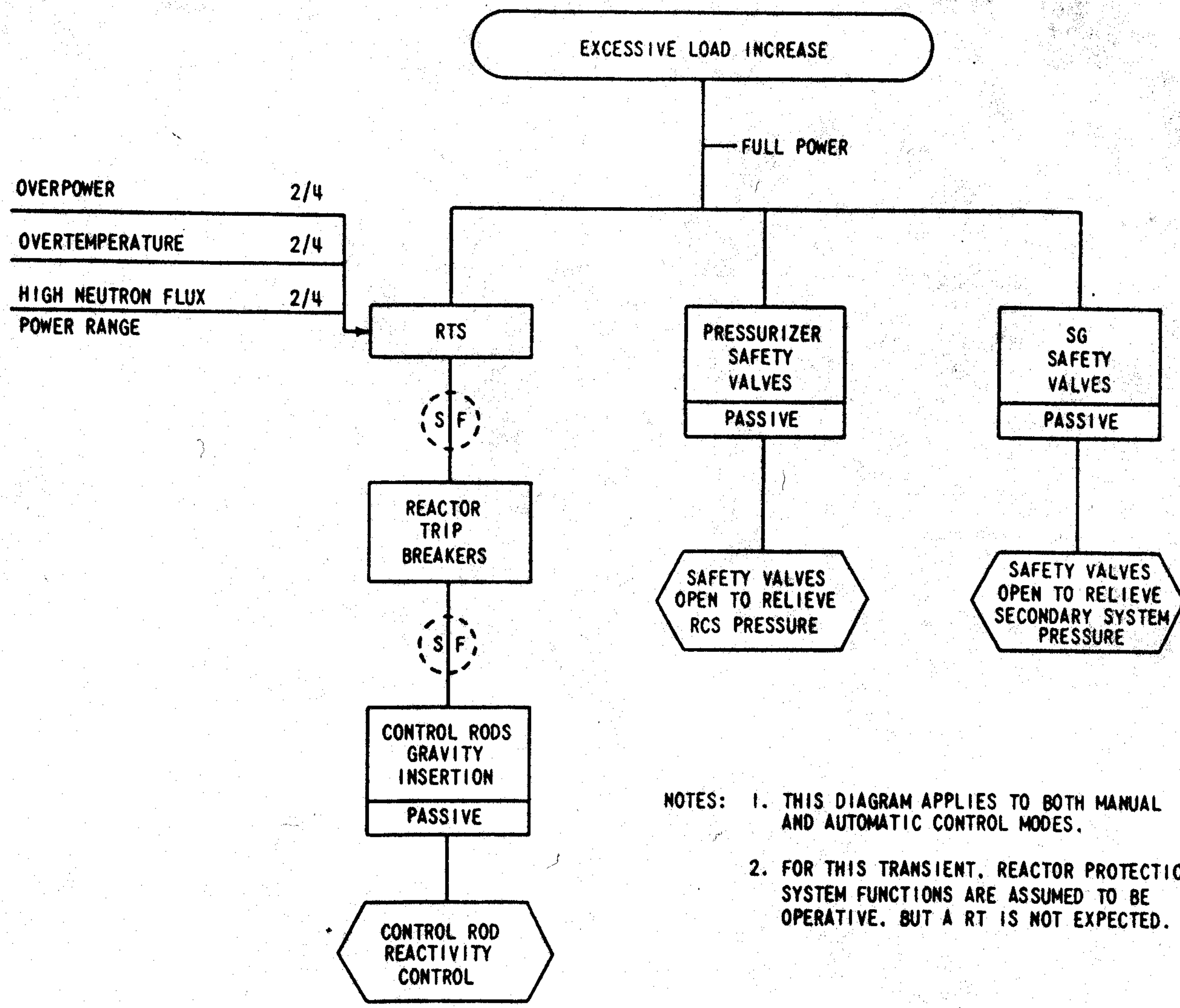
FIGURE 15.0-8

EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEMS MALFUNCTION



NOTES: 1. THIS DIAGRAM APPLIES TO BOTH MANUAL AND AUTOMATIC CONTROL MODES.

2. FOR THIS TRANSIENT, REACTOR PROTECTION SYSTEM FUNCTIONS ARE ASSUMED TO BE OPERATIVE, BUT A RT IS NOT EXPECTED.

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FIGURE 15.0-9

EXCESSIVE LOAD INCREASE

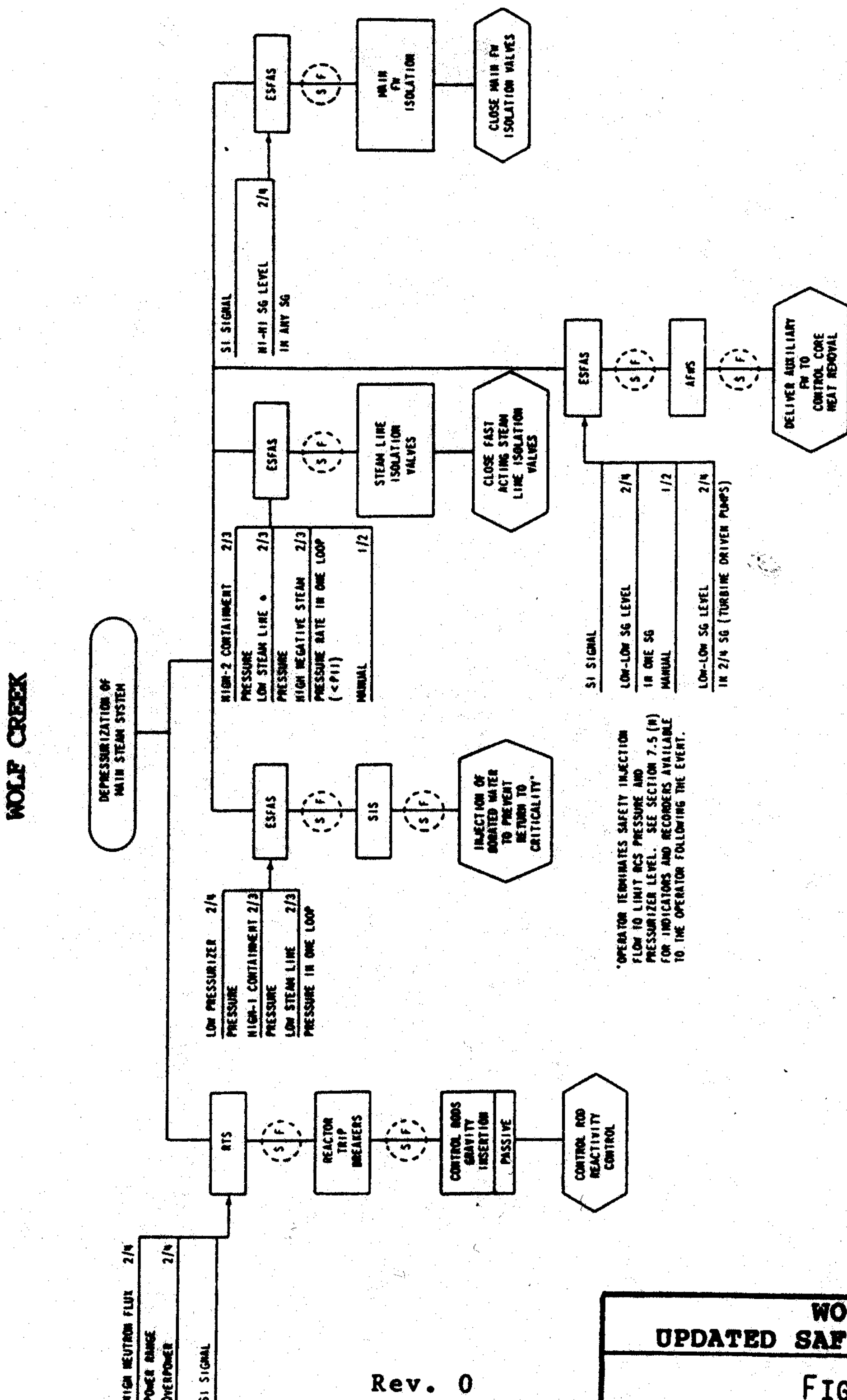

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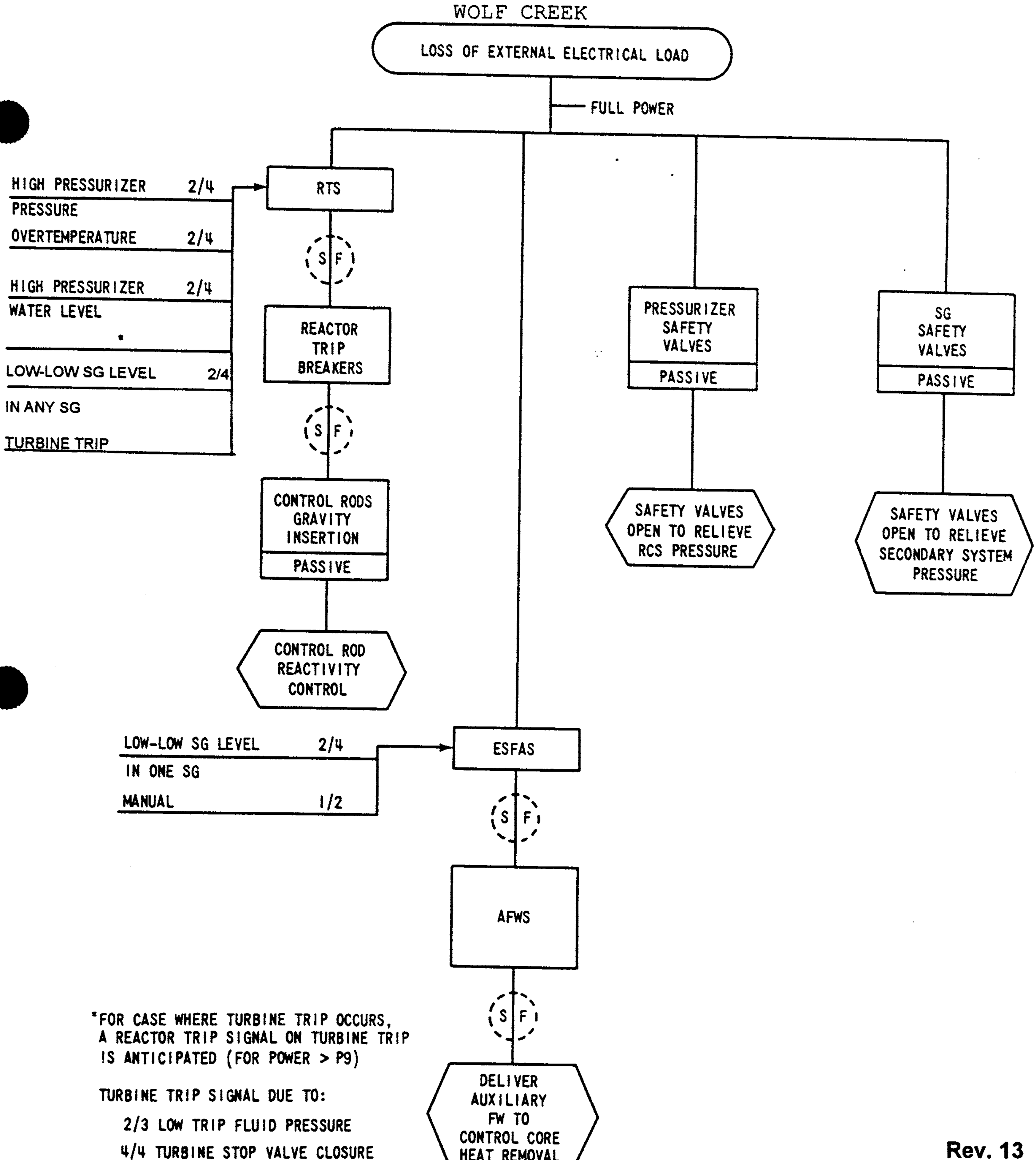
FIGURE 15.0-10

DEPRESSURIZATION OF MAIN STEAM SYSTEM



*FOR CASE WHERE TURBINE TRIP OCCURS, A REACTOR TRIP SIGNAL ON TURBINE TRIP IS ANTICIPATED (FOR POWER > P9)

TURBINE TRIP SIGNAL DUE TO:

2/3 LOW TRIP FLUID PRESSURE

4/4 TURBINE STOP VALVE CLOSURE

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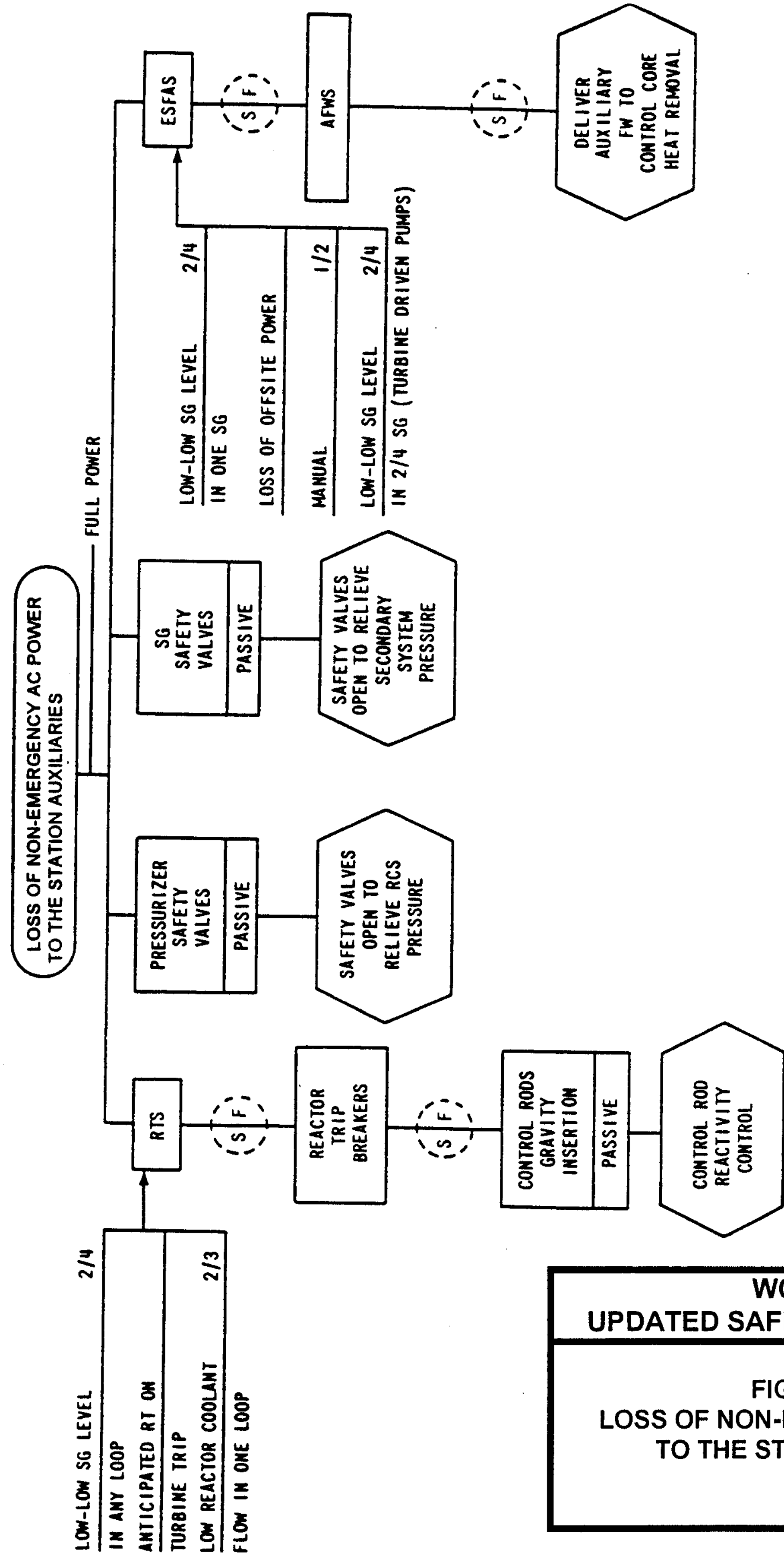
FIGURE 15.0-11

LOSS OF EXTERNAL LOAD



Rev. 14

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FIGURE 15.0-12
LOSS OF NON-EMERGENCY AC POWER
TO THE STATION AUXILIARIES

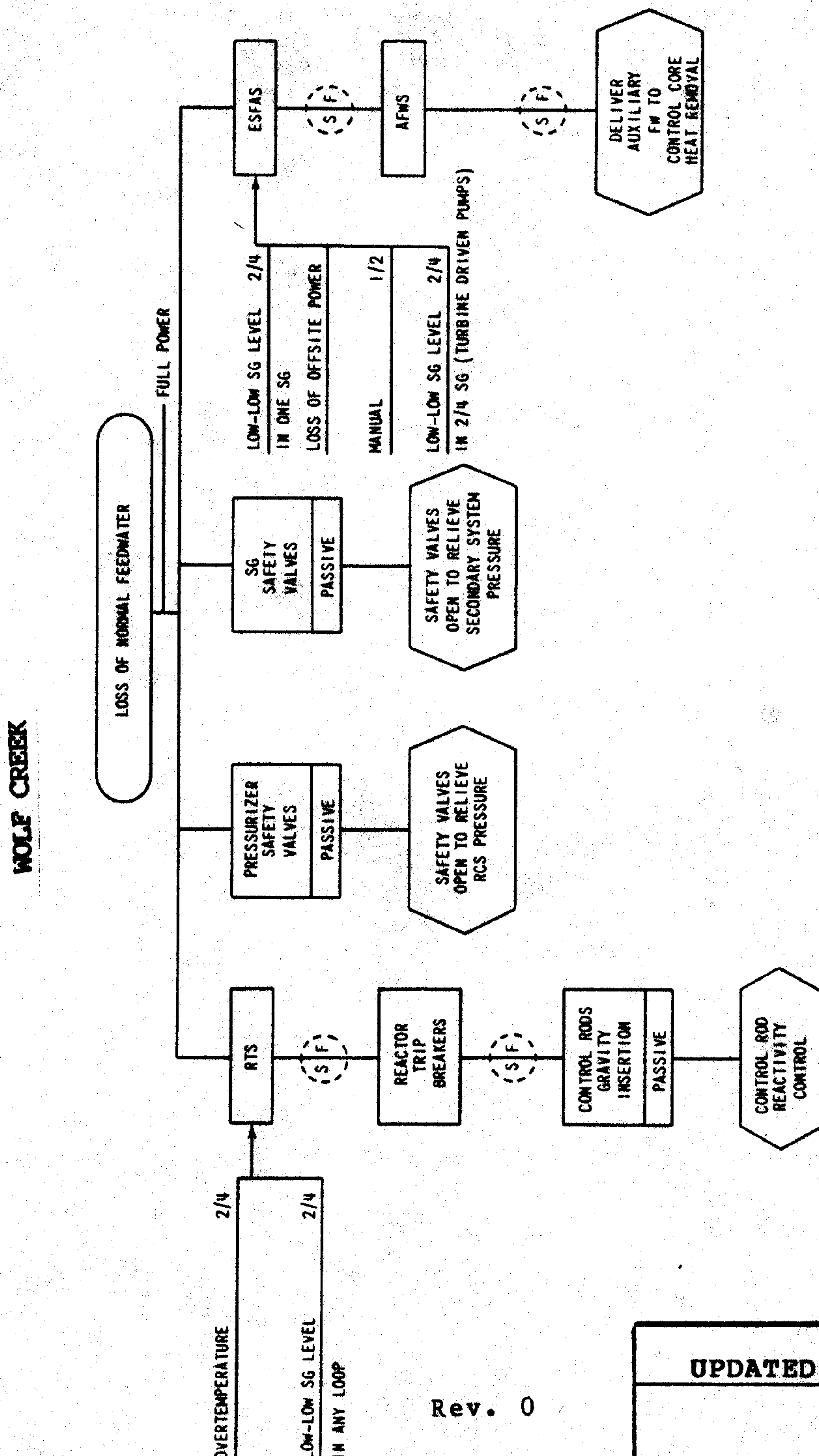

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FIGURE 15.0-13

LOSS OF NORMAL FEEDWATER

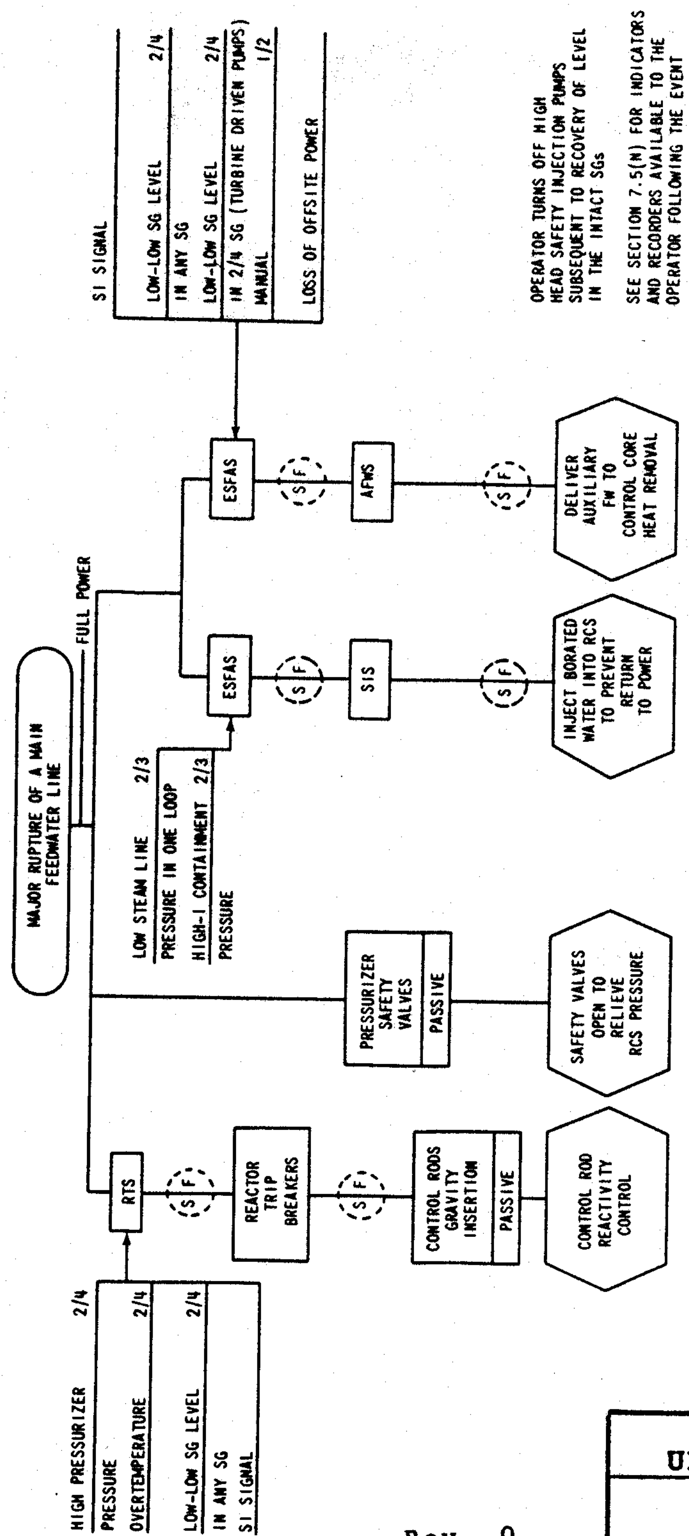

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FIGURE 15.0-14

MAJOR RUPTURE OF A MAIN FEEDWATER LINE

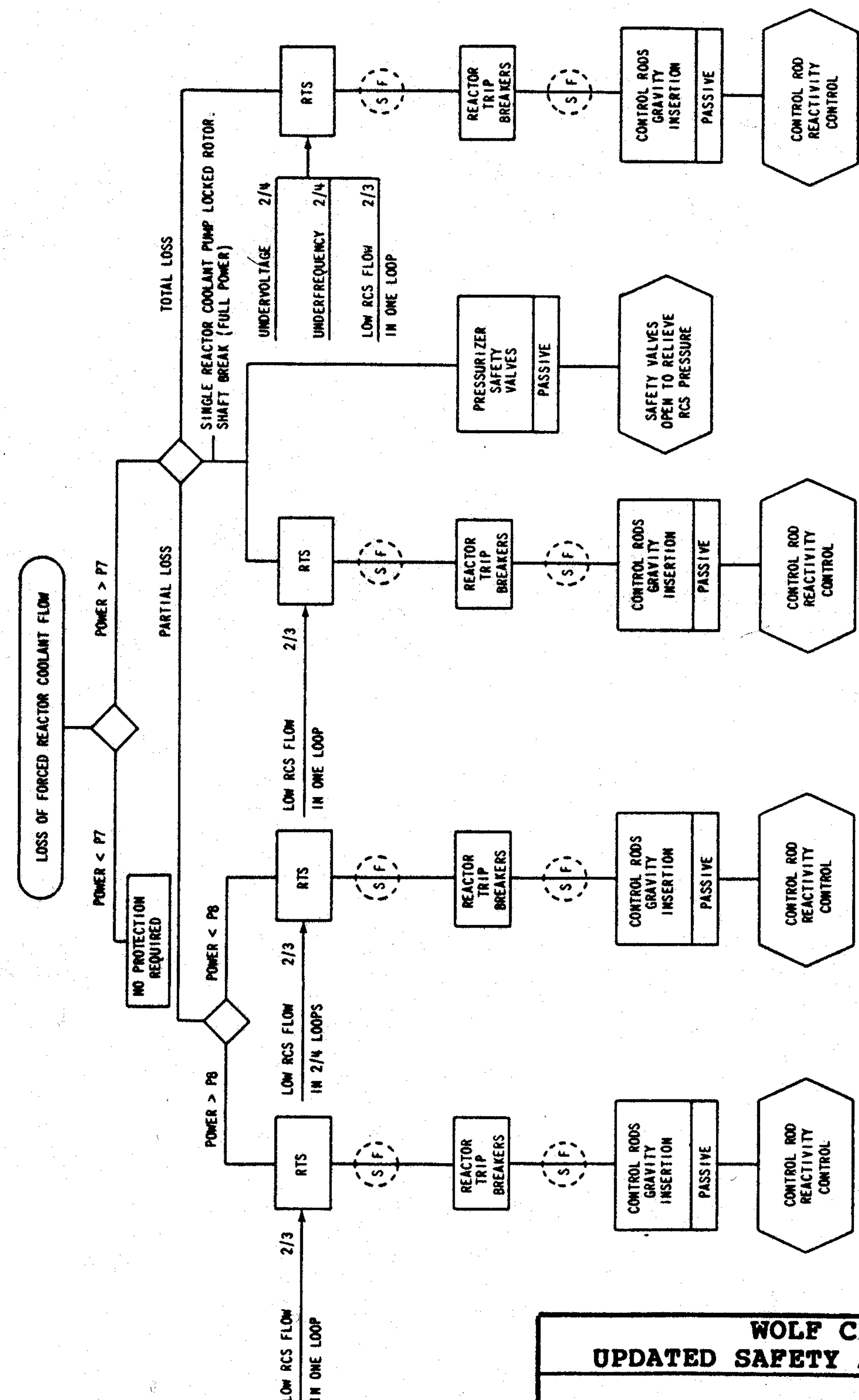

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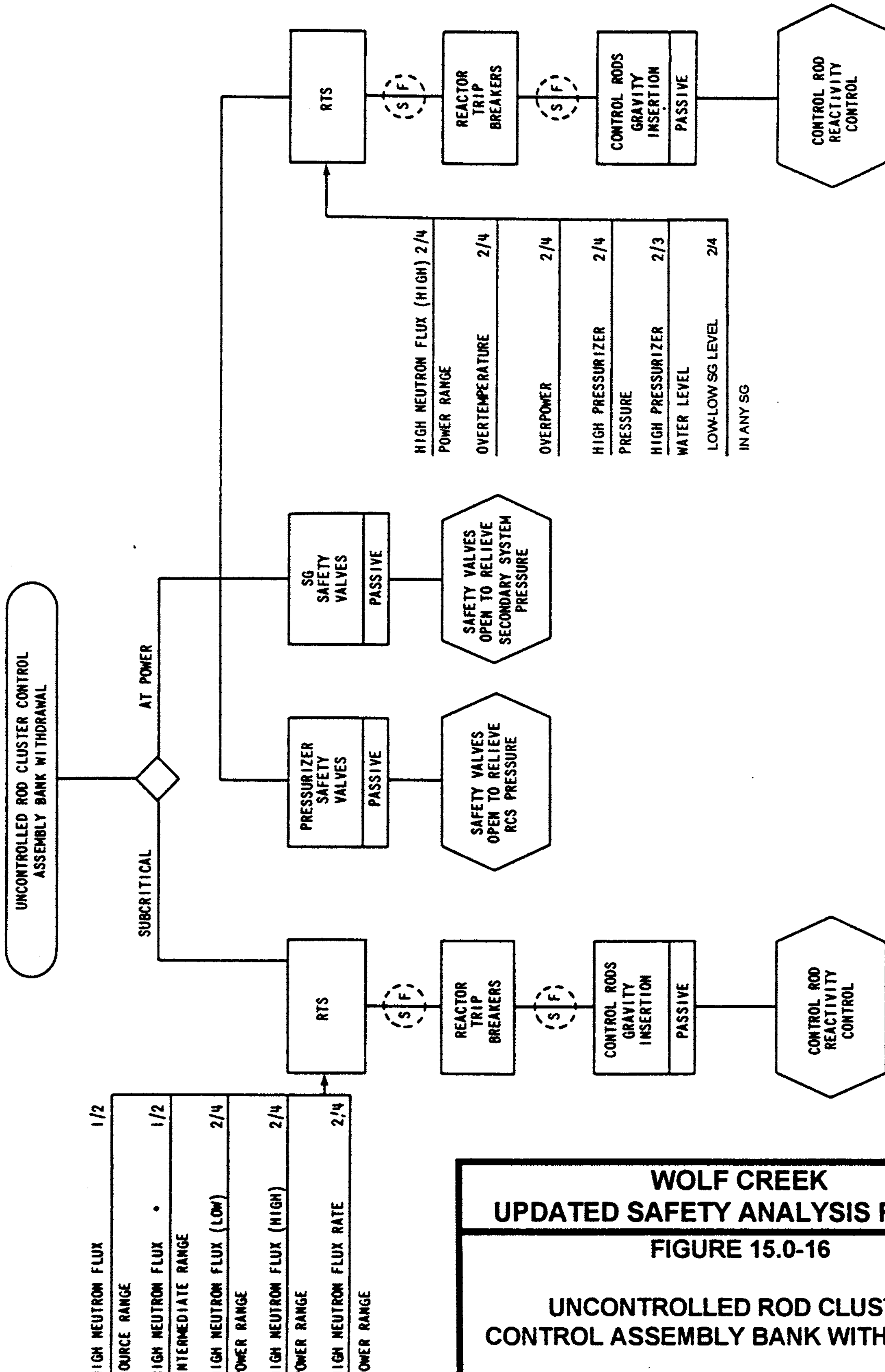
FIGURE 15.0-15

LOSS OF FORCED REACTOR COOLANT FLOW



Rev. 13

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FIGURE 15.0-16 UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL

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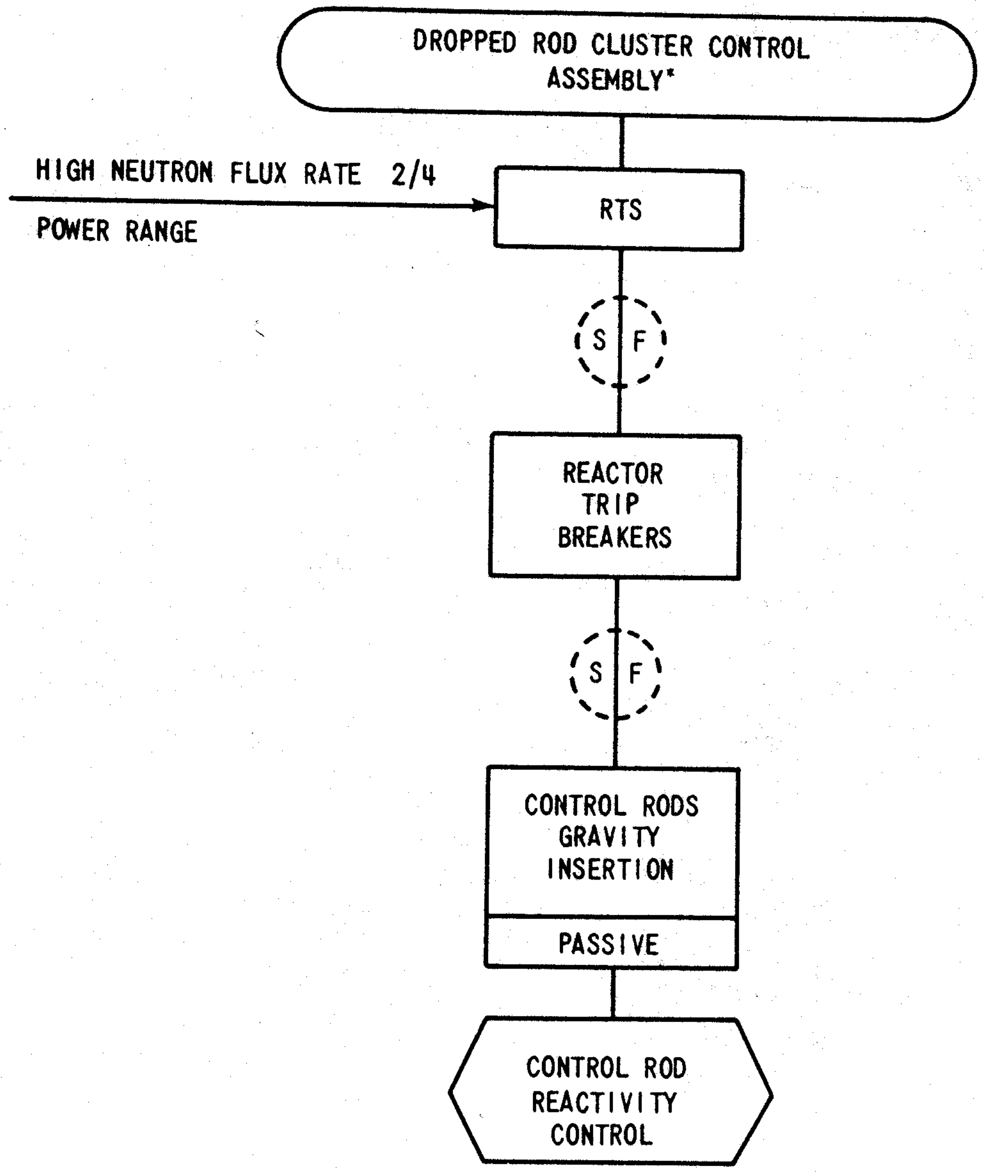

*TRIP SEQUENCE OCCURS ONLY IF ROD WORTH IS GREATER THAN MINIMUM REQUIRED TO TRIGGER THE FLUX RATE TRIP, OR IF AN ENTIRE ROD CLUSTER CONTROL ASSEMBLY BANK DROPS. FOR OTHER ROD WORTH, OR FOR MISALIGNMENT, NO PROTECTION REQUIRED.

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FIGURE 15.0-17

DROPPED ROD CLUSTER CONTROL ASSEMBLY

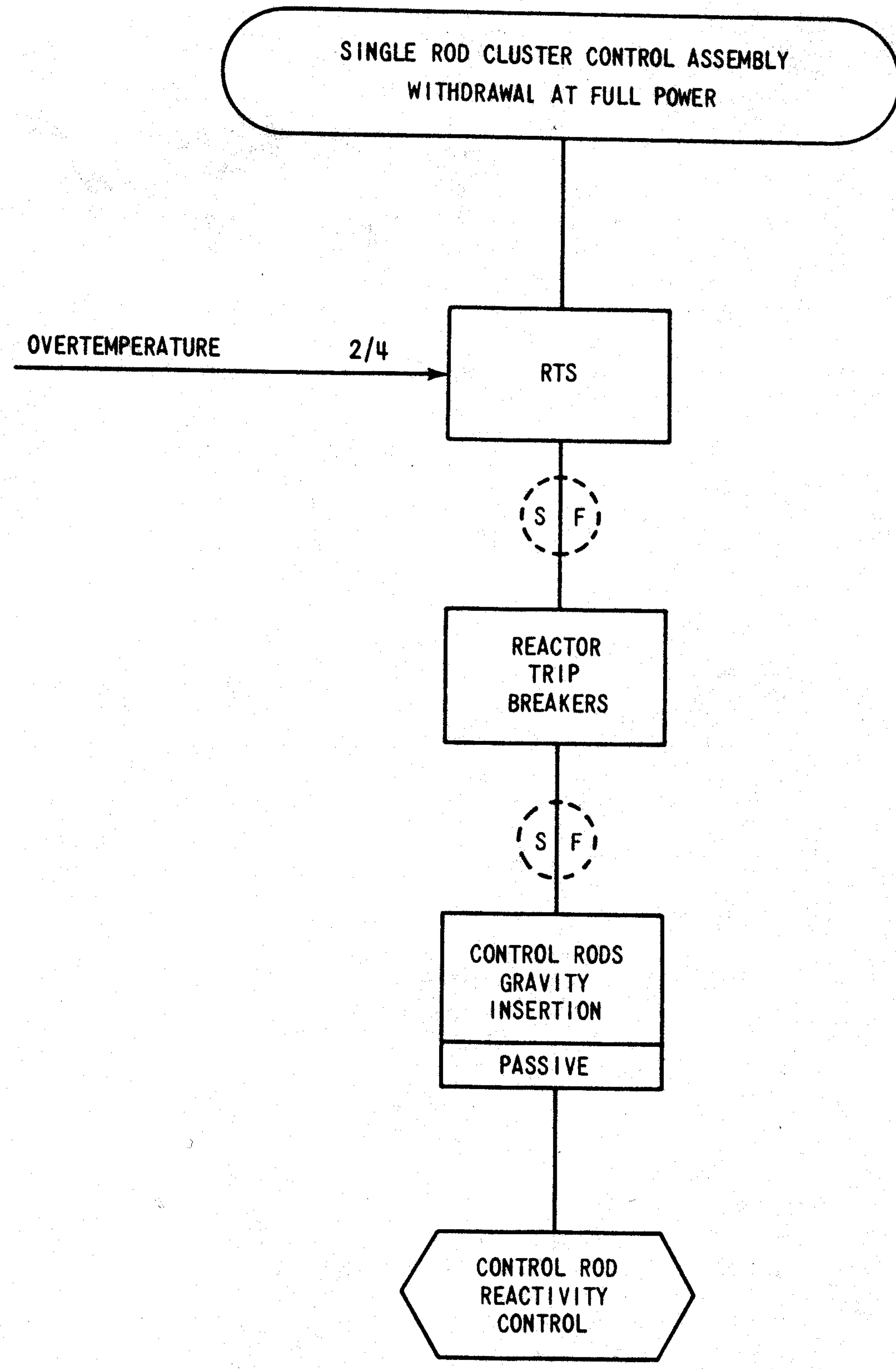

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FIGURE 15.0-18

SINGLE ROD CLUSTER CONTROL ASSEMBLY WITHDRAWAL AT FULL POWER

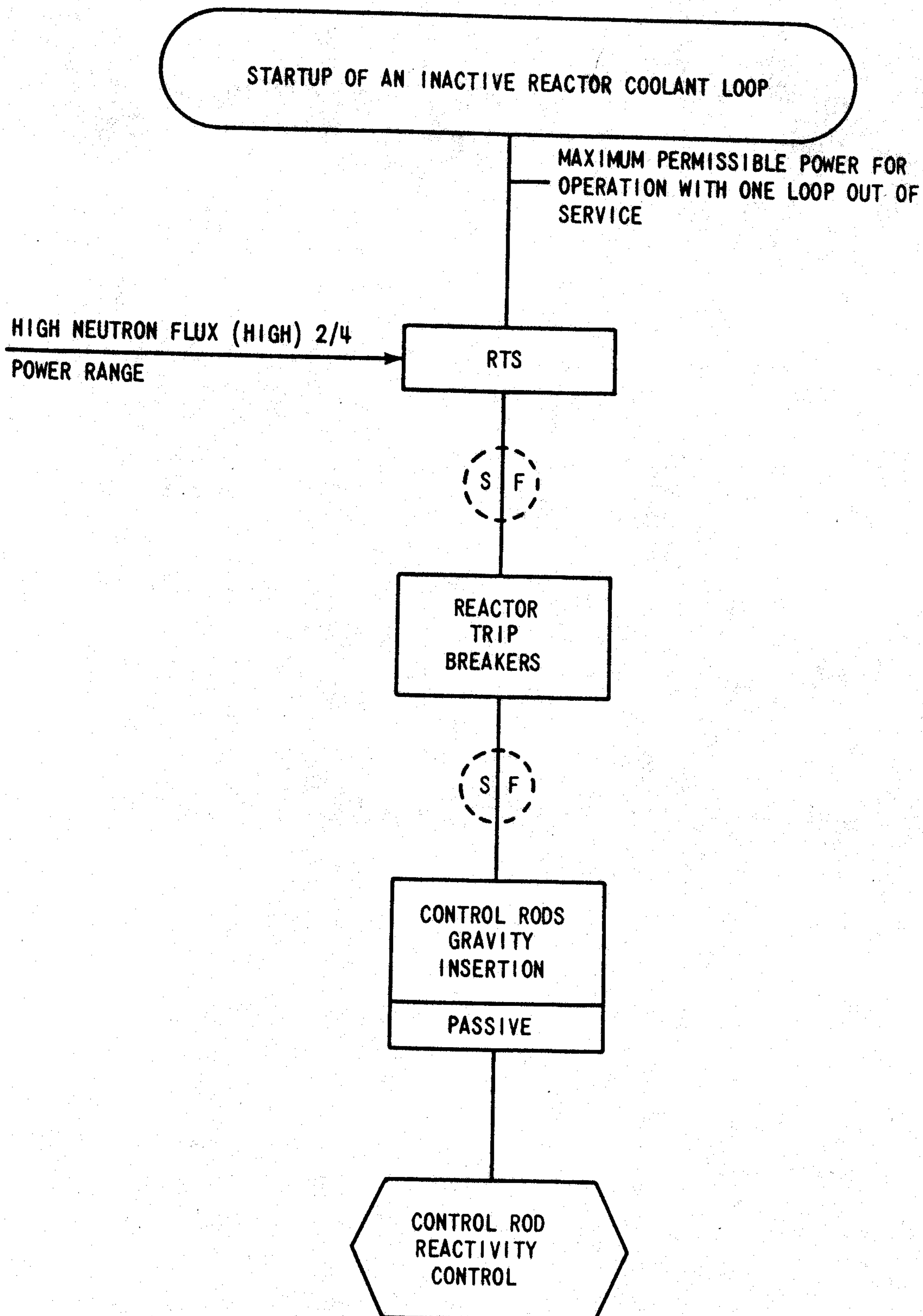

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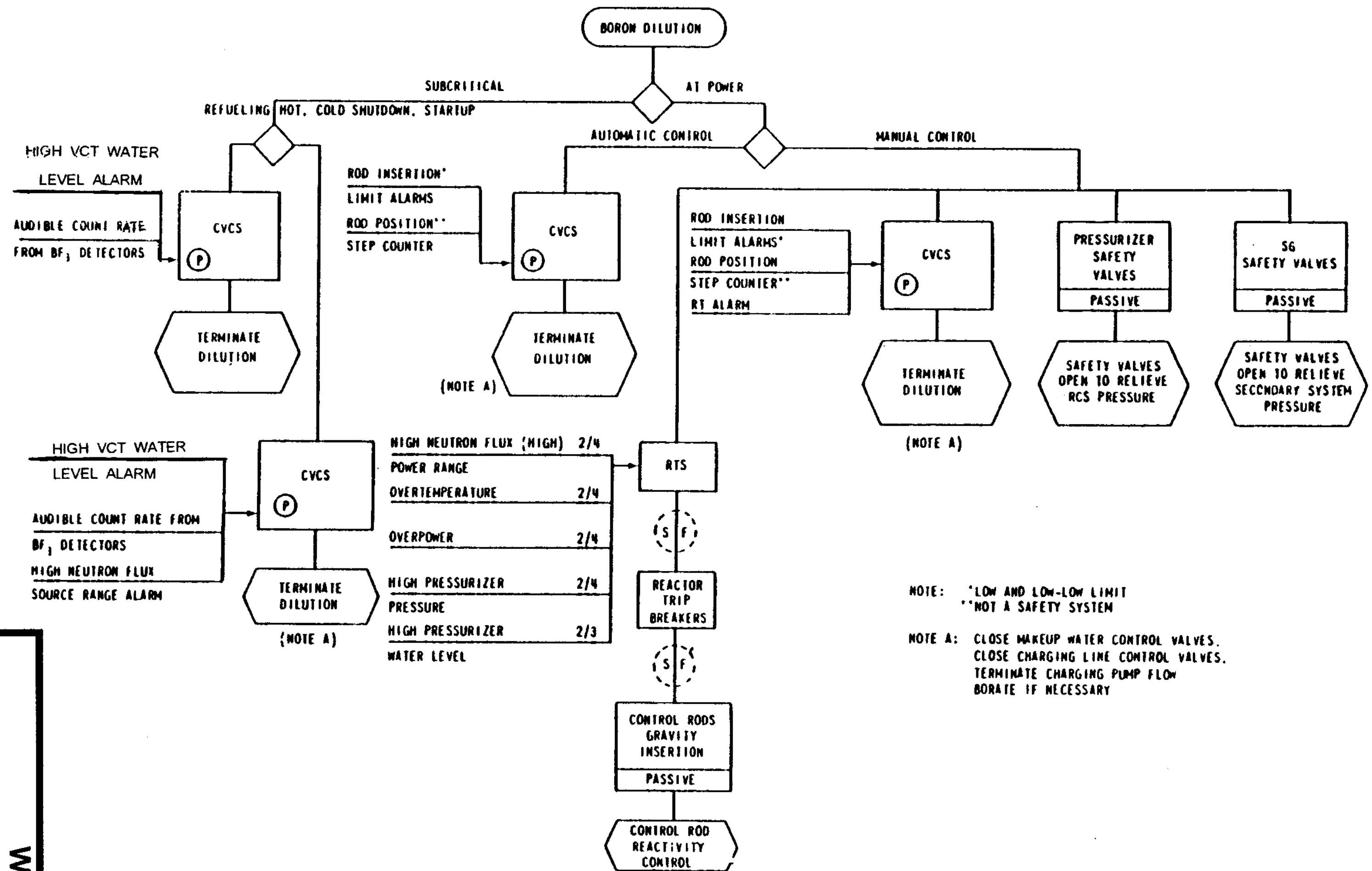
FIGURE 15.0-19

STARTUP OF AN INACTIVE REACTOR COOLANT LOOP



NOTE: *LOW AND LOW-LOW LIMIT
**NOT A SAFETY SYSTEM

NOTE A: CLOSE MAKEUP WATER CONTROL VALVES.
CLOSE CHARGING LINE CONTROL VALVES.
TERMINATE CHARGING PUMP FLOW
BORATE IF NECESSARY

Rev. 14

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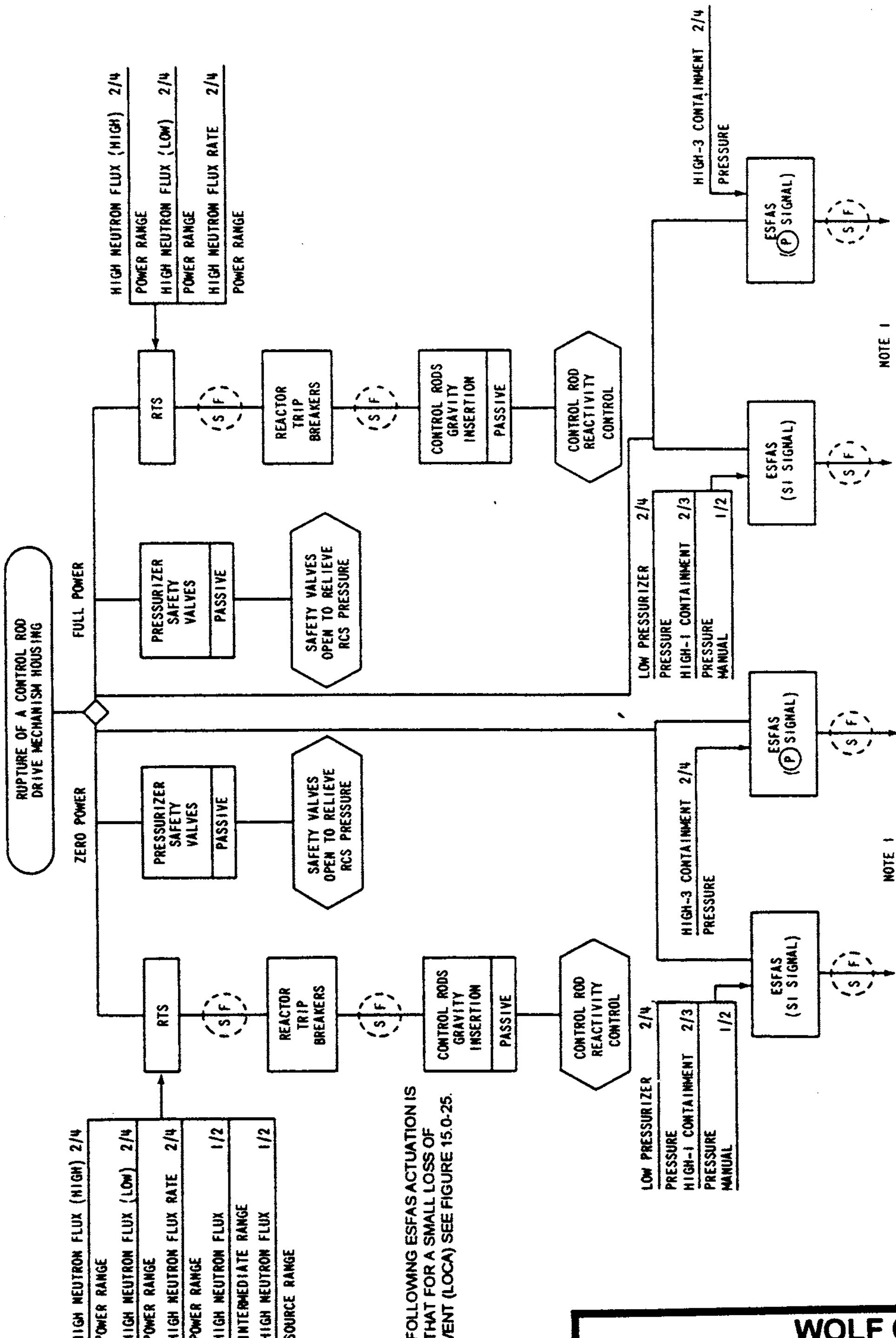
FIGURE 15.0-20
BORON DILUTION



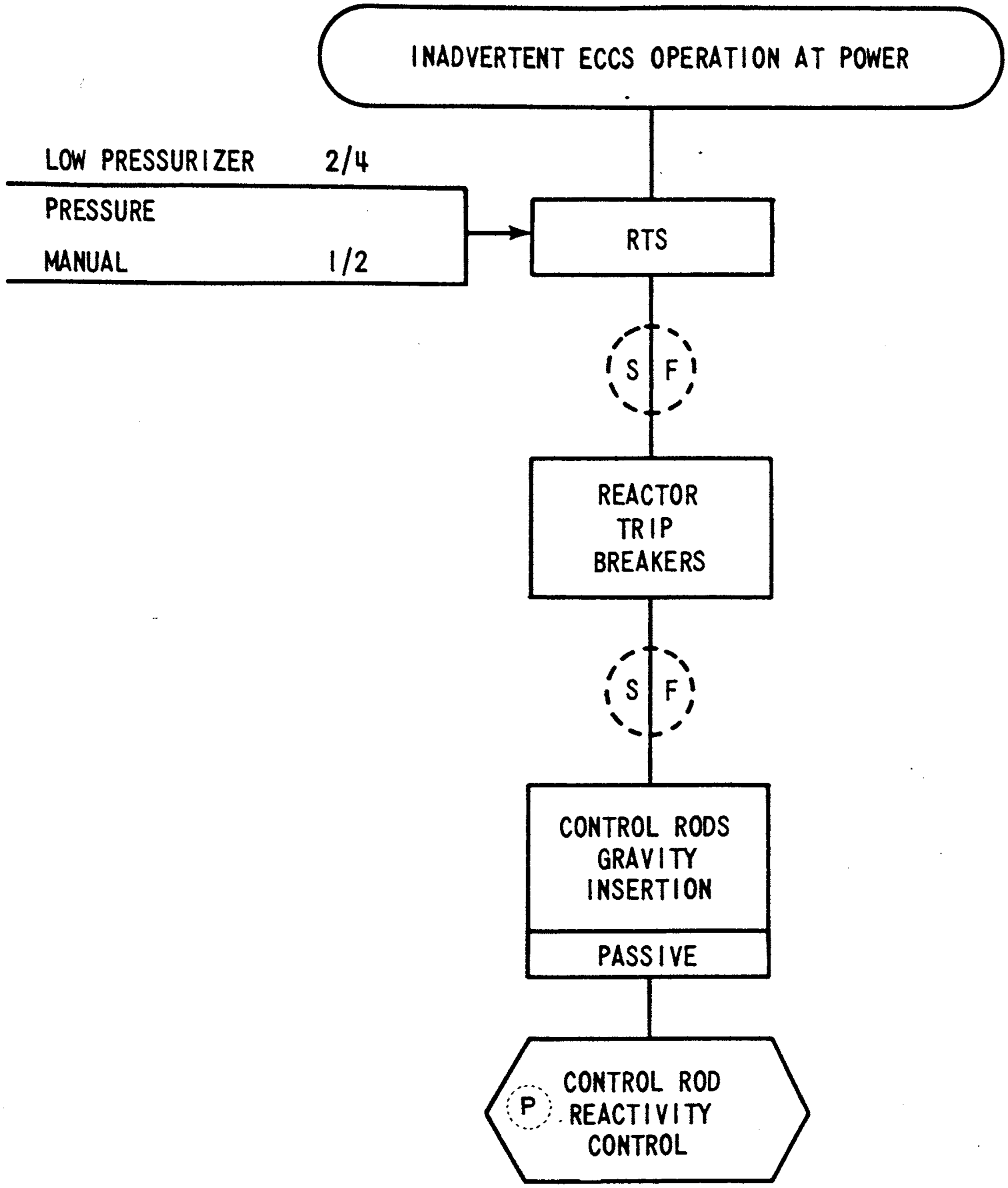
NOTE 1: SEQUENCE FOLLOWING ESFAS ACTUATION IS SIMILAR TO THAT FOR A SMALL LOSS OF COOLANT EVENT (LOCA) SEE FIGURE 15.0-25.

Rev. 13

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.0-21

RUPTURE OF CONTROL ROD DRIVE MECHANISM HOUSING

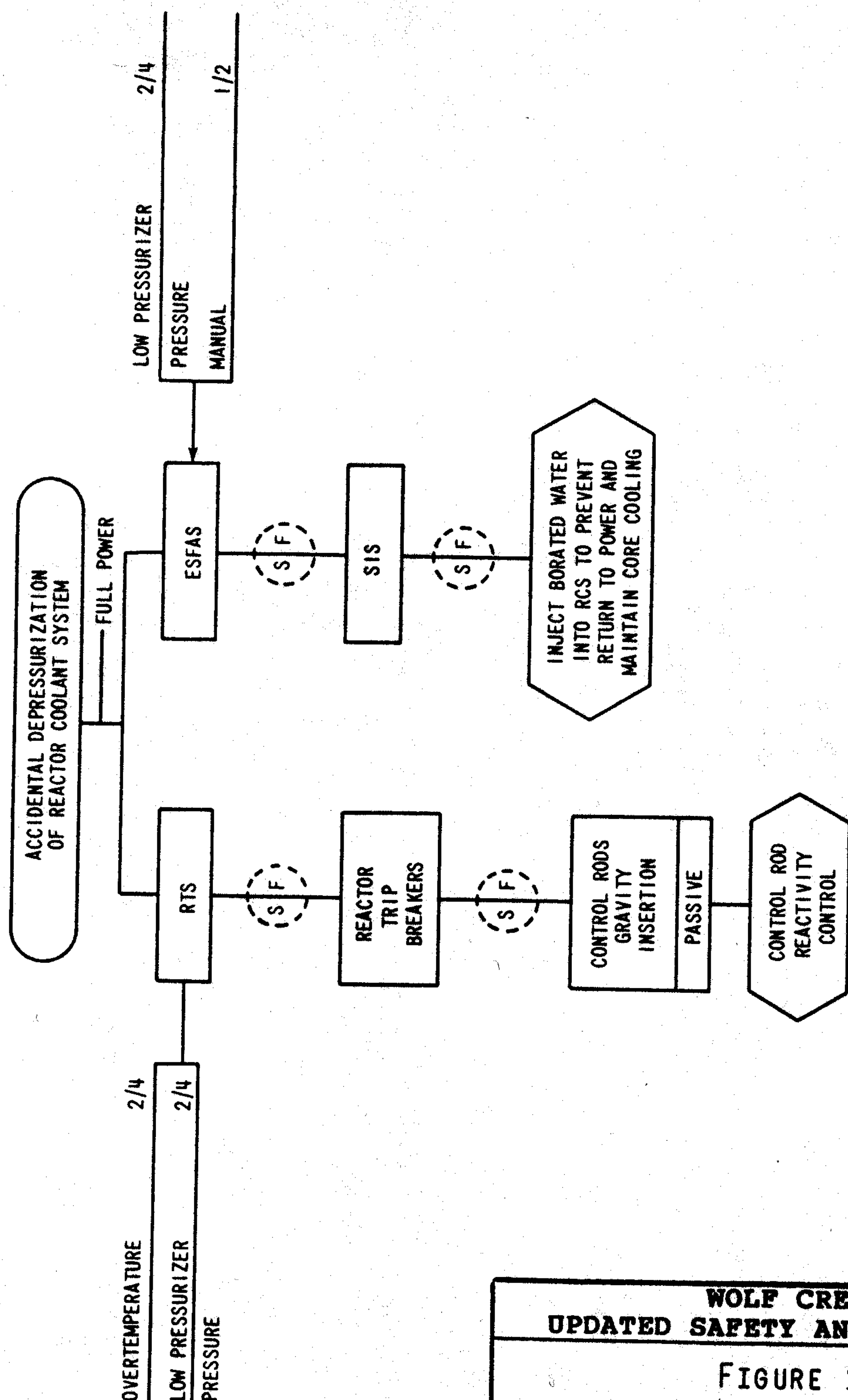


Rev. 13

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.0-22

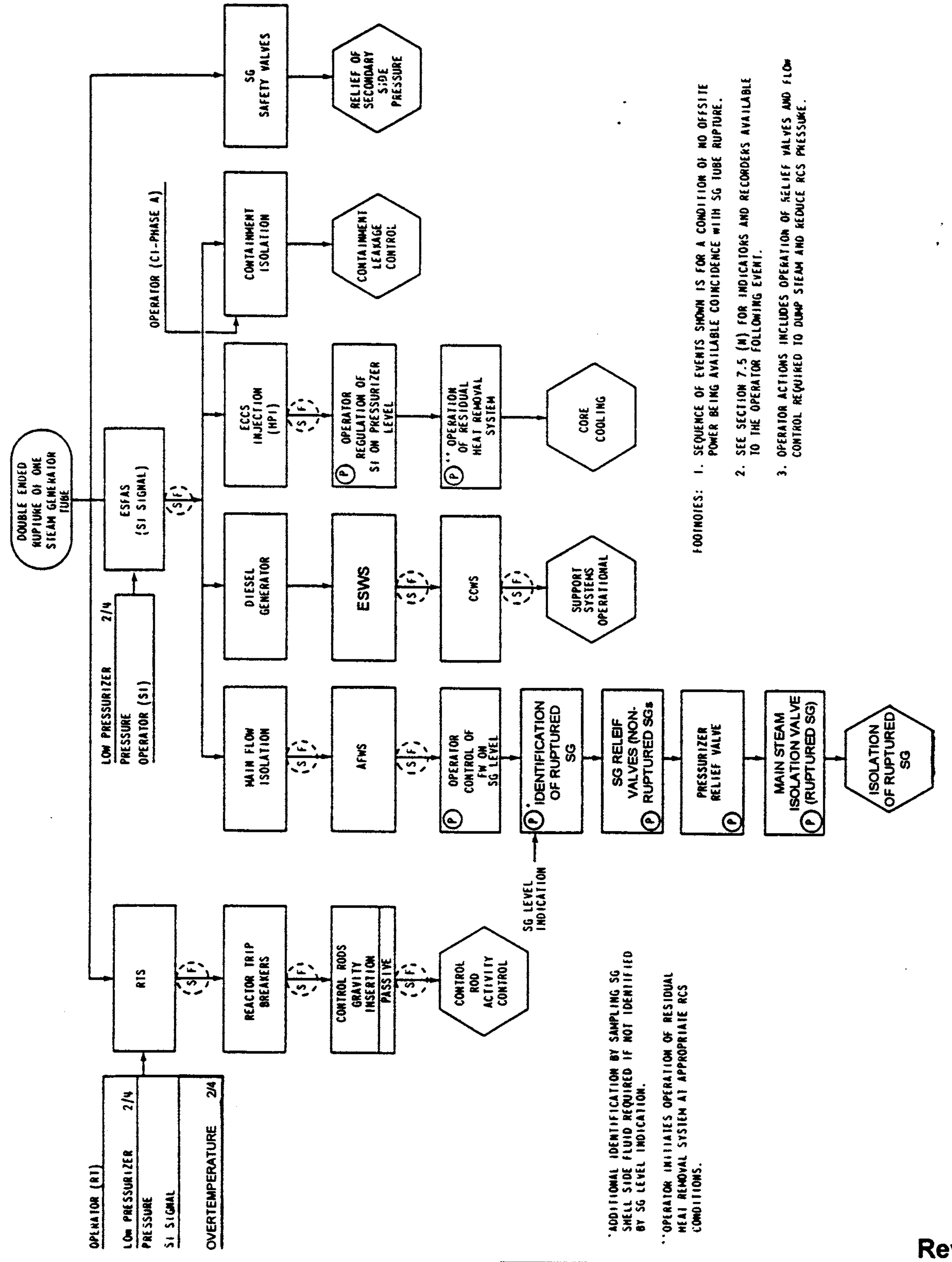
INADVERTENT ECCS OPERATION
AT POWER



WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.0-23

ACCIDENTAL DEPRESSURIZATION OF REACTOR COOLANT SYSTEM

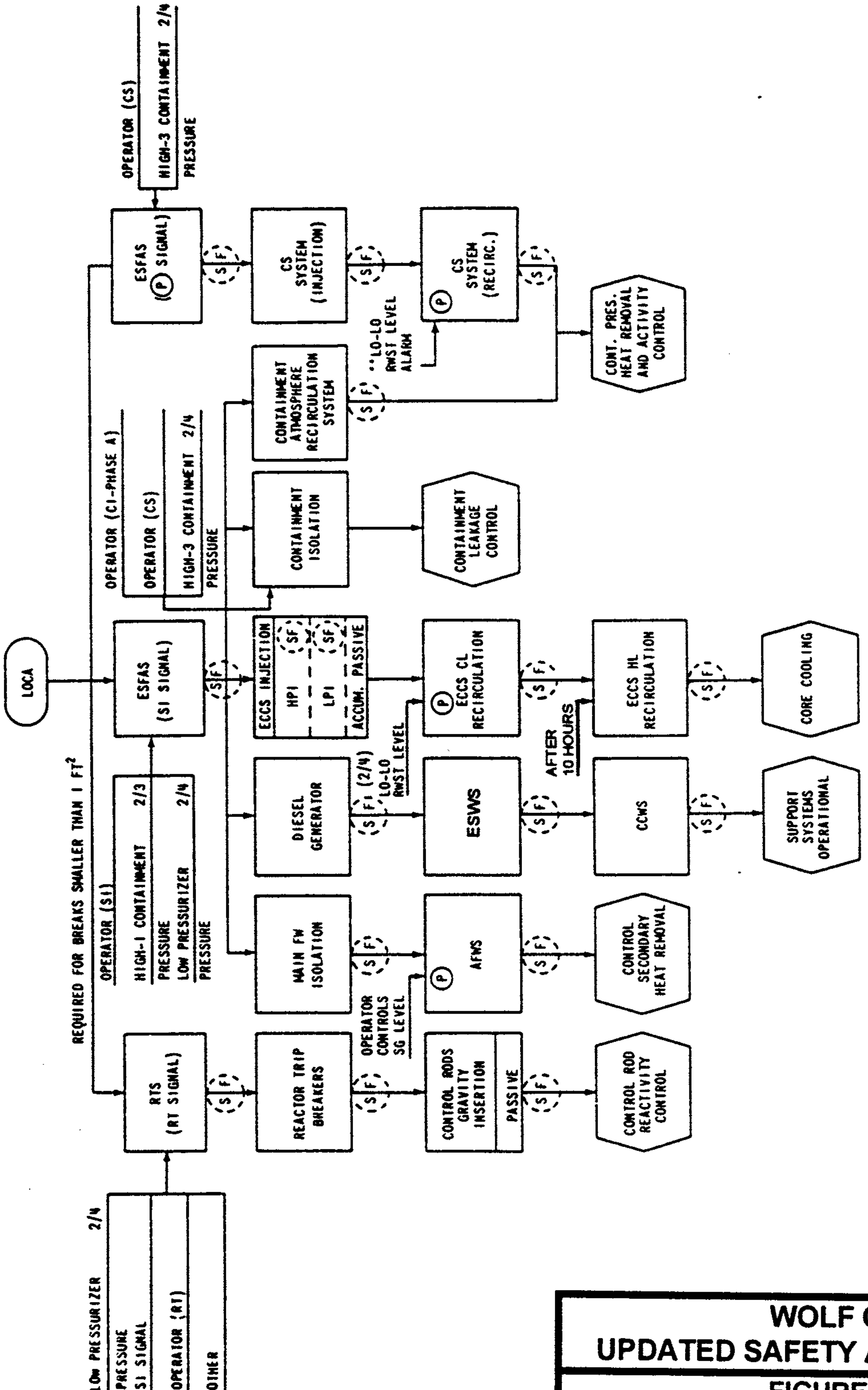


Rev. 13

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.0-24

STEAM GENERATOR TUBE RUPTURE



* OPERATOR ACTION INCLUDES VALVE ALIGNMENT AND PUMP CONTROL REQUIRED FOR RECIRCULATION

** OPERATOR ALIGNS SPRAY PUMP SUCTION FOR RECIRCULATION TO CONTAINMENT SUMP ON LO-LO RWST LEVEL ALARM

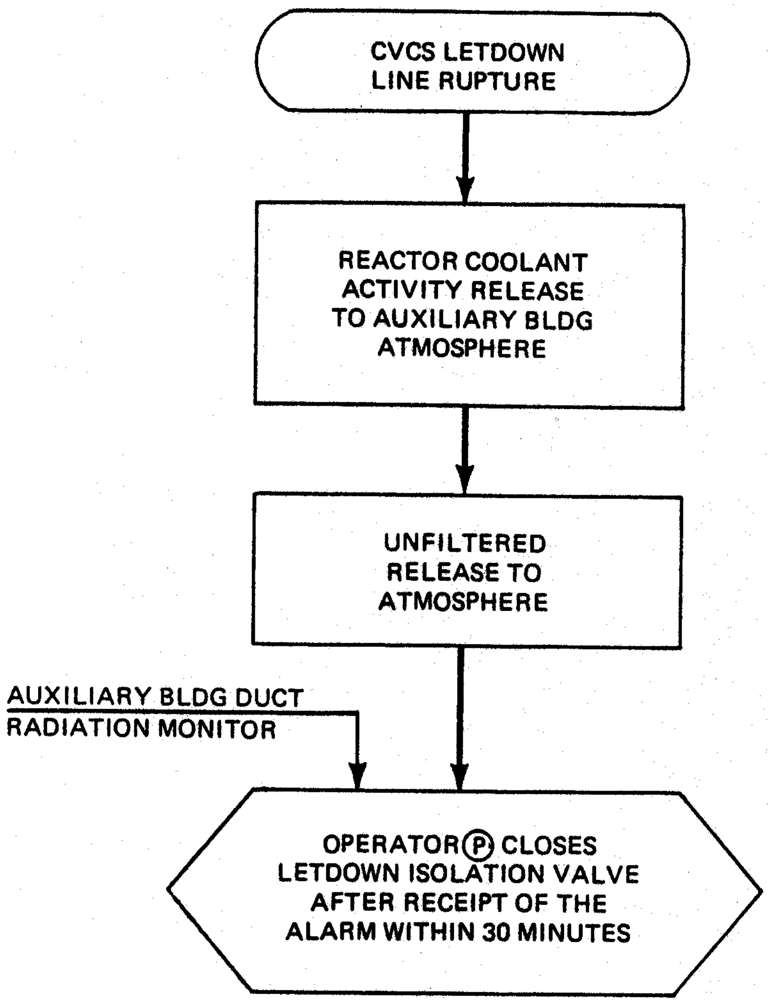
FOOTNOTE: SEE SECTION 7.5(N) FOR INDICATORS AND RECORDERS AVAILABLE TO THE OPERATOR FOLLOWING EVENT

Rev. 13

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.0-25

LOSS OF COOLANT ACCIDENT

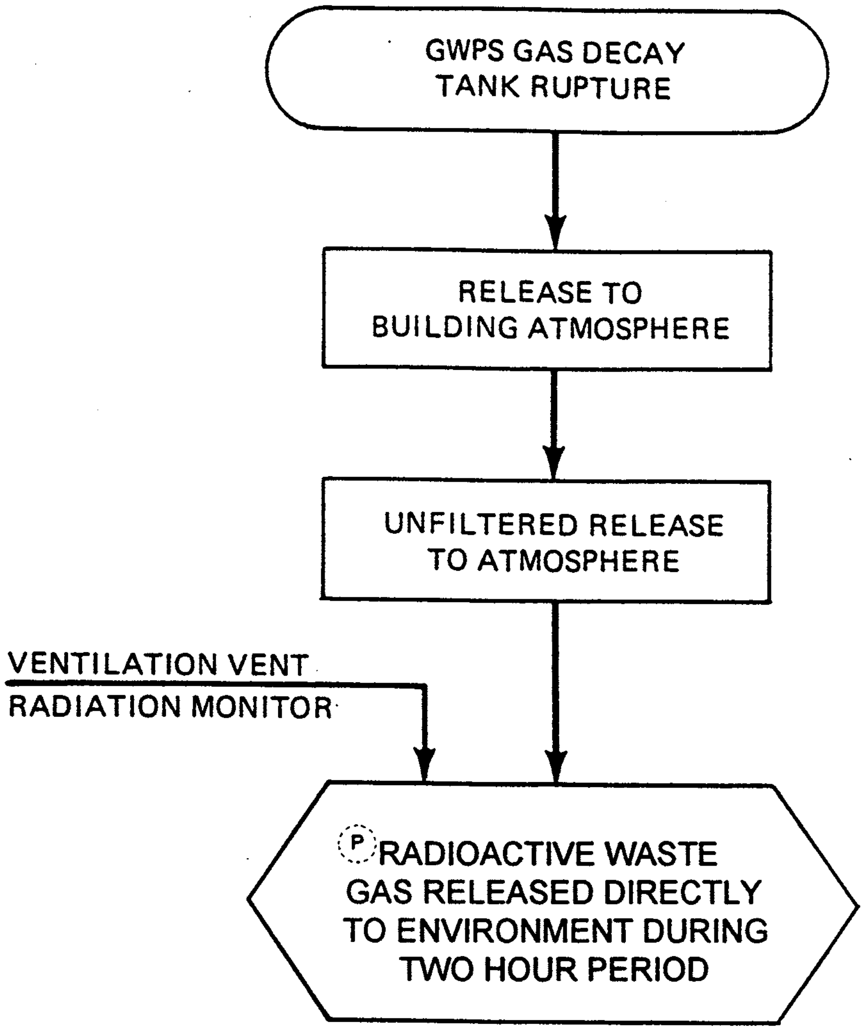


Rev. 0

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.0-26

CVCS LETDOWN LINE RUPTURE

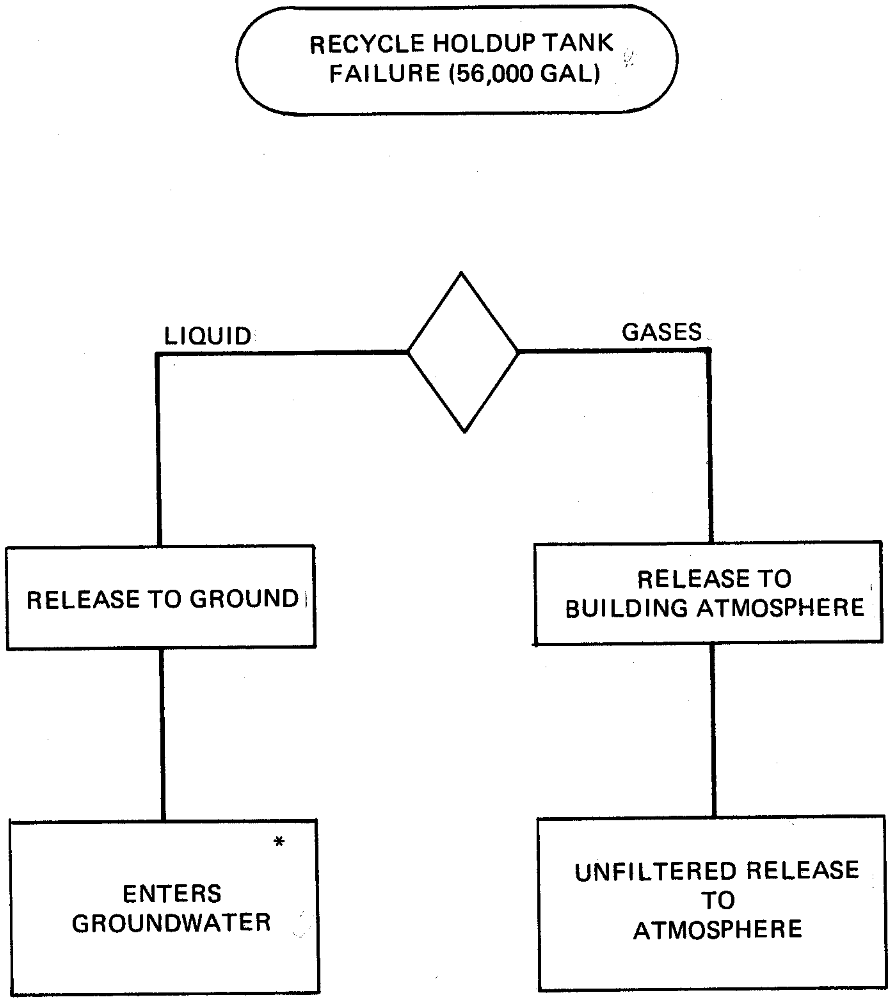


Rev. 13

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.0-27

GWPS GAS DECAY TANK RUPTURE



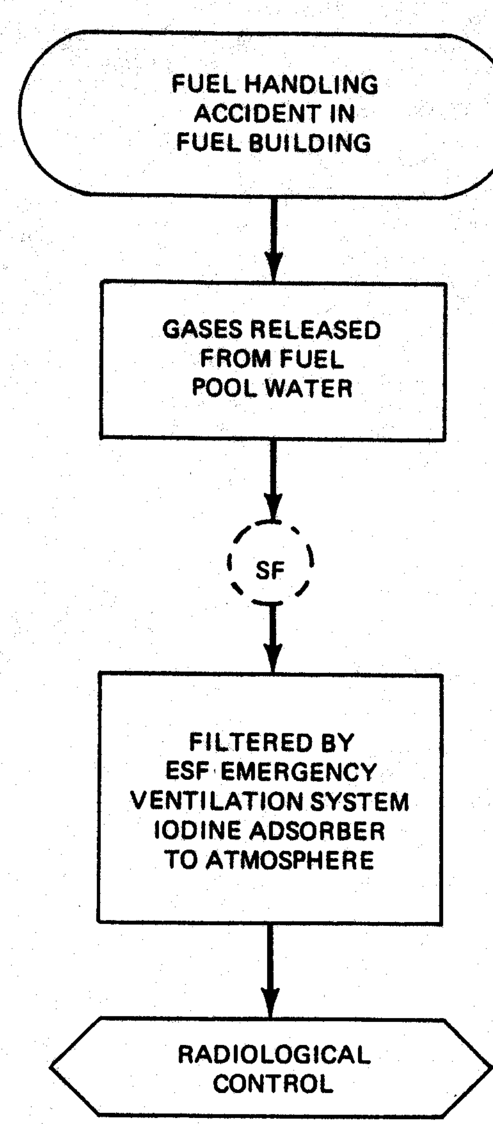
*See Section 2.4.13

Rev. 1

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.0-28

FLOOR DRAIN TANK FAILURE

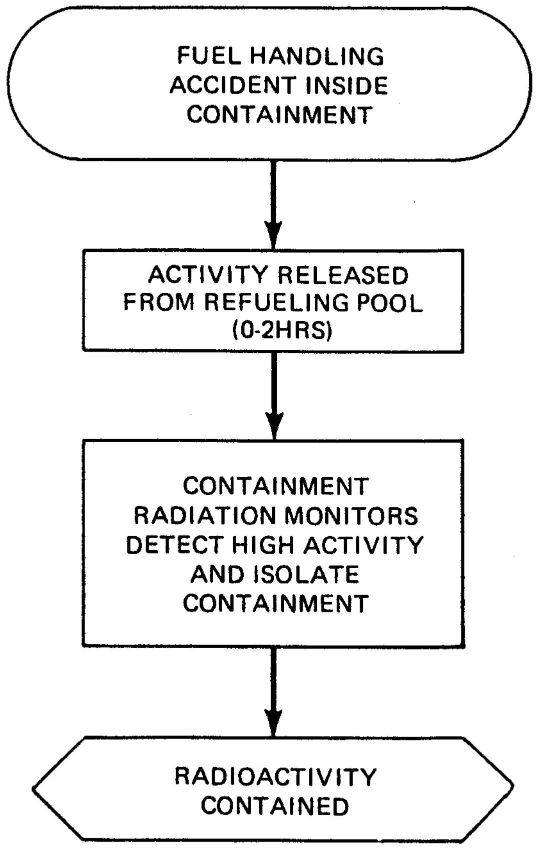


Rev. 0

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.0-29

FUEL HANDLING ACCIDENT IN FUEL BUILDING

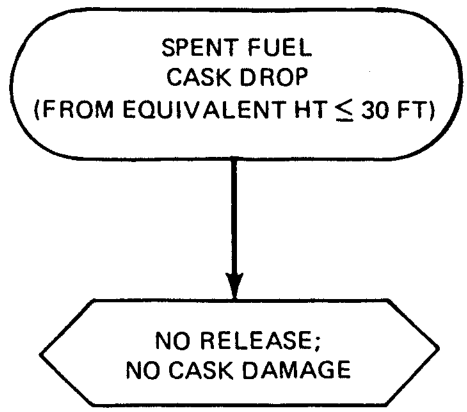


Rev. 0

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.0-30

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT



Rev. 0

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.0-31

SPENT FUEL CASK DROP ACCIDENT

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 (RCS) by the secondary system. Detailed analyses are presented for the following events, which have been identified as the most limiting cases:

a. Feedwater system malfunctions that result in a decrease in feedwater temperature

b. Feedwater system malfunctions that result in an increase in feedwater flow

c. Excessive increase in secondary steam flow

d. Inadvertent opening of a steam generator atmospheric relief or safety valve

e. Steam system piping failure

The above are considered to be ANS Condition II events, with the exception of steam system piping failures, which are considered to be ANS Condition III (minor) and Condition IV (major) events. Section 15.0.1 provides a discussion of ANS classifications and applicable acceptance criteria.

All of the accidents in this section have been analyzed. It has been determined that the most severe radiological consequences will result from the main steam line break accident discussed in Section 15.1.5. Therefore, the radiological consequences are only reported for that limiting case.

15.1.1 FEEDWATER SYSTEM MALFUNCTIONS THAT RESULT IN A DECREASE 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 the reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and the RCS. The overpower/overtemperature protection (neutron overpower, overtemperature, and overpower DT trips) prevents any power increase which could lead to a departure from nucleate boiling ratio (DNBR) less than the safety analysis limit.

A reduction in feedwater temperature may be caused by a spurious heater drain pump trip. In the event of a spurious heater drain pump trip, 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 RCS.

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

The net effect on the RCS 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 ÷ T.

A decrease in normal feedwater temperature is classified as an ANS Condition II event, fault of moderate frequency. See Section 15.0.1 for a discussion of Condition II events.

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 Section 15.0.8 and listed in Table 15.0-6.

15.1.1.2 Analysis of Effects and Consequences

Method of Analysis

This transient is analyzed by computing conditions at the feedwater pump inlet following a heater drain pump trip. These feedwater conditions are then used to recalculate 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:

a. Plant initial power level corresponding to guaranteed nuclear steam supply system thermal output

b. Heater drain pumps trip, resulting in a reduction in feedwater temperature

Plant characteristics and initial conditions are further discussed in Section 15.0.3.

Results

A trip of the heater drain pumps causes a reduction in feedwater temperature that increases the thermal load on the primary system. The calculated reduction in feedwater temperature is 63.0 °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 a spurious heater drain pump trip would result in a transient very similar (but of reduced magnitude) to that presented in Section 15.1.3 for an excessive increase in secondary steam flow, which evaluates the consequences of a 10-percent step load increase. Therefore, the transient results of this analysis are not presented.

The plant is expected to reach stabilized conditions at a power level slightly higher than the initial power level. Normal plant operating procedures would then be followed to reduce power.

15.1.1.3 Conclusions

The decrease in the feedwater temperature transient is less severe than the increase in the feedwater flow event (see Section 15.1.2) and the increase in the secondary steam flow event (see Section 15.1.3). Based on results presented in Sections 15.1.2 and 15.1.3, the applicable acceptance criteria for the decrease in feedwater temperature event have been met.

15.1.2 FEEDWATER SYSTEM MALFUNCTIONS THAT RESULT IN AN INCREASE IN FEEDWATER FLOW

15.1.2.1 Identification of Causes and Accident Description

Addition of excessive feedwater will cause an increase in core power by decreasing the reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The overpower/overtemperature protection (neutron overpower, overtemperature, and overpower DT trips) prevents any power increase which could lead to a DNBR less than the safety analysis limit.

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 RCS due to increased subcooling in the steam generator. With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in the RCS temperature and thus a reactivity insertion in the presence of a negative moderator temperature coefficient of reactivity.

Continuous addition of excessive feedwater is prevented by the steam generator high-high level trip, which initiates feedwater isolation and trips the turbine and main feedwater pumps.

An increase in the normal feedwater flow is classified as an ANS Condition II event, fault of moderate frequency. See Section 15.0.1 for a discussion of ANS Condition II events.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-8.

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 RETRAN (Ref. 5). This code simulates a multiloop system, 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. VIPRE-01 (Ref. 6) is used to determine the core thermal limits to determine DNBR. RETRAN generated statepoints are used as VIPRE-01 boundary conditions to perform the DNB analysis.

The system is analyzed to demonstrate plant behavior in the event that excessive feedwater addition occurs due to a control system malfunction or operator error that allows a feedwater control valve to open fully. Two cases are analyzed, assuming a conservatively large negative moderator temperature coefficient:

a. Accidental opening of two feedwater control valves with the reactor just critical at zero load conditions

b. Accidental opening of two feedwater control valves with the reactor in automatic control at full power

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

a. For the feedwater control valve accident at full power, two feedwater control valves are assumed to malfunction, resulting in a step increase to 200 percent of nominal feedwater flow to two steam generators.

b. For the feedwater control valve accident at zero load conditions, a feedwater control valve malfunction occurs, which results in an increase in flow to two steam generators from zero to 250 percent of the nominal full load value for two steam generators.

c. For the zero load condition, feedwater temperature is at a conservatively low value of 32 °F.

d. No credit is taken for the heat capacity of the RCS and steam generator thick metal structure in attenuating the resulting plant cooldown.

e. The feedwater flow resulting from two fully open control valves is terminated by a steam generator high-high level trip signal, which initiates feedwater isolation and trips the main feedwater pumps.

Plant characteristics and initial conditions are further discussed in Section 15.0.3.

Subsequent to feedwater isolation initiated by a steam generator high-high level trip, the reactor continues to operate until the low-low steam generator level setpoint is reached. No credit is taken in the licensing basis analysis for a reactor trip on turbine trip. No single active failure will prevent operation of the reactor protection system. A discussion of anticipated transients without scram (ATWS) considerations is presented in Section 15.8.

Results

In the case of an accidental full opening of two feedwater control valves 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 Section 15.4.1 for an uncontrolled rod cluster control assembly (RCCA) bank withdrawal from a subcritical or low power startup condition. It should be noted that if the incident occurs with the unit just critical at no-load, 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 (maximum reactivity feedback coefficients, with rod control) results in the greatest power increase. Assuming no rod control, the case results in a similar transient.

The feedwater flow resulting from two fully open control valves is terminated by a steam generator high-high level trip signal, which initiates feedwater isolation and trips the main feedwater pumps. This prevents continuous addition of the feedwater. Once main feedwater is isolated, the reactor continues to operate until the lo-lo steam generator trip setpoint is reached.

When the lo-lo steam generator level setpoint is reached the reactor is tripped and the control rods fall into the core terminating the event.

Transient results presented in Figures 15.1-1 and 15.1-2, and 15.1-2A for the full power case show the increase in nuclear power and ÷T associated with the increased thermal load on the reactor. The DNBR does not drop below the safety analysis limit. Following the reactor trip, the plant approaches a stabilized condition. Standard plant shutdown procedures may then be followed to further cool down the plant.

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 of heat flux does not exceed 118 percent of its nominal value (i.e., the assumed high neutron flux trip setpoint). 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 reactor 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.

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

15.1.2.3 Conclusions

The results of the analysis show that the DNBRs encountered for an excessive feedwater addition at power are above the limiting value. The DNBR design basis is described in Section 4.4. Additionally, it has been determined 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 subcritical condition analysis, presented in Section 15.4.1.

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 Sections 15.1.4 and 15.1.5.

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

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

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

a. Overpower ÷T
b. Overtemperature ÷T
c. Power range high neutron flux

An excessive load increase incident is considered to be an ANS Condition II event, fault of moderate frequency. See Section 15.0.1 for a discussion of Condition II events.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-9.

15.1.3.2 Analysis of Effects and Consequences

Method of Analysis

This accident is analyzed using the LOFTRAN code (Ref. 1). The code simulates the neutron kinetics, RCS, 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. VIPRE-01 (Ref. 6) is used to determine the core thermal limits to determine DNBR. LOFTRAN generated statepoints are used as VIPRE-01 boundary conditions to perform the DNB analysis.

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

a. Reactor control in manual with minimum moderator reactivity feedback

b. Reactor control in manual with maximum moderator reactivity feedback

c. Reactor control in automatic with minimum moderator reactivity feedback

d. Reactor control in automatic with maximum moderator reactivity feedback

For the minimum feedback cases, the core has the most positive moderator temperature coefficient of reactivity, therefore, reductions in coolant temperature will have the least impact on core power. For the maximum feedback cases, the moderator temperature coefficient is assumed at the most negative value. This results in the largest amount of reactivity feedback due to changes in coolant temperature. For the cases with automatic rod control, no credit was taken for ÷T trips on overtemperature or overpower in order to demonstrate the inherent transient capability of the plant. Under actual operating conditions, such a trip may occur, after which the plant would quickly stabilize.

A 10-percent step increase in steam demand is assumed, and all cases are studied without credit being taken for pressurizer heaters. Initial operating conditions are assumed at nominal values, consistent with steady-state full power operation, following statistical core design (SCD) methodology. Plant characteristics and initial conditions are further discussed in Section 15.0.3.

Normal reactor control systems and engineered safety 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.

Results

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 safety analysis limit.

Figures 15.1-7 through 15.1-10 illustrate the transient, assuming that the reactor is in the automatic control mode and no reactor trip signals occur. Both the minimum and maximum moderator feedback cases show that core power increases, thereby reducing the rate of decrease in coolant average temperature and pressurizer pressure. For both of these cases, the minimum DNBR remains above the safety analysis limit.

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. Note that due to the measurement errors assumed in the setpoints, it is possible that reactor trip could actually occur for the automatic control cases. The plant would then reach a stabilized condition following the trip.

The excessive load increase incident is an overpower transient for which the fuel temperatures will rise. Reactor trip may not occur for some 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 reactor 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.

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

15.1.3.3 Conclusions

The analysis presented above shows that for a 10-percent step load increase, the DNBR remains above the safety analysis limit; the design basis for DNBR is described in Section 4.4. The plant rapidly reaches a stabilized condition, following the load increase.

15.1.4 INADVERTENT OPENING OF A STEAM GENERATOR ATMOSPHERIC RELIEF OR SAFETY VALVE

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 are associated with an inadvertent opening of a single turbine bypass, atmospheric relief, or safety valve. The analyses performed, assuming a rupture of a main steam line, are given in Section 15.1.5.

The steam release, as a consequence of this accident, results in an initial increase in steam flow followed by a decrease in steam flow during the rest of the accident as the steam generator pressure decreases. The energy removal from the RCS 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.

The analysis is performed to demonstrate that the following criterion is satisfied:

Assuming a stuck RCCA, with offsite power available, and assuming a single failure in the engineered safety features system, 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, atmospheric relief, or safety valve.

Accidental depressurization of the secondary system is classified as an ANS Condition II event. See Section 15.0.1 for a discussion of Condition II events.

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

a. Safety injection system actuation from any of the following:

   1. Two out of four low pressurizer pressure signals

   2. Two out of three low steam line pressure signals in any one loop

b. The overpower reactor trips (neutron flux and ÷T) and the reactor trip occurring in conjunction with receipt of the SIS

c. 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, an SIS will rapidly close all feedwater control valves and back up feedwater isolation valves, trip the main feedwater pumps, and close the feedwater pump discharge valves.

d. Trip of the fast-acting main steam line isolation valves (designed to close in less than 5 seconds) on:

1. Safety injection system actuation derived from two out of three low steam line pressure signals in any one loop (above Permissive P-11)

2. Two out of three high negative steam pressure rates in any loop (below Permissive P-11)

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-10.

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:

a. A full plant digital computer simulation, using the RETRAN code (Ref. 5) to determine RCS temperature and pressure during cooldown, and the effect of safety injection. VIPER-01 (Ref. 6) is used to determine the core thermal limits to determine DNBR. RETRAN generated statepoints are used as VIPRE-01 boundary conditions to perform the DNB analysis.

b. Analyses to determine that there will be no consequential damage to the core or reactor coolant system.

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

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

b. 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 is included. The K_{eff} versus temperature at 1,000 psia corresponding to the negative moderator temperature coefficient used is shown in Figure 15.1-11.

c. Minimum capability for injection of RWST boron solution corresponding to the most restrictive single failure in the safety injection system. This corresponds to the flow delivered by one centrifugal charging pump delivering flow to the cold leg header and taking suction from the RWST. Reactor coolant seal injection flow is not included in the total core delivery. The volume downstream of the RWST must be swept prior to delivery of boric acid to the reactor coolant loops.

d. The case studied is a steam flow of 268 pounds per second at 1,200 psia, with offsite power available. This is the maximum capacity of any single turbine bypass, atmospheric relief, or safety valve. Initial hot standby conditions (557 F average coolant temperature) at time zero are assumed, since this represents the most conservative initial condition.

   Should the reactor be just critical or operating at power at the time of a steam release, the reactor will be tripped by the normal overpower protection when power level reaches a trip point. Following a trip at power, the RCS 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 steam release before the no-load conditions of RCS 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. However, since the initial steam generator water inventory is greatest at no-load, the magnitude and duration of the RCS cooldown are less for steamline release occurring at power.

e. In computing the steam flow, the Moody Curve (Ref. 2) for fL/D = 0 is used.

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

g. Offsite power is assumed, since this would maximize the cooldown.

h. Maximum cold auxiliary feedwater flow is assumed.

i. Four reactor coolant pumps are operating.

Results

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 15.1-12 and 15.1-13 show the transient results for a steam flow of 268 lb/sec at 1,200 psia.

The assumed steam release is greater than or equal to the capacity of any single steam dump, atmospheric relief, or safety valve. Safety injection is initiated automatically by low pressurizer pressure. Operation of one centrifugal charging pump is assumed. Boron solution from the RWST enters the RCS, providing sufficient negative reactivity to prevent core damage. The 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 and steam generator tubes.

Since the transient occurs over a period of about 600 seconds, the neglected stored energy is likely to have a significant effect in slowing the cooldown. The calculated time sequence of events for this accident is listed in Table 15.1-1.

15.1.4.3 Conclusions

The analysis shows that the criteria stated earlier in this section are satisfied. For an accidental depressurization of the main steam system, the DNB design limits are not exceeded. This case is less limiting than the steamline rupture case described in Section 15.1.5.

15.1.5 STEAM SYSTEM PIPING FAILURE

15.1.5.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steam line would result in an initial increase in steam flow that decreases during the accident as the steam generator pressure decreases.

The energy removal from the RCS 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 RCCA is assumed stuck in its fully withdrawn position after reactor trip, there is possibility that the core will become critical and return to power. A return to power following a steam line rupture is a potential problem mainly because of the high power peaking factors which exist, assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid solution delivered by the safety injection system.

The analysis of a main steam line rupture is performed to demonstrate that the following criteria are satisfied:

Assuming a stuck RCCA with or without offsite power, and assuming a single failure in the engineered safety features system, the core cooling capability is maintained. Radiation doses do not exceed the guidelines of 10 CFR 100.

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that the DNB design basis is not exceeded for any rupture, assuming the most reactive control rod assembly stuck in its fully withdrawn position. The DNBR design basis is discussed in Section 4.4.

A major steam line rupture is classified as an ANS Condition IV event. See Section 15.0.1 for a discussion of Condition IV events.

Effects of minor secondary system pipe breaks are bounded by the analysis presented in this section. Minor secondary system pipe breaks are classified as Condition III events, as described in Section 15.0.1.3.

The major rupture of a steam line is the most limiting cooldown transient, and is analyzed at zero power with no decay heat. Decay heat would retard the cooldown, thereby reducing the likelihood that the reactor will return to power. A detailed analysis of this transient with the most limiting break size, a double ended rupture, is presented here. The assumptions used in this analysis are discussed in Reference 3. Reference 3 also contains a discussion of the spectrum of break sizes and power levels analyzed.

During startup or shutdown evolutions, when the operator manually blocks the safety injection on low pressurizer pressure or low steamline pressure and steamline isolation on low steamline pressure when pressurizer pressure is less than P-11 setpoint (i.e., 1970 psig), the steamline pressure-negative rate-high signal is automatically enabled to provide steamline isolation. For inside containment breaks, steamline isolation may also be provided by the containment pressure High-2 signal and safety injection would be actuated by the containment pressure High-1 signal. For a steamline break occurring outside containment, an automatic actuation signal for safety injection would not be available. Since the steamline break could occur outside containment, it is possible to have a steamline break event below the P-11 interlock setpoint that does not generate a safety injection actuation of borated ECCS flow. With no borated ECCS flow supplied to the core, a return to criticality and subsequent power excursion in the core would result.

However, the combined effect of the negative reactivity associated with the initial RCS boration requirement to meet the shutdown margin and the steamline isolation provided by the steamline high pressure negative rate trip function to limit the steam blowdown would be more than sufficient to limit the core power excursion following a return to criticality. Analysis results confirm that the consequences of a postulated steamline break event occurring in Mode 3 below P-11 with the safety injection being blocked, would be bounded by the limiting steamline break scenario initiating from the HZP conditions with a 0 ppm boron concentration. Therefore, boration to cold shutdown conditions prior to SI blocking is not necessary

The following functions provide the protection for a steam line rupture:

a. Safety injection system actuation from any of the following:

    1. Two out of four low pressurizer pressure signals

    2. Two out of three high-1 containment pressure signals

    3. Two out of three low steam line pressure signals in any one loop

b. The overpower reactor trips (neutron flux and ÷T) and the reactor trip occurring in conjunction with receipt of the SIS.

c. 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 a reactor trip, an SIS will rapidly close all feedwater control valves and back up feedwater isolation valves, trip the main feedwater pumps, and close the feedwater pump discharge valves.

d. Trip of the fast acting main steam line isolation valves (designed to close in less than 5 seconds) on:

    1. High-2 containment pressure

    2. Safety injection system actuation derived from two out of three low steam line pressure signal in any one loop (above Permissive-11)

    3. Two out of three high negative steam pressure rate in any one loop (below Permissive-11)

Fast-acting isolation valves are provided in each steam line; these valves will fully close within 5 seconds of actuation, following a large break in the steam line. 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 steam line isolation is included in Chapter 10.0.

Flow restrictors are installed in the steam generator outlet nozzle, an integral part of the steam generator. The effective throat area of the nozzles is 1.4 square feet, which is considerably less than the main steam pipe area; thus, the nozzles also serve to limit the maximum steam flow for a break at any location.

Table 15.1-2 lists the equipment required in the recovery from a high energy line rupture. Not all equipment is required for any one particular break, since the requirements will vary, depending upon postulated break size and location. Design criteria and methods of protection of safety-related equipment from the dynamic effects of postulated piping ruptures are provided in Section 3.6.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-10.

15.1.5.2 Analysis of Effects and Consequences

Method of Analysis

The analysis of the steam pipe rupture has been performed to determine:

a. The core heat flux and RCS temperature and pressure resulting from the cooldown following the steam line break. The RETRAN code (Ref. 5) has been used.

b. The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital-computer code, VIPRE (Ref. 6), has been used to determine if DNB occurs for the core conditions computed in item a above.

The following conditions were assumed to exist at the time of a main steamline break accident:

a. 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 each operating cycle is restricted by the insertion limits so that addition of positive reactivity in a steam line break accident will not lead to a more adverse condition than the case analyzed.

b. 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 temperature at 1,000 psia, 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 15.1-14.

The core properties associated with the sector nearest the affected steam generator and those associated with the remaining sector 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. The stuck rod is assumed in the region of the core of lowest temperature.

To verify the conservatism of this method, the reactivity, as well as the power distribution, was checked for the limiting conditions during the transient 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 surface boiling occurs in the high heat 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 limiting conditions during the transient. These results verify conservatism, i.e., underprediction of negative reactivity feedback from power generation.

c. Minimum capability for injection of boron solution corresponding to the most restrictive single failure in the safety injection system. The emergency core cooling system consists of three systems: 1) the passive accumulators, 2) the residual heat removal system, and 3) the safety injection system. Only the safety injection system is modeled for the steam line break accident analysis.

The actual modeling of the safety injection system in RETRAN is as follows:
The flow corresponds to full flow (less seal injection flow) of one charging pump delivering to the RCS via the cold leg header. No credit has been taken for the borated water that must be swept from the lines downstream of the RWST prior to the delivery of the boron solution from the RWST to the reactor coolant loops. The RWST Boron concentration is assumed to be 2000 ppm.

For the cases where offsite power is assumed, the sequence of events in the safety injection system is as follows. After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate, and the high head safety injection pump starts. In 12 seconds, the valves are assumed to be in their final position, and the pump is assumed to be at full speed (see note in 15.1.5.5). The volume downstream of the RWST must be swept prior to delivery of boric acid to the reactor coolant loops. This delay is included in the calculations.

In cases where offsite power is not available, an additional 12-second delay is assumed to start the diesels and to load the necessary safety injection equipment onto them.

d. Since the steam generators are provided with integral flow restrictors with a 1.4-square-foot throat area, any rupture with a break area greater than 1.4 square feet, regardless of location, would have the same effect on the NSSS as the 1.4-square-foot break. The following cases have been considered in determining the core power and RCS transients:

   1. Complete severance of a pipe, with the plant initially at no-load conditions, full reactor coolant flow with offsite power available.

   2. Case (1) with loss of offsite power simultaneous with the steam line break, which causes initiation of the safety injection signal. Loss of offsite power results in reactor coolant pump coastdown.

e. Power peaking factors corresponding to one stuck RCCA and nonuniform core inlet coolant temperatures 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 RCCA during the return to power phase following the steam line break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck 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 standby conditions (557 F average coolant temperature) at time zero, including the application of conservative uncertainties to the initial RCS pressure, flow, and temperature, values, since this represents the most pessimistic initial condition. Should the reactor be just critical or operating at power at the time of a steam line 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 RCS 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 steam line break before the no-load conditions of RCS 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.

f. In computing the steam flow during a steam line break, the Moody Curve (Ref. 2) for fL/D = 0 is used.

Results

The calculated sequence of events for both cases analyzed is shown on Table 15.1-1.

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

Core Power and Reactor Coolant System Transient

Figures 15.1-15 through 15.1-26 show the RCS transient and core heat flux following a main steam line rupture (complete severance of a pipe) at initial no-load condition (case 1).

Offsite power is assumed to be 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 steam line 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 steam lines 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 main steam line isolation 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 RCCAs inserted (with the design shutdown assuming one stuck RCCA) before the RWST boron solution at enters the RCS. A peak core power significantly lower than the nominal full power value is attained.

The calculation assumes that the boric acid is mixed with, and diluted by, the water flowing in the RCS prior to entering the reactor core. The concentration after mixing depends upon the relative flow rates in the RCS and in the safety injection system. The variation of mass flow rate in the RCS due to water density changes is included in the calculation, as is the variation of flow rate in the safety injection system due to changes in the RCS pressure. The safety injection system flow calculation includes the line losses in the system as well as the pump head curve.

Figures 15.1-21 through 15.1-26 show the salient parameters for case 2, which corresponds to the case discussed above with the additional loss of offsite power at the time the safety injection signal is generated. The safety injection system delay time includes 12 seconds to start the diesel in addition to 12 seconds to start the safety injection pump and open the valves (see note in 15.1.5.5). 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 RCS is reduced by the decreased flow in the RCS. The peak power remains well below the nominal full power value.

It should be noted that following a steam line break only one steam generator blows down completely. Thus, the remaining steam generators are still available for the 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 atmospheric relief or safety valves.

Margin to Critical Heat Flux

A DNB analysis was performed for both of these cases. It was found that both cases had a minimum DNBR greater than the safety analysis limit.

15.1.5.3 Radiological Consequences

15.1.5.3.1 Method Of Analysis

15.1.5.3.1.1 Physical Model

The radiological consequences of a MSLB inside the containment are less severe than the one outside the containment because the radioactivity released will be held up inside the containment, allowing decay and plateout of the radionuclides. To evaluate the radiological consequences due to a postulated MSLB (outside the containment), it is assumed that there is a complete severance of a main steam line outside the containment.

It is also assumed that there is a simultaneous loss of offsite power, resulting in reactor coolant pump coastdown. The safety injection system is actuated and the reactor trips.

The main steam line isolation valves, their bypass valves, and the steam line drain valves isolate the steam generators and the main steam lines upon a signal initiated by the engineered safety features actuation system under the conditions of high steam negative pressure rate or low steam line pressure. The main steam isolation valves are installed in the main steam lines from each steam generator downstream from the safety and atmospheric relief valves outside the containment. The break in the main steam line is assumed to occur outside of the containment. The affected steam generator (steam generator connected to a broken steam line) blows down completely. The steam is vented directly to the atmosphere.

Each of the steam generators incorporates integral flow restrictors, which are designed to limit the rate of steam blowdown from the steam generators following a rupture of the main steam line. This, in turn, reduces the cooling rate of the reactor coolant system to preclude departure from nucleate boiling (DNB).

In case of loss of offsite power, the remaining steam generators are available for dissipation of core decay heat by venting steam to the atmosphere via the atmospheric relief valves. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently so that the RHR system can be utilized to cool the reactor.

15.1.5.3.1.2 Assumptions and Conditions

The major assumptions and parameters assumed in the analysis are itemized in Tables 15.1-3 and 15A-1.

The assumptions used to determine the concentrations of radioactive isotopes within the secondary system for this accident are as follows:

a. The secondary system initial concentrations of radioactive isotopes are assumed to be the dose equivalent of 0.1 σCi/gm of 1-131.

b. A primary-to-secondary leakage rate of 1 gpm is assumed to exist and is assumed to be in the affected steam generator.

c. The reactor coolant concentration of radioactive isotopes is determined by two methods, and both cases are analyzed. These are:

   1. The initial reactor coolant concentrations of radioactive isotopes are assumed to be the dose equivalent of 1.0 σCi/gm of I-131 with an iodine spike that increases the rate of iodine release into the reactor coolant by a factor of 500.

   2. An assumed reactor coolant concentration of radioactive isotopes with a dose equivalent of 60 σCi/gm of I-131 as a result of preaccident iodine spikes.

d. The reactor coolant concentrations of noble gas correspond to 1-percent failed fuel.

e. Partition factors used to determine the secondary system activities are given in Table 15.1-3.

The following specific assumptions and parameters are used to calculate the activity release:

a. Offsite power is lost, resulting in reactor coolant pump coastdown.

b. No condenser air removal system release and no normal operating steam generator blowdown is assumed to occur during the course of the accident.

c. Eight hours after the occurrence of the accident, the residual heat-removal system (RHRS) starts operation to cool down the plant.

d. After the accident, the primary-to-secondary leakage continues for 8 hours, at which time the reactor coolant system is depressurized.

e. The affected steam generator (steam generator connected to the broken steam line) is allowed to blow down completely.

f. Steam release to the atmosphere and the associated activity release from the safety and atmospheric relief valves and the broken steam line is terminated 8 hours after the accident, when the RHRS is activated to complete cooldown.

g. The amount of noble gas activity released is equal to the amount present in the reactor coolant, which leaks to the secondary during the accident. The amount of iodine activity released is based on the activity present in the secondary system and the amount of leaked reactor coolant which is entrained in the steam that is discharged to the environment via the safety and atmospheric relief valves and the broken steam line. Partition factors used for the unaffected steam generators after the accident occurs are given in Table 15.1-3. An iodine partition factor of 1 is used for the affected steam generator.

h. The activity released from the broken steam line and the safety and atmospheric relief valves during the 8-hour duration of the accident is immediately vented to the atmosphere.

15.1.5.3.1.3 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 the onsite meteorological measurement programs described in Section 2.3.

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.

15.1.5.3.1.4 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 (steam generator connected to the broken steam line) is released directly to the environment. The unaffected steam generators are assumed to continually discharge steam and entrained activity via the safety and atmospheric relief valves up to the time initiation of the RHRS can be accomplished.

Since the activity is released directly to the environment with no credit for plateout or retention, the results of the analysis are based on the most direct leakage pathway available. Therefore, 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 Conservatisms in the Analysis

a. Reactor coolant activities are based on the Technical Specification limit of 1.0 σCi/gm I-131 dose equivalent with extremely large iodine spike values persisting for the entire duration of the accident, resulting in equivalent concentrations many times greater than the reactor coolant activities based on 0.12 percent failed fuel associated with normal operating conditions.

b. A 1-gpm steam generator primary-to-secondary leakage is assumed, which is significantly greater than that anticipated during normal operation. Furthermore, it was conservatively assumed that all leakage is to the affected steam generator only.

c. 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.

15.1.5.3.3.1 Filter Loadings

The only ESF filtration system considered in the analysis which limits the consequences of the MSLB is the control room filtration system. Activity loadings on the control room charcoal filter are based on flow rate through the filter, the concentration of activity at the filter inlet, and the filter efficiency.

Activity in the control room filter as a function of time has been evaluated for the more limiting LOCA analysis, as discussed in Section 15.6.5.4.3.1. Since the control room filters are capable of accommodating the potential design basis LOCA fission product iodine loadings, more than adequate design margin is available with respect to postulated MSLB releases.

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 doses due to immersion from direct radiation and the thyroid dose due to inhalation have been analyzed for the 0-2 hour dose at the exclusion area boundary and for the duration of the accident (0 to 8 hrs) at the low-population zone outer boundary. The results are listed in Table 15.1-4. The resultant doses are within the acceptance limits, a small fraction (10 percent) of exposure limits of 10CFR100, i.e., 2.5 rem and 30 rem respectively for the whole body and thyroid doses for the case of concurrent iodine spike and the exposure limits of 10CFR100 i.e., 25 rem and 300 rem respectively for the whole body and thyroid doses for the case of pre-accident iodine spike.

15.1.5.4 <u>Conclusions</u>

The analysis has shown that the criteria stated earlier in Section 15.1.5.1 are satisfied.

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable and not precluded by the criteria, the above analysis shows that the DNB design basis is met for any rupture, assuming the most reactive RCCA stuck in its fully withdrawn position.

15.1.5.5 <u>Notes</u>

(1) As discussed in Reference 4, the SI response time has an additional delay of 15 seconds. This makes the SI response time 27 and 39 seconds for the cases with and without offsite power respectively.

15.1.6 REFERENCES

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

2. Moody, F. J., "Transactions of the ASME, Journal of Heat Transfer," Figure 3, page 134, February 1965.

3. Wood, D. C., and Hollingsworth, S. D., "Reactor Core Response to Excessive Secondary Steam Releases," WCAP-9226 (Proprietary), WCAP-9227 (Non-proprietary), January 1978.

4. Letter from D. P. Dominicis, Westinghouse, to O. Maynard, WCNOC, SAP-87-155, "Increased SI Delay Safety Evaluation." April 21, 1987.

5. McFadden, J. H., et. al., "RETRAN-02 - A Program for Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," EPRI NP-1850-CCM-A, October 1984.

6. Stewart, C. W., et. al.,"VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores," Battelle, Pacific Northwest Laboratories, EPRI NP-2511-CCM-A, August 1989.

TABLE 15.1-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT RESULT IN AN INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (sec)
Feedwater system malfunctions that result in an increase in feedwater flow at full power	Two main feedwater control valves fail fully open	0.0
	Feedwater Isolation due to high-high steam generator level	28.7
	Reactor trip on lo-lo steam generator level	57.9
	Rods begin to drop	59.9
	Minimum DNBR occurs	36.0
Excessive increase in secondary steam flow		
1. Manual reactor control (minimum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	140
2. Manual reactor control (maximum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	50
3. Automatic reactor control (minimum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	110

TABLE 15.1-1 (Sheet 2)

Accident	Event	Time (sec)
4. Automatic reactor control (maximum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	50
Inadvertent opening of a steam generator atmospheric relief or safety valve		
	Inadvertent opening of one main steam safety or atmospheric relief valve	0.0
	Auxiliary Feedwater Actuation	0.0
	Low Pressure Trip (SIS)	185
	Peak Reactor power	372
	RWST boron solution reaches core	400
Steam system piping failure		
1. Case 1 (offsite power available)	Steam line ruptures	0
	SI actuation	12.9
	Criticality attained	18.0
	Pressurizer empties	14.0
	RWST boron solution reaches core	64.9
2. Case 2 (concurrent loss of offsite power)	Steam line ruptures	0.0
	SI actuation	18.0
	Criticality attained	22.0
	Pressurizer empties	18.0
	RWST boron solution reaches core	82.0

TABLE 15.1-2

EQUIPMENT REQUIRED FOLLOWING A RUPTURE OF A MAIN STEAM LINE

Short-Term (Required for Mitigation of Accident)	Hot Standby	Required for Cooldown
Reactor trip and safeguard actuation channels, including sensors, circuitry, and processing equipment (the protection circuits used to trip the reactor on undervoltage, underfrequency, and turbine trip may be excluded)	Auxiliary feedwater system, including pumps, water supply steam generator atmospheric relief valves 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 atmospheric relief valves Control for defeating automatic safety injection actuation during a cooldown and depressurization (i.e., SIS is reset)
Safety injection system, including the pumps, the refueling water storage tank, the boron injection tank, and the system valves and piping	Containment air coolers Capability for obtaining reactor coolant system sample	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
Containment spray system	Capability for boration to required hot standby concentration	Capability to depressurize the reactor coolant system to allow residual heat removal system operation
Diesel generators and emergency power distribution equipment		
Essential service water system		
Containment air coolers		

TABLE 15.1-2 (Sheet 2)

Short-Term (Required for Mitigation of Accident)	Hot Standby	Required for Cooldown
Auxiliary feedwater system, including pumps, water supplies, piping and valves		
Main feedwater control valves (trip closed feature)		
Bypass feedwater control valves (trip closed feature)		
Primary and secondary safety valves		Capability to borate to cold shutdown concentration
Associated pump room coolers		
Circuits and/or equipment required to trip the main feedwater pumps		
Main feedwater isolation valves (trip closed feature)		
Main steamline isolation valves (trip closed feature)		
Main steamline isolation valve bypass valves (trip closed feature)		
Steam generator blowdown isolation valves (automatic closure feature)		
Batteries (Class 1E)		

TABLE 15.1-2 (Sheet 3)

Short-Term (Required for Mitigation of Accident)	Hot Standby	Required for Cooldown
Control room ventilation		
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*		

*See Section 7.5 for a discussion of the post-accident monitoring systems.

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	3565
b.	Steam generator tube leakage, gpm	1
c.	Reactor coolant iodine activity:	
	1) Case 1	Dose equivalent of 1.0σCi/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
	2) Case 2	An assumed pre-accident iodine spike which has resulted in the dose equivalent of 60 σCi/gm of I-131
d.	Reactor coolant noble gas activity:	
	1) Case 1	Based on 1-percent failed fuel as provided in Table 11.1-5
	2) Case 2	Based on 1-percent failed fuel as provided in Table 11.1-5
e.	Secondary system initial activity	Dose equivalent of 0.1 σCi/gm of I-131
f.	Iodine partition factors	
	1) Affected steam generator	1.0
	2) Unaffected steam generator	0.01
g.	Reactor coolant mass, lbs	4.94E+5
h.	Steam generator mass	
	1) Affected steam generator, lbs	164,500
	2) Each unaffected steam generator, lbs	95,500

II. Atmospheric Dispersion Factors — See Table 15A-2

III. Activity Release Data:

a.	Affected steam generator	
	1) Initial steam release, 0-30 min, lbs	164,500
	2) Reactor coolant release, 0-8 hr, lbs	4,000
b.	Unaffected steam generator	
	1) Steam release, 0-2 hr, lbs	404,452
	2) Steam release, 2-8 hr, lbs	945,973
	3) Reactor coolant release, 0-8 hr, lbs	0

TABLE 15.1-3 (Sheet 2)

c. Activity released to the environment

1) Case 1

Isotope	0-2 hr (Ci)	0-8 hr (Ci)
I-131	2.45E+1	3.00E+2
I-132	5.81E+1	2.99E+2
I-133	5.01E+1	5.69E+2
I-134	1.66E+1	3.32E+1
I-135	3.73E+1	3.48E+2
Xe-131m	1.54	6.11
Xe-133m	2.40	9.23
Xe-133	1.31E+2	5.13E+2
Xe-135m	5.13E-2	5.15E-2
Xe-135	4.13	1.33E+1
Xe-137	4.68E-3	4.68E-3
Xe-138	6.29E-2	6.31E-2
Kr-83m	1.76E-1	3.14E-1
Kr-85m	8.81E-1	2.34
Kr-85	4.27	1.70E+1
Kr-87	4.06E-1	6.03E-1
Kr-88	1.53	3.39
Kr-89	2.09E-3	2.10E-3

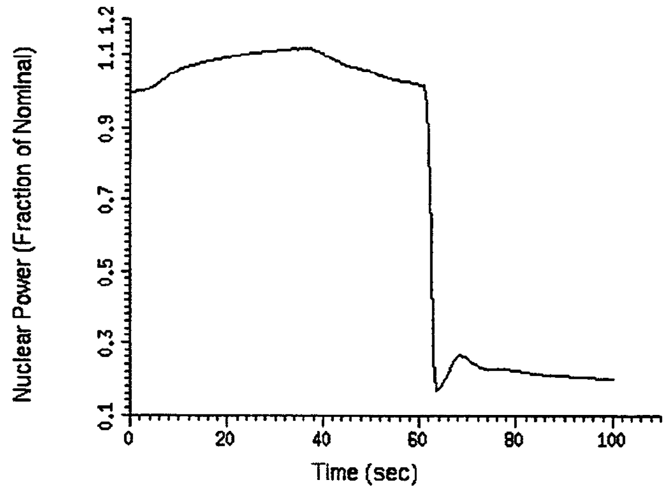
2) Case 2

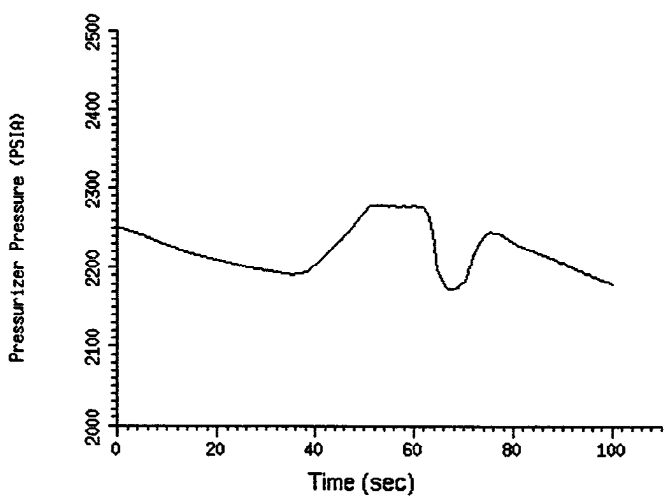
Isotope	0-2 hr (Ci)	0-8 hr (Ci)
I-131	2.54E+1	8.41E+1
I-132	2.30E+1	4.01E+1
I-133	4.37E+1	1.33E+2
I-134	4.17	4.86
I-135	2.43E+1	6.21E+1
Xe-131m	1.54	6.11
Xe-133m	2.40	9.23
Xe-133	1.31E+2	5.13E+2
Xe-135m	5.13E-2	5.15E-2
Xe-135	4.13	1.33E+1
Xe-137	4.68E-3	4.68E-3
Xe-138	6.29E-2	6.31E-2
Kr-83m	1.76E-1	3.14E-1
Kr-85m	8.81E-1	2.34
Kr-85	4.27	1.70E+1
Kr-87	4.06E-1	6.03E-1
Kr-88	1.53	3.39
Kr-89	2.09E-3	2.10E-3

TABLE 15.1-4

RADIOLOGICAL CONSEQUENCES OF A MAIN STEAM LINE BREAK

	Dose (rem)
CASE 1, 1.0 σCi/gm I-131 equivalent w/I spike	
Exclusion area boundary (0-2 hr)	
Thyroid	2.76
Whole body	9.98E-3
Low-population zone outer boundary (duration)	
Thyroid	4.33
Whole body	8.78E-3
CASE 2, 60 σCi/gm I-131 equivalent	
Exclusion area boundary (0-2 hr)	
Thyroid	2.67
Whole body	5.34E-3
Low population zone outer boundary (duration)	
Thyroid	1.15
Whole body	1.69E-3



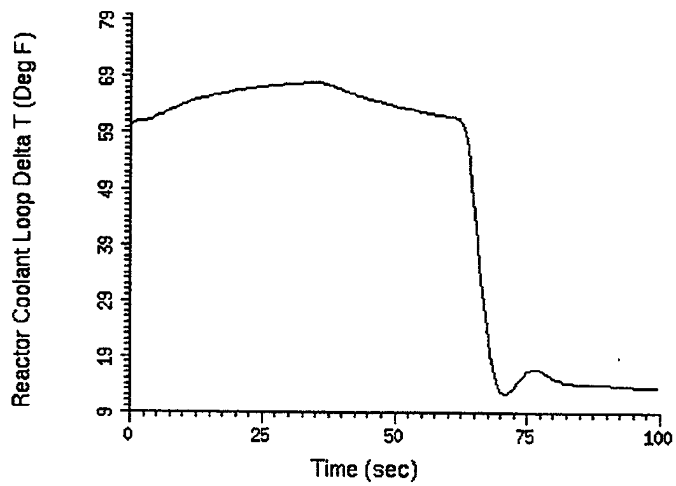


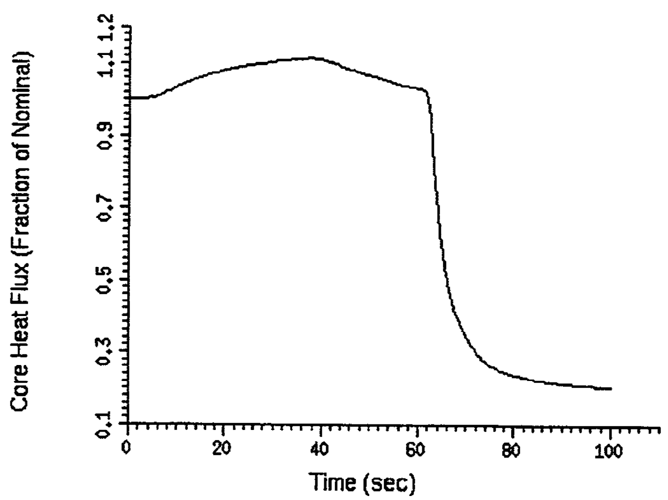
Rev. 10

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.1-1

NUCLEAR POWER, AND PRESSURIZER PRESSURE TRANSIENTS FOR FEEDWATER CONTROL VALVE MALFUNCTION



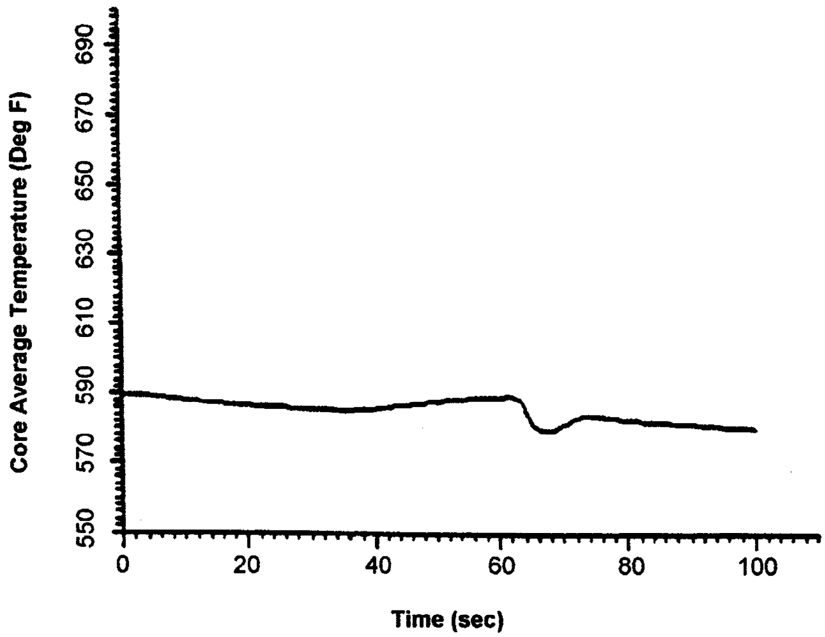


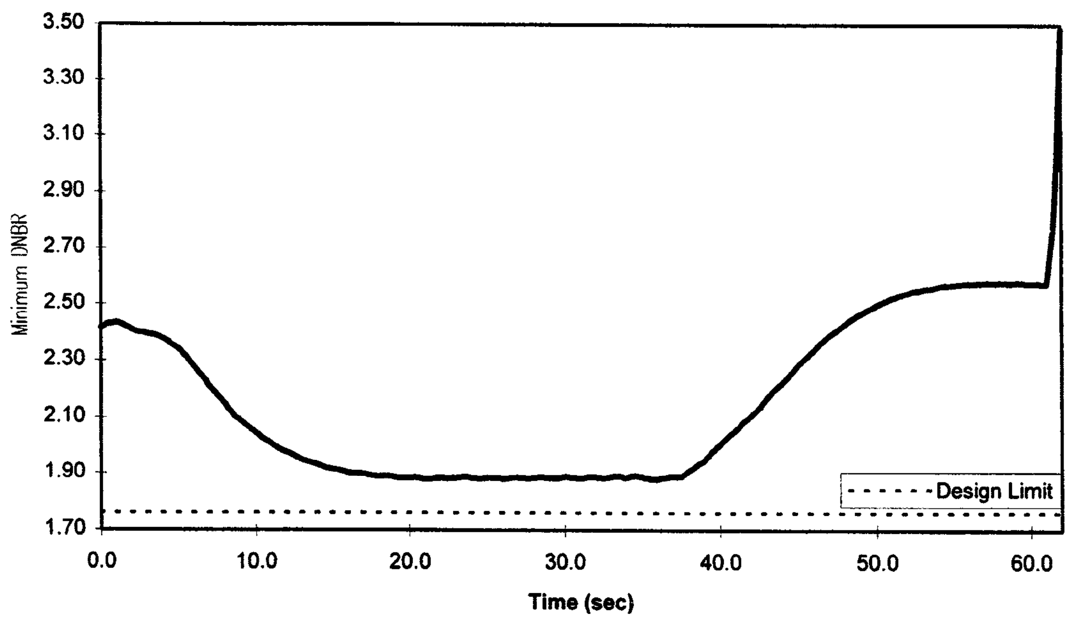
Rev. 10

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.1-2

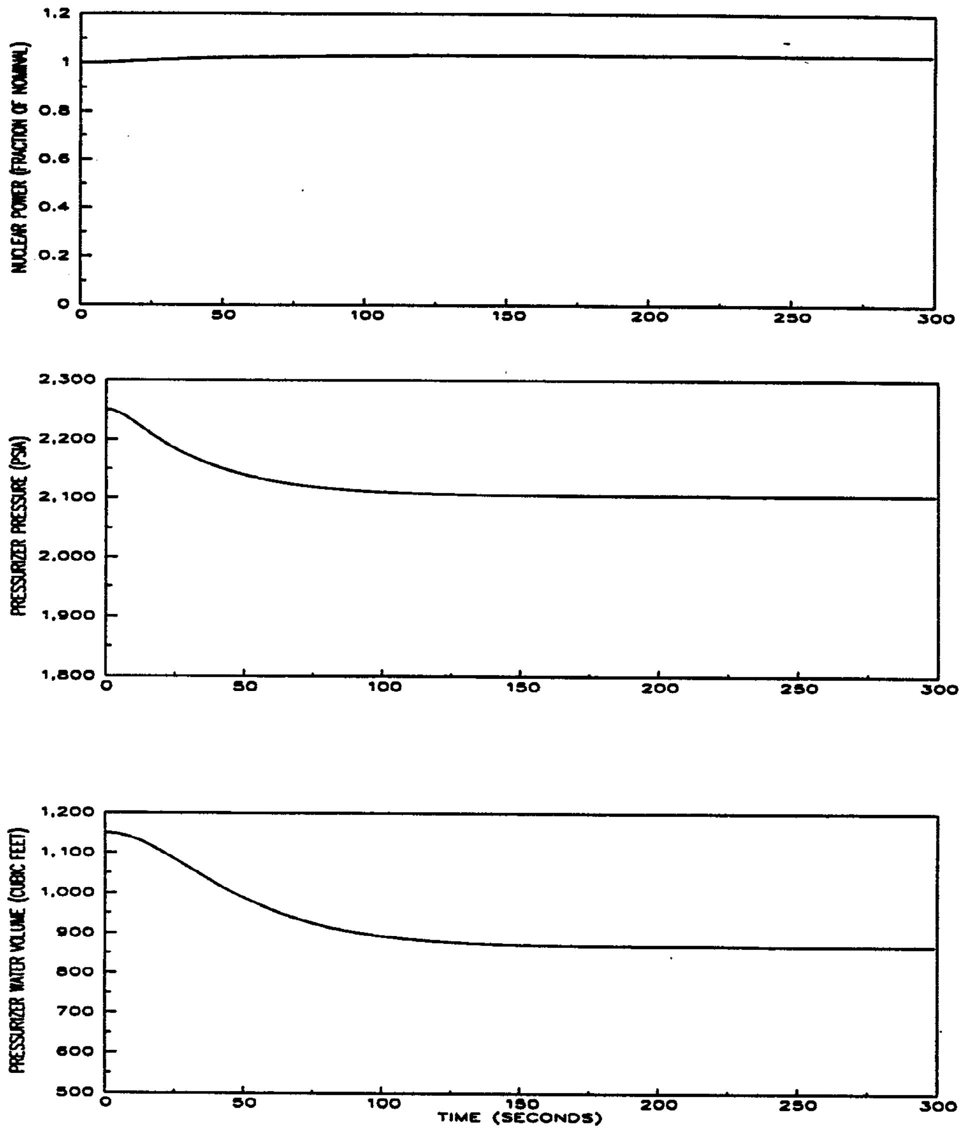
REACTOR COOLANT LOOP ΔT, AND CORE HEAT FLUX TRANSIENTS FOR FEEDWATER CONTROL VALVE MALFUNCTION





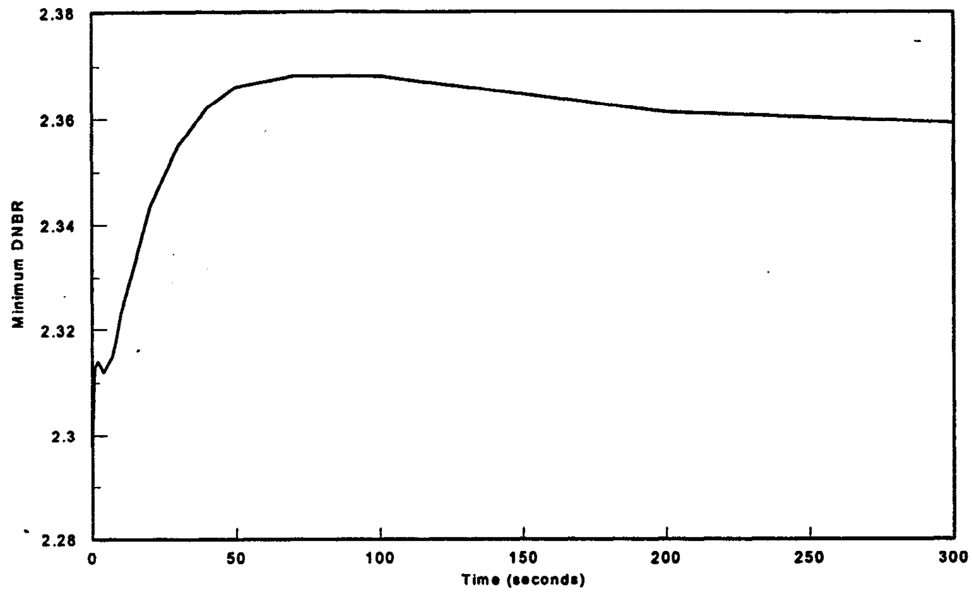
Rev. 13

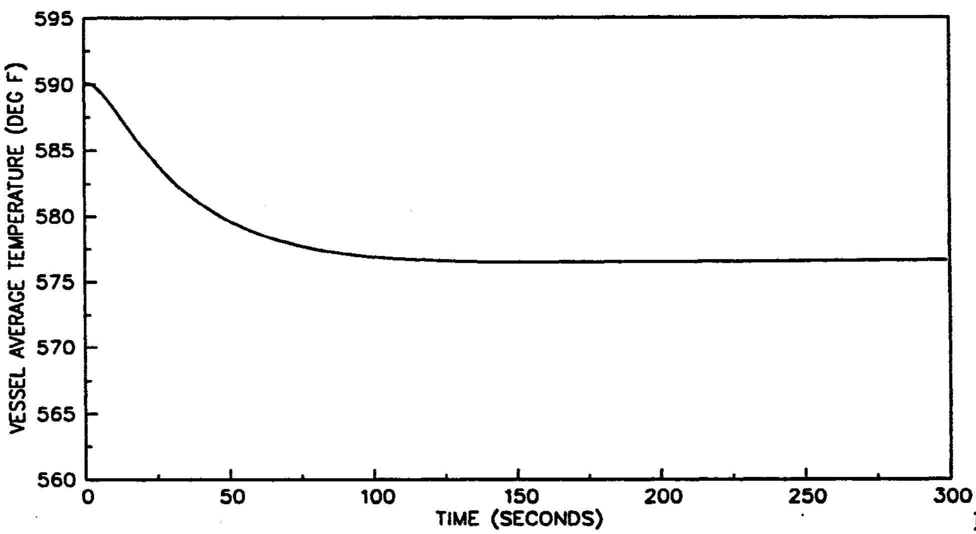
WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.1-2A CORE AVERAGE TEMPERATURE TRANSIENT AND MINIMUM DNBR FOR FEEDWATER CONTROL VALVE MALFUNCTION



Rev. 7

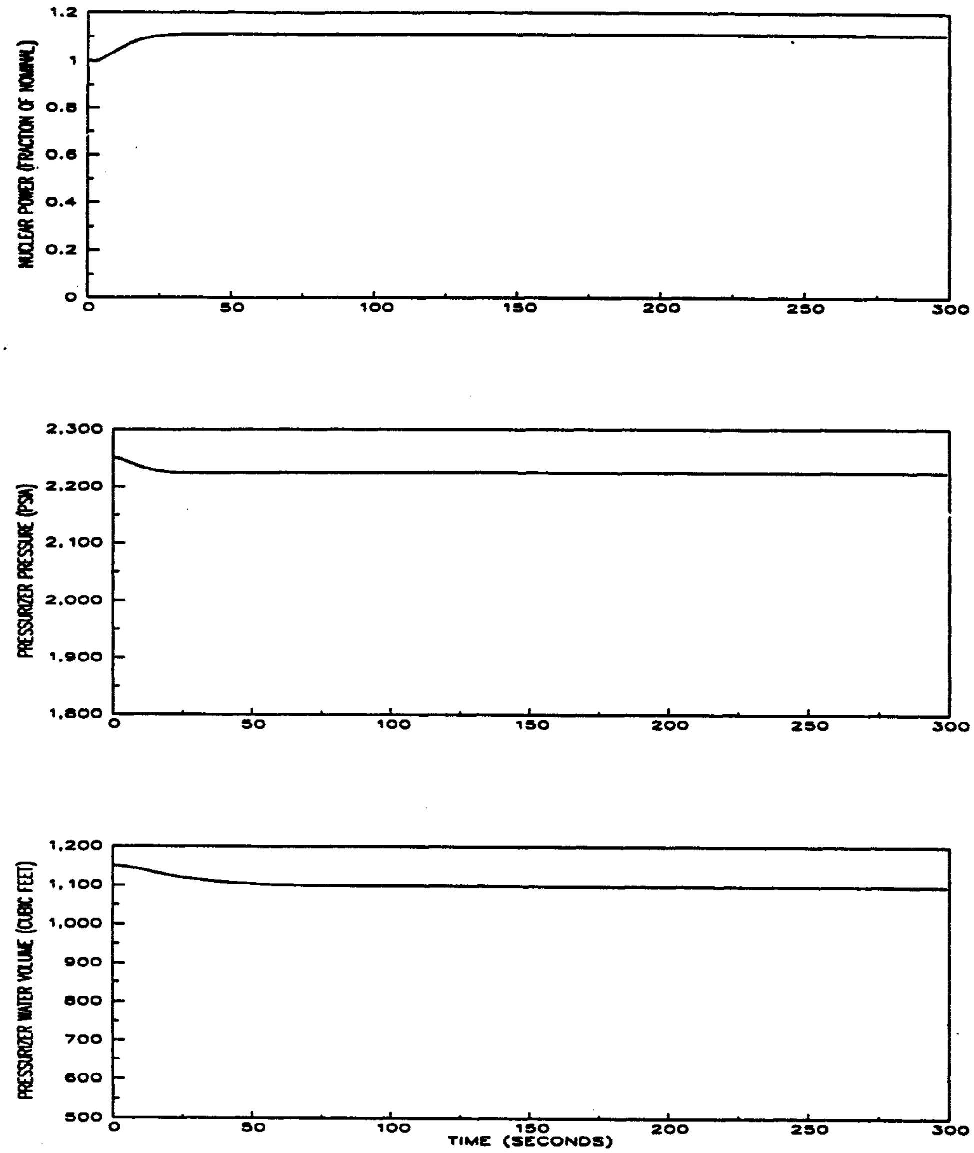
WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.1-3 NUCLEAR POWER, PRESSURIZER PRESSURE AND WATER VOLUME TRANSIENTS FOR TEN PERCENT STEP LOAD INCREASE MINIMUM REACTIVITY FEEDBACK, MANUAL REACTOR CONTROL





Rev.7

WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.1-4 DNBR AND VESSEL AVERAGE TEMPERATURE TRANSIENTS FOR TEN PERCENT STEP LOAD INCREASE, MINIMUM REACTIVITY FEEDBACK, MANUAL REACTOR CONTROL

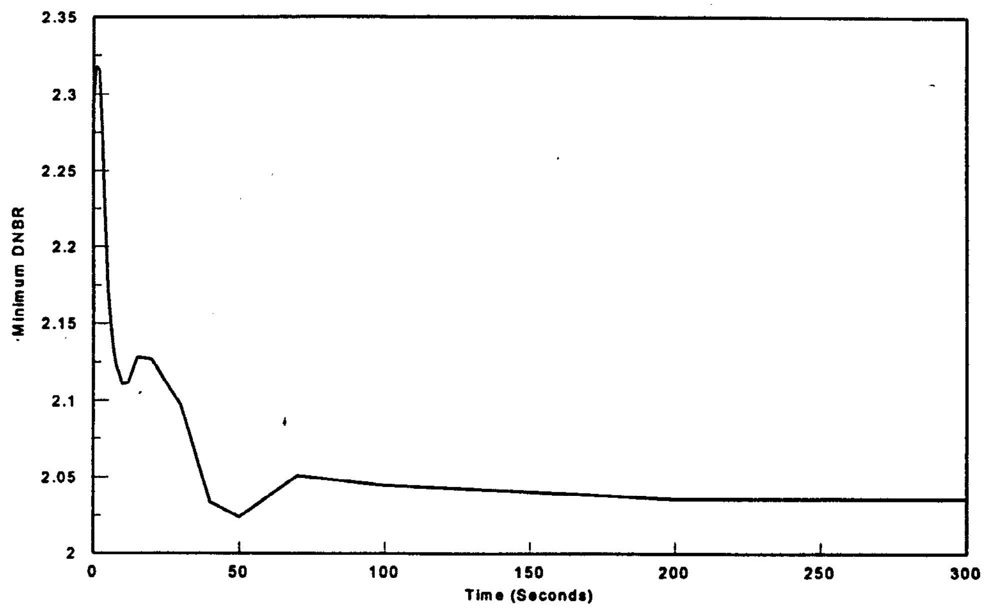


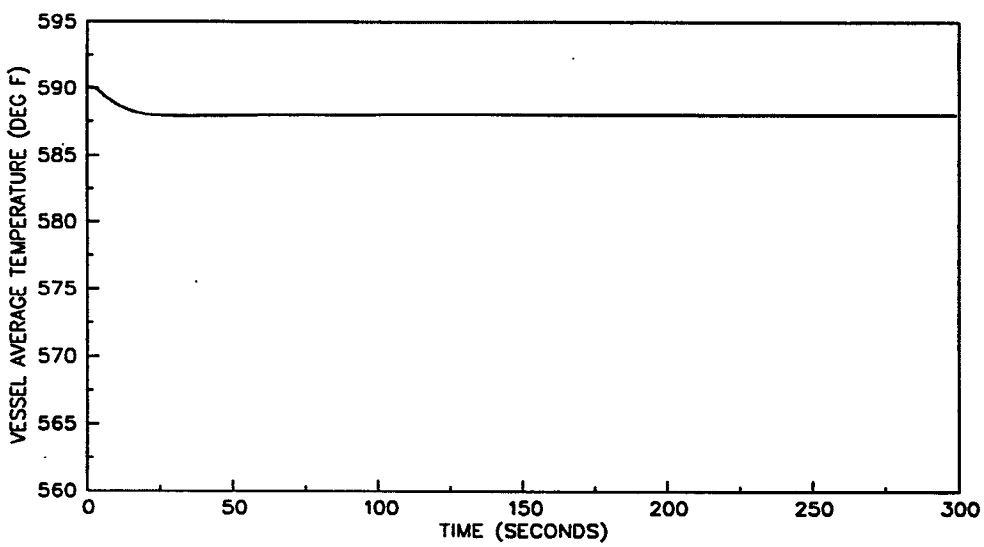
Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.1-5

NUCLEAR POWER, PRESSURIZER PRESSURE AND WATER VOLUME TRANSIENTS FOR TEN PERCENT STEP LOAD INCREASE MAXIMUM REACTIVITY FEEDBACK, MANUAL REACTOR CONTROL



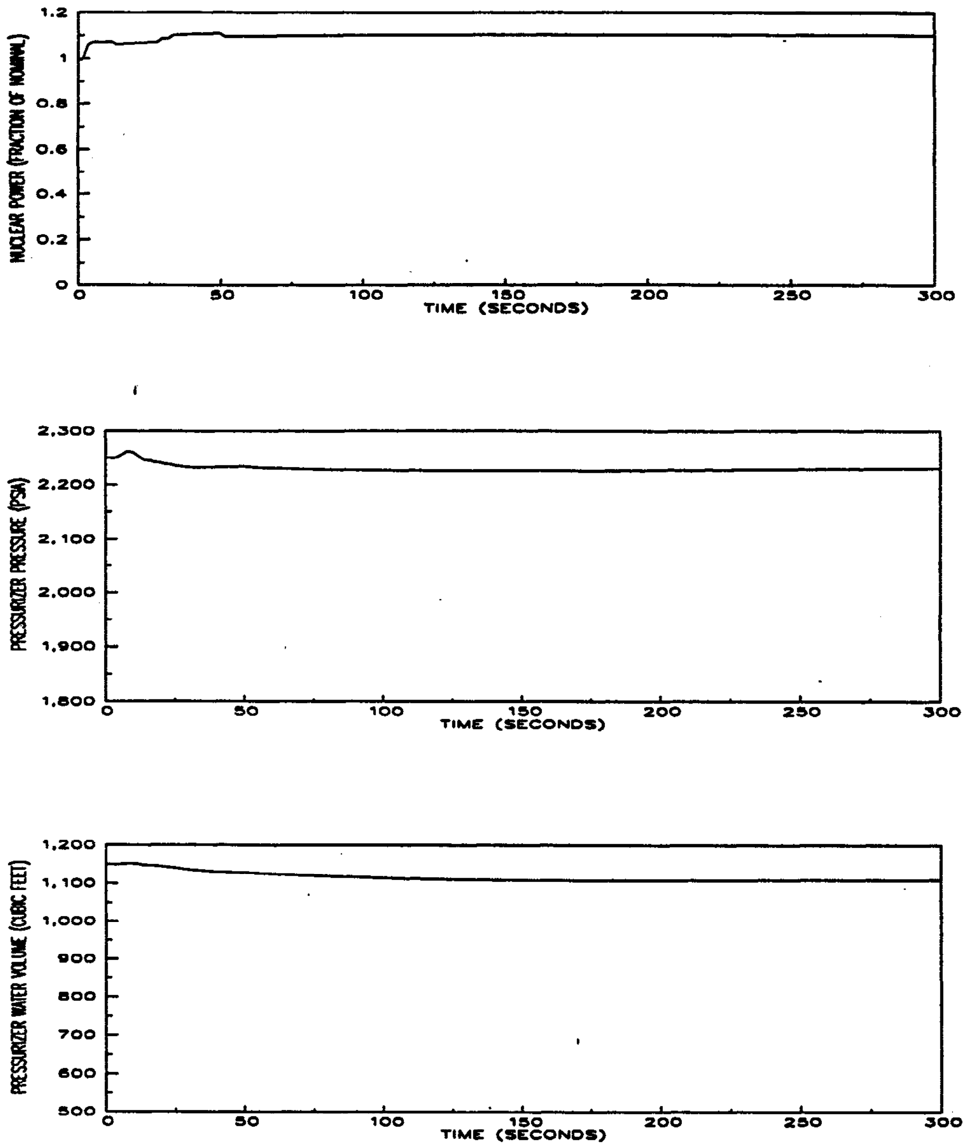


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.1-6

DNBR AND VESSEL AVERAGE TEMPERATURE TRANSIENTS FOR TEN PERCENT STEP LOAD INCREASE, MINIMUM REACTIVITY FEEDBACK, MANUAL REACTOR CONTROL

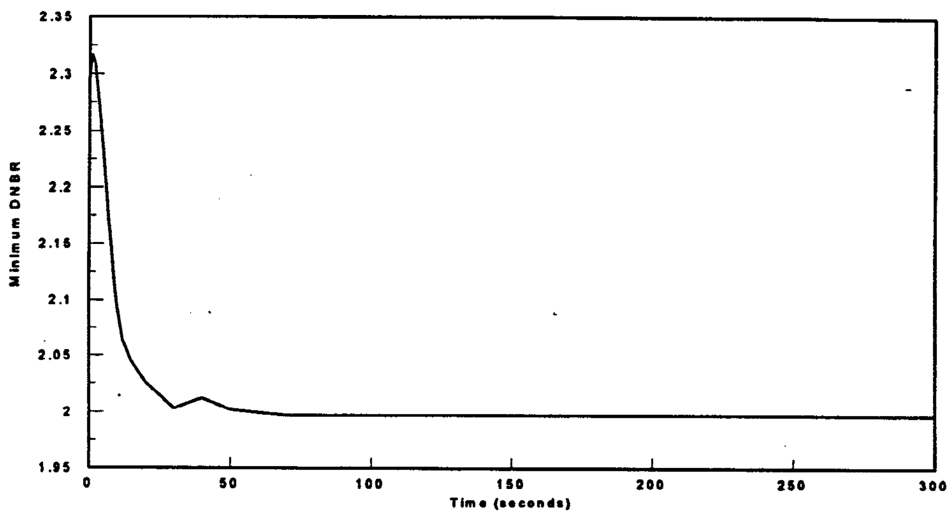


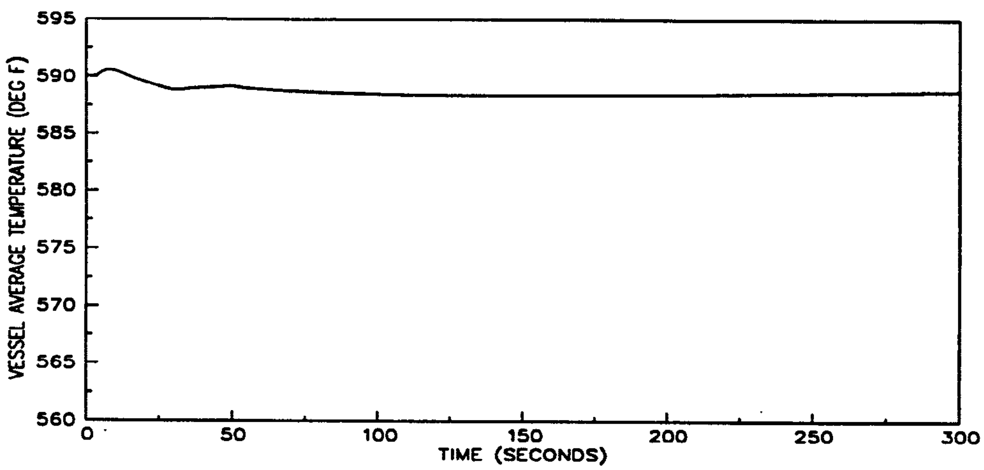
Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.1-7

NUCLEAR POWER, PRESSURIZER PRESSURE AND WATER VOLUME TRANSIENTS FOR TEN PERCENT STEP LOAD INCREASE MINIMUM REACTIVITY FEEDBACK, AUTOMATIC REACTOR CONTROL



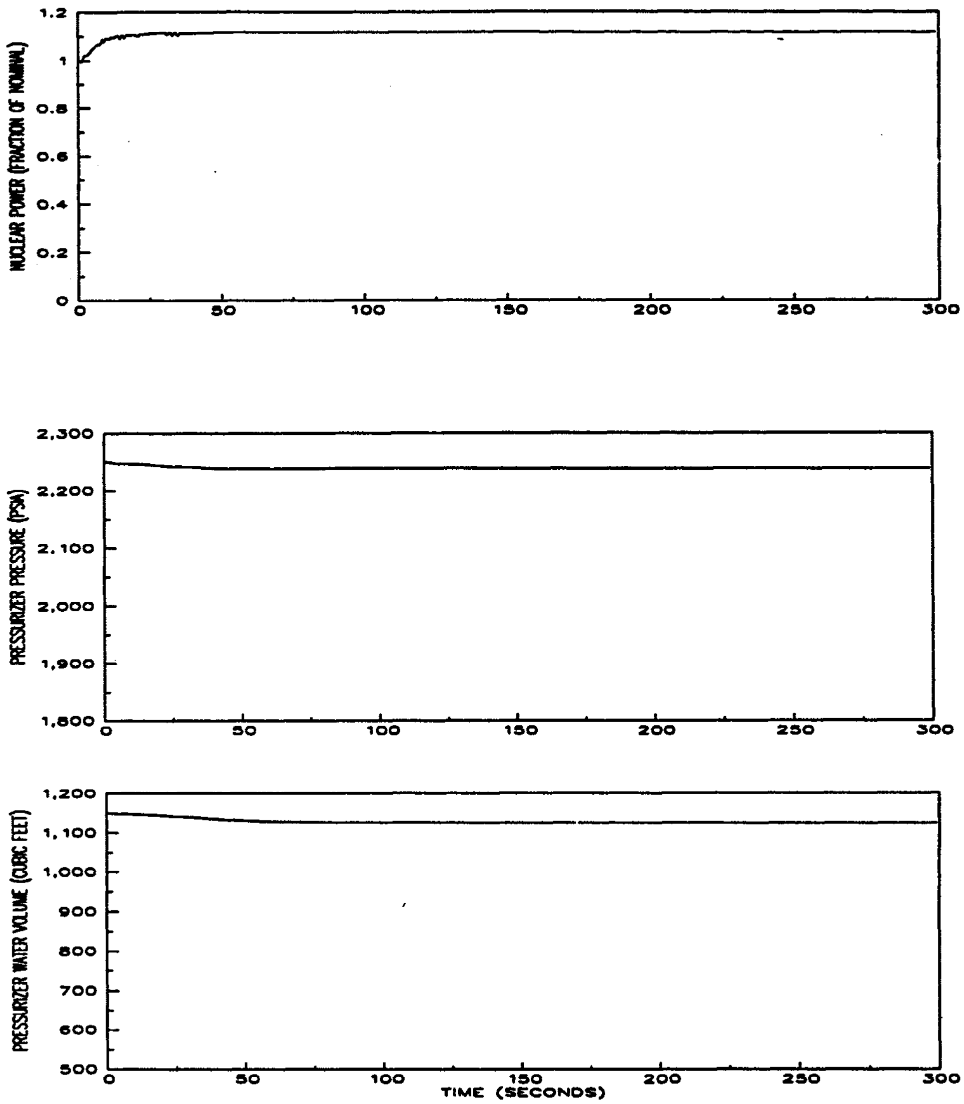


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.1-8

DNBR AND VESSEL AVERAGE TEMPERATURE TRANSIENTS FOR TEN PERCENT STEP LOAD INCREASE, MINIMUM REACTIVITY FEEDBACK, AUTOMATIC REACTOR CONTROL

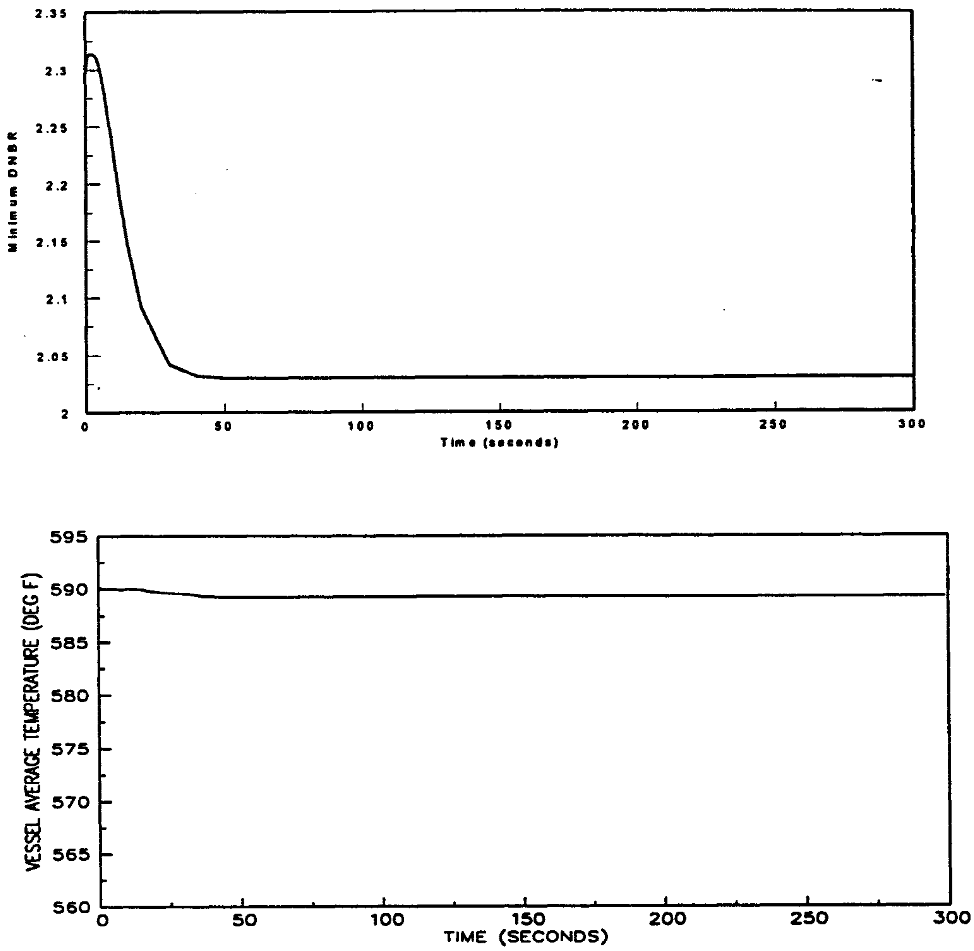


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.1-9

NUCLEAR POWER, PRESSURIZER PRESSURE AND WATER VOLUME TRANSIENTS FOR TEN PERCENT STEP LOAD INCREASE MAXIMUM REACTIVITY FEEDBACK, AUTOMATIC REACTOR CONTROL

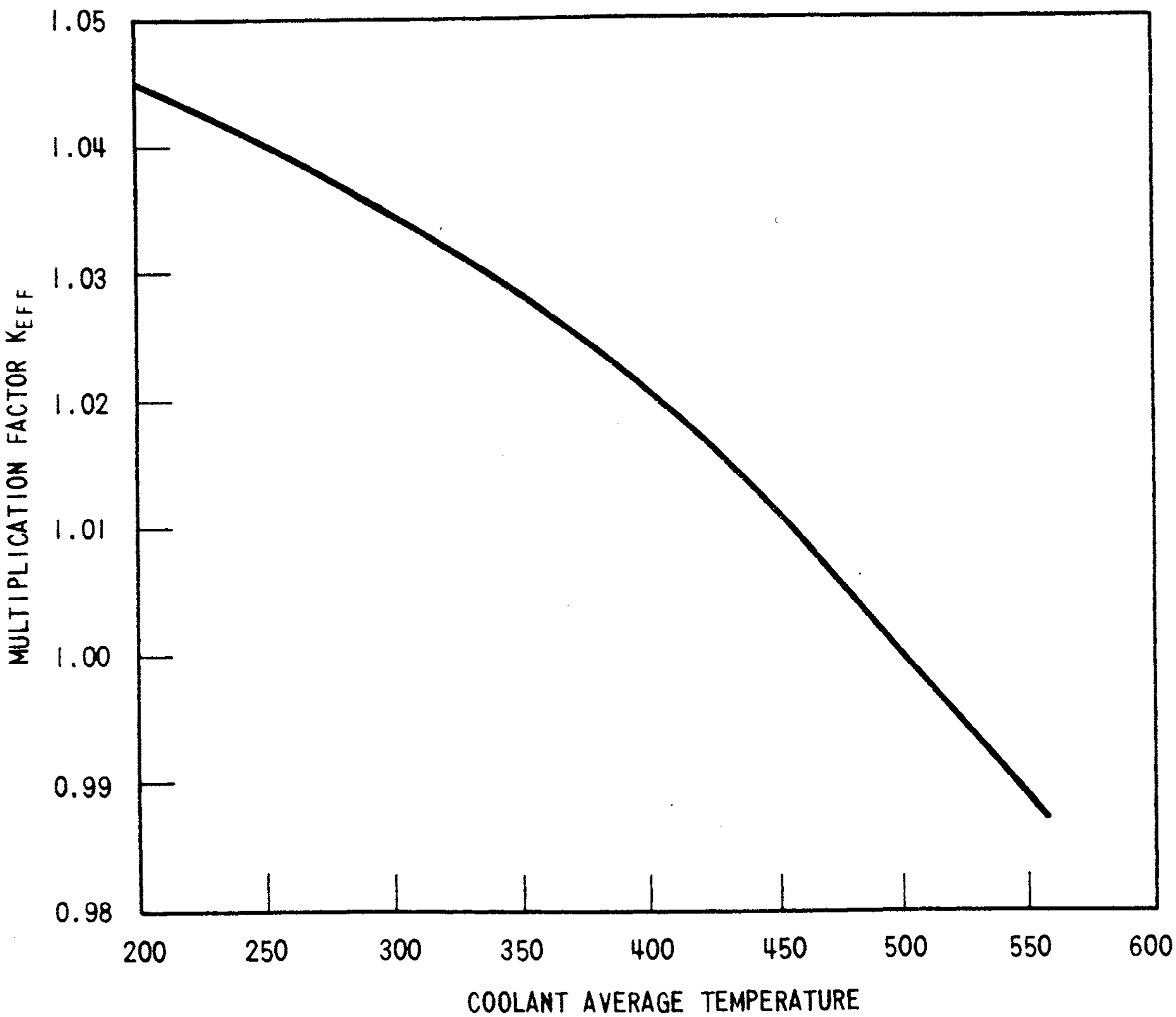


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.1-10

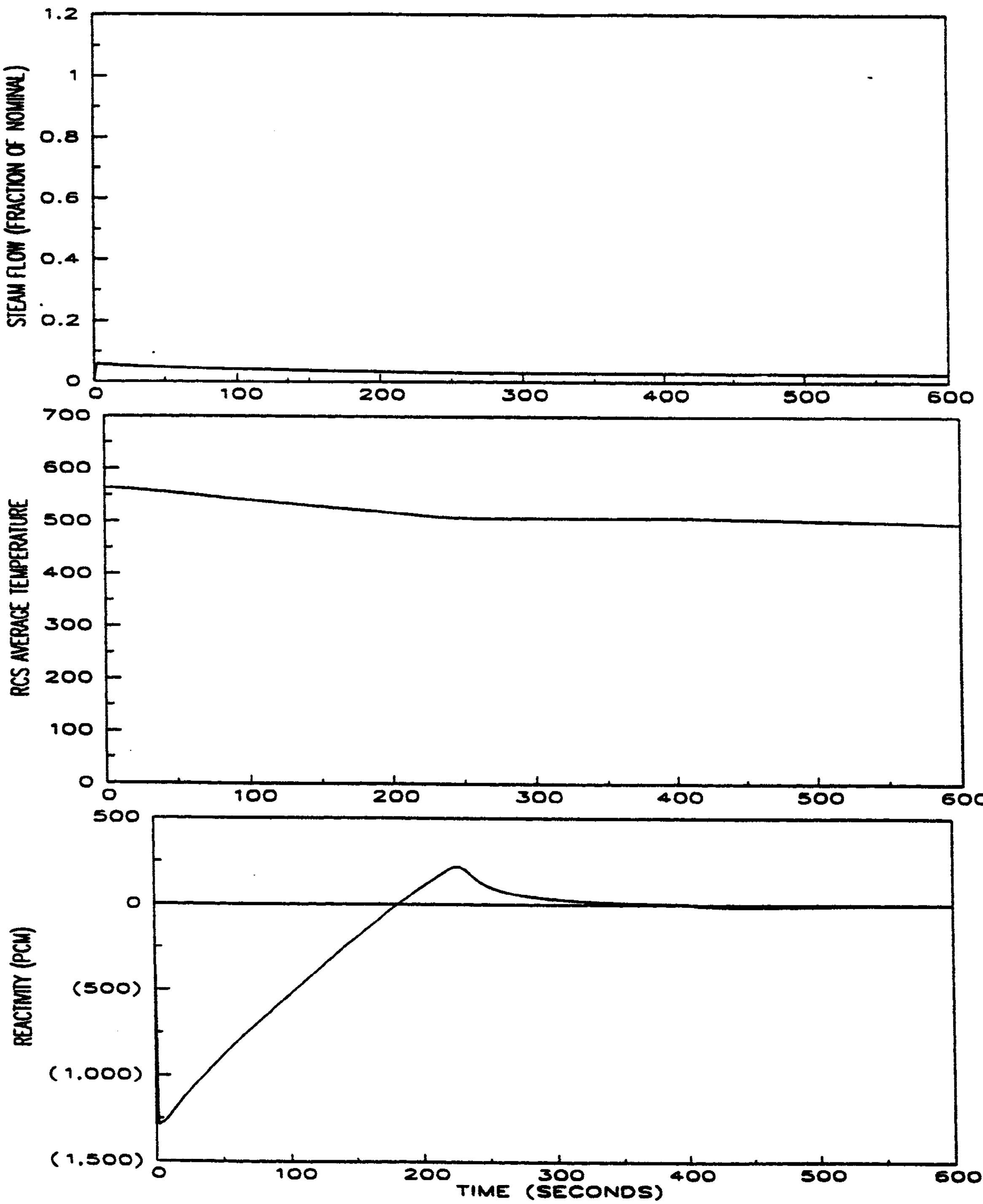
DNBR AND VESSEL AVERAGE TEMPERATURE TRANSIENTS FOR TEN PERCENT STEP LOAD INCREASE, MAXIMUM REACTIVITY FEEDBACK, AUTOMATIC REACTOR CONTROL



Rev. 0

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.1-11
K_{EFF} VERSUS TEMPERATURE



Rev. 13

WOLF CREEK
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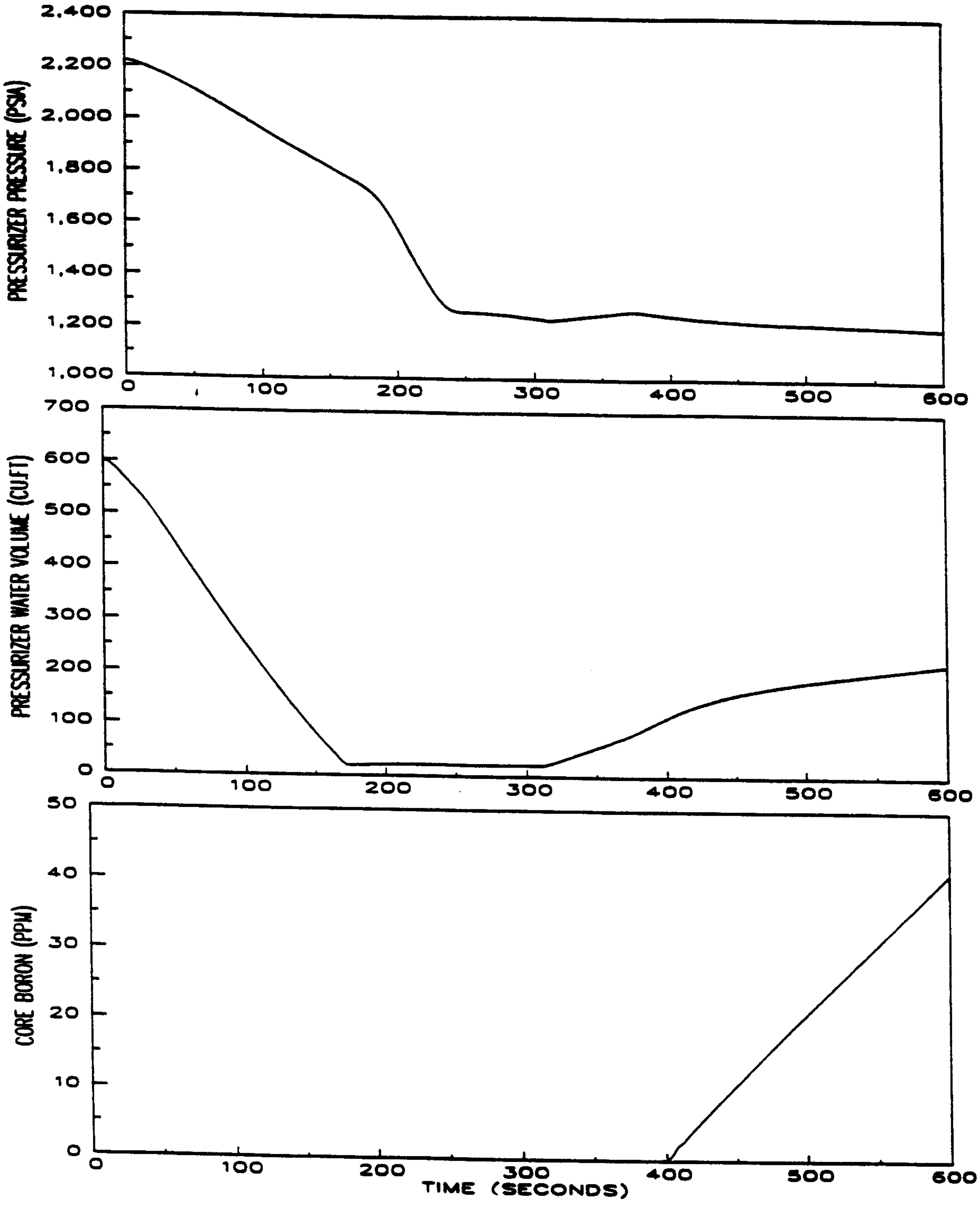
FIGURE 15.1-12

FAILURE OF A STEAM GENERATOR SAFETY OR ATMOSPHERIC RELIEF VALVE



Rev. 14

WOLF CREEK
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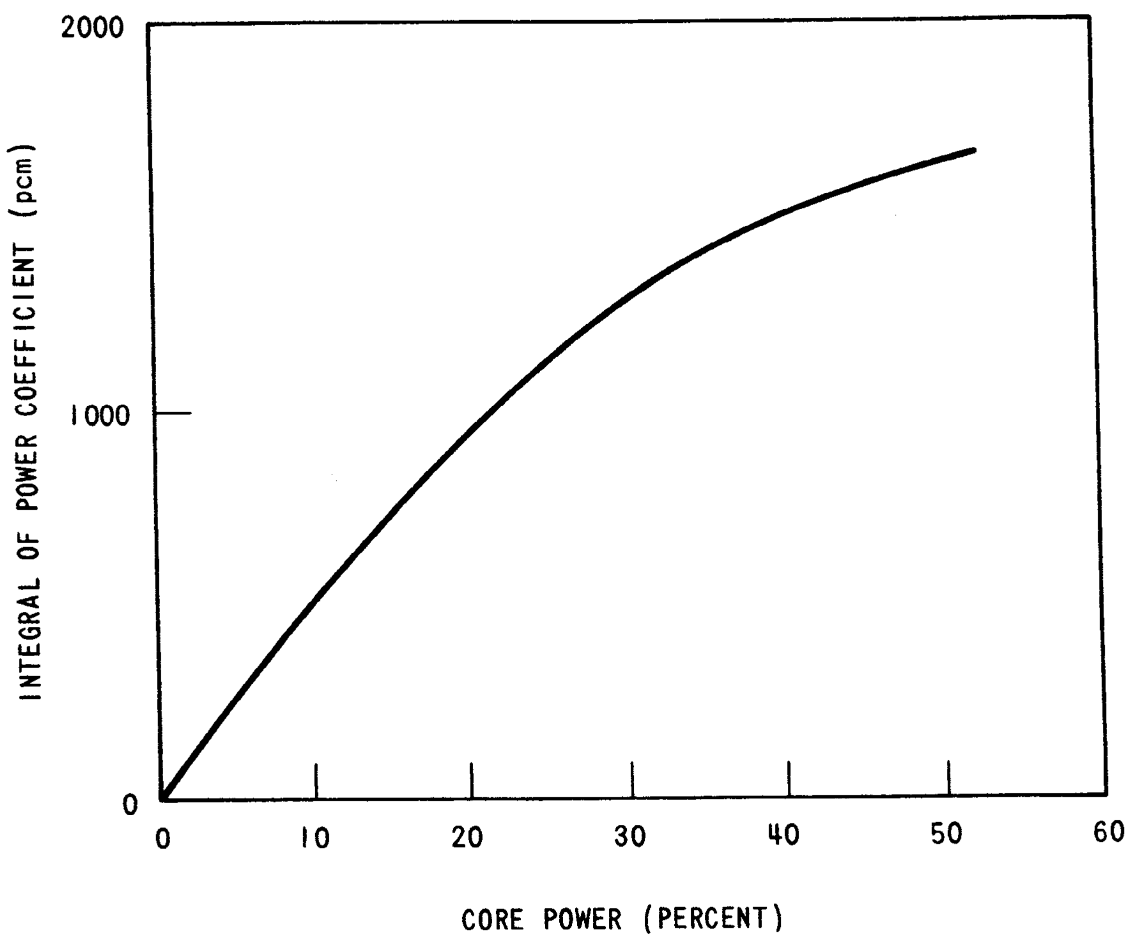
FIGURE 15.1-13

FAILURE OF A STEAM GENERATOR
SAFETY OR ATMOSPHERIC
RELIEF VALVE



Rev. 0

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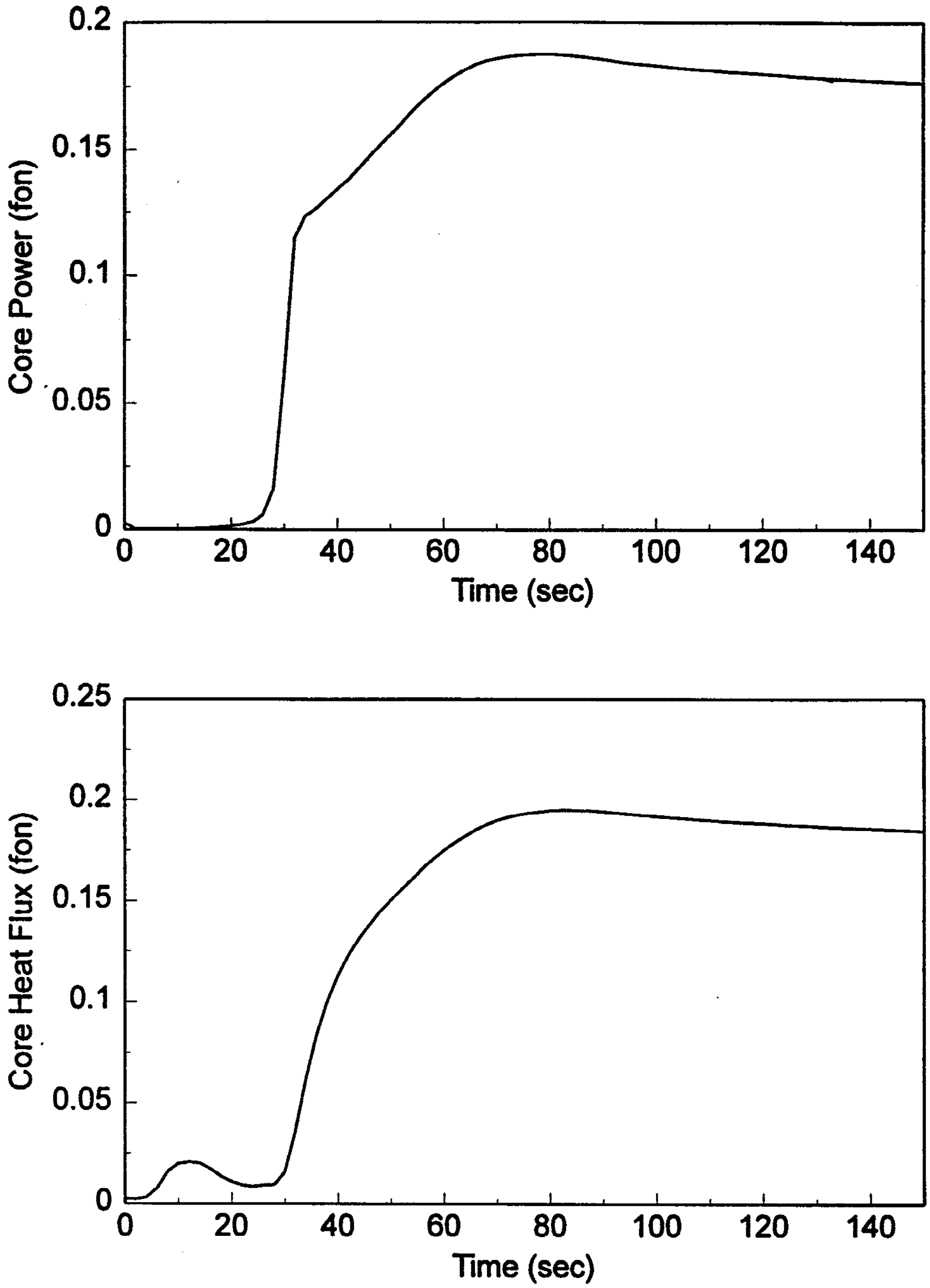
FIGURE 15.1-14

DOPPLER POWER FEEDBACK



Rev. 7

WOLF CREEK

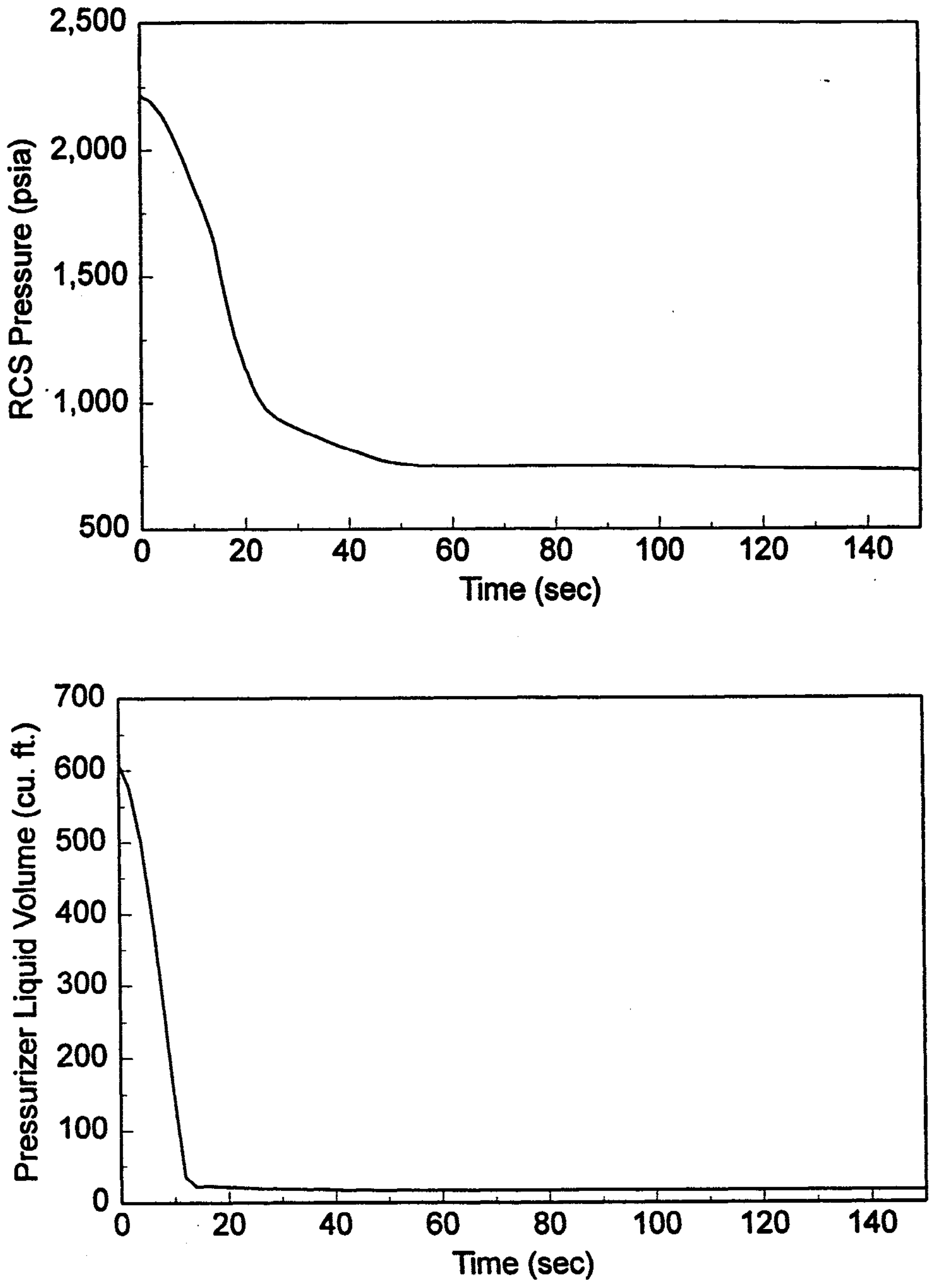
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.1-15:

NORMALIZED CORE POWER AND

NORMALIZED CORE HEAT FLUX

SLB WITH OFFSITE POWER



Rev. 7

WOLF CREEK

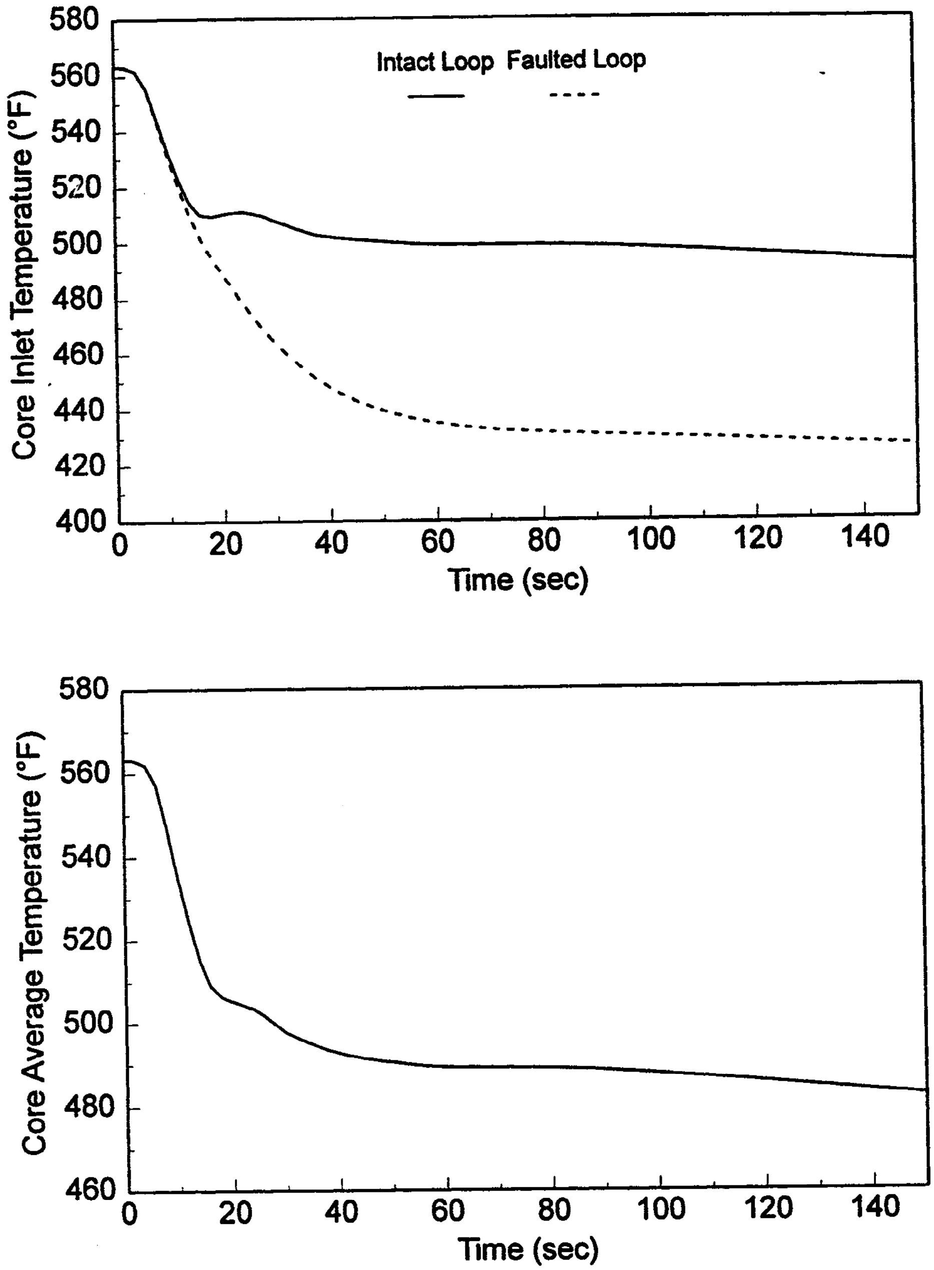
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.1-16:

PRESSURIZER PRESSURE AND

PRESSURIZER LIQUID VOLUME

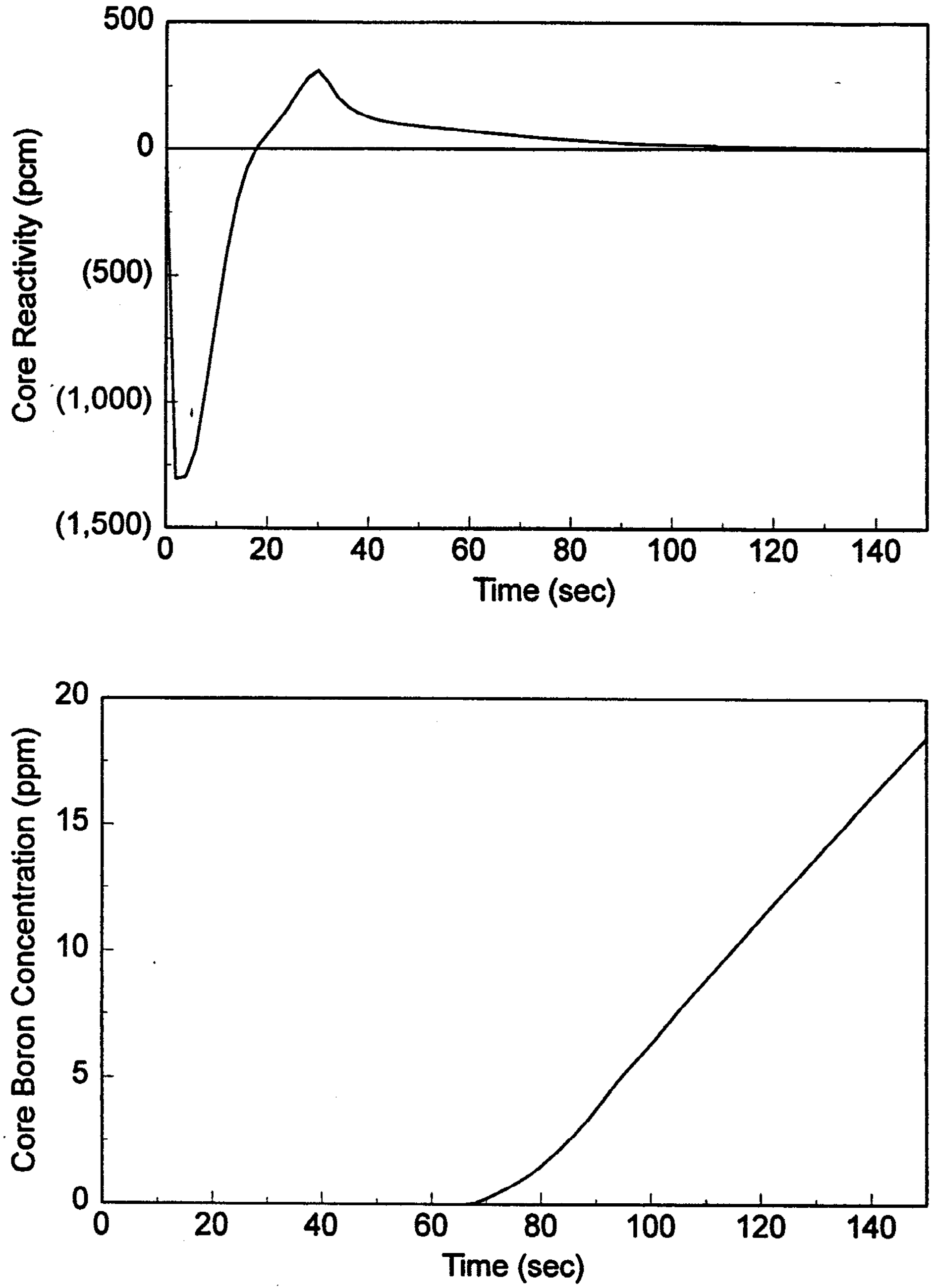
SLB WITH OFFSITE POWER



Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.1-17:
CORE INLET TEMPERATURE AND
CORE AVERAGE TEMPERATURE
SLB WITH OFFSITE POWER

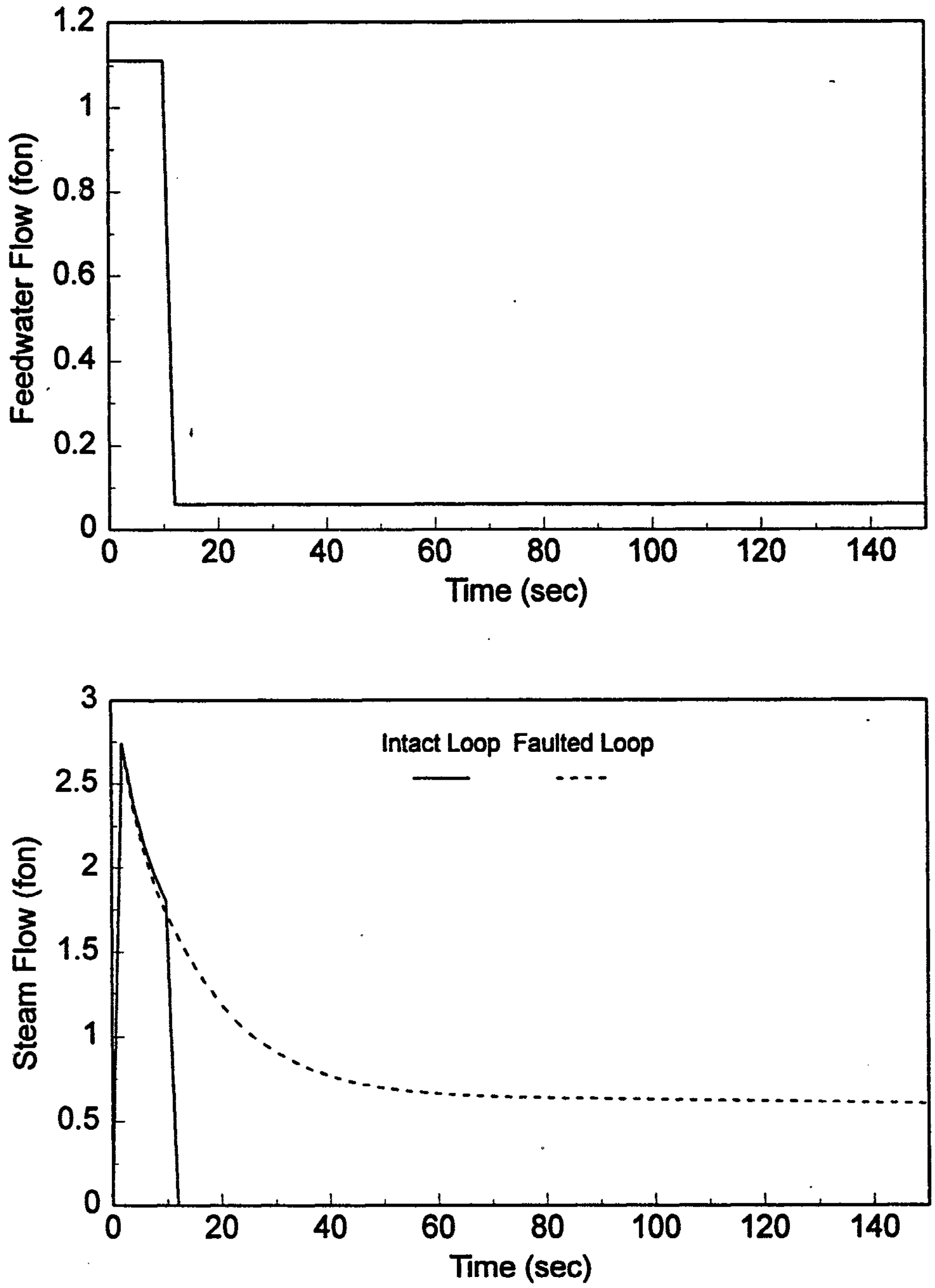


Rev. 7

WOLF CREEK

UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.1-18:
TOTAL CORE REACTIVITY AND CORE BORON CONCENTRATION
SLB WITH OFFSITE POWER



Rev. 7

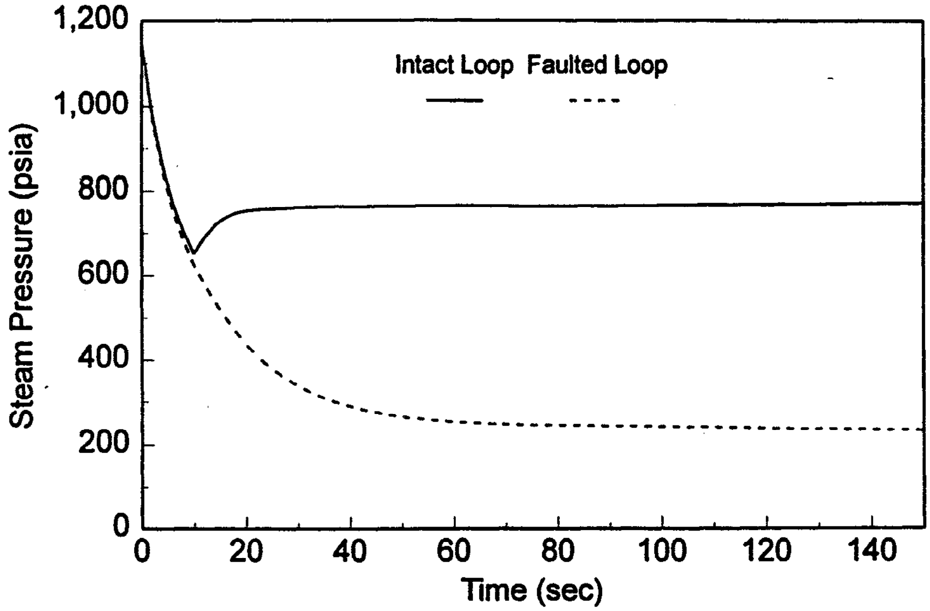
WOLF CREEK

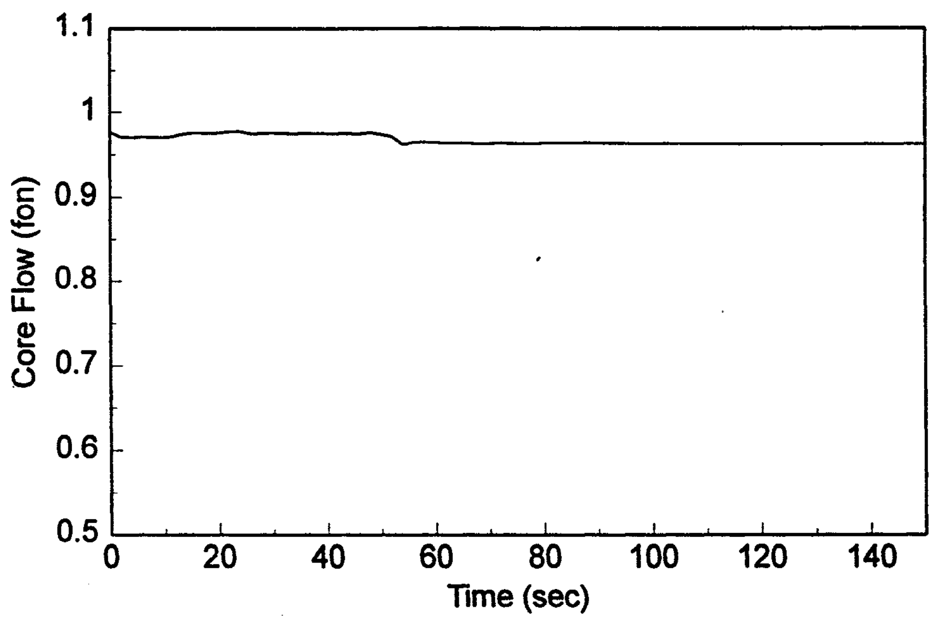
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.1-19

NORMALIZED FEEDWATER FLOW AND BREAK FLOW

SLB WITH OFFSITE POWER

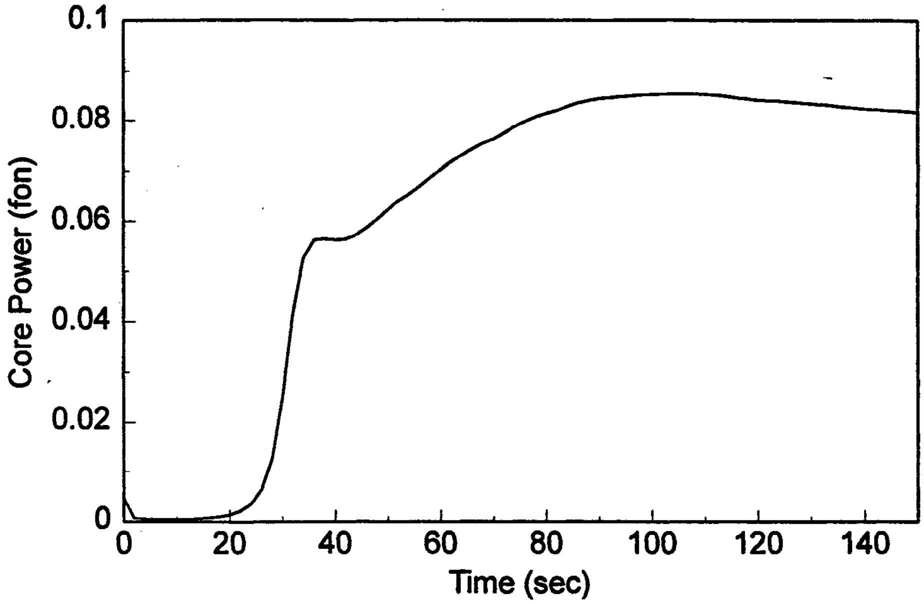


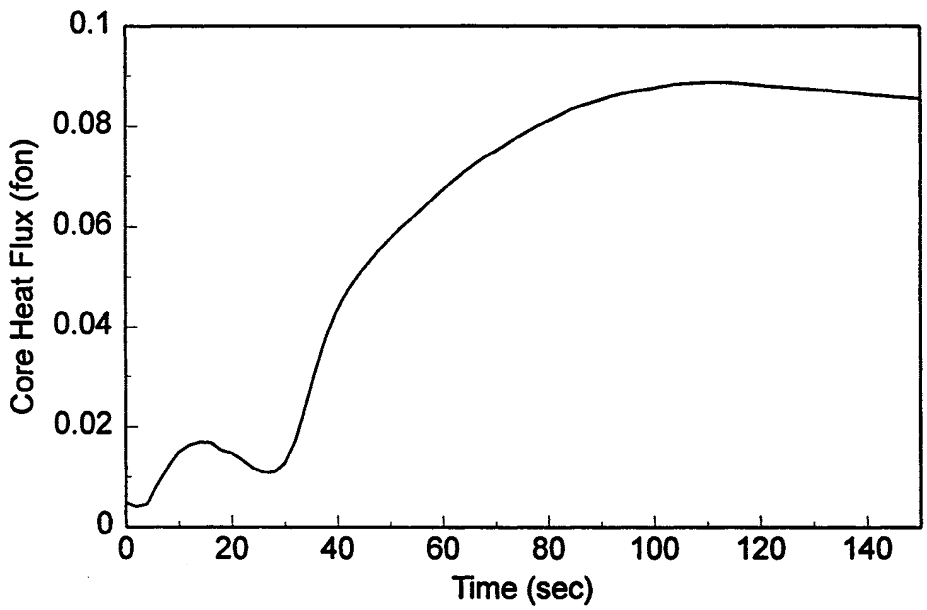


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

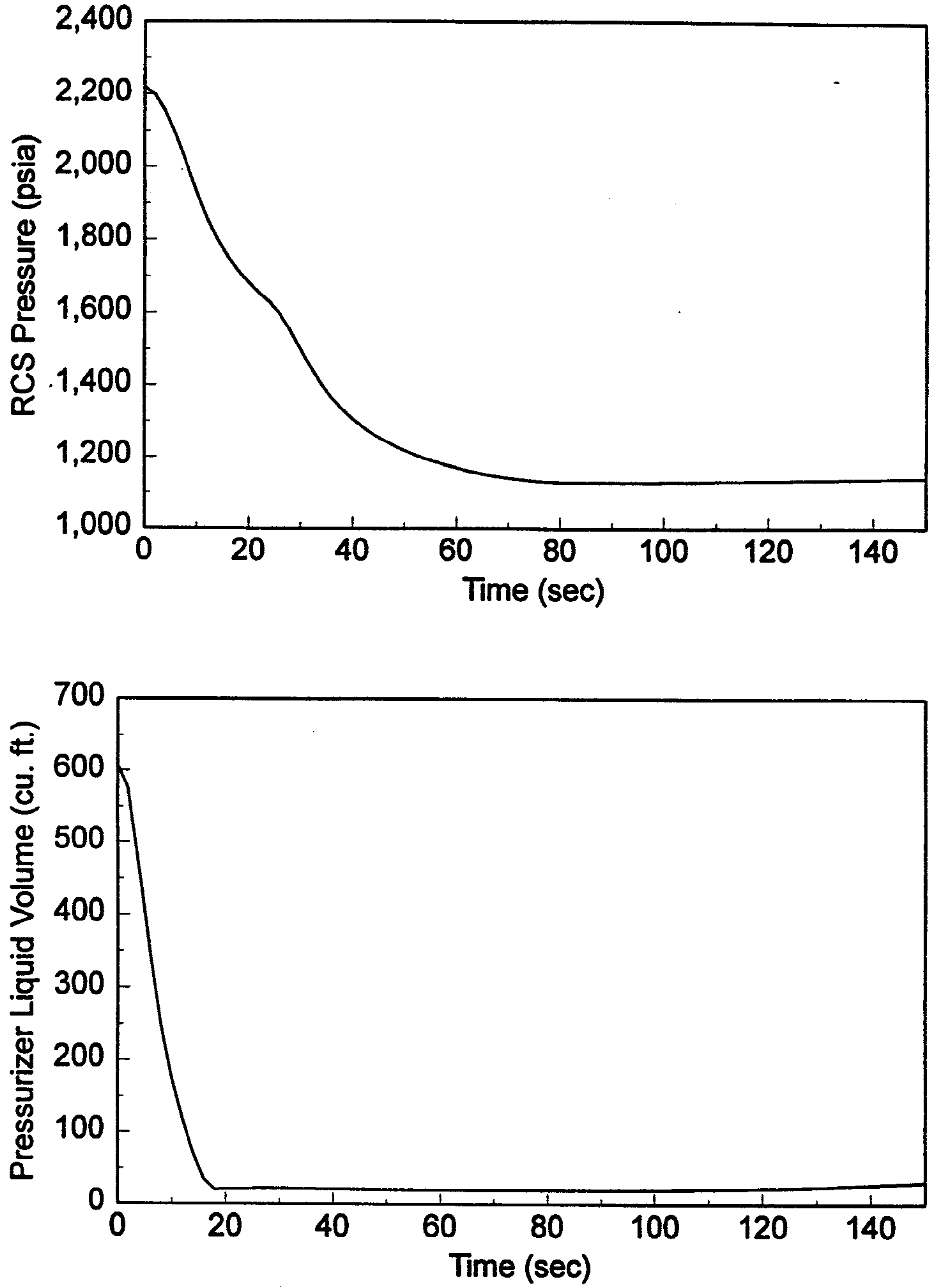
FIGURE 15.1-20:
STEAMLINE PRESSURE AND CORE
INLET FLOW
SLB WITH OFFSITE POWER





Rev. 7

WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.1-21: NORMALIZED CORE POWER AND NORMALIZED CORE HEAT FLUX SLB WITHOUT OFFSITE POWER



Rev. 7

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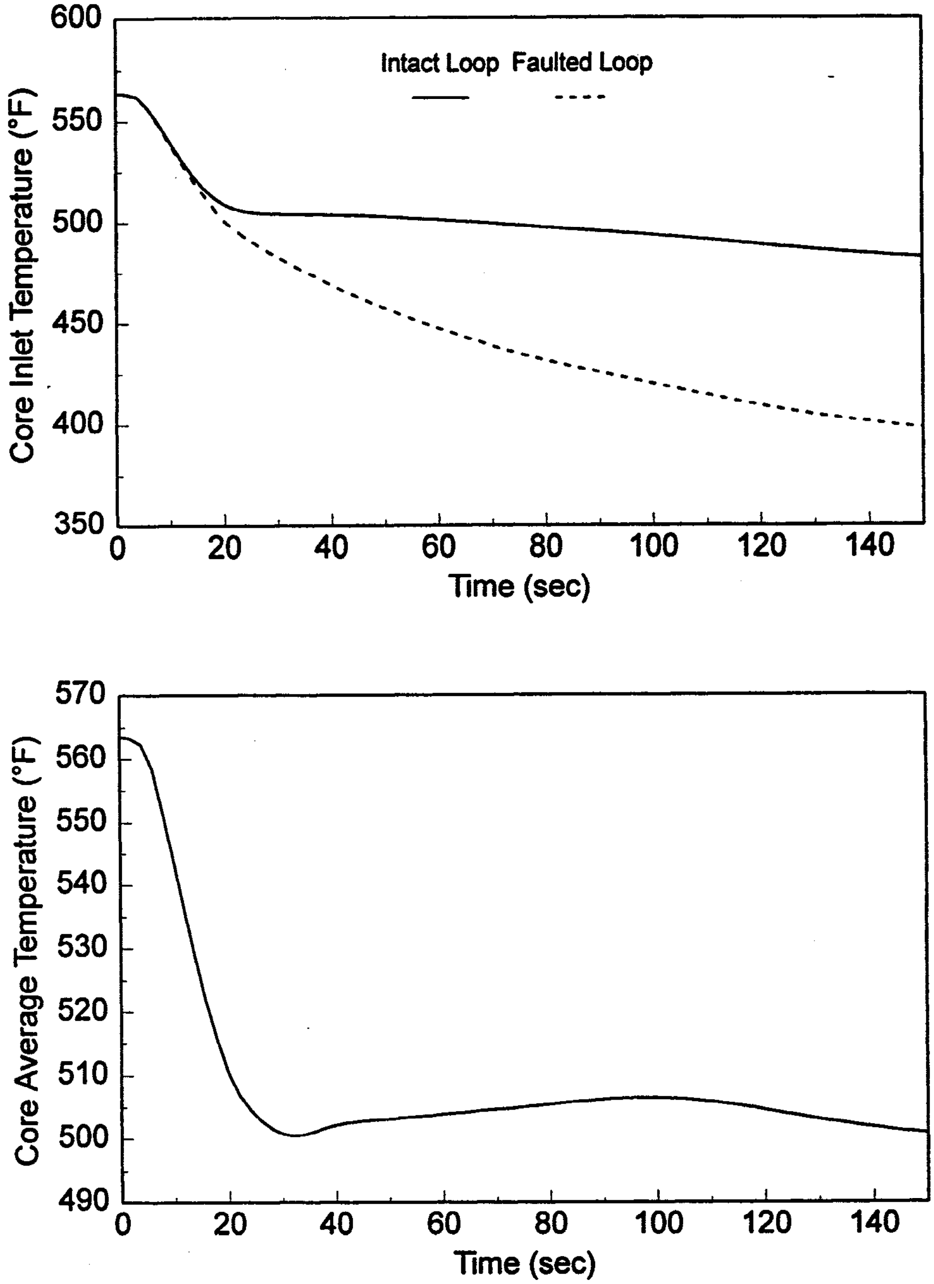
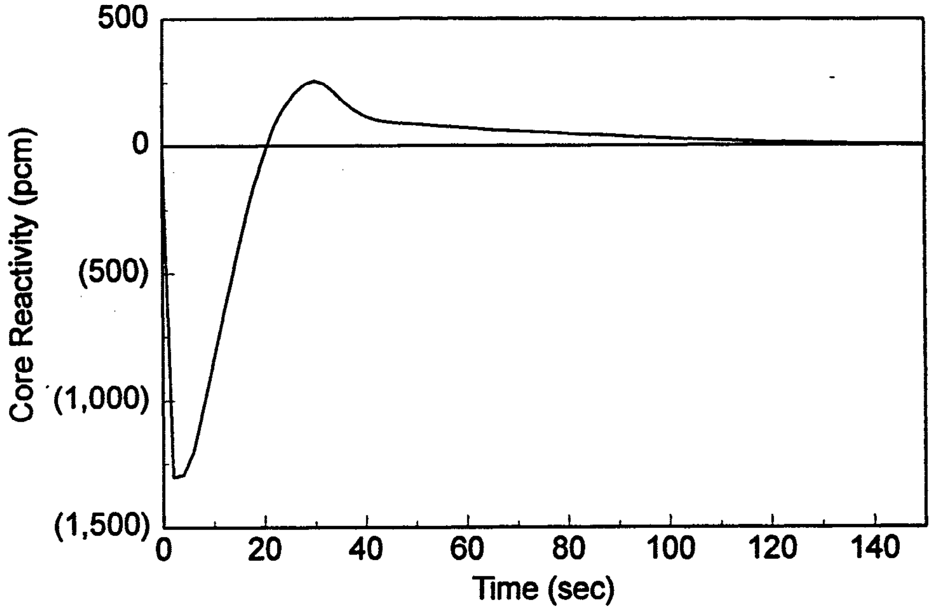
FIGURE 15.1-22: PRESSURIZER PRESSURE AND PRESSURIZER LIQUID VOLUME SLB WITHOUT OFFSITE POWER



Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

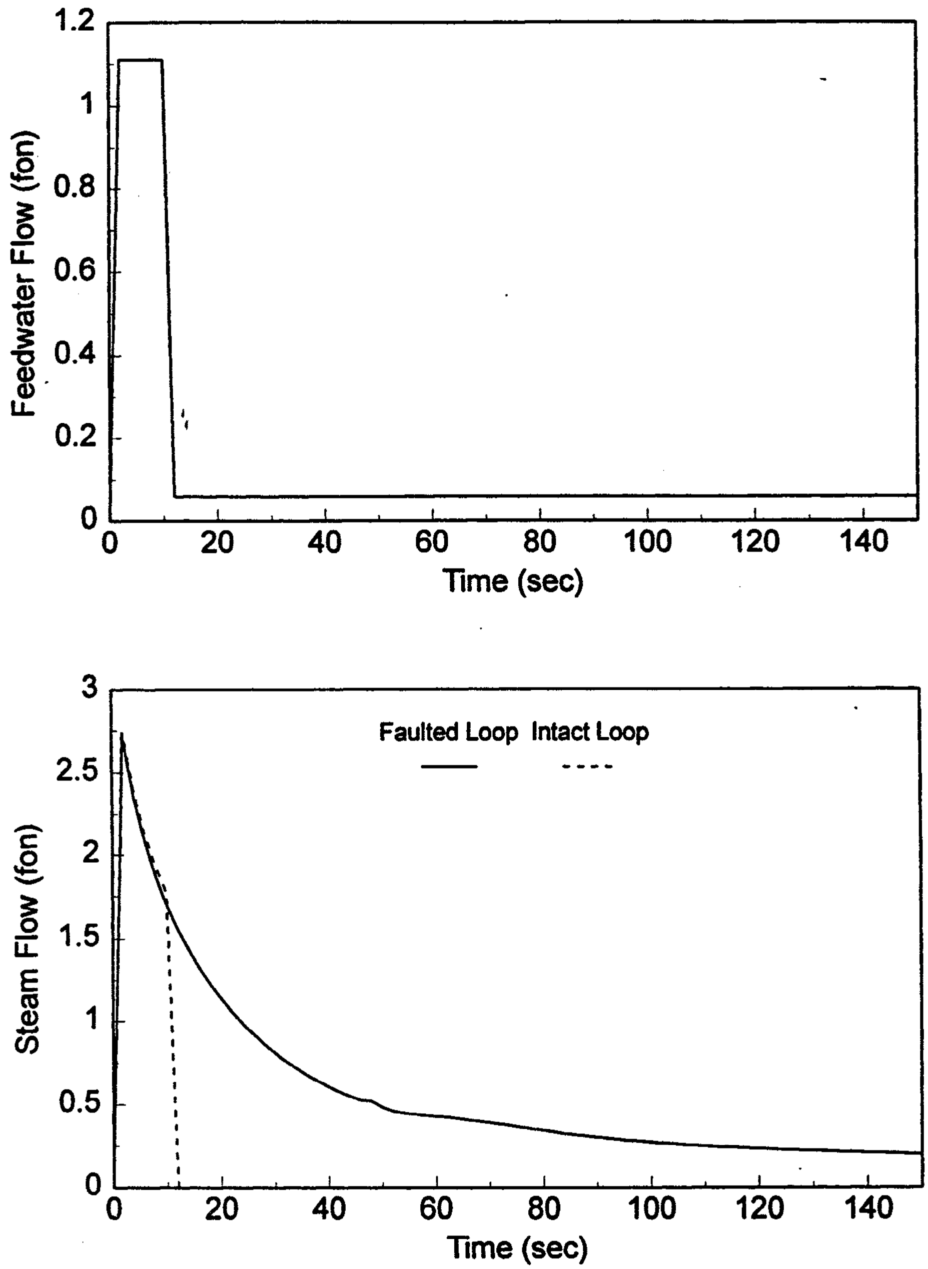
FIGURE 15.1-23:
CORE INLET TEMPERATURE AND
CORE AVERAGE TEMPERATURE
SLB WITHOUT OFFSITE POWER



Rev. 7

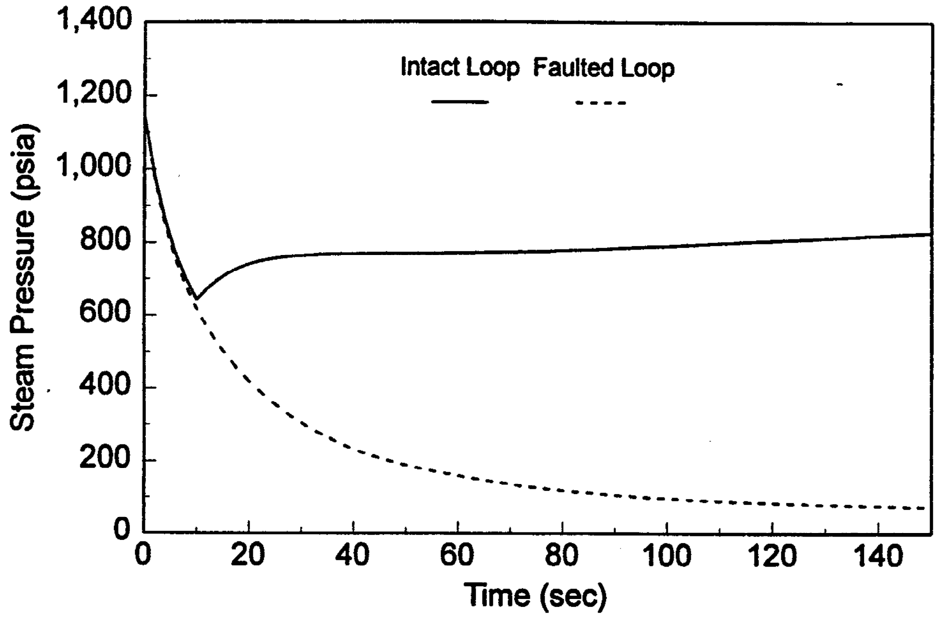
WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

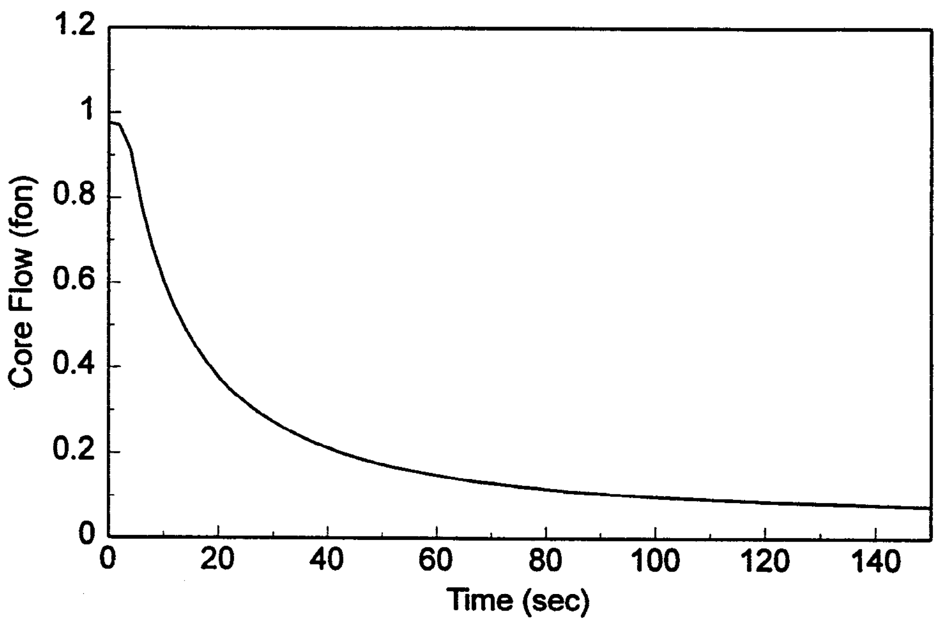
FIGURE 15.1-24:
TOTAL CORE REACTIVITY AND CORE
BORON CONCENTRATION
SLB WITHOUT OFFSITE POWER



Rev. 7

WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.1-25: NORMALIZED FEEDWATER FLOW AND BREAK FLOW SLB WITHOUT OFFSITE POWER





Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.1-26:
STEAMLINE PRESSURE AND CORE
INLET FLOW
SLB WITHOUT OFFSITE POWER

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 the following events, which have been identified as more limiting than the others:

a. Steam pressure regulator malfunction or failure that results in decreasing steam flow

b. Loss of external electrical load

c. Turbine trip

d. Inadvertent closure of main steam isolation valves

e. Loss of condenser vacuum and other events resulting in turbine trip

f. Loss of nonemergency ac power to the station auxiliaries

g. Loss of normal feedwater flow

h. Feedwater system pipe break

The above items are considered to be ANS Condition II events, with the exception of a feedwater system pipe break, which is considered to be an ANS Condition IV event. Section 15.0.1 contains a discussion of ANS classifications and applicable acceptance criteria.

All of the accidents in this section have been analyzed. It has been determined that the most severe radiological consequences will result from the loss of ac power accident of Section 15.2.6. Therefore, the radiological consequences are only reported for that limiting case.

15.2.1 STEAM PRESSURE REGULATOR MALFUNCTION OR FAILURE THAT RESULTS IN DECREASING STEAM FLOW

There are no steam pressure regulators in the Wolf Creek unit whose failure or malfunction could cause a steam flow transient. Therefore, this event is not applicable to the Wolf Creek Generating Station.

15.2.2 LOSS OF EXTERNAL ELECTRICAL LOAD

15.2.2.1 Identification of Causes and Accident Description

A major load loss on the plant can result from loss of external electrical load 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 emergency 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. The automatic turbine bypass system would accommodate the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if the turbine bypass system and pressurizer pressure control system are functioning properly. If the condenser were not available, the excess steam generation would be relieved to the atmosphere. Additionally, main feedwater flow would be lost if the condenser were not available. For this situation, feedwater flow would be maintained by the auxiliary feedwater system.

For a loss of external electrical load without subsequent turbine trip, no direct reactor trip signal would be generated. The plant would 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.

Following the loss of load, protection would be provided by high pressurizer pressure, high pressurizer water level, and overtemperature DT trips should a safety limit be approached. Voltage and frequency relays associated with the reactor coolant pump provide no additional safety function for this event. Following a complete loss of external load, the maximum turbine overspeed would be approximately 8 to 9 percent, resulting in an overfrequency of less than 6 Hz. This resulting overfrequency is not expected to damage the voltage and frequency sensors in any way. Any degradation in their performance could be ascertained at that time. Any increased frequency to the reactor coolant pump motors will result in slightly increased flow rate and subsequent additional margin to safety limits. For postulated loss of load and subsequent turbine-generator overspeed, any overfrequency condition is not seen by other safety-related pump motors, reactor protection system equipment, or other safety-related loads. Safety-related loads are supplied from offsite power or, alternatively, from emergency diesels. Reactor protection system equipment is supplied from the 120-Volt ac instrument power supply system which, in turn, is supplied from the inverters; the inverters are supplied from a dc bus energized from batteries or by a rectified ac voltage from safety-related busses.

In the event that the turbine bypass 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 overtemperature DT signal. The steam generator shell side pressure and reactor coolant temperature will increase rapidly. The pressurizer safety valves and steam generator safety valves are, however, sized to protect the RCS and steam generator against overpressure for all load losses without assuming the operation of the turbine bypass system, pressurizer spray, pressurizer power-operated relief valves, automatic rod cluster control assembly (RCCA) control, or direct reactor trip on turbine trip.

The steam generator safety valve capacity is sized to remove the steam flow at the engineered safety features rating (see section 5.2.2.1) from the steam generator, without exceeding 110 percent of the steam system design pressure. The pressurizer safety valve capacity is sized to accommodate a complete loss of heat sink, with the plant initially operating at the maximum calculated turbine load along with operation of the steam generator safety valves. The pressurizer safety valves are then 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 Section 15.0.1 for a discussion of Condition II events.

A loss of external load event results in a nuclear steam supply system transient that is bounded by the turbine trip event analyzed in Section 15.2.3. Therefore, a detailed transient analysis is not presented for the loss of external load event.

The primary side transient is caused by a decrease in 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 feedwater flow not be reduced, a larger heat sink would 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 (see section 10.2.2.2). The analysis performed for the turbine trip event assumes an instantaneous loss of steam flow, 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

Refer to Section 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 bound those expected for the loss of external load, as discussed in Section 15.2.2.1.

Plant systems and equipment which may be required to function to mitigate the effects of a complete loss of load are discussed in Section 15.0.8 and listed in Table 15.0-6.

The reactor protection system may be required to function to terminate core heat input and to 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 pressures below allowable limits. No single active failure will prevent operation of any system required to function. Normal reactor control systems and engineered safety systems are not required to function. The auxiliary feedwater system may, however, be automatically actuated following a loss of main feedwater; this will further mitigate the effects of the transient.

15.2.2.3 Conclusions

Based on results obtained for the turbine trip event (see Section 15.2.3) and considerations described in Section 15.2.2.1, the applicable acceptance criteria for a loss of external load event are met.

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 (Direct reactor trip on turbine trip is blocked below 50% power by the P-9 interlock) from a signal derived from the turbine stop emergency trip fluid pressure and turbine stop valves. The turbine stop valves close rapidly (see section 10.2.2.2) on loss of trip fluid pressure actuated by one of a number of possible turbine trip signals. Turbine trip initiation signals include:

a. Generator trip
b. Low condenser vacuum
c. Loss of lubricating oil
d. Turbine thrust bearing failure
e. Turbine overspeed
f. 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 initiate turbine bypass, and, if above 50-percent power, a reactor trip. The loss of steam flow results in an almost immediate rise in secondary system temperature and pressure, with a resultant primary system transient as described in Section 15.2.2.1 for the loss of external load event. A slightly more severe transient may occur for the turbine trip event, due to the more rapid loss of steam flow caused by the more rapid valve closure (see section 10.2.2.2), hence a more rapid loss of primary-to-secondary heat transfer.

The automatic turbine bypass system would accommodate up to 40 percent of rated steam flow. Reactor coolant temperatures and pressure do not increase significantly if the turbine bypass system and pressurizer pressure control system are functioning properly. If the condenser were not available, the excess steam generation would be relieved to the atmosphere, and main feedwater flow would be lost. For this situation, feedwater flow would be maintained by the auxiliary feedwater system to ensure adequate residual and decay heat removal capability. Should the turbine bypass system fail to operate, the steam generator safety valves may lift to provide pressure control. See Section 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 Section 15.0.1 for a discussion of Condition II events.

A turbine trip is more limiting than loss of external load, loss of condenser vacuum, and other events which result in a turbine trip as a result of the potential for a more rapid loss of steam flow during the turbine trip. As such, this event has been analyzed in detail. Results and discussion of the analysis are presented in Section 15.2.3.2.

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 102 percent of 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. This assumption delays reactor trip until conditions in the RCS result in a trip due to other signals. Thus, the analysis assumes a worst transient. In addition, no credit is taken for the turbine bypass system. 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 RETRAN-02 (Ref 5). RETRAN-02 has been found acceptable by the NRC for use as a licensing basis safety analysis code. RETRAN-02 is a thermal-hydraulic systems analysis code employing a one-dimensional, homogeneous equilibrium mixture thermal-hydraulic model for the RCS, a point kinetics model for the reactor core, special component and auxiliary models (ex., pumps, temperature transport, non-equilibrium pressurizer) and control system models. The code computes pertinent plant transient information including core power level, RCS pressure and temperature. VIPRE-01 (Ref 6) is used to evaluate the core thermal limits to determine DNBR. RETRAN-02 generated state points are used as VIPRE-01 boundary conditions to perform a Statistical Core Design (SCD) DNB analysis.

The turbine trip event is analyzed for RCS overpressurization assuming nominal initial conditions including allowances for measurement errors. DNBR is evaluated using Westinghouse RTDP methodology (Ref. 7) which assumes nominal initial conditions.

The major assumptions used in the analysis are summarized below:

a. Initial operating conditions (Overpressure)
   Initial reactor power and RCS temperatures are assumed at their maximum values consistent with steady state full power operation, including allowances for calibration and instrument errors. The initial RCS pressure is assumed at a minimum value consistent with steady state full power operation, including allowances for calibration and instrument errors. This is to maximize the heat up time prior to reactor trip and therefore, maximizing the peak RCS pressure and peak pressurizer level during the transient.

b. Initial operating conditions (DNB)
   The initial pressure, reactor power and RCS temperatures are assumed at their nominal values consistent with steady state full power operation. Allowances for calibration and instrument errors are treated statistically by the DNBR evaluation code.

c. Reactivity coefficients (two cases are analyzed):

1. Minimum reactivity feedback

   A most positive moderator temperature coefficient and a least negative Doppler-only power coefficient are assumed.

2. Maximum reactivity feedback

   A conservatively large negative moderator temperature coefficient and a most negative Doppler-only power coefficient are assumed.

Cases for DNB and system overpressurization are analyzed for both the minimum and maximum reactivity feedback conditions.

d. 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.

e. Steam release

   No credit is taken for the operation of the turbine bypass system or steam generator atmospheric 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.

f. Pressurizer spray and power-operated relief valves

1. DNB

   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.

2. Overpressure

   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.

g. 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 auxiliary feedwater flow, since a stabilized plant condition will be reached before auxiliary feedwater initiation is normally assumed to occur; however, the auxiliary feedwater pumps would be expected to start on a trip of the main feedwater pumps. The auxiliary feedwater flow would remove core decay heat following plant stabilization.

h. 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, overtemperature ΔT, and low-low steam generator water level.

Plant characteristics and initial conditions are further discussed in Section 15.0.3.

Major plant systems and equipment available for mitigation of transient and accident conditions which may be required to function to mitigate the effects of a turbine trip event are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-11.

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. Normal reactor control system and engineered safety systems are not required to function.

Results

The transient responses for a turbine trip from 102 percent of full power operation are shown for four cases in Figures 15.2-1 through 15.2-8. Two cases are presented without pressure control (i.e., PORVs, pressurizer spray) to ensure 110% of the design pressure (2750 psia) is not exceeded and two cases using nominal initial plant conditions and pressure control are analyzed to provide boundary conditions for DNB analysis. Each case is analyzed assuming minimum and maximum reactivity feedback to ensure the worst case is analyzed. 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 total loss of steam load overpressure evaluation with minimum reactivity feedback, assuming no credit is taken for the steam bypass. The reactor is tripped on the high pressurizer pressure signal. The neutron flux increases slightly above 102 percent of full power until the reactor is tripped, due to the positive moderator temperature coefficient. In this case, the pressurizer safety valves are actuated, and maintain RCS pressure below 110 percent of the design value.

Figures 15.2-3 and 15.2-4 show the responses for the total loss of steam load overpressure evaluation with maximum reactivity feedback. All other plant parameters are assumed to be the same as in the previous case. Again, the reactor is tripped on high pressurizer pressure and the pressurizer safety valves are actuated to limit primary pressure.

The DNB evaluation for the turbine trip event was analyzed assuming full pressure control and nominal initial conditions. The errors in initial conditions are treated statistically in the VIPRE model (Ref 5). Figures 15.2-5 and 15.2-6 show the transient responses for the total loss of steam load with minimum reactivity feedback for the DNB evaluation. No credit is taken for the steam bypass. Due to the positive moderator temperature coefficient, the power increases to approximately 110 percent of nominal before the reactor is tripped by the high pressurizer pressure trip channel. Steam is released through the pressurizer safety valves for this case and the maximum RCS pressure is maintained below 110 percent of the design pressure. The steam generator safety valves limit the secondary steam conditions to saturation at the safety valve setpoint. Minimum DNBR remains above the safety analysis limit.

Figures 15.2-7 and 15.2-8 show the response for the DNB evaluation with maximum reactivity feedback. The reactor is tripped by the high pressurizer pressure trip channel but the pressurizer safety valves are not actuated for this case. Reactor power remains essentially constant until the trip occurs. The steam generator safety valves limit the secondary steam conditions to saturation at the safety valve setpoint. Minimum DNBR remains above the safety analysis limit.

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 Conclusions

Results of the analyses 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 analyses show that the DNBR will not decrease below the safety analysis limit at any time during the transient. Thus, the DNB design basis, as described in Section 4.4, is met.

15.2.4 INADVERTENT CLOSURE OF MAIN STEAM ISOLATION VALVES

Inadvertent closure of the main steam isolation valves would result in a turbine trip with no credit taken for the turbine bypass system. Turbine trips are discussed in Section 15.2.3.

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 Section 15.2.3. A loss of condenser vacuum would preclude the use of steam dump to the condenser; however, since steam dump is assumed to be unavailable 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 Section 15.2.3 apply to the loss of the condenser vacuum. In addition, analyses for the other possible causes of a turbine trip, as listed in Section 15.2.3.1, are covered by Section 15.2.3. Possible overfrequency effects due to a turbine overspeed condition are discussed in Section 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 STATION AUXILIARIES (BLACKOUT)

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 plant 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 station, or by a loss of the onsite ac distribution system.

The DNB transient is more severe than the turbine trip event analyzed in Section 15.2.3 because, for this case, the decrease in heat removal by the secondary system 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: 1) due to 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 the 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:

a. Plant vital instruments are supplied from emergency dc power sources.

b. As the steam system pressure rises following the trip, the steam generator atmospheric relief valves are automatically opened to the atmosphere. The condenser is assumed to be unavailable for turbine bypass. If the steam flow path through the atmospheric 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.

c. As the no-load temperature is approached, the steam generator atmospheric relief valves (or safety valves, if the atmospheric relief relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot shutdown condition.

d. The standby diesel generators, started on loss of voltage on the plant emergency busses, begin to supply plant vital loads.

The auxiliary feedwater system is started automatically, as follows:

a. Two motor-driven auxiliary feedwater pumps are started on any of the following:

   1. Low-low level in any steam generator

   2. Any safety injection signal

   3. Loss of offsite power

   4. Manual actuation

b. The turbine-driven auxiliary feedwater pump is started on any of the following:

   1. Low-low level in any two steam generators

   2. Loss of offsite power

   3. Manual actuation

The motor-driven auxiliary feedwater pumps are supplied power by the diesels, and the turbine-driven pump utilizes steam from the secondary system. The turbine exhausts the secondary steam to the atmosphere. The auxiliary feedwater pumps take suction from the condensate storage tank (or the essential service water system, if the condensate storage tank is unavailable) 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 nonemergency ac power to the station auxiliaries is classified as an ANS Condition II event, fault of moderate frequency. This event is more limiting than the turbine-trip-initiated decrease in secondary heat removal without loss of ac power, which was analyzed in Section 15.2.3. However, a loss of ac power to the station auxiliaries, as postulated above, could also result in a loss of normal feedwater if the condensate pumps lose their 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 auxiliary feedwater in the secondary system. The DNB transient for this event is bounded by the complete loss of flow event. Thus the 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.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-12.

15.2.6.2 Analysis of Effects and Consequences

Method of Analysis

The loss of non-emergency AC transient is analyzed by employing the detailed digital computer program RETRAN-02 (Ref 5). RETRAN-02 has been found acceptable by the NRC for use as a licensing basis safety analysis code. RETRAN-02 is a thermal-hydraulic systems analysis code employing a one-dimensional, homogeneous equilibrium mixture thermal-hydraulic model for the RCS, a point kinetics model for the reactor core, special component and auxiliary models (ex., pumps, temperature transport, non-equilibrium pressurizer) and control system models. The code computes pertinent plant transient information including core power level, RCS pressure and temperature. VIPRE-01 (Ref 6) is used to evaluate the core thermal limits to determine DNBR. RETRAN-02 generated state points are used as VIPRE-01 boundary conditions to perform a Statistical Core Design (SCD) DNB analysis.

The assumptions used in the analysis are as follows:

a. The plant is initially operating at 102 percent of the rated core power.
b. A conservative core residual heat generation based upon long-term operation at the initial power level preceding the trip.
c. A heat transfer coefficient (see Ref. 2) in the steam generator associated with RCS natural circulation.
d. The auxiliary feedwater system is actuated by the low-low steam generator water level signal, assumed to occur at 0% of narrow range span. The analysis assumes a total of 700 gpm is evenly delivered to 4 steam generators. This is a conservative minimum value for the AFW flow and bounds either a single failure of the turbine-driven AFW pump or one motor-driven AFW pump (e.g., diesel generator failure). A delay of 392 seconds is assumed, including 60 seconds for diesel generator and pump start, before delivering relatively cold auxiliary feedwater to the steam generators. The delay allows for filling the associated feedwater piping with no credit for the hotter main feedwater being purged through the steam generators.
e. Reactor trip occurs when the steam generator level reaches 0.0% of narrow range span.
f. The pressurizer spray valves and PORVS are assumed operable. This assumption maximizes the transient pressurizer water volume.
g. The steam dump system and steam generator atmospheric relief valves are assumed to be unavailable. Secondary system steam relief is assumed through the main steam safety valves (MSSVs) only.

Plant characteristics and initial conditions are further discussed in Section 15.0.3.

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 calculated sequence of events for this event is listed in Table 15.2-1.

The RETRAN code results demonstrate that the natural circulation flow available is sufficient to provide adequate core decay heat removal following reactor trip and reactor coolant pump coastdown. Table 15.2-4 lists the natural circulation flow rates (in terms of percentage of full power) at no-load conditions and each steam generator removing heat.

The analysis begins with a loss of normal feedwater resulting from a loss of non-emergency AC power. This causes the steam generator level to fall, the RCS temperature and pressurizer pressure to rise. When the reactor trips on low-low level, the reactor power drops toward decay heat levels, and the RCPs subsequently trip. Due to the reactor power reduction following reactor trip, the RCS temperature falls, as do the pressurizer pressure and level.

Following reactor trip, the focus of the analysis becomes the falling steam generator inventory. While the inventory is high, there will be sufficient heat transfer to remove the decay heat. However, the timely arrival of auxiliary feedwater at 433 seconds is sufficient to reverse the steam generator inventory reduction.

Results of the analysis show that, for the loss of non-emergency AC to plant auxiliaries, all safety criteria are met. Since the DNBR remains above the design limit, the core is not adversely affected. Auxiliary feedwater capacity is sufficient to prevent water relief through the pressruizer relief and safety valves; this assures that the RCS is not overpressurized. The calculated peak RCS pressure is approximately 2680 psia, which is less than 110% of the design pressure (2750 psia).

15.2.6.3 Radiological Consequences

15.2.6.3.1 Method of Analysis

15.2.6.3.1.1 Physical Model

The dose calculation for loss of ac power is based on the sequence of events described in Table 15.2-1. It is assumed that heat removal from the nuclear steam supply system is achieved by venting the steam for 8 hours.

The reactor coolant is assumed to be contaminated by radioactive fission products introduced through fuel cladding defects. The secondary system is contaminated by the inleakage of reactor coolant through postulated steam generator tube leaks.

The radioactivity in the vented steam is dispersed in the atmosphere without any reduction due to plateout, fallout, filtering, etc.

15.2.6.3.1.2 Assumptions and Conditions

The major assumptions and parameters assumed in the analysis are found in Tables 15.2-2 and 15.A-1. The assumptions used to determine the activity released are as follows:

a. The reactor coolant activity assumed is the Technical Specification limit of 1.0 μCi/gm I-131 dose equivalent.

b. The initial steam generator activity assumed is the Technical Specification limit of 0.1 μCi/gm I-131 dose equivalent.

c. A 1-gpm steam generator primary-to-secondary leakage is assumed for the duration of steam venting.

d. For noble gases, the activity released is taken to be the activity introduced by reactor coolant inleakage without holdup in the steam system.

e. The iodine activity present in the primary-to-secondary leakage is assumed to be homogeneously mixed with the iodine activity initially present in the steam generators. The iodine partition factor provided in Table 15.2-2 is utilized to determine the iodine activity released via steam venting from the steam generators.

f. The atmospheric dispersion factors are given in Table 15A-2.

15.2.6.3.1.3 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 using the onsite meteorological measurement programs described in Section 2.3.

c. The thyroid inhalation and total-body immersion doses to a receptor at the exclusion area boundary or outer boundary of the low population zone were analyzed using the models described in Appendix 15A.

15.2.6.3.1.4 Identification of Leakage Pathways and Resultant Leakage Activities

Normal activity paths from the secondary system, such as the condenser air removal system and steam generator blowdown, cease during station blackout. The steam is released to the atmosphere through the:

a. Atmospheric relief valves

b. Main steam safety valves

Since all these paths are taken as direct to the atmosphere without any form of decontamination, they are all radiologically equivalent and need not be distinguished.

15.2.6.3.2 Identification of Uncertainties in, and Conservative Aspects of, the Analysis

The principal uncertainties in the dose calculation arise from the uncertainties in the accident circumstances, particularly the extent of steam contamination, the weather at the time, and delay before preferred ac power is restored. Each of these uncertainties is handled by making very conservative or worst-case assumptions.

a. Reactor coolant activities are based on the Technical Specification limit, which is significantly higher than the activities associated with normal operating conditions, based on 0.12-percent failed fuel.

b. A 1-gpm steam generator primary-to-secondary leakage is assumed, which is significantly greater than that anticipated during normal operation.

c. 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 evaluated radiological consequences, based on the meteorological conditions assumed, will be conservative.

15.2.6.3.3 Conclusions

15.2.6.3.3.1 Filter Loadings

No filter serves to limit the release of radioactivity in this accident. There is no significant activity buildup on any filters as a consequence of loss of ac power.

15.2.6.3.3.2 Doses to Receptor at Exclusion Area Boundary and Low Population Zone Outer Boundary

The maximum doses to an individual who spends the first 2 hours after loss of ac power at the exclusion area boundary, and the maximum doses for a long-term exposure (8 hours or longer) at the outer boundary of the low-population zone, are given in Table 15.2-3. These doses are within a small fraction of the guideline values of 10 CFR 100.

15.2.6.4 Conclusions

Results of the analysis show that, for the loss of non-emergency ac power to plant auxiliaries event, all safety criteria are met. Since the DNBR remains above the design limit, the core is not adversely affected.

Analysis of the natural circulation capability of the RCS demonstrates that sufficient long term heat removal capability exists following reactor coolant pump coastdown to prevent fuel or clad damage.

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 the 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.
The reactor trip on low-low water level in any steam generator provides the necessary protection against a loss of normal feedwater.

The following occur upon loss of normal feedwater (assuming main feedwater pump failures or valve malfunctions):

a. As the steam system pressure rises following the trip, the steam generator atmospheric relief valves are automatically opened to the atmosphere. The condenser is assumed to be unavailable for turbine bypass. If the steam flow path through the atmospheric 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.

b. As the no-load temperature is approached, the steam generator atmospheric relief valves (or safety valves, if the atmospheric 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 Section 15.0.1 for a discussion of Condition II events.

The auxiliary feedwater system is started automatically, as discussed in Section 15.2.6.1. The turbine-driven auxiliary feedwater pump utilizes steam from the secondary system and exhausts to the atmosphere. The motor-driven auxiliary feedwater pumps are supplied power by the diesel generators. The auxiliary feedwater pumps take suction directly from the condensate storage tank (or essential service water system, if the condensate storage tank is unavailable) 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 auxiliary feedwater system 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 of the loss of normal feedwater event employing the digital computer program RETRAN-02 (Ref 5) has been performed. RETRAN-02 has been found acceptable by the NRC for use as a licensing basis safety analysis code. RETRAN-02 is a thermal-hydraulic systems analysis code employing a one-dimensional, homogeneous equilibrium mixture thermal-hydraulic model for the RCS, a point kinetics model for the reactor core, special component and auxiliary models (ex., pumps, temperature transport, non-equilibrium pressurizer) and control system models. The code computes pertinent plant transient information including core power level, RCS pressure and temperature. VIPRE-01 (Ref. 6) is used to evaluate the core thermal limits to determine DNBR. RETRAn-02 generated state points are used as VIPRE-01 boundary conditions to perform a Statistical Core Design (SCD) DNB analysis.

The assumptions used in the analysis are as follows:

a. The plant is initially operating at 102 percent of the engineered safety features design rating.

b. A conservative core residual heat generation, based upon long-term operation at the initial power level preceding the trip.

c. A heat transfer coefficient (see Ref. 2) in the steam generator associated with RCS natural circulation.

d. Reactor trip occurs on steam generator low-low level at 0.0% of narrow range span.

e. The auxiliary feedwater system is actuated by the low-low steam generator water level signal, assumed to occur at 0% of narrow range span. The analysis assumes a total of 700 gpm is evenly delivered to 4 steam generators. This is a conservative minimum value for the AFW flow and bounds either a single failure of the turbine-driven AFW pump or one motor-driven AFW pump (e.g., diesel generator failure). A delay of 392 seconds is assumed, including 60 seconds for diesel generator and pump start, before delivering relatively cold auxiliary feedwater to the steam generators. The delay allows for filling the associated feedwater piping with no credit for the hotter main feedwater being purged through the steam generators.

f. The auxiliary feedwater system is actuated by the low-low steam generator water level signal.

g. Secondary system steam relief is achieved through the steam generator safety valves.

h. The pressurizer sprays and PORVS are assumed operable. This maximizes the peak transient pressurizer water volume.

The loss of normal feedwater analysis is performed to demonstrate the adequacy of the reactor protection and engineered safeguards systems (e.g., the auxiliary feedwater system) 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-low steam generator level trip. The reactor coolant pumps may be manually tripped at some later time to reduce heat addition to the RCS.

An additional assumption made for the loss of normal feedwater evaluation is that only the pressurizer safety valves are assumed to function normally. Operation of the valves, if required, maintains peak RCS pressure below 110% of design pressure throughout the transient.

Plant characteristics and initial conditions are further discussed in Section 15.0.3.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-13. 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 auxiliary feedwater system is required to deliver a minimum auxiliary feedwater flow rate. No single active failure will prevent operation of any system required to function. A discussion of anticipated transients without scram (ATWS) considerations is presented in Section 15.8.

Results

Figures 15.2-11 through 15.2-12 show the significant plant parameters following a loss of normal feedwater.

The analysis begins with a loss of normal feedwater. This causes the steam generator level to fall, the RCS temperature to increase gradually and the pressurizer pressue and liquid level to rise. Due to the positive moderator temperature coefficent assumed, the reactor power also begins to rise. When the reactor trips on low-low steam generator level, the reactor power drops toward decay heat levels. Due to the reactor power reduction, the RCS temperature falls, as do the pressurizer pressure and level.

Following reactor trip, the focus of the transient analysis becomes the falling steam generator inventory. While the inventory is high, there will be sufficient heat transfer to remove the core decay heat. However, the timely arrival of auxiliary feedwater at 433 seconds is sufficient to reverse the steam generator inventory reduction. The capacity of the auxiliary feedwater pumps is such that the water level in the steam generators is sufficient to dissipate core residual heat without water relief from the RCS safety valves. Figure 15.2-12 shows that at no time is there water relief from the pressurizer.

The results of the analysis show that, for the loss of normal feedwater event, all safety analysis criteria are met. Since the DNBR remains above the design limit, the core is not adversely affected. The AFW capacity is also sufficient to assure that the RCS does not overpressurize.

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 approaches a stabilized condition following reactor trip and auxiliary feedwater initiation. Plant procedures may be followed to further cool down the plant.

15.2.7.3 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 auxiliary feedwater capacity is such that reactor coolant water is not relieved from the pressurizer relief or safety valves.

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 feedwater line between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. Further, a break in this location could preclude the subsequent addition of auxiliary feedwater to the affected steam generator. (A break upstream of the feedwater line check valve would affect the NSSS only as a loss of feedwater. This case is covered by the evaluation in Sections 15.2.6 and 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 an RCS cooldown (by excessive energy discharge through the break) or an RCS heatup. Potential RCS cooldown resulting from a secondary pipe rupture is evaluated in Section 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:

a. 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.

b. Fluid in the steam generator may be discharged through the break, and would then not be available for decay heat removal after trip.

c. The break may be large enough to prevent the addition of any main feedwater after trip.

An auxiliary feedwater system functions to ensure the availability of adequate feedwater 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.

A major feedwater line rupture is classified as an ANS Condition IV event. See Section 15.0.1 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. Sensitivity studies presented in Reference 3 illustrate many of the limiting assumptions for the feedwater line rupture. In addition, the major assumptions pertinent to this analysis are defined below.

The main feedwater control system is assumed to fail due to an adverse environment. The water levels in all steam generators are assumed to decrease equally until the low-low steam generator level reactor trip setpoint is reached. After reactor trip, a double-ended rupture of the largest feedwater line is assumed. These assumptions conservatively bound the most limiting feedwater line rupture that can occur. Analyses have been performed at full power, with and without loss of offsite power, and with no credit taken for the pressurizer power-operated relief valves. For the case without offsite power available, the power is assumed to be lost at the time of reactor trip. This is more conservative than the case where power is lost at the initiation of the event. These cases are analyzed below.

The following provides the protection for a main feedwater line rupture:

a. A reactor trip on any of the following conditions:

    1. High pressurizer pressure
    2. Overtemperature ΔT

3. Low-low steam generator water level in any steam generator
4. Safety injection signals from any of the following: 1) two-out-of-three low steam line pressure in any one loop or 2) two-out-of-three high containment pressure (hi-1)

Refer to Chapter 7.0 for a description of the actuation system.

b. The auxiliary feedwater system provides an assured source of feedwater to the steam generators for decay heat removal. Refer to Section 10.4.9 for a description of the auxiliary feedwater system.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-14.

15.2.8.2 Analysis of Effects and Consequences

Method of Analysis

A detailed analysis of the feedwater line break event employing the digital computer program RETRAN-02 (Ref 5) has been performed. RETRAN-02 has been found acceptable by the NRC for use as a licensing basis safety analysis code. RETRAN-02 is a thermal-hydraulic systems analysis code employing a one-dimensional, homogeneous equilibrium mixture thermal-hydraulic model for the RCS, a point kinetics model for the reactor core, special component and auxiliary models (ex., pumps, temperature transport, non-equilibrium pressurizer) and control system models. The code computes pertinent plant transient information including core power level, RCS pressure and temperature. VIPRE-01 (Ref. 6) is used to evaluate the core thermal limits to determine DNBR. RETRAN-02 generated state points are used as VIPRE-01 boundary conditions to perform a Statistical Core Design (SCD) DNB analysis.

Major assumptions used in the analysis are as follows:

a. The initial Plant Power is assumed to be 102% of rated core power. Plant characteristics and initial conditions are further discussed in Section 15.0.

b. No credit is taken for the pressurizer power-operated relief valves or pressurizer spray.

c. Initial pressurizer level is at the nominal programmed value plus 5 percent (error); initial steam generator water level is at the nominal value.

d. No credit is taken for the high pressurizer pressure reactor trip.

e. Main feedwater is assumed to be lost to all steam generators at event initiation due to a malfunction in the feedwater control system. The feedline break, and subsequent reverse blowdown of the faulted steam generator, is conservatively assumed to occur when the steam generator inventory reaches 0% narrow range span.

f. The worst possible break area is assumed. This maximizes the blowdown discharge rate following the time of trip, which maximizes the resultant heatup of the reactor coolant.

g. A bounding feedwater line break discharge quality is assumed.

h. Reactor trip is assumed to be initiated when the low-low steam generator level reaches 0 percent of narrow range span in the ruptured steam generator.

i. The auxiliary feedwater system is actuated by the low-low steam generator water level signal. The auxiliary feedwater system is assumed to supply a total of 563 gpm to three unaffected steam generators, including allowance for possible 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 auxiliary feedwater pumps. An additional 314 seconds was assumed before the feedwater lines were purged and the relatively cold (120 F) auxiliary feedwater entered the unaffected steam generators.

j. No credit is taken for heat energy deposited in RCS metal during the RCS heatup.

k. No credit is taken for charging or letdown.

l. Steam generator heat transfer area is assumed to decrease as the shell side liquid inventory decreases.

m. Conservative core residual heat generation is assumed based upon long-term operation at the initial power level preceding the trip.

n. No credit is taken for the following potential protection logic signals to mitigate the consequences of the accident:

1. High pressurizer pressure
2. Overtemperature ΔT
3. High pressurizer level
4. High containment pressure

Receipt of a low-low steam generator water level signal in at least one steam generator starts the motor-driven auxiliary feedwater pumps, which in turn initiate auxiliary feedwater flow to the steam generators. The turbine-driven auxiliary feedwater pump is initiated if the low-low steam generator water level signal is reached in at least two steam generators. Similarly, receipt of a low steam line pressure signal in at least one steam line initiates a steam line isolation signal which closes all main steam line isolation valves.

Emergency operating procedures following a feedwater system pipe rupture require the following actions to be taken by the reactor operator:

a. Isolate feedwater flow spilling from the ruptured feedwater line and align the system so that the level in the intact steam generators is recovered.

b. High head safety injection should be terminated in accordance with the emergency operating procedures.

Subsequent to terminating high head safety injection, plant operating procedures are followed in cooling the plant to a safe shutdown condition.

Plant characteristics and initial conditions are further discussed in Section 15.0.3.

No reactor control systems are assumed to function. 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 auxiliary feedwater system and the safety injection system. For the auxiliary feedwater system, the worst case configuration has been used, i.e., only three intact steam generators receive auxiliary feedwater following the break. A discharge flow control device, located on the auxiliary feedwater line to each steam generator, is assumed to regulate the flow from the motor-driven auxiliary feedwater pump to the affected steam generator. This ensures that a minimum flow of 250 gpm, from both the motor-driven and turbine-driven auxiliary feedwater pumps, is delivered to the intact steam generator feed by the same motor-driven auxiliary feedwater pump that is feeding the affected steam generator. The second motor-driven auxiliary feedwater pump has been assumed to fail. The turbine-driven auxiliary feedwater pump delivers 470 gpm equally split to the three intact steam generators. This assumption is conservative because it maximizes the purge time in the feedwater lines before auxiliary feedwater enters the unaffected steam generators. Thus, a total flow of 563 gpm is delivered to the intact steam generators.

For the case without offsite power, there is a flow coastdown until flow in the loops reaches the natural circulation value. The natural circulation capability of the RCS has been shown (in Section 15.2.6) to be sufficient to remove core decay heat following reactor trip, for the loss of ac power transient. Pump coastdown characteristics are demonstrated in Sections 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. The auxiliary feedwater system is described in Section 10.4.9.

Results

Calculated plant parameters following a major feedwater line rupture are shown in Figures 15.2-13 through 15.2-22. Results for the case with offsite power available are presented in Figures 15.2-13 through 15.2-17. Results for the case where offsite power is lost are presented in Figures 15.2-18 through 15.2-22. 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 15.2-15 (with offsite power available) and Figures 15.2-19 and 15.2-20 (without offsite power) show that pressures in the RCS and main steam system remain below 110 percent of the respective design pressures. Pressurizer pressure decreases after reactor trip on low-low steam generator level. Pressurizer pressure decreases due to the loss of heat input, until coolant expansion occurs due to reduced heat transfer capability in the steam generators; the pressurizer safety valves open to maintain primary coolant system pressure at an acceptable value.

Figures 15.2-13 and 15.2-18 show that following reactor trip the plant remains subcritical.

Figures 15.2-14 and 15.2-19 show that the pressurizer does not empty throughout the transient so that the core remains covered at all times and that no boiling occurs in the reactor coolant loops. Although not required for this Condition IV event, a DNB calculation was also performed and demonstrated that the DNBR remains above the design limit. Therefore, the core is not adversely affected.

The major difference between the two cases analyzed can be seen in the plots of hot and cold leg temperatures, Figure 15.2-16 (with offsite power available) and Figure 15.2-21 (without offsite power). It is apparent that for the initial transient (~150 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 until the auxiliary feedwater system is realigned. AFWS realignment is not assumed, however, in these cases. The pressurizer fills more rapidly for the case with power due to the increased coolant expansion resulting from the pump heat addition. As previously stated, the core remains covered with water for both cases.

15.2.8.3 Conclusions

Results of the analyses show that for the postulated feedwater line rupture, the assumed auxiliary feedwater system capacity is adequate to remove decay heat, to prevent overpressurizing the RCS, and to prevent uncovering the reactor core. Radioactivity doses from the postulated feedwater lines rupture are less than those previously presented for the postulated steam line break. All applicable acceptance criteria are therefore met.

15.2.9 REFERENCES

1. Cooper, L., Miselis, V. and Starek, R. M., "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, Revision 1, June, 1972 (also letter NS-CE-622, dated April 16, 1975, C. Eicheldinger (Westinghouse) to D.B. Vassallo (NRC), additional information on WCAP-7769, Revision 1).

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

3. Lang, G. E. and Cunningham, J. P., "Report on the Consequences of a Postulated Main Feedline Rupture," WCAP-9230 (Proprietary) and WCAP-9231 (Non-Proprietary), January 1978.

5. McFadden, J. H., et. al., "RETRAN-02 - A Program for Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," EPRI NP-1850-CCM-A, October 1984.

6. Stewart, C. W., et. al.,"VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores," Battelle, Pacific Northwest Laboratories, EPRI NP-2511-CCM-A, August 1989.

7. WCAP-11397-P-A, "Revised Thermal Design Procedure," A. J. Fiedland, et al., April 1989.

TABLE 15.2-1
TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (sec)
Turbine Trip		
1. Overpressure evaluation without pressurizer control (minimum feedback)	Turbine trip; Loss of main feedwater flow	0.025
	Initiation of steam release from pressurizer relief valve	NONE
	High pressurizer pressure reactor trip setpoint reached	5.45
	Initiation of steam release from steam generator safety valves	5.55
	Rods begin to drop	7.45
	Initiation of steam release from pressurizer safety valve	7.65
	Peak pressurizer pressure occurs	8.00
2. Overpressure evaluation without pressurizer control (maximum feedback)	Turbine trip; Loss of main feedwater flow	0.025
	Initiation of steam release from pressurizer relief valve	NONE
	High pressurizer pressure reactor trip setpoint reached	5.45
	Initiation of steam release from steam generator safety valves	5.55
	Rods begin to drop	7.45
	Initiation of steam release from pressurizer safety valve	7.60
	Peak pressurizer pressure occurs	7.90

TABLE 15.2-1 (Sheet 2)

Accident	Event	Time (sec)
Turbine Trip		
3. DNB evaluation with pressurizer control (minimum feedback)	Turbine trip; Loss of main feedwater flow	0.025
	Initiation of steam release from pressurizer relief valve	3.85
	Initiation of steam release from steam generator safety valves	7.74
	High pressurizer pressure reactor trip setpoint reached	9.04
	Rods begin to drop	11.04
	Initiation of steam release from pressurizer safety valve	11.90
	Peak system pressure occurs	12.05
4. DNB evaluation with pressurizer control (maximum feedback)	Turbine trip; Loss of main feedwater flow	0.025
	Initiation of steam release from pressurizer relief valve	3.85
	Initiation of steam release from steam generator safety valves	7.74
	High pressurizer pressure reactor trip setpoint reached	9.50
	Rods begin to drop	11.50
	Peak system pressure occurs	13.20

Accident	Event	Time (sec)
Loss of nonemergency ac power to the station auxiliaries	Main feedwater flow stops	0.1
	Low-low steam generator water level trip	39.3
	Rods begin to drop	41.3
	Reactor coolant pumps begin to coast down	43.3
	Auxiliary feedwater delivered from two motor-driven auxiliary feedwater pumps or one motor-driven auxiliary feedwater pump and the turbine driven pump	433.3
Loss of normal feedwater flow	Main feedwater flow stops	0.1
	Low-low steam generator water level trip	39.3
	Rods begin to drop	41.3
	Auxiliary feedwater delivered from two motor-driven auxiliary feedwater pumps or one motor-driven auxiliary feedwater pump and the turbine driven pump	433.3

TABLE 15.2-1 (Sheet 4)

Accident	Event	Time (sec)
Feedwater system pipe break		
1. With offsite power available	Feedwater control system fails	10.0
	Low-low steam generator level reactor trip set point reached in all steam generators	43.92
	Rods begin to drop and feedwater line rupture occurs	45.92
	Steam generator safety valve setpoint reached in intact steam generators	48.25
	Low steam line pressure setpoint reached in ruptured steam generator	75.02
	All main steam line isolation valves close	82.02
	Auxiliary feedwater to intact steam generators is initiated	417.85
	Pressurizer safety valve setpoint reached following feedwater line rupture	598.92
	Core decay heat decreases to auxiliary feedwater heat removal capacity	1,820

TABLE 15.2-1 (Sheet 5)

Accident	Event	Time (sec)
2. Without offsite power		
	Feedwater control system fails	10.0
	Low-low steam generator level reactor trip setpoint reached in all steam generators	43.92
	Rods begin to drop and feedwater line rupture occurs	45.92
	Steam generator safey valve setpoint reached in intact steam generators	48.47
	Low steam line pressure setpoint reached in ruptured steam generator	71.97
	All main steam line isolation valves close	78.97
	Auxiliary feedwater to intact steam generators is initiated	417.85
	Core decay heat decreases to auxiliary feedwater heat removal capacity	860.0

WOLF CREEK

TABLE 15.2-2

PARAMETERS USED IN EVALUATING RADIOLOGICAL CONSEQUENCES OF LOSS OF NONEMERGENCY AC POWER

I.	Source Data		
	a.	Steam generator type leakage, gpm	1
	b.	Reactor coolant initial iodine activity	1.0 μCi/gm of I-131 dose equivalent
	c.	Secondary system initial iodine activity	0.1 μCi/gm of I-131 dose equivalent
	d.	Reactor coolant initial noble gas activity	Based on 1% failed fuel, see Table 11.1-5
	e.	Iodine partition factor in the steam generator	0.01
	f.	Each steam generator water mass, lb	9.55E+4
II.	Atmospheric Dispersion Factors		See Table 15A-2
III.	Activity Release Data		
	a.	Total primary to secondary leakage 0-8 hr, lb	4000
	b.	Steam release from all steam generators 0-2 hours, lb	5.49E+5
		2-8 hours, lb	1.03E+6
	c.	Activity released to the environment	

Isotope	0-2 hr (Ci)	0-8 hr (Ci)
I-131	1.83E-1	5.31E-1
I-132	1.54E-1	2.55E-1
I-133	3.12E-1	8.42E-1
I-134	2.47E-2	2.87E-2
I-135	1.71E-1	3.95E-1
Xe-131m	1.54	6.13
Xe-133m	2.41	9.26
Xe-133	1.31E+2	5.15E+2
Xe-135m	5.13E-2	5.15E-2
Xe-135	4.14	1.34E+1
Xe-137	4.68E-3	4.68E-3
Xe-138	6.30E-2	6.31E-2
Kr-83m	1.76E-1	3.15E-1
Kr-85m	8.82E-1	2.35
Kr-85	4.27	1.71E+1
Kr-87	4.06E-1	6.04E-1
Kr-88	1.53	3.40
Kr-89	2.09E-3	2.09E-3

TABLE 15.2-3

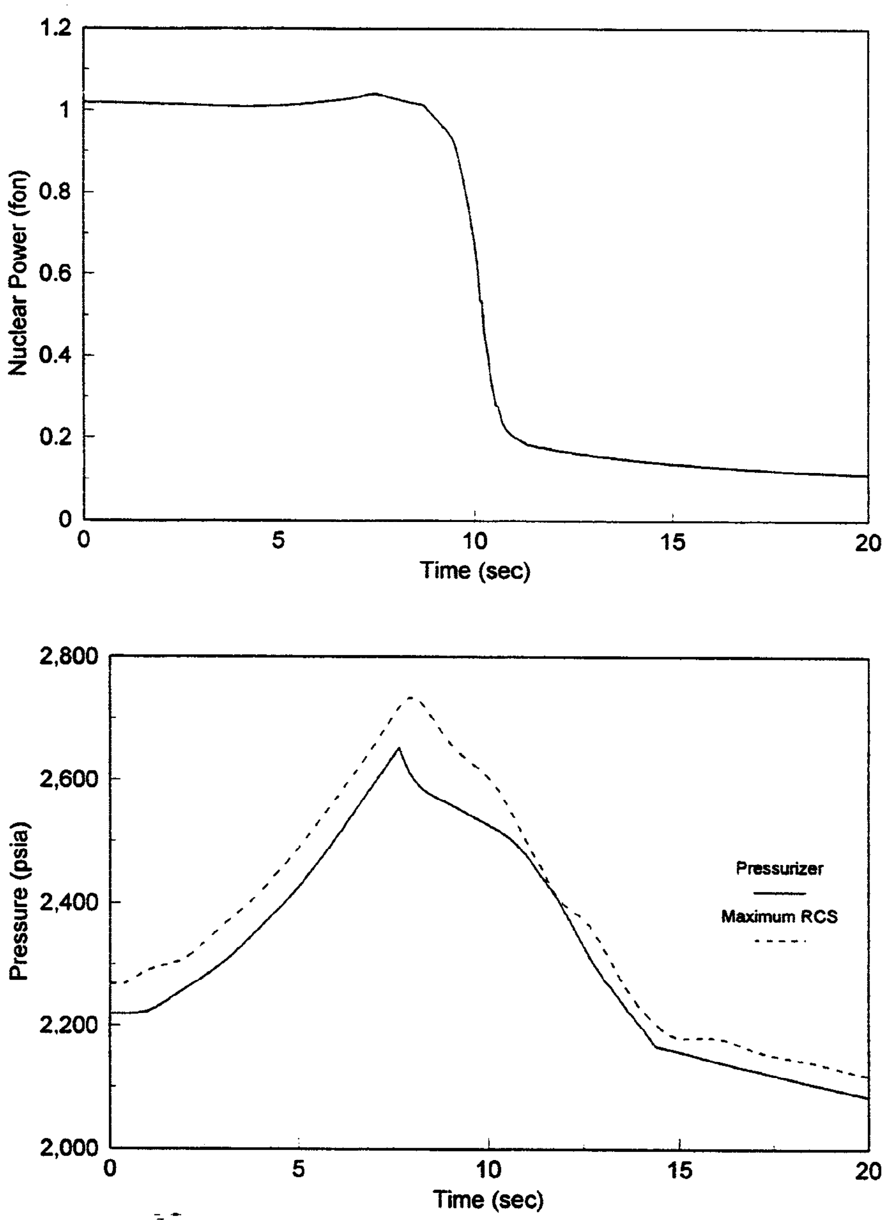
RADIOLOGICAL CONSEQUENCES OF LOSS OF NON-EMERGENCY AC POWER

	Wolf Creek Dose (rem)
Exclusion area boundary (0-2 hr)	
Thyroid, rem	1.92E-2
Whole body, rem	3.84E-4
Low-population zone, outer boundary (duration)	
Thyroid, rem	7.24E-3
Whole body, rem	1.60E-4

TABLE 15.2-4

NATURAL CIRCULATION FLOW

Core Power (percent)	RCS Natural Circulation Flow (percent)
5.0	5.9
4.0	5.4
3.0	4.8
2.0	4.1
1.0	3.2



Rev. 10

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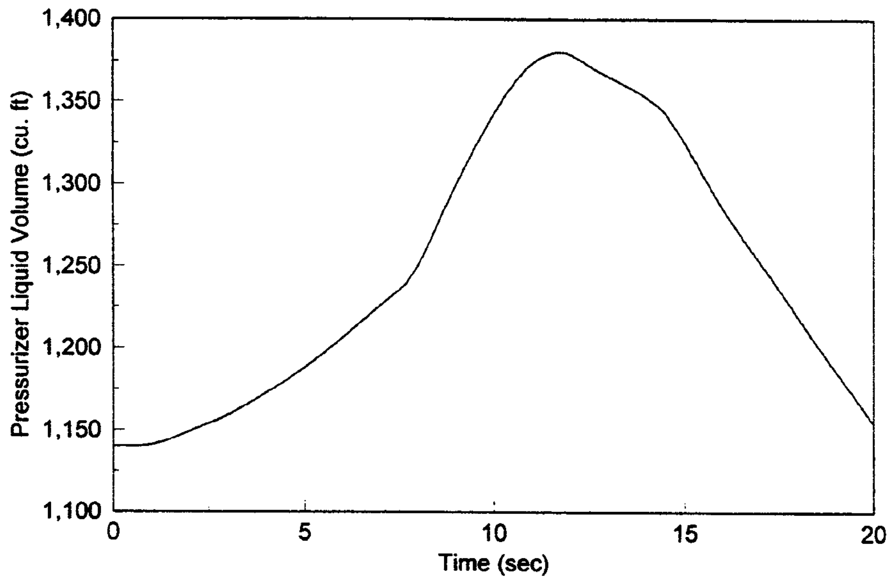
FIGURE 15.2-1
NUCLEAR POWER AND PRESSURE FOR TURBINE TRIP EVENT WITH MINIMUM REACTIVITY FEEDBACK OVERPRESSURE EVALUATION



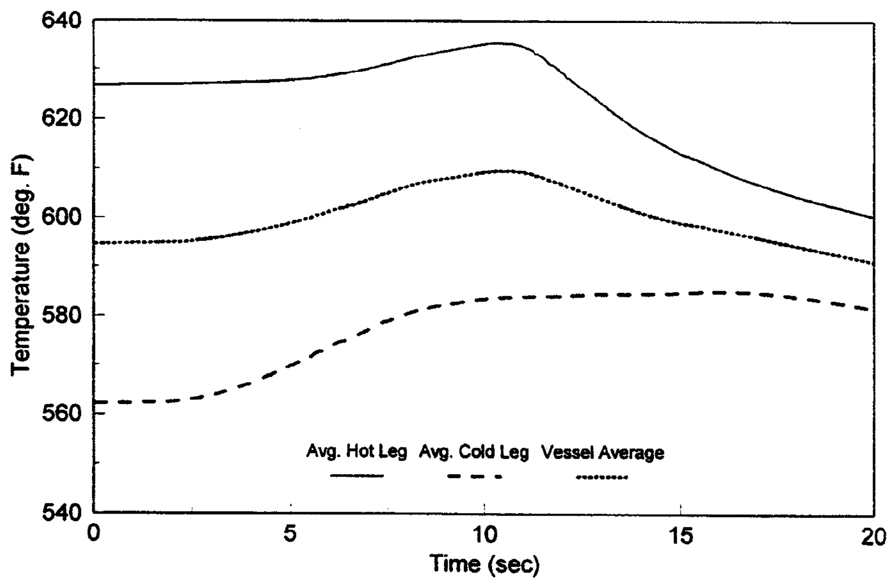


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FIGURE 15.2-2
PRESSURIZER LIQUID VOLUME AND RCS TEMPERATURE FOR TURBINE TRIP EVENT WITH MINIMUM REACTIVITY FEEDBACK OVERPRESSURE EVALUATION

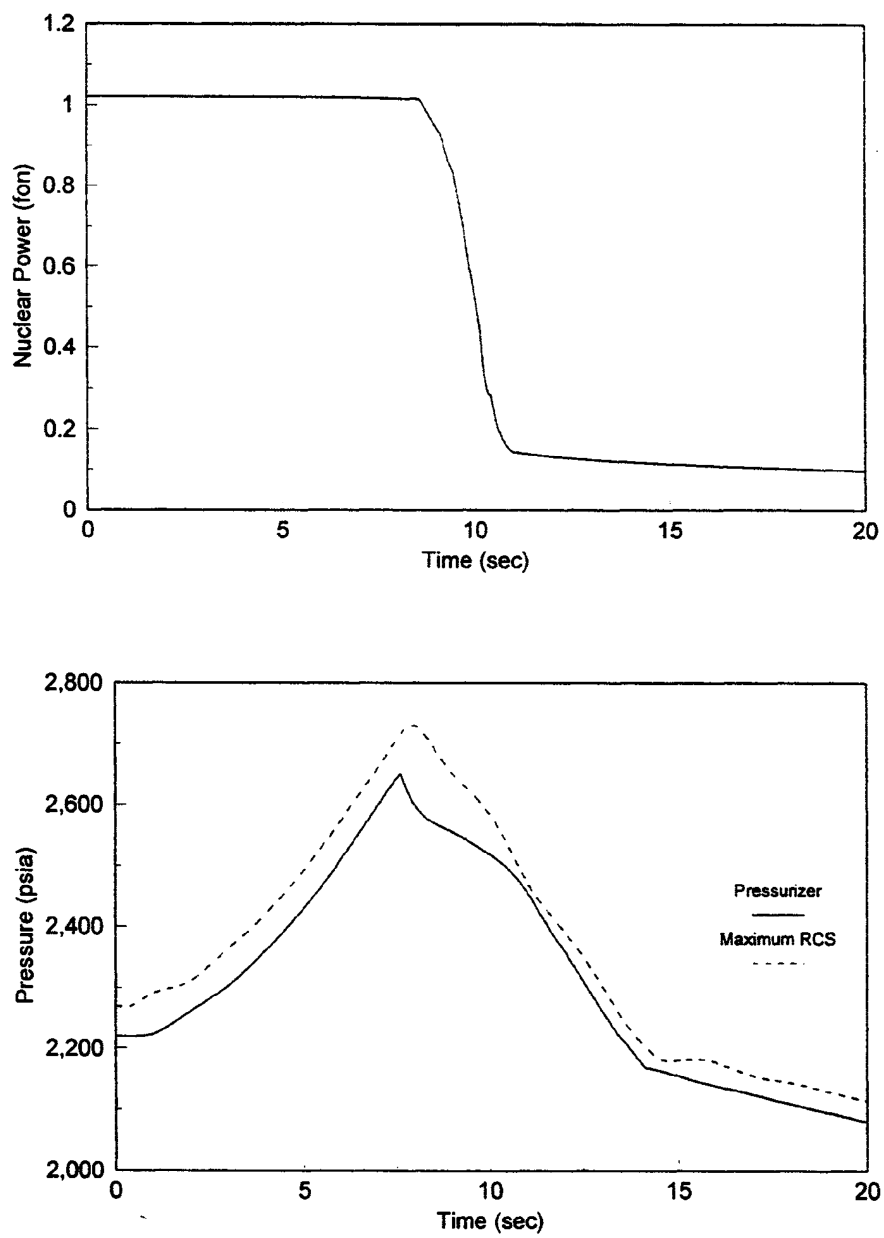

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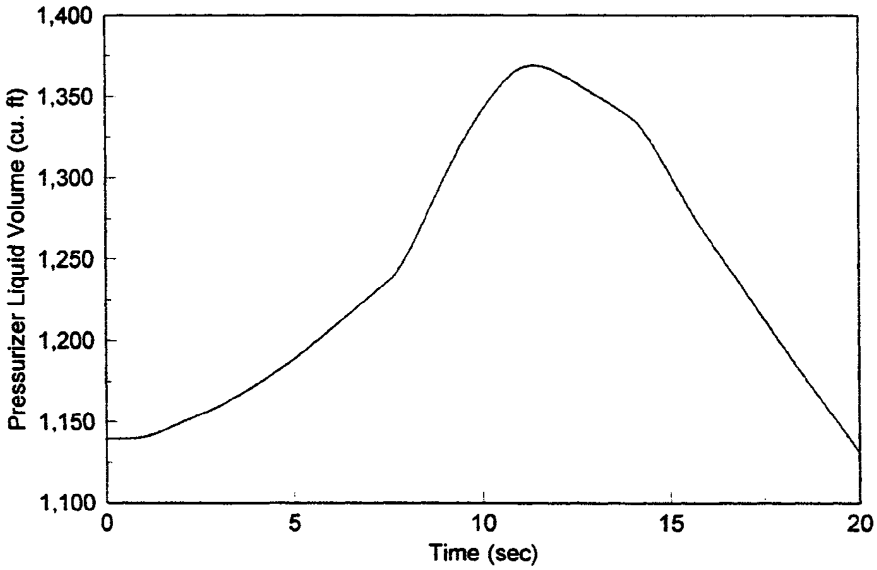
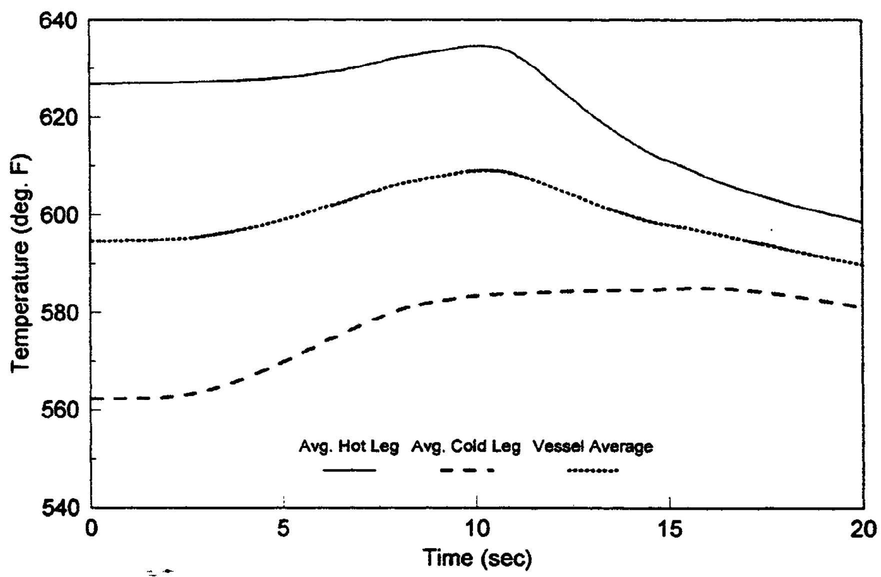
FIGURE 15.2-3 NUCLEAR POWER AND PRESSURE FOR TURBINE TRIP EVENT WITH MAXIMUM REACTIVITY FEEDBACK OVERPRESSURE EVALUATION





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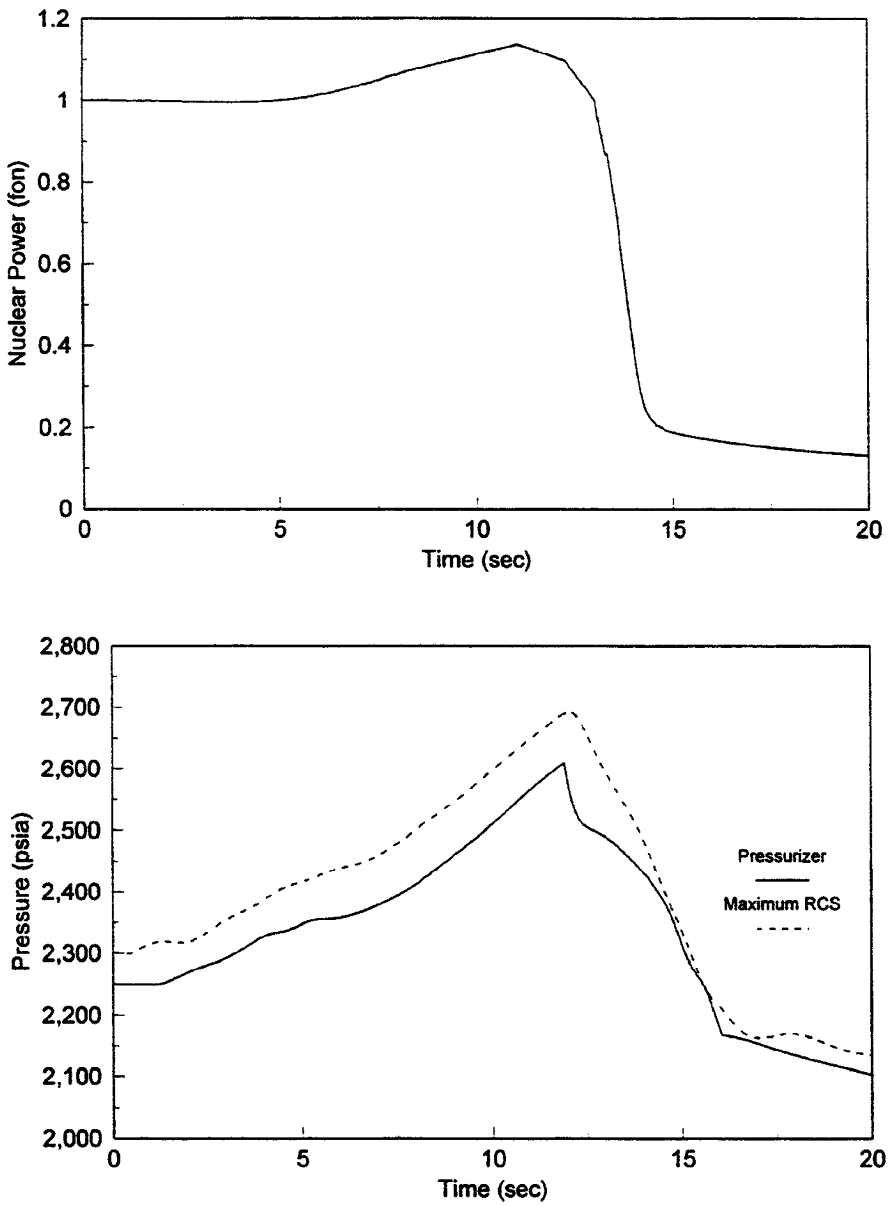
WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.2-4
PRESSURIZER LIQUID VOLUME AND RCS TEMPERATURE FOR TURBINE TRIP EVENT WITH MAXIMUM REACTIVITY FEEDBACK
OVERPRESSURE EVALUATION



Rev. 10

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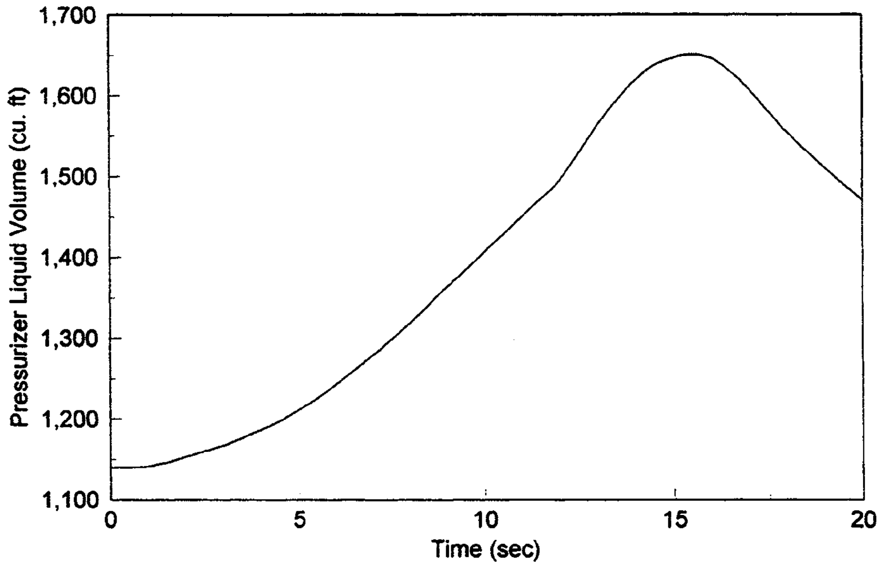
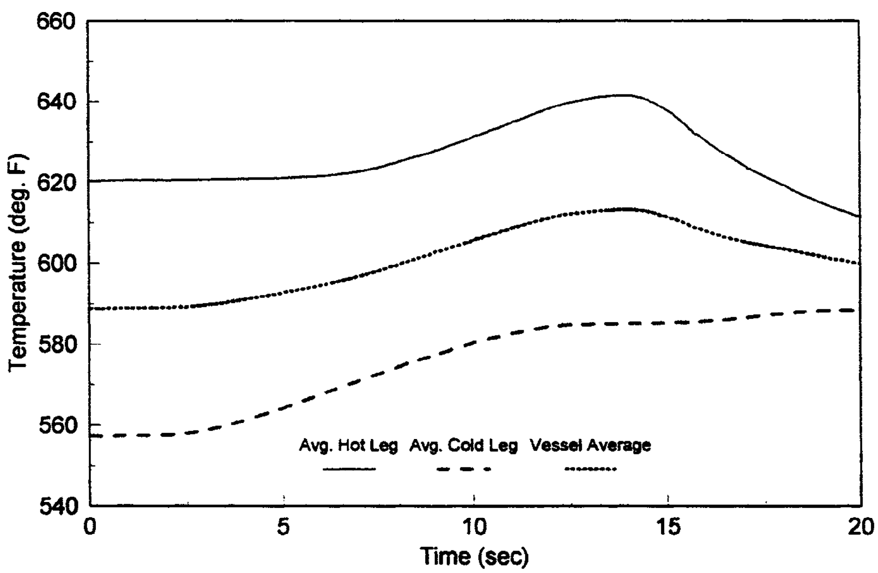
FIGURE 15.2-5
NUCLEAR POWER AND PRESSURE FOR TURBINE TRIP EVENT WITH MINIMUM REACTIVITY FEEDBACK DNB EVALUATION





Rev. 10

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FIGURE 15.2-6 PRESSURIZER LIQUID VOLUME AND RCS TEMPERATURE FOR TURBINE TRIP EVENT WITH MINIMUM REACTIVITY FEEDBACK DNB EVALUATION

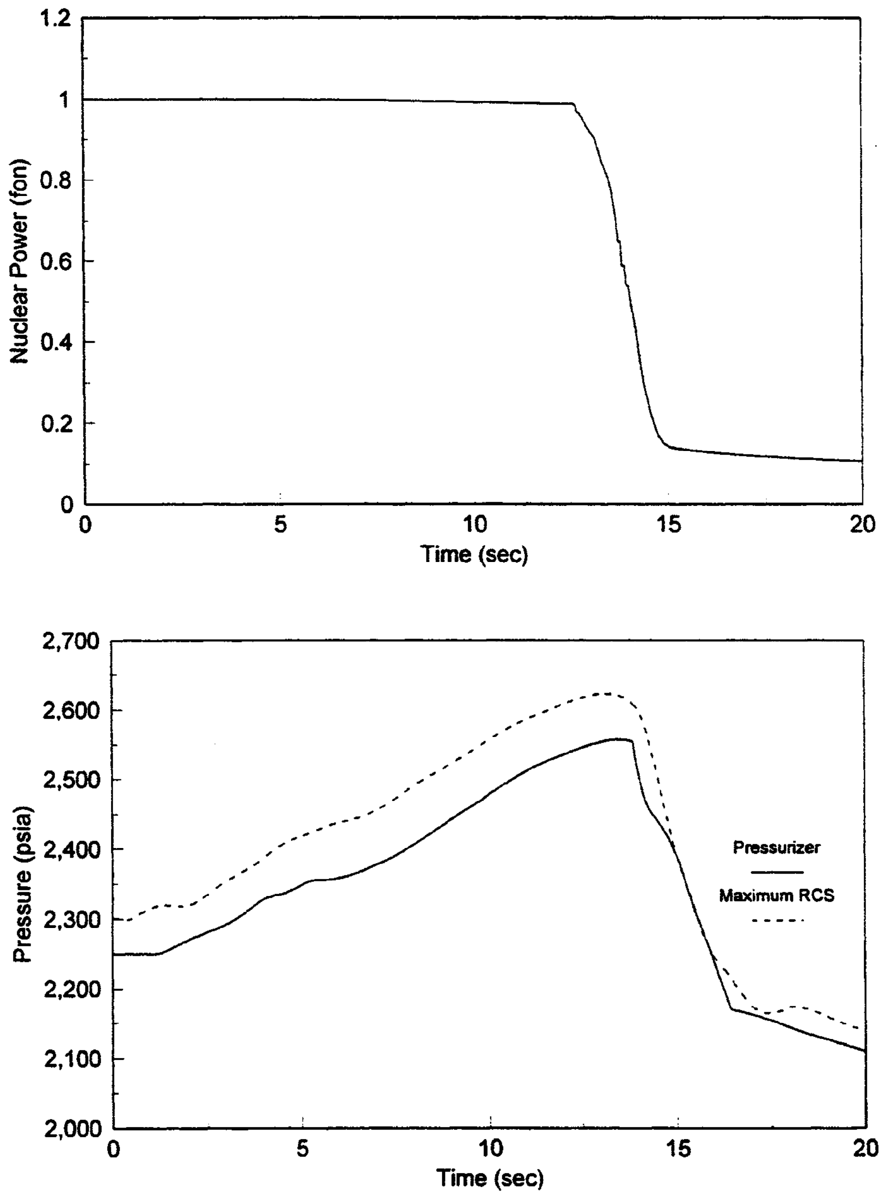

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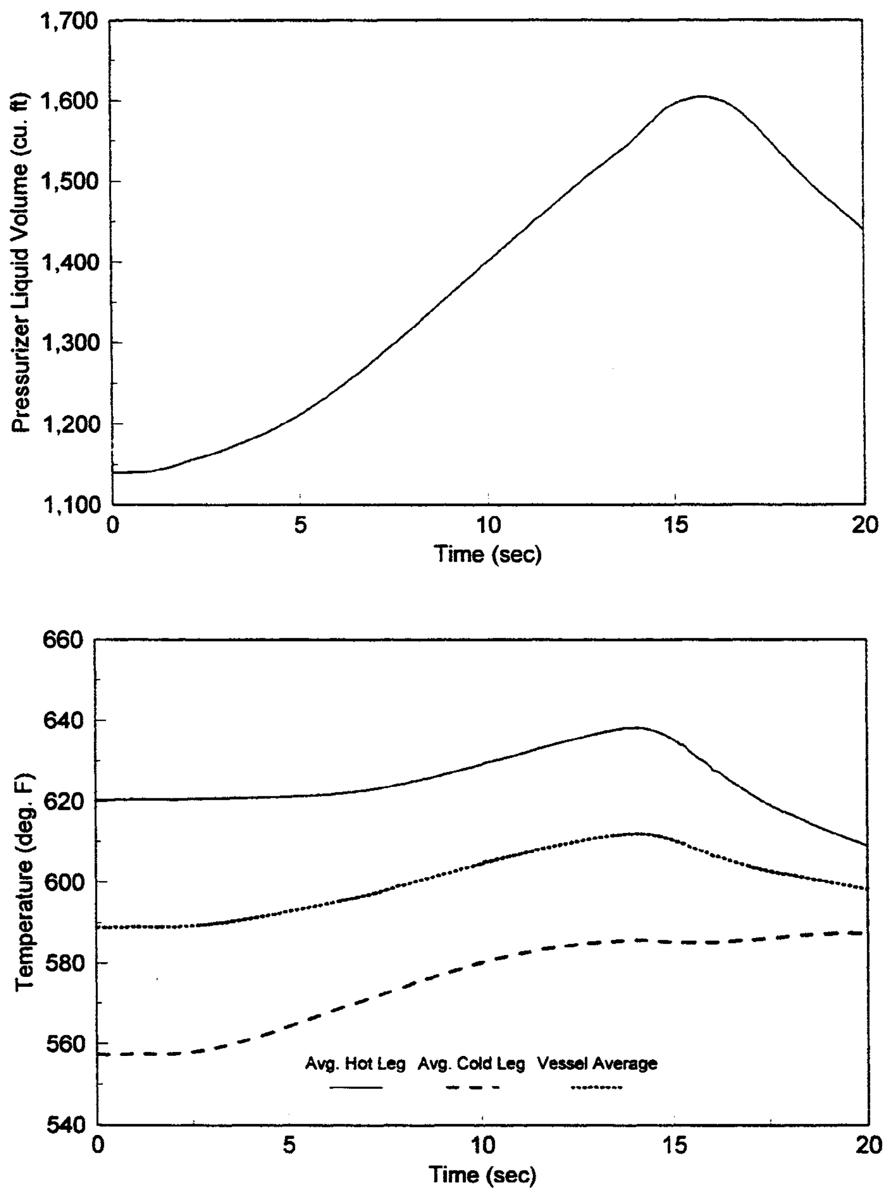
FIGURE 15.2-7 NUCLEAR POWER AND PRESSURE FOR TURBINE TRIP EVENT WITH MAXIMUM REACTIVITY FEEDBACK DNB EVALUATION



Rev. 10

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.2-8
PRESSURIZER LIQUID VOLUME AND RCS TEMPERATURE FOR TURBINE TRIP EVENT WITH MAXIMUM REACTIVITY FEEDBACK
DNB EVALUATION

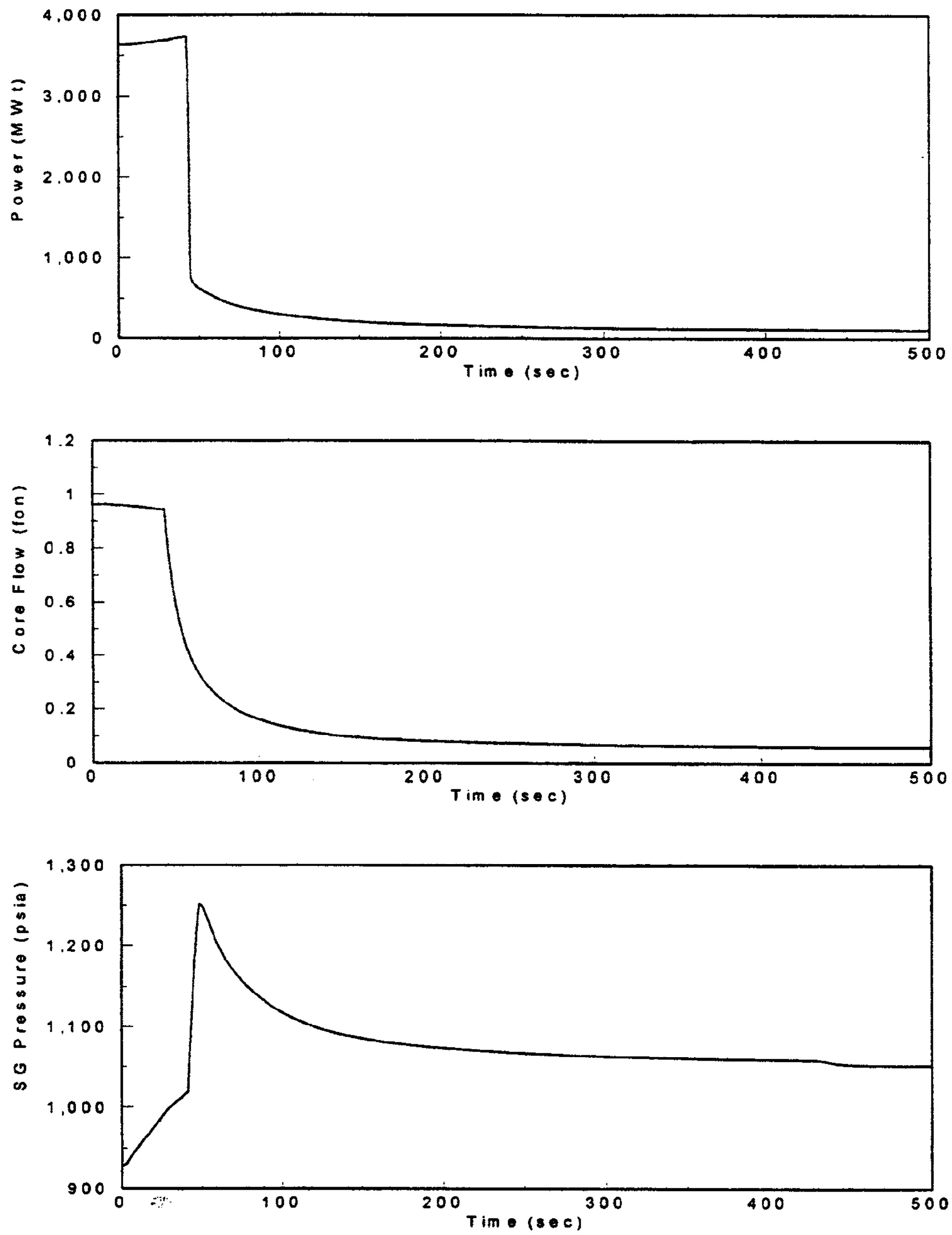


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FIGURE 15.2-9 NUCLEAR POWER, REACTOR COOLANT FLOW RATE AND STEAM GENERATOR PRESSURE TRANSIENTS FOR LOSS OF AC POWER

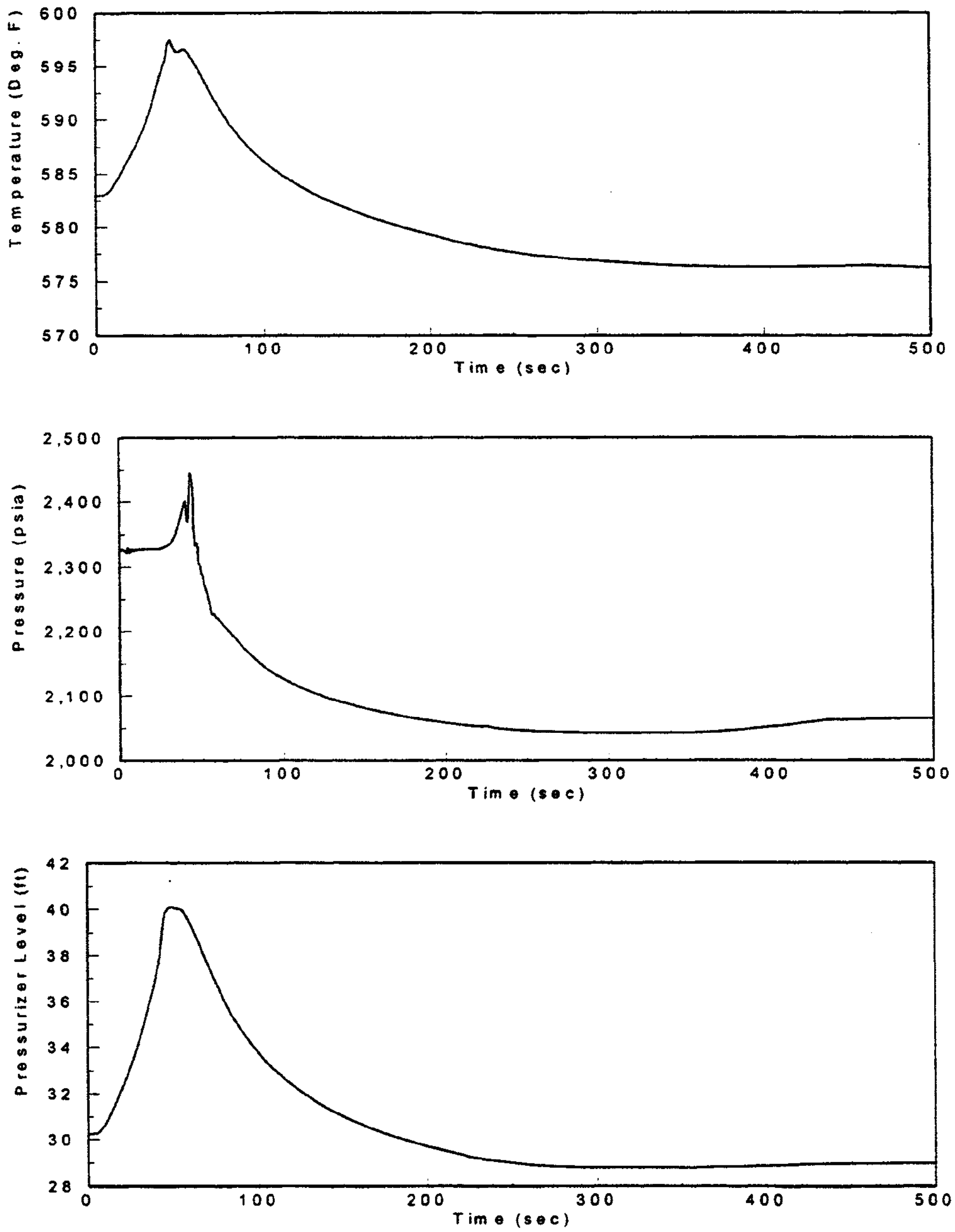

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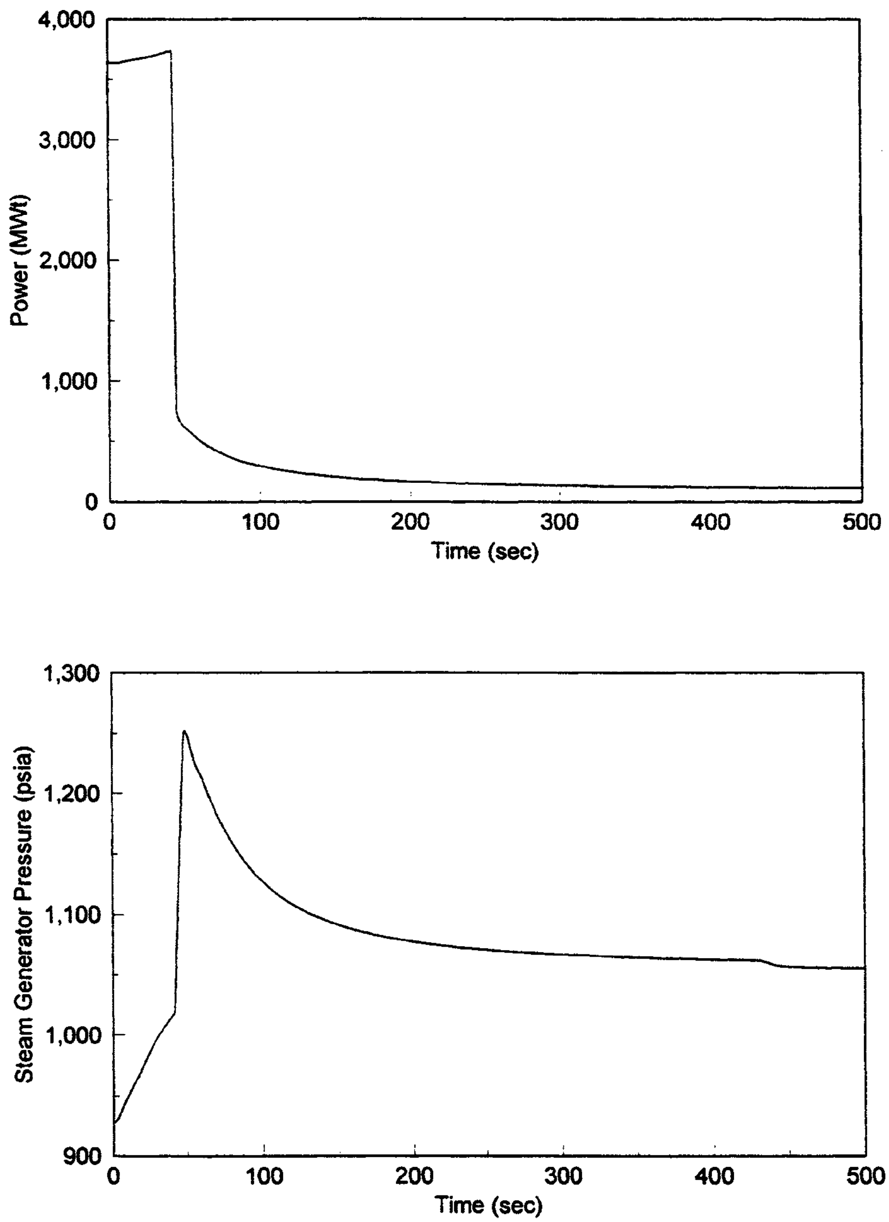
FIGURE 15.2-10 REACTOR COOLANT TEMPERATURE, MAXIMUM RCS PRESSURE AND PRESSURIZER LIQUID LEVEL FOR LOSS OF AC POWER



Rev. 10

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FIGURE 15.2-11
NUCLEAR POWER AND STEAM GENERATOR PRESSURE TRANSIENTS FOR LOSS OF NORMAL FEEDWATER

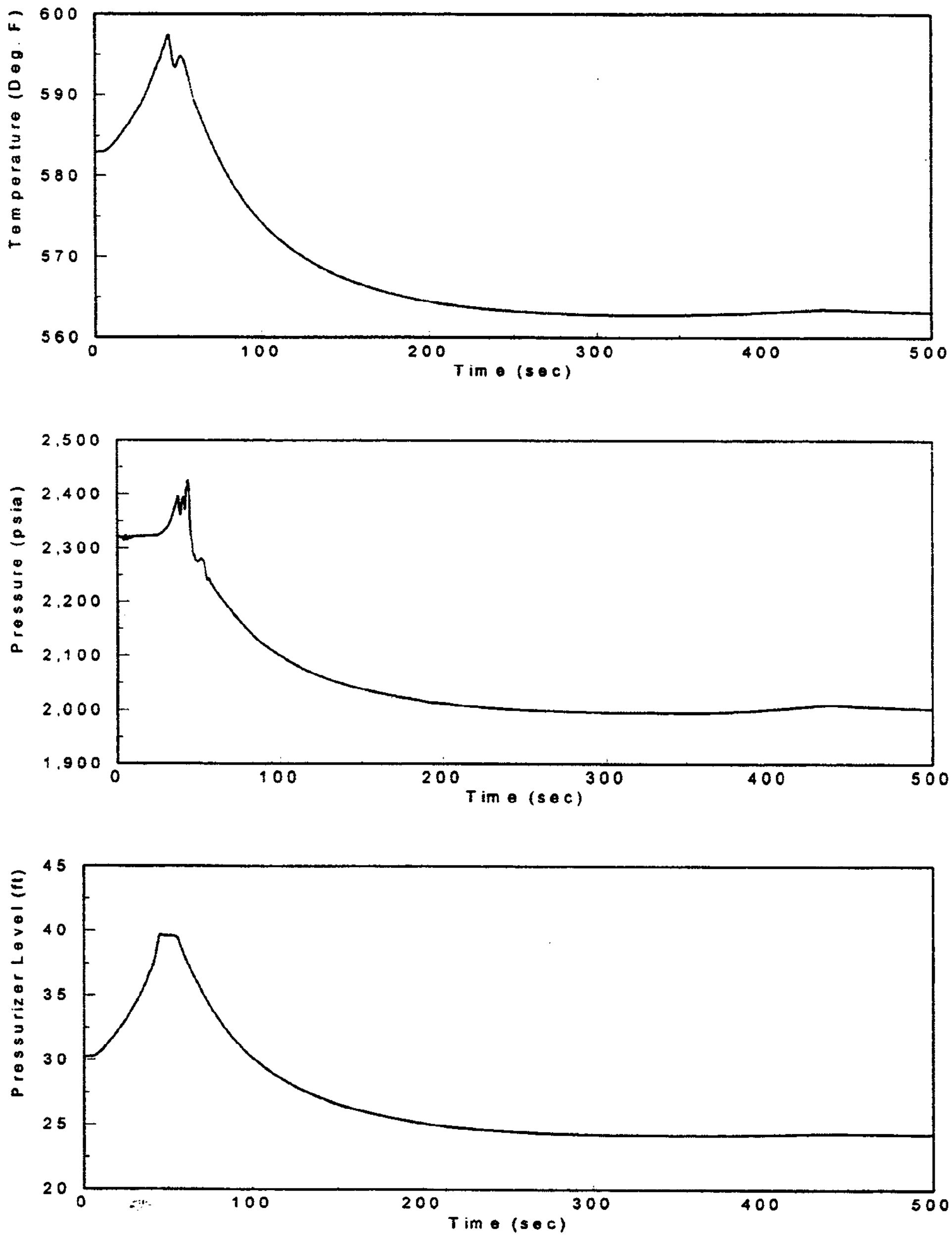

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FIGURE 15.2-12
CORE AVERAGE TEMPERATURE, MAXIMUM RCS PRESSURE AND PRESSURIZER LIQUID LEVEL TRANSIENTS FOR LOSS OF NORMAL FEEDWATER

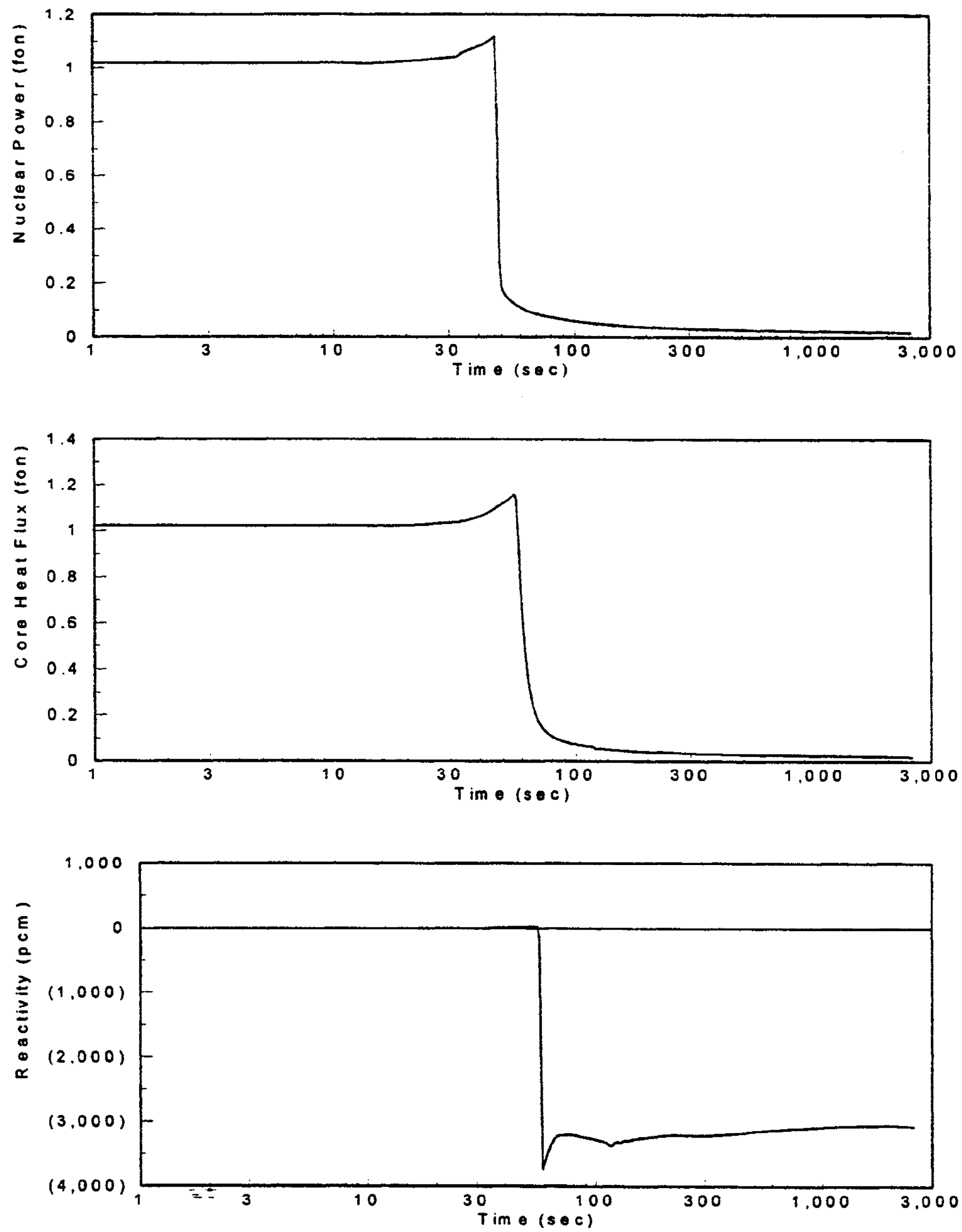

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FIGURE 15.2-13
NUCLEAR POWER, CORE HEAT FLUX AND TOTAL CORE REACTIVITY FOR MAIN FEEDWATER LINE RUPTURE WITH OFFSITE POWER AVAILABLE

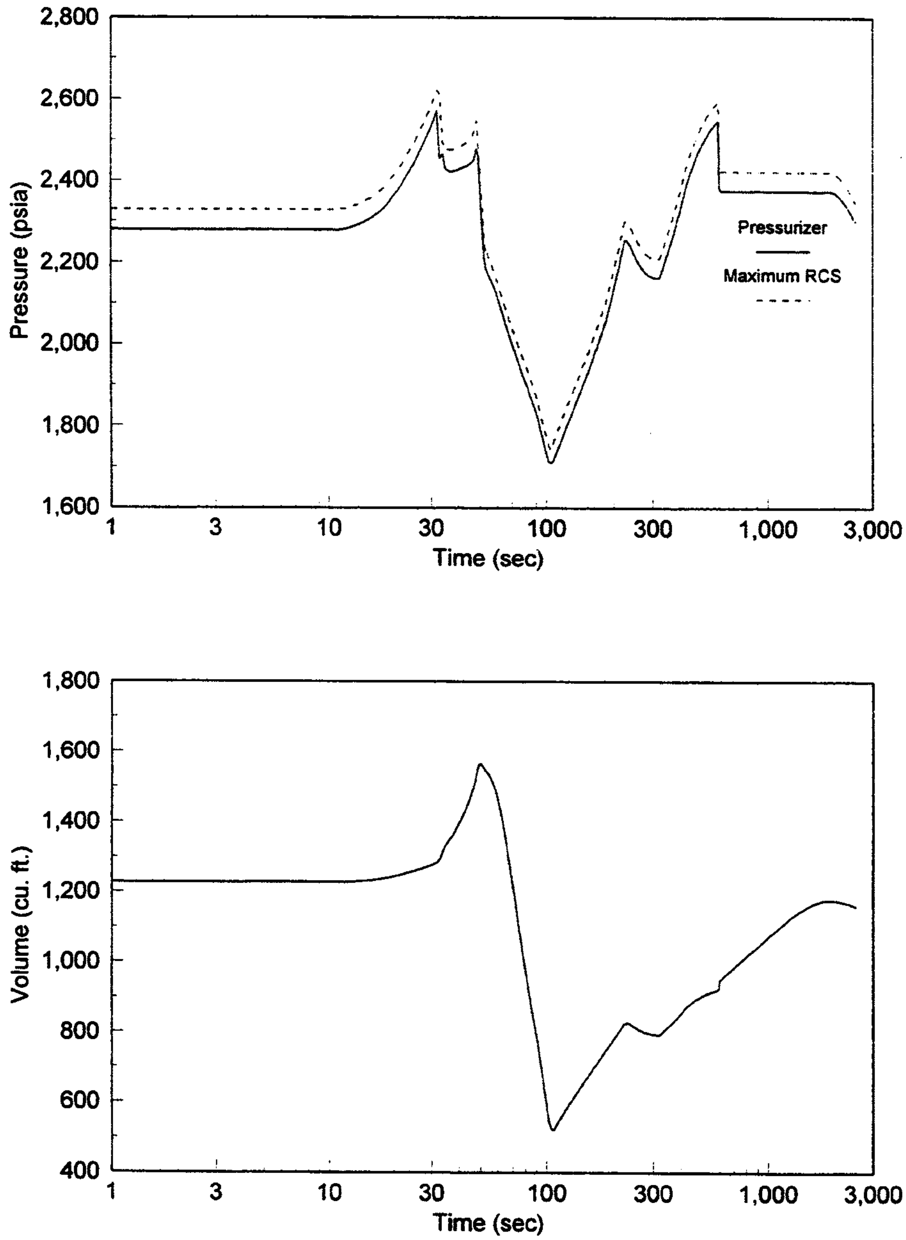

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FIGURE 15.2-14 PRESSURIZER AND MAXIMUM SYSTEM PRESSURE AND PRESSURIZER WATER VOLUME FOR MAIN FEEDWATER LINE RUPTURE WITH OFFSITE POWER AVAILABLE

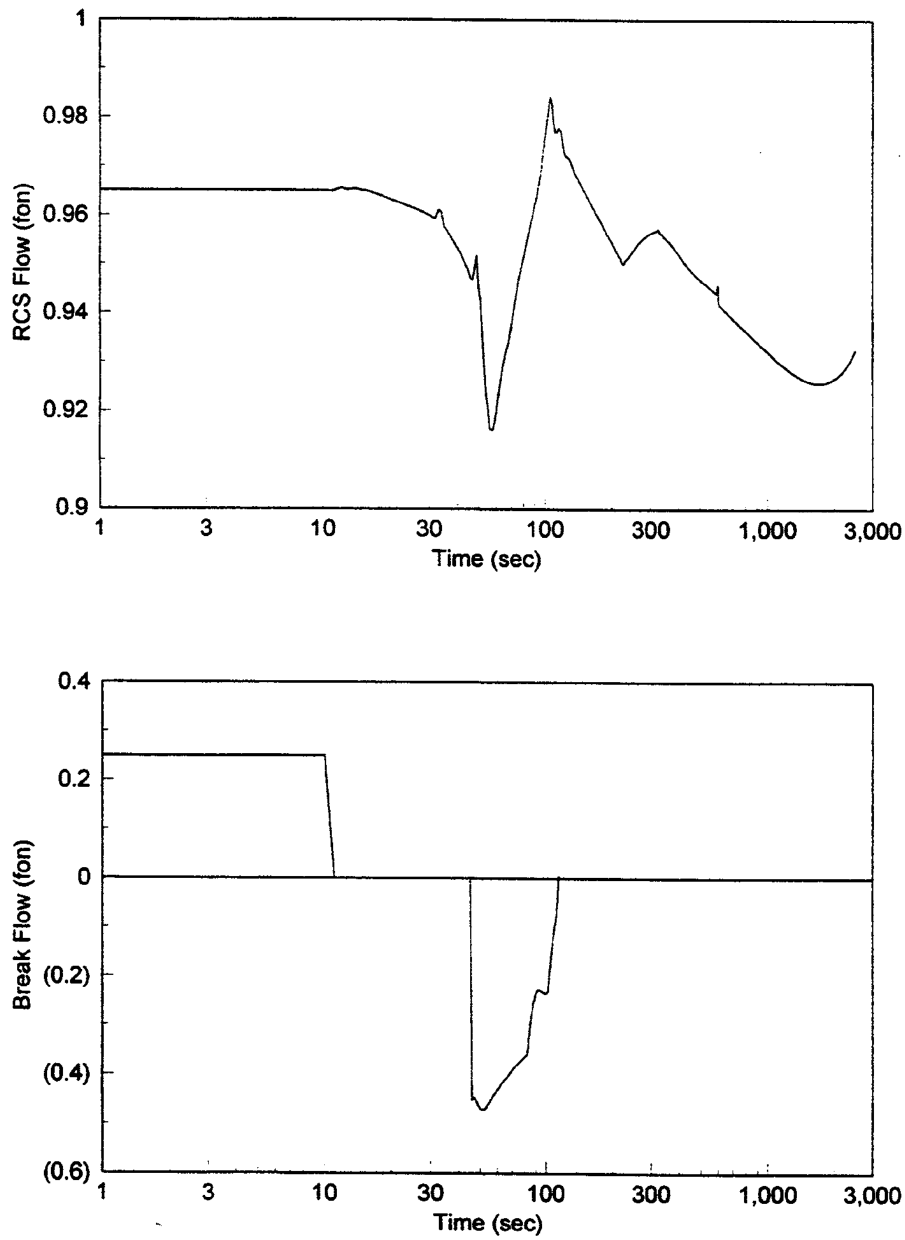

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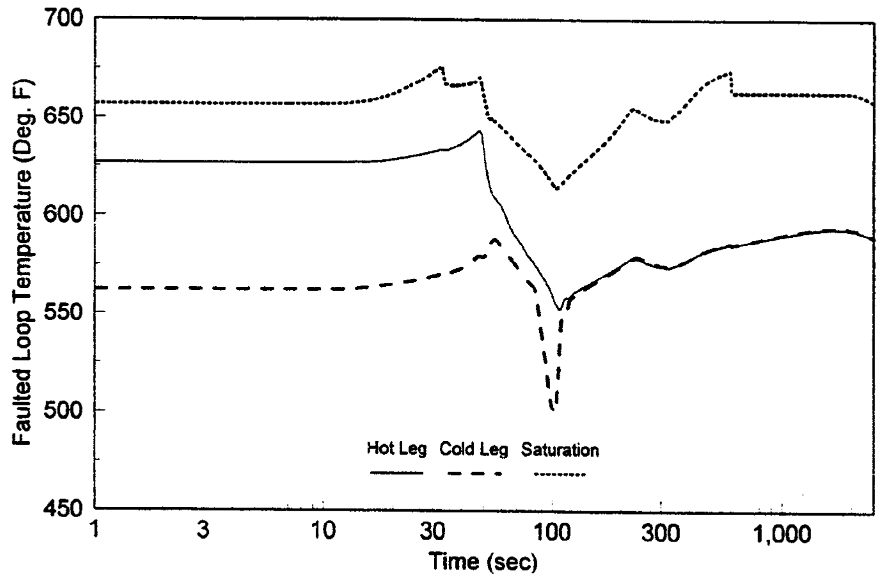
FIGURE 15.2-15
REACTOR COOLANT FLOW AND FEEDWATER LINE BREAK FLOW FOR MAIN FEEDWATER LINE RUPTURE WITH OFFSITE POWER AVAILABLE



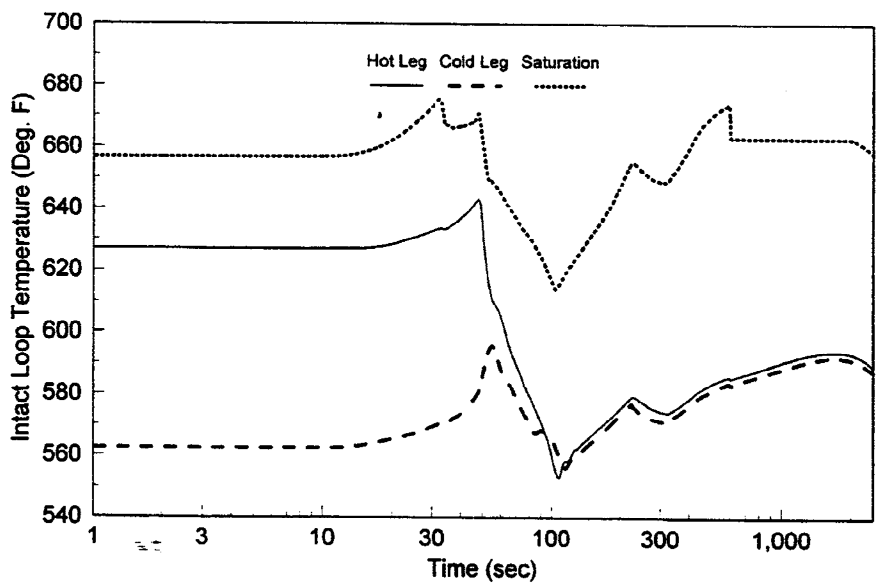


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FIGURE 15.2-16 FAULTED LOOP AND INTACT LOOP REACTOR COOLANT TEMPERATURE FOR MAIN FEEDWATER LINE RUPTURE WITH OFFSITE POWER AVAILABLE

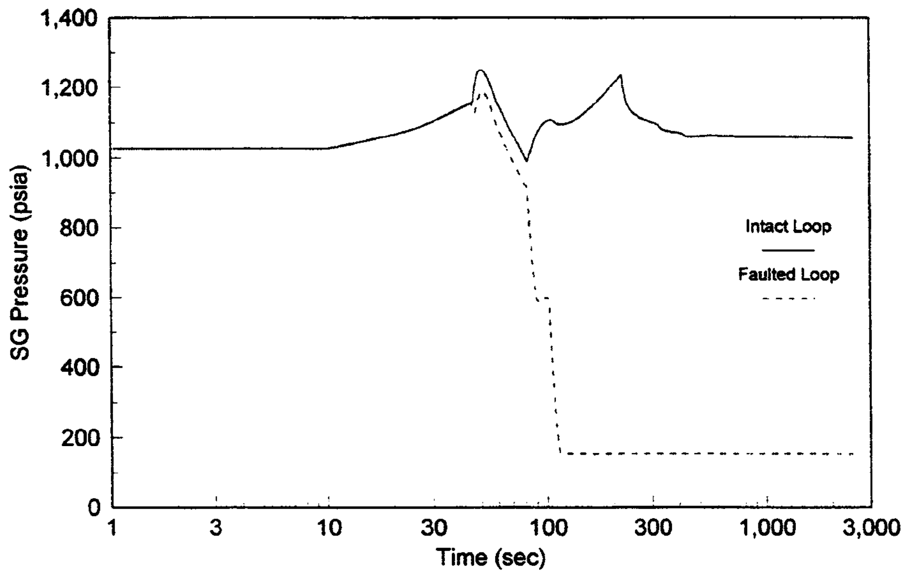

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FIGURE 15.2-17 STEAM GENERATOR PRESSURE FOR MAIN FEEDWATER LINE RUPTURE WITH OFFSITE POWER AVAILABLE

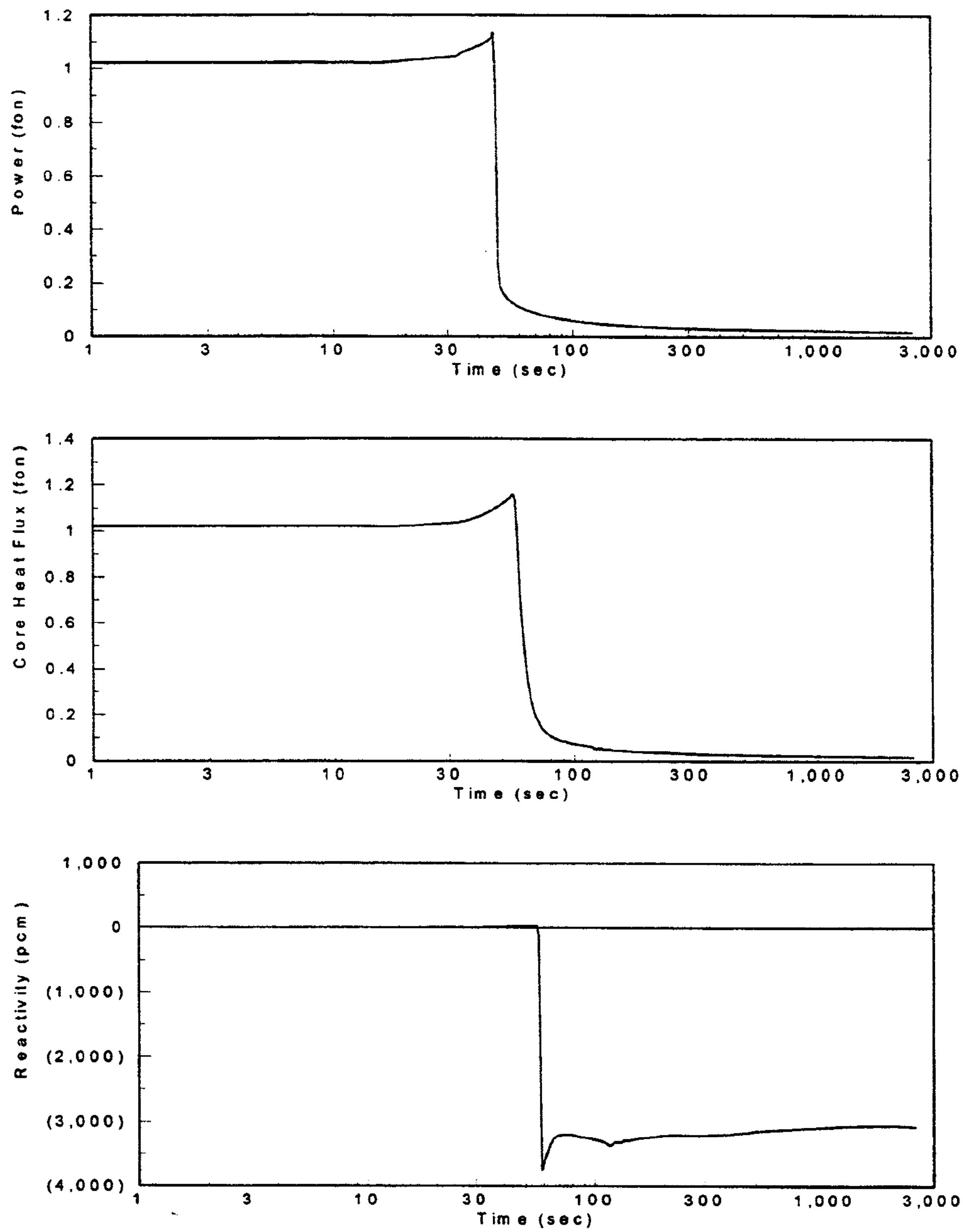

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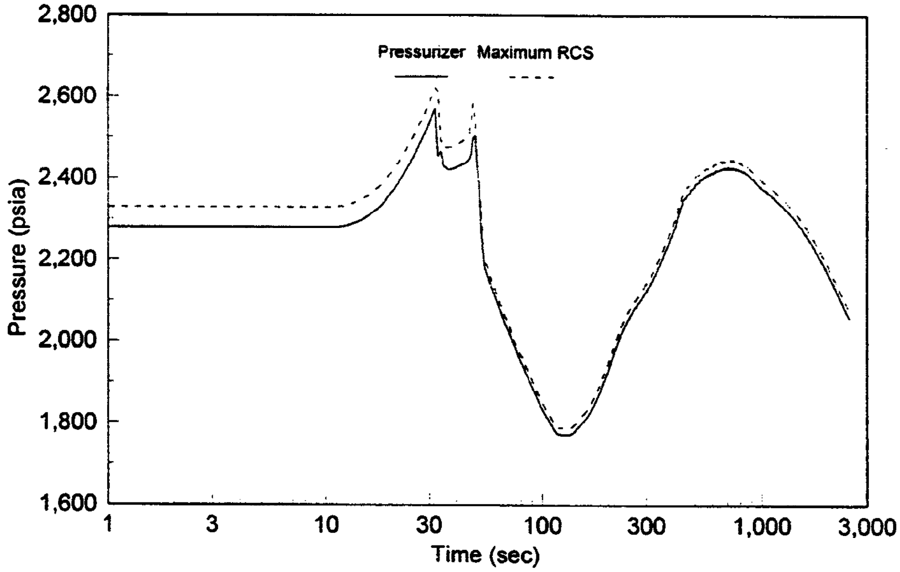
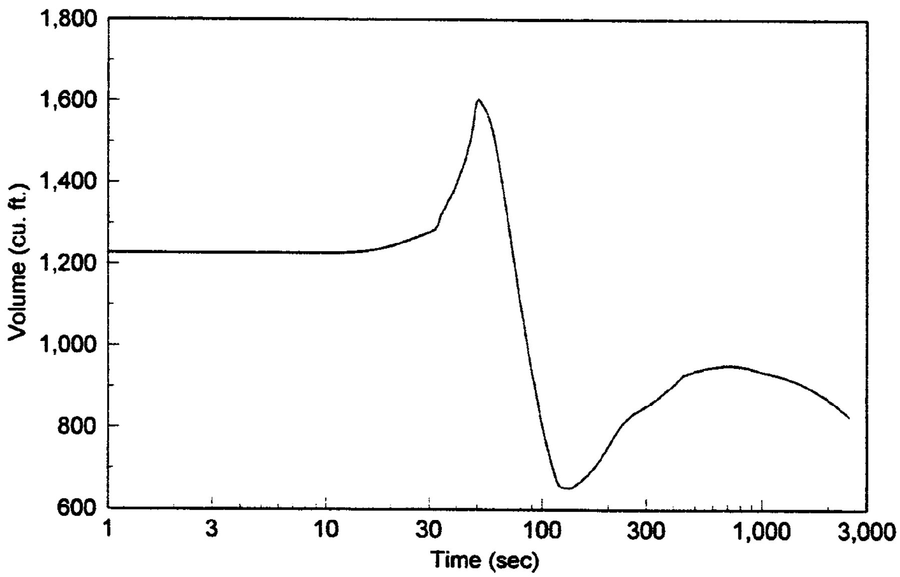
FIGURE 15.2-18 NUCLEAR POWER, CORE HEAT FLUX AND TOTAL CORE REACTIVITY FOR MAIN FEEDWATER LINE RUPTURE WITHOUT OFFSITE POWER





Rev. 10

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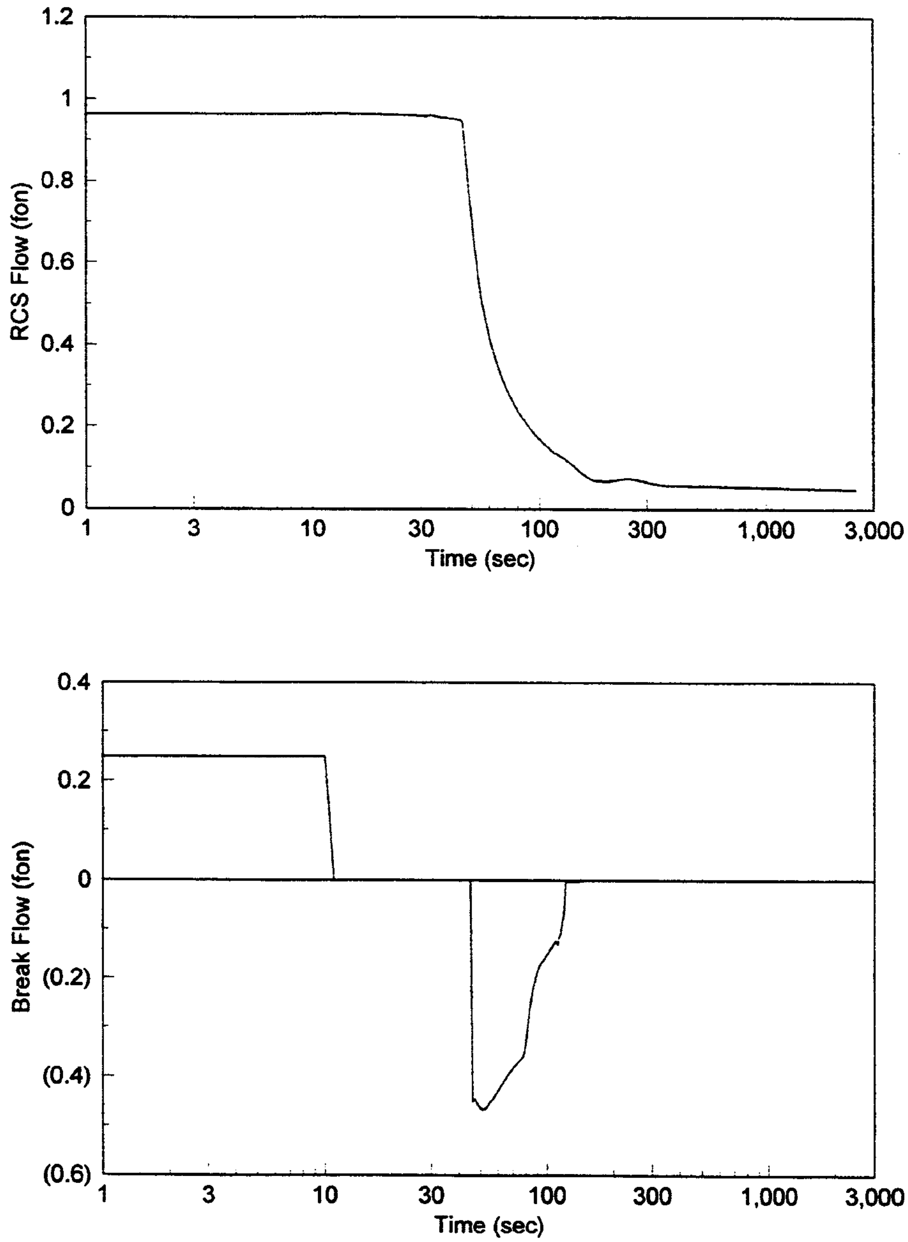
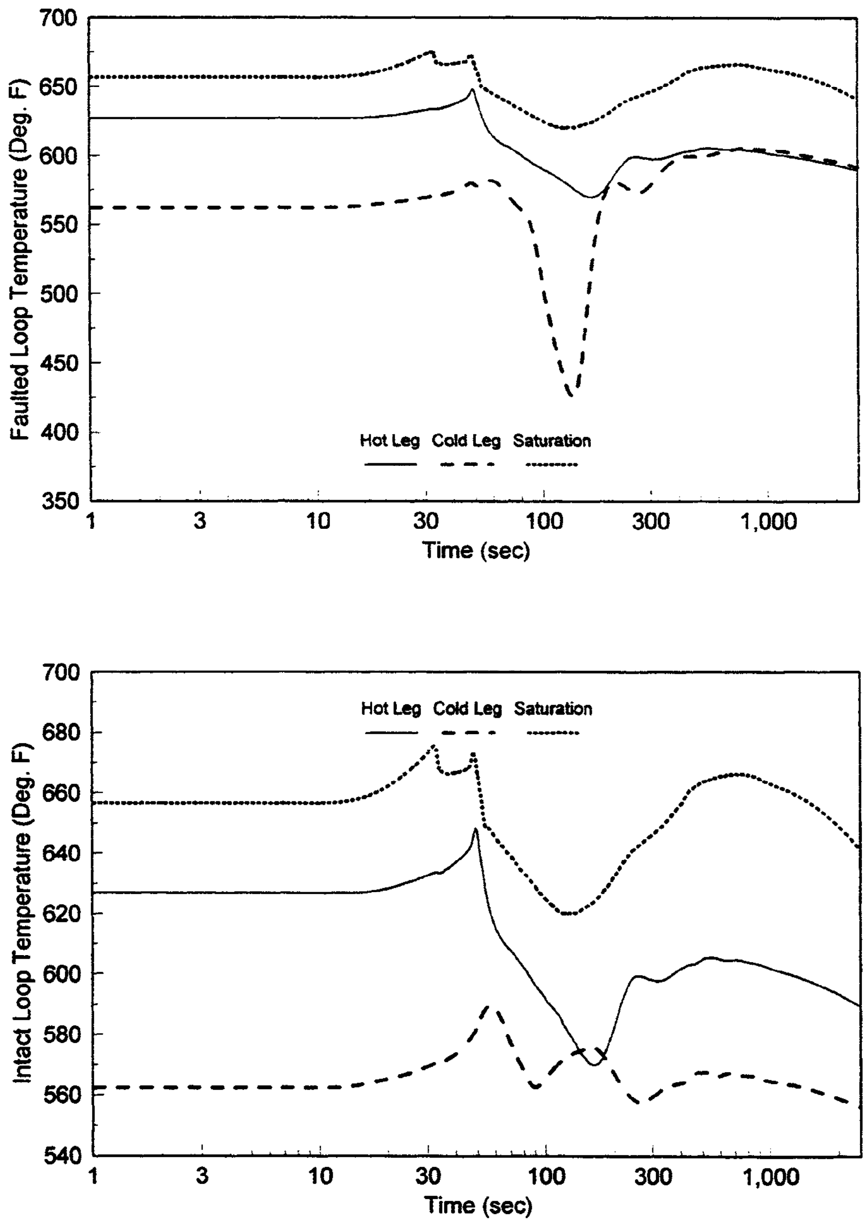
FIGURE 15.2-19 PRESSURIZER AND MAXIMUM SYSTEM PRESSURE AND PRESSURIZER WATER VOLUME FOR MAIN FEEDWATER LINE RUPTURE WITHOUT OFFSITE POWER



Rev. 10

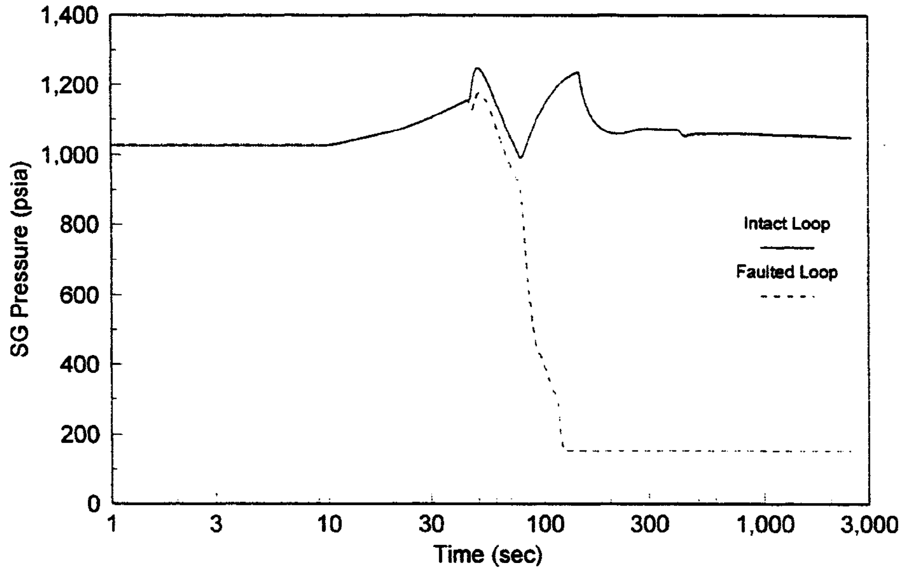
WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.2-20
REACTOR COOLANT FLOW AND FEEDWATER LINE BREAK FLOW FOR MAIN FEEDWATER LINE RUPTURE WITHOUT OFFSITE POWER



Rev. 10

WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.2-21 FAULTED LOOP AND INTACT LOOP REACTOR COOLANT TEMPERATURE FOR MAIN FEEDWATER LINE RUPTURE WITHOUT OFFSITE POWER



Rev. 10

WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.2-22 STEAM GENERATOR PRESSURE FOR MAIN FEEDWATER LINE RUPTURE WITHOUT OFFSITE POWER

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

A number of faults which could result in a decrease in the reactor coolant system (RCS) flow rate are postulated. These events are discussed in this section. Detailed analyses are presented for the most limiting of the following flow decrease events:

a. Partial loss of forced reactor coolant flow
b. Complete loss of forced reactor coolant flow
c. Reactor coolant pump shaft seizure (locked rotor)
d. Reactor coolant pump shaft break

Item a above is considered to be an ANS Condition II event, item b an ANS Condition III event, and items c and d ANS Condition IV events. Section 15.0.1 contains a discussion of ANS classifications.

All of the accidents in this section have been analyzed. It has been determined that the most severe radiological consequences will result from the reactor coolant pump shaft seizure accident of Section 15.3.3. Therefore, doses are reported only for that limiting case.

15.3.1 PARTIAL LOSS OF FORCED REACTOR COOLANT FLOW

15.3.1.1 Identification of Causes and Accident Description

A partial loss-of-coolant flow accident can result from a mechanical or electrical failure in a reactor coolant pump, or from a fault in the power supply to the pump or pumps supplied by a reactor coolant pump bus. If the reactor is at power at the time of the accident, the immediate effect of the loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in departure from nucleate boiling (DNB) with subsequent fuel damage if the reactor is not tripped promptly.

Normal power for the pumps is supplied through individual busses connected to the generator. When a generator trip occurs, the busses are automatically transferred to a transformer supplied from external power lines, and the pumps will continue to operate. Following any turbine trip where there are no electrical faults which require tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds. The reactor coolant pumps remain connected to the generator, thus ensuring full flow for approximately 30 seconds after the reactor trip before any transfer is made.

A partial loss of coolant flow is classified as an ANS Condition II incident (a fault of moderate frequency), as defined in Section 15.0.1.

The necessary protection against this event is provided by the low primary coolant flow reactor trip signal, which is actuated in any reactor coolant loop by two out of three low flow signals. Above Permissive 8 (see Section 7.2), low flow in any loop will actuate a reactor trip. Between approximately 10-percent power (Permissive 7) and the power level corresponding to Permissive 8, low flow in any two loops will actuate a reactor trip. Above Permissive 7, two or more reactor coolant pump circuit breakers opening will actuate the corresponding undervoltage relays. This results in a reactor trip which serves as a backup to the low flow trip.

15.3.1.2 Analysis of Effects and Consequences

Method of Analysis

The case for a loss of two reactor coolant pumps with four loops in operation is analyzed here.

This event is analyzed by employing the detailed digital computer program RETRAN-02 (Ref 1). RETRAN-02 has been found acceptable by the NRC for use as a licensing basis safety analysis code. RETRAN-02 is a thermal-hydraulic systems analysis code employing a one-dimensional, homogeneous equilibrium mixture thermal-hydraulic model for the RCS, a point kinetics model for the reactor core, special component and auxiliary models (ex., pumps, temperature transport, non-equilibrium pressurizer) and control system models. The code computes pertinent plant transient information including core power level, RCS pressure and temperature. VIPRE-01 (Ref 2) is used to evaluate the core thermal limits to determine DNBR. RETRAN-02 generated state points are used as VIPRE-01 boundary conditions to perform a Statistical Core Design (SCD) DNB analysis.

Initial Conditions

Plant characteristics and initial conditions are discussed in Section 15.0.3. This event is initiated at nominal operating conditions in accordance with Westinghouse RTDP methodology (Ref. 4) to perform the DNB analysis

Reactivity Coefficients

A conservatively large absolute value of the Doppler-only power coefficient is used (see Figure 15.0-2), and the most positive moderator temperature coefficient is assumed, since this results in the maximum core power during the initial part of the transient, when the minimum DNBR is reached.

For this analysis, the curve of trip reactivity insertion versus time (Figure 15.0-5) was used.

Flow Coastdown

The flow coastdown analysis is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance, and the pump characteristics, and is based on high estimates of system pressure losses.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-15.

Results

Figures 15.3-1 through 15.3-5 show the transient response for the loss of two reactor coolant pumps with four loops in operation. Figure 15.3-6 shows the DNBR to be always greater than the safety analysis limit.

The plant is tripped by the low flow trip rapidly enough to ensure that the ability of the reactor coolant to remove heat from the core is not greatly reduced for the cases analyzed. Thus, the average fuel and clad temperatures do not increase significantly above their respective initial values.

The calculated sequence of events table for the cases analyzed is shown on Table 15.3-1. The affected reactor coolant pump(s) will continue to coast down, and the core flow will reach a new equilibrium value corresponding to the two pumps still in operation. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

15.3.1.3 Conclusions

The analysis shows that the DNBR will not decrease below the safety limit at any time during the transient. The DNBR analysis design basis is described in Section 4.4. All applicable acceptance criteria are met.

15.3.2 COMPLETE LOSS OF FORCED REACTOR COOLANT FLOW

15.3.2.1 Identification of Causes and Accident Description

A complete loss of flow accident may result from a simultaneous loss of electrical supplies to all reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of loss-of-coolant flow is a rapid increase in the coolant temperature. This increase could result in a DNB with subsequent fuel damage if the reactor were not tripped promptly.

Normal power for the reactor coolant pumps is supplied through busses from a transformer connected to the generator. When a generator trip occurs, the busses are automatically transferred to a transformer supplied from external power lines, and the pumps will continue to supply coolant flow to the core. Following any turbine trip where there are no electrical faults which require tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds. The reactor coolant pumps remain connected to the generator, thus ensuring full flow for 30 seconds after the reactor trip before any transfer is made.

A complete loss of flow accident is classified as an ANS Condition III event (an infrequent fault), as defined in Section 15.0.1. The following signals provide protection against this event:

a. Reactor coolant pump power supply undervoltage or underfrequency
b. Low reactor coolant loop flow

The reactor trip on reactor coolant pump undervoltage is provided to protect against conditions which can cause a loss of voltage to all reactor coolant pumps. This function is blocked below approximately 10-percent power (Permissive 7).

The reactor trip on reactor coolant pump underfrequency is provided to trip the reactor for an underfrequency condition, resulting from frequency disturbances on the power grid. Reference 3 provides analyses of grid frequency disturbances and the resulting NSSS protection requirements which are applicable to WCGS.

Reference 3 shows that the underfrequency trip of the reactor coolant pump breakers is not required for grid decay rates up to 5 Hz/sec. Grid stability and transient analyses for both the Wolf Creek and Callaway sites show maximum grid decay rates of less than 5 Hz/sec. Therefore, the reactor coolant pump breaker trip on under frequency (Figure 7.2-1, Sheet 5) is not a safety function in the WCGS design.

The reactor trip on low primary coolant loop flow is provided to protect against loss of flow conditions which affect only one reactor coolant loop. This function is generated by two out of three low flow signals per reactor coolant loop. Above Permissive 8, low flow in any loop will actuate a reactor trip. Between approximately 10-percent power (Permissive 7) and the power level corresponding to Permissive 8, low flow in any two loops will actuate a reactor trip. If the maximum grid frequency decay rate is less than approximately 2.5 Hertz per second, this trip function will protect the core from underfrequency events. This effect is fully described in Reference 3.

15.3.2.2 Analysis of Effects and Consequences

Method of Analysis

The following case has been analyzed:

Loss of four pumps with four loops in operation

The method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in Section 15.3.1, except that, following the loss of power supply to all pumps at power, a reactor trip is actuated by either reactor coolant pump power supply undervoltage or underfrequency.

Results

Figure 15.3-8, based on the Pressure Evaluation Model, and Figures 15.3-7 and 15.3-9 through 15.3-12, based on the Statistical Core Design Model, show the transient response for the loss of power to all reactor coolant pumps with four loops in operation. The reactor is assumed to be tripped on an undervoltage signal. Figure 15.3-12 shows the DNBR to be always greater than the safety analysis limit.

For the case analyzed, the plant is tripped by the undervoltage trip sufficiently fast to ensure that the ability of the primary coolant to remove heat from the fuel rod is not greatly reduced. Thus, the average fuel and clad temperatures do not increase significantly above their respective initial values.

The calculated sequence of events for the case analyzed is shown on Table 15.3-1. The reactor coolant pumps will continue to coast down, and natural circulation flow will eventually be established, as demonstrated in Section 15.2.6. With the reactor tripped, a stable plant condition will be attained. Normal plant shutdown may then proceed.

15.3.2.3 Conclusions

The analysis performed has demonstrated that, for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the safety analysis limit at any time during the transient. The design basis for the DNBR is described in Section 4.4. All applicable acceptance criteria are met.

15.3.3 REACTOR COOLANT PUMP SHAFT SEIZURE (LOCKED ROTOR)

15.3.3.1 Identification of Causes and Accident Description

The accident postulated is an instantaneous seizure of a reactor coolant pump rotor, as discussed in Section 5.4. Flow through the affected reactor coolant loop is rapidly reduced, leading to an initiation of a reactor trip on a low flow signal.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced - first, because the reduced flow results in a decreased tube side film coefficient, and then, because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves, in that sequence. The two power-operated relief valves are designed for reliable operation, and would be expected to function properly during the accident. However, for conservatism, their pressure-reducing effect, as well as the pressure-reducing effect of the spray, are not included in the analysis.

This event is classified as an ANS Condition IV incident (a limiting fault), as defined in Section 15.0.1.

15.3.3.2 Analysis of Effects and Consequences

Method of Analysis

Two digital computer codes are used to analyze this transient. The RETRAN code (Ref. 1) is used to calculate the resulting loop and core flow transients following the pump seizure, the time of reactor trip based on the loop flow transients, the nuclear power following reactor trip, and to determine the peak pressure. The thermal behavior of the fuel located at the core hot spot is investigated by using the FACTRAN code (Ref. 5), which uses Bishop-Sandberg-Ton film boiling correlation to evaluate fluid properties.

Accident Scenario

Two cases are analyzed:

a. One locked rotor with four loops in operation with offsite power.
b. One locked rotor with four loops in operation without offsite power.

The locked rotor transient is postulated to occur in the following manner:

a. Reactor coolant pump rotor locks and flow in that loop begins to coast down.
b. The reactor is tripped on low RCS flow in one loop.
c. Turbine generator trips.
d. Offsite power is lost for the without offsite power case.

   NOTE: Grid stability analyses show that the grid will remain stable and offsite power will not be lost because of a unit trip from 100 percent power. Refer to Section 8.2.2. The following analysis assumes no time delay between reactor trip and loss of offsite power. This is a conservative assumption based on the grid stability analyses.

e. The loss of offsite power causes the three remaining reactor coolant pumps to coast down.

At the beginning of the postulated locked rotor accident, i.e., at the time the shaft in one of the reactor coolant pumps is assumed to seize, the plant is assumed to be in operation under the most adverse steady state operating conditions, i.e., maximum guaranteed steady state thermal power, maximum steady state pressure, and maximum steady state coolant average temperature. Plant characteristics and initial conditions are further discussed in Section 15.0.3.

For the peak pressure evaluation, the initial pressure is conservatively set at 30 psi above nominal pressure (2,250 psia) to allow for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. To obtain the maximum pressure in the primary side, conservatively high loop pressure drops are added to the calculated pressurizer pressure.

Evaluation of the Pressure Transient

After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion is assumed to begin 1 second after the flow in the affected loop reaches 87 percent of nominal flow. No credit is taken for the pressure-reducing effect of the pressurizer relief valves, pressurizer spray, steam dump, or controlled feedwater flow after plant trip. Although these operations are expected to occur and would result in a lower peak pressure, an additional degree of conservatism is provided by ignoring their effect.

Evaluation of DNB in the Core During the Accident

For this accident, DNB is calculated to occur in the core and, therefore, an evaluation of the consequences with respect to fuel rod thermal transients is performed. Results obtained from analysis of this "hot spot" condition represent the upper limit with respect to clad temperature and zirconium-water reaction.

Film Boiling Coefficient

The film boiling coefficient is calculated in the FACTRAN code (Ref. 5), using the Bishop-Sandberg-Tong film boiling correlation. The fluid properties are evaluated at film temperature (average between wall and bulk temperatures). The program calculates the film coefficient at every time step, based upon the actual heat transfer conditions at the time. The nuclear power, system pressure, bulk density, and mass flow rate as a function of time are used as program input.

For this analysis, the initial values of the pressure and the bulk density are used throughout the transient, since they are the most conservative with respect to clad temperature response. For conservatism, DNB was assumed to start at the beginning of the accident.

Fuel Clad Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and clad (gap coefficient) have a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between pellet and clad. Based on investigations on the effect of the gap coefficient upon the maximum clad temperature during the transient, the gap coefficient was assumed to increase from a steady state value consistent with initial fuel temperature to 10,000 Btu/hr-ft^2-°F at the initiation of the transient. Thus, the large amount of energy stored in the fuel because of the small initial value is released to the clad at the initiation of the transient.

Zirconium-Steam Reaction

The zirconium-steam reaction can become significant above 1,800 °F (clad temperature). The Baker-Just parabolic rate equation shown below is used to define the rate of the zirconium-steam reaction.

$$\frac{d(w^2)}{dt} = 33.3 \times 10^6 \exp\left(-\frac{45500}{1.986T}\right)$$

where:

w = amount reacted, mg/cm^2
t = time, sec
T = temperature, K

The reaction heat is 1,510 cal/gm.

The effect of zirconium-steam reaction is included in the calculation of the "hot spot" clad temperature transient.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-15. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

Results and Conclusions

Figures 15.3-13, based on the Pressure Evaluation Model, and Figures 15.3-14 through 15.3-19, based on the Statistical Core Design Model, show the transient results for one locked rotor with and without offsite power with four loops in operation. The results of these calculations are also summarized in Table 15.3-2. The peak RCS pressure reached during the transient is less than 110% of the design pressure.

Also, the peak clad surface temperature is considerably less than 2,700 °F. It should be noted that the clad temperature was conservatively calculated, assuming that DNB occurs at the initiation of the transient.

The calculated sequence of events for the two cases analyzed is shown on Table 15.3-1. Figures 15.3-14 and 15.3-15 shows that the core flow rapidly reaches a new equilibrium value. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

15.3.3.3 Radiological Consequences

15.3.3.3.1 Method of Analysis

15.3.3.3.1.1 Physical Model

The instantaneous seizure of a reactor coolant pump rotor results in a reactor trip on a low coolant flow signal. With the coincident loss of offsite power, the condensers are not available, so the excess heat is removed from the secondary system by a steam dump through the steam generator safety and atmospheric relief valves. Steam generator tube leakage is assumed to continue until the pressures in the reactor coolant and secondary systems are equalized. The reactor coolant will contain the gap activities of the fraction of the fuel which undergoes DNB in addition to its assumed equilibrium activity.

15.3.3.3.1.2 Assumptions and Conditions

The major assumptions and parameters used in the analysis are itemized in Tables 15.3-3 and 15A-1 and summarized below.

The assumption used to determine the initial concentrations of isotopes in the reactor coolant and secondary coolant prior to the accident are as follows:

a. The reactor coolant iodine activity is based on the dose equivalent of 1.0 μCi/gm of I-131.

b. The noble gas activity in the reactor coolant is based on 1-percent failed fuel.

c. The secondary coolant activity is based on the dose equivalent of 0.1 μCi/gm of I-131.

The following conditions are used to calculate the activity released.

a. 5 percent of fuel rod gap activity is additionally released to the reactor coolant.

b. Offsite power is lost.

c. Following the incident, secondary steam is released to the environment for heat removal. The total quantity of steam released is given in Table 15.3-3.

d. Primary-to-secondary leakage continues after the accident for a period of 8 hours. At that time, reactor coolant and secondary system pressures are equalized. Until the pressure equalizes, the leakage rate is assumed to be constant and equal to the rate existing prior to the incident of 1 gpm (500 lbs/hr).

e. Fission products released from the fuel-cladding gap of the damaged fuel rods are assumed to be instantaneously and homogeneously mixed with the reactor coolant.

f. The noble gas activity released is equal to the amount present in the reactor coolant which leaks into the secondary system after the accident.

g. The iodine activity present in the primary to secondary leakage is assumed to mix homogeneously with the iodine activity initially present in the steam generators.

h. A partition factor of 0.01 between the vapor and liquid phases for radioiodine in the steam generators is utilized to determine iodine releases to the environment via steam venting from the steam generators.

i. The activity released from the steam generators is immediately vented to the environment.

j. No credit is taken for radioactive decay or ground deposition during radioactivity transport to offsite location.

k. Short-term accident atmospheric dispersion factors corresponding to ground level releases, breathing rates, and dose conversion factors are given in Table 15A-2.

15.3.3.3.1.3 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 the onsite meteorological measurement programs described in Section 2.3 and are provided in Table 15A-2.

c. The thyroid inhalation and total-body immersion doses to a receptor at the exclusion area boundary or outer boundary of the low-population zone were analyzed using the models described in Appendix 15A.

15.3.3.3.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

The leakage pathways are:

a. Direct steam dump to the atmosphere through the secondary system atmospheric relief and safety valves for the secondary steam

b. Primary-to-secondary steam generator tube leakage and subsequent steam dump to the atmosphere through the secondary system atmospheric relief and safety valves

Table 15.3-3 shows the total curies released.

15.3.3.3.2 Identification of Uncertainties and Conservative Elements in the Analysis

a. Reactor coolant and secondary coolant activities of 1.0 mCi/gm and 0.1 mCi/gm I-131 dose equivalent, respectively, are many times greater than assumed for normal operation conditions.

b. A 1-gpm steam generator primary-to-secondary leakage, which is significantly greater than that anticipated during normal operation, is assumed.

c. The coincident loss of offsite power with the occurrence of a reactor coolant pump locked rotor is a highly conservative assumption. In the event of the availability of offsite station power, the condenser steam dump valves will open, permitting steam dump to the condenser. Thus there is no direct release to the environment.

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 meteorological conditions assumed will 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.

15.3.3.3.3 Conclusions

15.3.3.3.3.1 Filter Loadings

The only ESF filtration system considered in the analysis which limits the consequences of the reactor coolant pump locked rotor accident is the control room filtration system. Activity loadings on the control room charcoal filter are based on the flow rate through the filter, the concentration of activity at the filter inlet, and the filter efficiency.

The activity in the control room filter as a function of time has been evaluated for the loss-of-coolant accident, Section 15.6.5. Since the control room filters are capable of accommodating the potential design-basis loss-of-coolant accident fission product iodine loadings, more than adequate design margin is available with respect to postulated reactor coolant pump locked rotor accident releases.

15.3.3.3.3.2 Doses to Receptor at the Exclusion Area Boundary and Low-Population Zone Outer Boundary

The potential radiological consequences resulting from the occurrence of a postulated reactor coolant pump locked rotor have been conservatively analyzed, using assumptions and models described in previous sections.

The total-body doses due to immersion from direct radiation and the thyroid dose due to inhalation have been analyzed for the 0-2 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 results are listed in Table 15.3-4. The resultant doses are well within the guideline values of 10 CFR 100.

15.3.3.4 Conclusions

a. Since the peak RCS pressure reached during any of the transients is less than that which would cause stresses to exceed the faulted condition stress limits, the integrity of the primary coolant system is not endangered.

b. Since the peak clad surface temperature calculated for the hot spot during the worst transient remains considerably less than 2,700 $^{\circ}$F, the core will remain in place and intact with no loss of core cooling capability.

15.3.4 REACTOR COOLANT PUMP SHAFT BREAK

15.3.4.1 Identification of Causes and Accident Description

The accident is postulated as an instantaneous failure of a reactor coolant pump shaft, as discussed in Section 5.4. Flow through the affected reactor coolant loop is rapidly reduced, though the initial rate of reduction of coolant flow is greater for the reactor coolant pump rotor seizure event. Reactor trip is initiated on a low flow signal in the affected loop.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced - first, because the reduced flow results in a decreased tube side film coefficient and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves, in that sequence. The two power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure-reducing effect, as well as the pressure-reducing effect of the spray, are not included in the analysis.

This event is classified as an ANS Condition IV incident (a limiting fault), as defined in Section 15.0.1.

15.3.4.2 Conclusions

The consequences of a reactor coolant pump shaft break are no worse than those calculated for the locked rotor incident (see Section 15.3.3). With a failed shaft, the impeller could conceivably be free to spin in a reverse direction as opposed to being fixed in position as assumed in the locked rotor analysis. However, the net effect on core flow is negligible, resulting in only a slight decrease in the end point (steady state) core flow. For both the shaft break and locked rotor incidents, reactor trip occurs very early in the transient. In addition, the locked rotor analysis conservatively assumes that DNB occurs at the beginning of the transient.

15.3.5 REFERENCES

1. McFadden, J. H., et. al., "RETRAN-02 - A Program for Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," EPRI NP-1850-CCM-A, October 1984.

2. Stewart, C. W., et. al.,"VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores," Battelle, Pacific Northwest Laboratories, EPRI NP-2511-CCM-A, August 1989.

3. Baldwin, M. S., Merrian, M. M., Schenkel, H. S. and VanDeWalle, D. J., "An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs," WCAP-8424, Revision 1, May 1975.

4. WCAP-11397-P-A, "Revised Thermal Design Procedure," A. J. Fiedland, et al., April 1989

5. Hargrove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO2 Fuel Rod," WCAP-7908A, December 1989.

TABLE 15.3-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

Accident	Event	Time (sec)
Partial loss of forced reactor coolant flow		
1. Loss of two pumps with four loops in operation	Coastdown begins	0.0
	Low flow reactor trip	2.21
	Rods begin to drop	3.21
	Minimum DNBR occurs	3.8
Complete loss of forced reactor coolant flow		
1. Loss of four pumps with four loops in operation	All operating pumps lose power and begin coasting down	0.0
	Reactor coolant pump undervoltage trip point reached	0.0
	Rods begin to drop	1.5
	Minimum DNBR occurs	3.5

TABLE 15.3-1 (Sheet 2)

Accident	Event	Time (sec)
Reactor coolant pump shaft seizure (locked rotor)		
1. One locked rotor with four loops in operation W/ offsite power	Rotor on one pump locks	0.0
	Low flow trip point reached	1.04
	Rods begin to drop	2.04
	Maximum RCS pressure occurs	3.8
2. One locked rotor with four loops in operation W/O offsite power	Rotor on one pump locks	0.0
	Low flow trip point reached	1.04
	Rods begin to drop	2.04
	Power lost to remaining pumps	3.04
	Maximum RCS pressure occurs	4.7

TABLE 15.3-2

SUMMARY OF RESULTS FOR LOCKED ROTOR TRANSIENTS

	With Offsite Power	Without Offsite Power
Maximum reactor coolant system pressure, psia	2580	2591
Maximum clad temperature, °F, core hot spot	1938	1942
$Zr-H_2O$ reaction at core hot spot, percent by weight	0.36	0.40

TABLE 15.3-3

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LOCKED ROTOR ACCIDENT

I.	Source Data	
	a. Power level, MWt	3,565
	b. Steam generator tube leakage, gpm	1
	c. Reactor coolant iodine activity	Dose equivalent of 1.0 μCi/gm of I-131
	d. Reactor coolant noble gas activity	Based on 1-percent failed fuel, as provided in Table 11.1-5
	e. Secondary system activity	Dose equivalent of 0.1 μCi/gm of I-131
	f. Activity released to reactor coolant from failed fuel	
	1. Noble gas, percent of gap inventory	5
	2. Iodine, percent of gap inventory	5
	3. Gap inventory	Table 15A-3
	g. Iodine partition factor for steam generators	0.01
	h. Reactor coolant mass, lbs	4.94E+5
	i. Steam generator mass, per generator, lbs	9.55E+4
II.	Atmospheric Dispersion Factors	see Table 15A-2
III.	Activity Release Data	
	a. Steam discharge, 0-2 hrs	
	1. Reactor coolant leakage, lbs	1,000
	2. Mass released from steam generators, lbs	5.49E+5
	b. Steam discharge, 2-8 hrs	
	1. Reactor coolant leakage, lbs	3,000
	2. Mass released from steam generators, lbs	1.03E+6

TABLE 15.3-3 (Sheet 2)

c. Activity released to the environment

Isotope	0-2 hr (Ci)	0-8 hr (Ci)
I-131	8.37	8.36E+1
I-132	6.86	2.63E+1
I-133	1.38E+1	1.23E+2
I-134	5.87	9.88
I-135	1.17E+1	8.15E+1
Xe-131m	1.17E+1	4.66E+1
Xe-133m	6.30E+1	2.42E+2
Xe-133	2.09E+3	8.24E+3
Xe-135m	6.99E+1	7.02E+1
Xe-135	4.45E+2	1.44E+3
Xe-137	8.02E+1	8.02E+1
Xe-138	2.81E+2	2.82E+2
Kr-83m	8.82E+1	1.58E+2
Kr-85m	2.33E+2	6.23E+2
Kr-85	3.52E+1	1.41E+2
Kr-87	3.19E+2	4.74E+2
Kr-88	5.85E+2	1.30E+3
Kr-89	3.43E+1	3.43E+1

TABLE 15.3-4

RADIOLOGICAL CONSEQUENCES OF A LOCKED ROTOR ACCIDENT

	Wolf Creek Dose (rem)
Exclusion Area Boundary (0-2 hr)	
Thyroid, rem	0.882
Whole body, rem	0.076
Low Population ZoneOuter Boundary (duration)	
Thyroid, rem	1.130
Whole body, rem	0.021

TABLE 15.3-5

THIS TABLE HAS BEEN DELETED

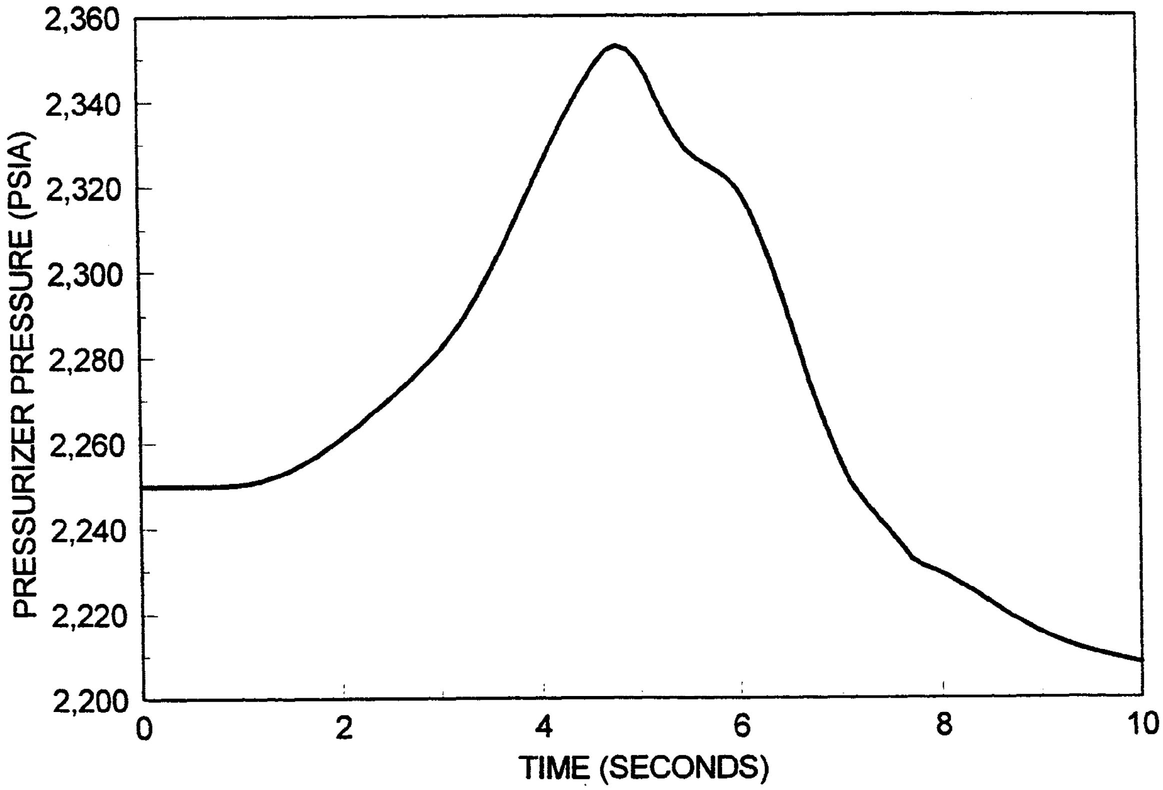


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FIGURE 15.3-1
PRESSURIZER PRESSURE TRANSIENT FOR FOUR LOOPS IN OPERATION, TWO PUMPS COASTING DOWN

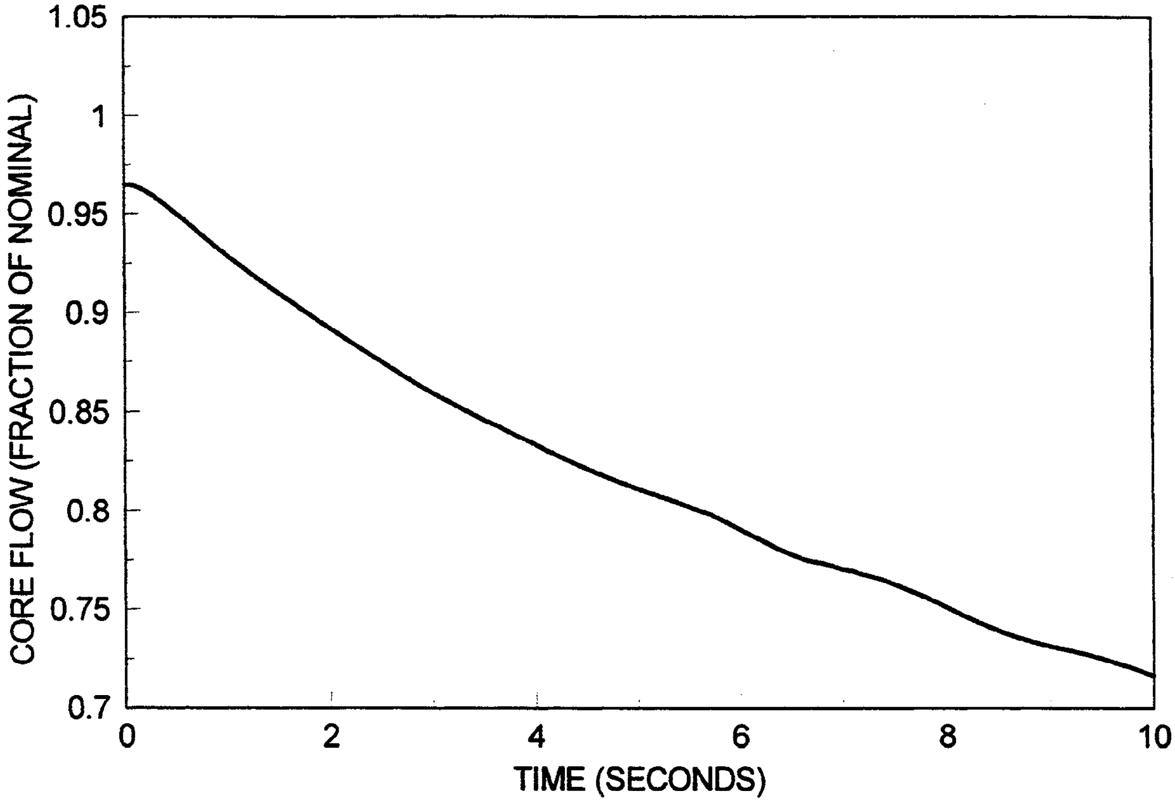

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FIGURE 15.3-2
TOTAL CORE FLOW TRANSIENT FOR FOUR LOOPS IN OPERATION, TWO PUMPS COASTING DOWN

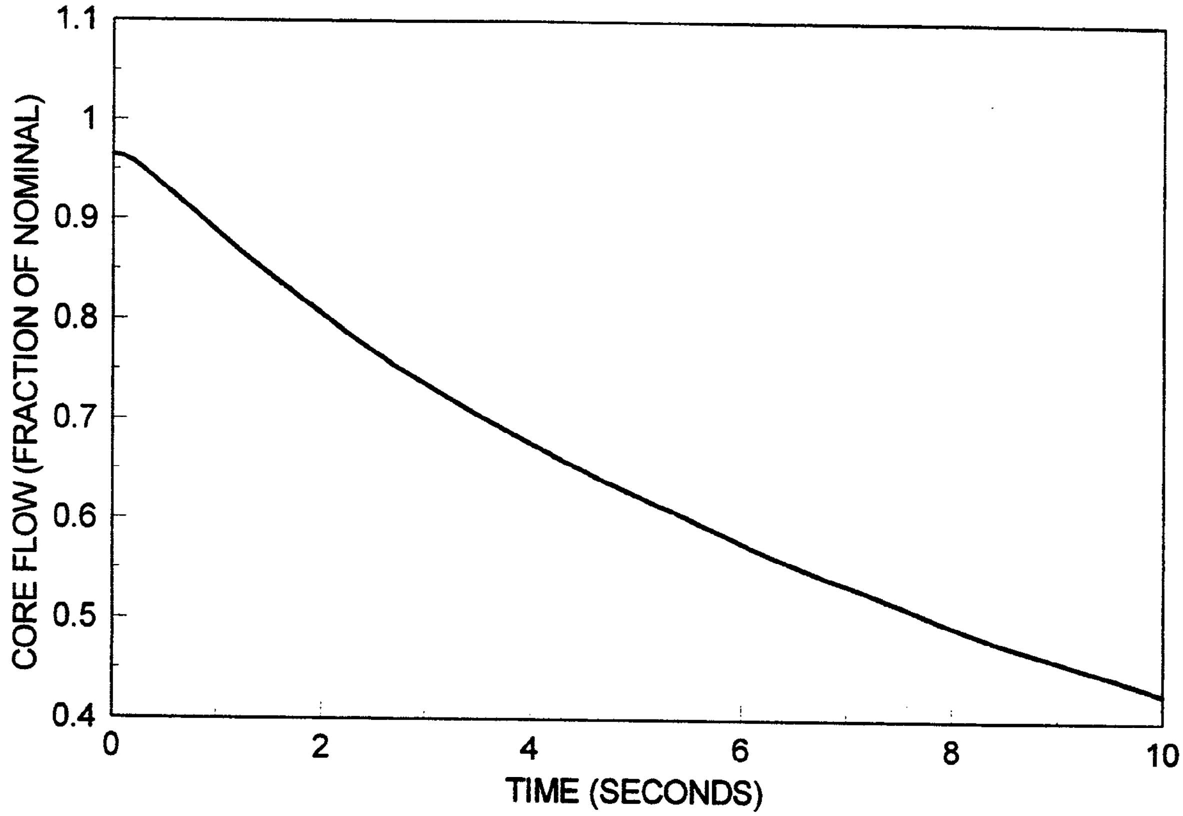

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FIGURE 15.3-3
FAULTED LOOP FLOW TRANSIENT FOR FOUR LOOPS IN OPERATION, TWO PUMPS COASTING DOWN

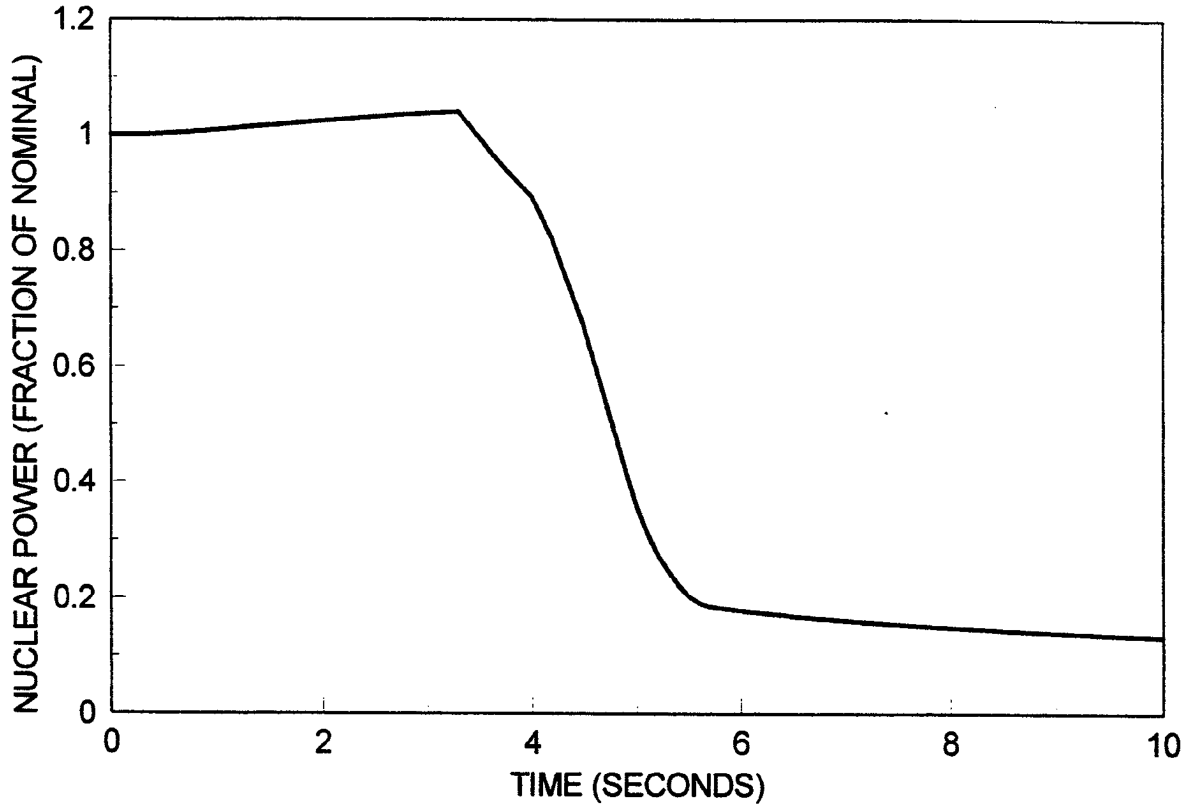

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FIGURE 15.3-4 NUCLEAR POWER TRANSIENT FOR FOUR LOOPS IN OPERATION, TWO PUMPS COASTING DOWN

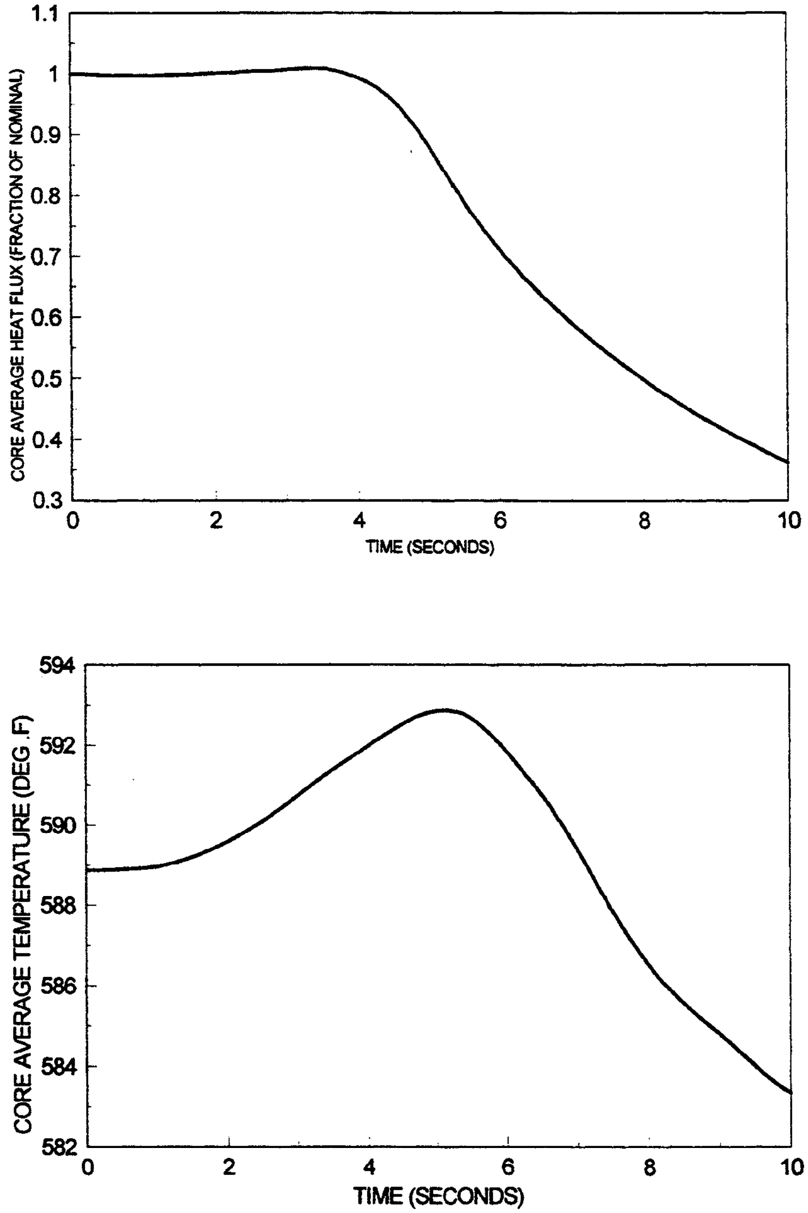

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FIGURE 15.3-5
CORE HEAT FLUX AND CORE AVERAGE TEMPERATURE FOR FOUR LOOPS IN OPERATION, TWO PUMPS COASTING DOWN

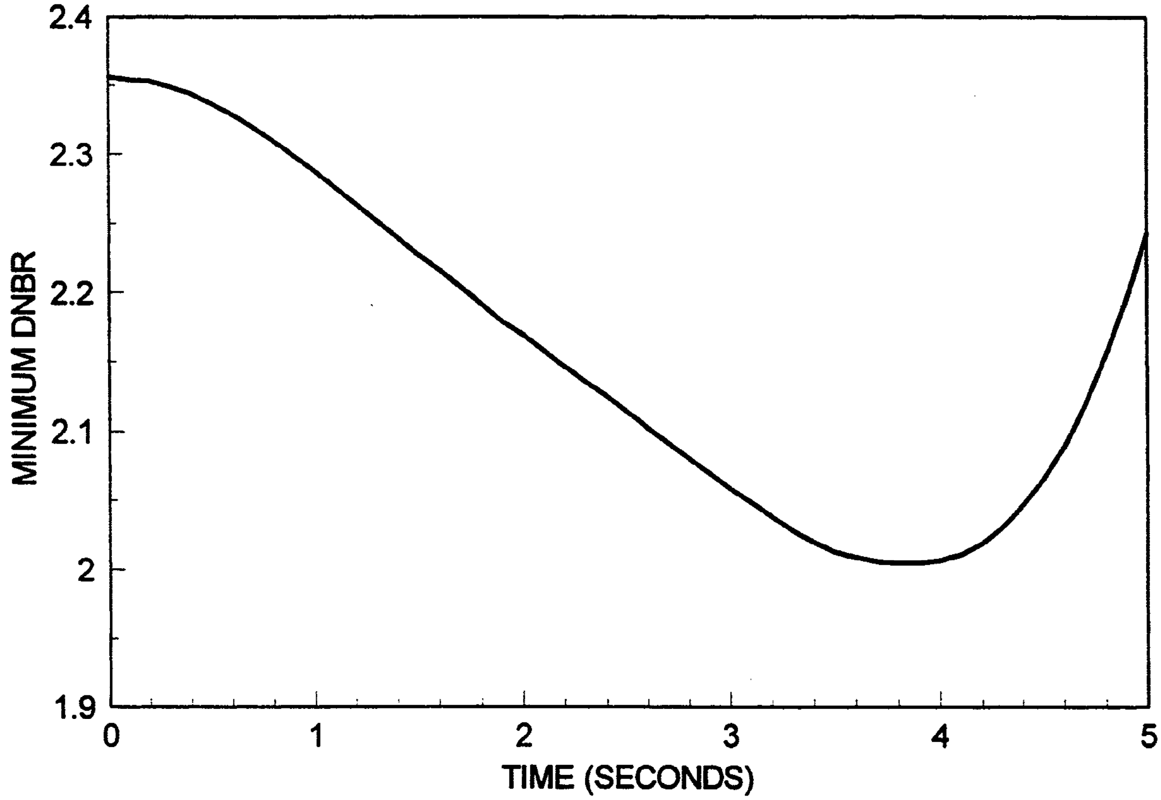

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FIGURE 15.3-6 DNBR TRANSIENT FOR FOUR LOOPS IN OPERATION, TWO PUMPS COASTING DOWN

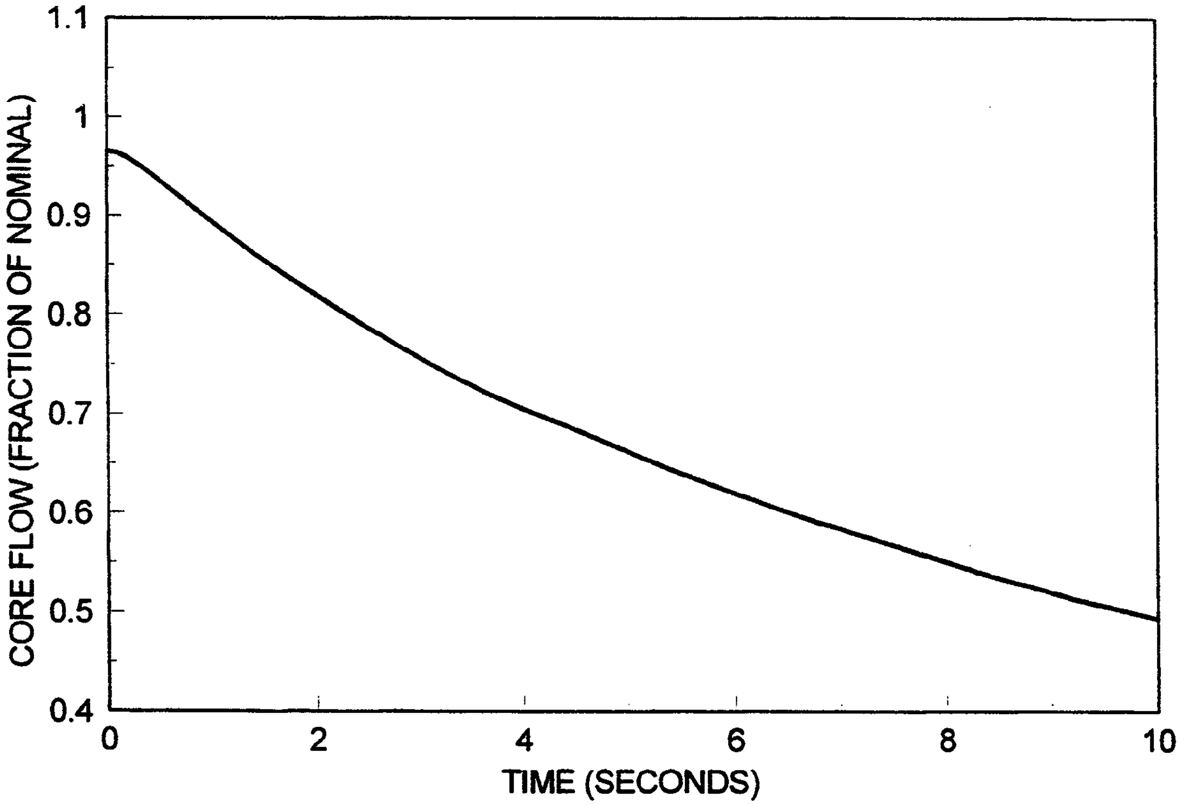

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FIGURE 15.3-7 TOTAL CORE FLOW TRANSIENT FOR FOUR LOOPS IN OPERATION, FOUR PUMPS COASTING DOWN

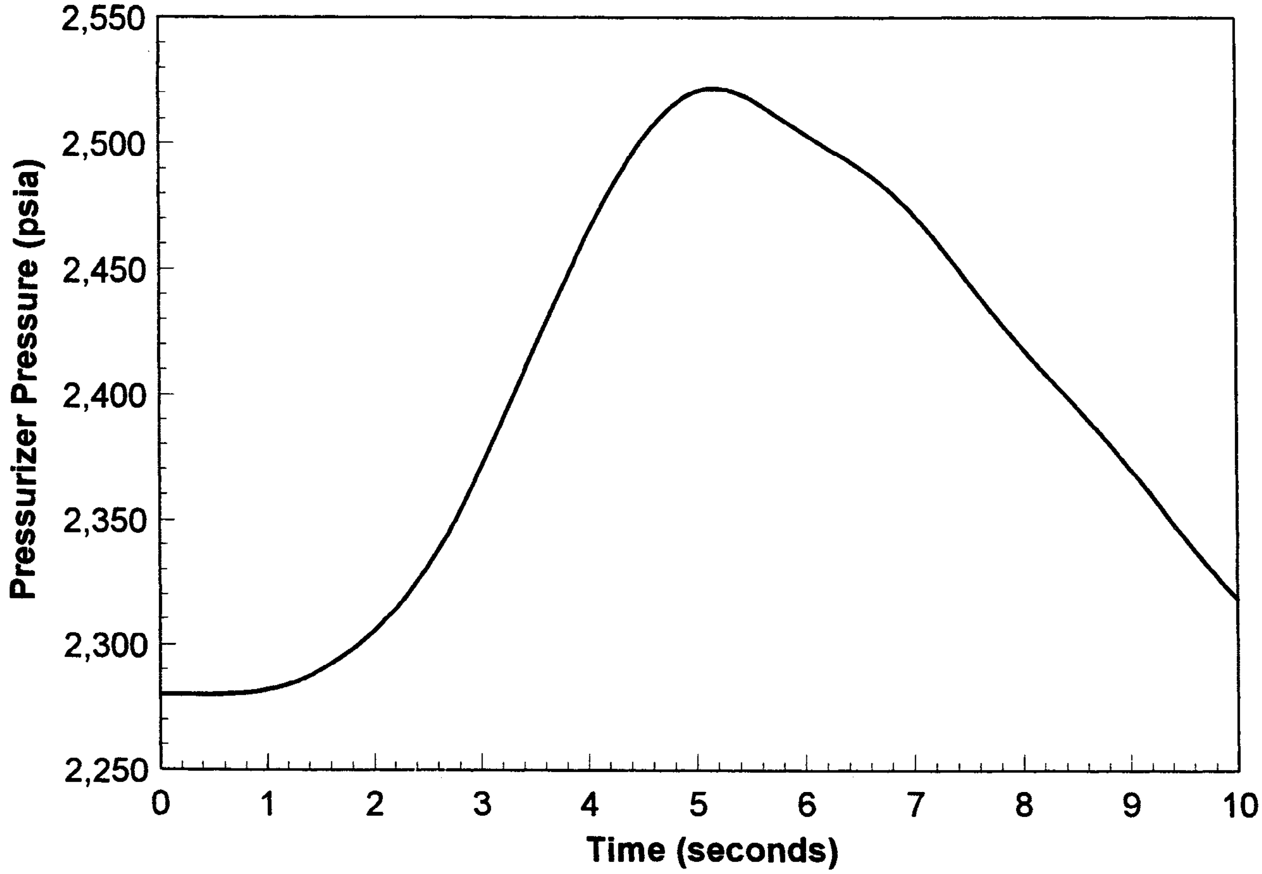

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FIGURE 15.3-8 PRESSURIZER PRESSURE TRANSIENT FOR FOUR LOOPS IN OPERATION, FOUR PUMPS COASTING DOWN

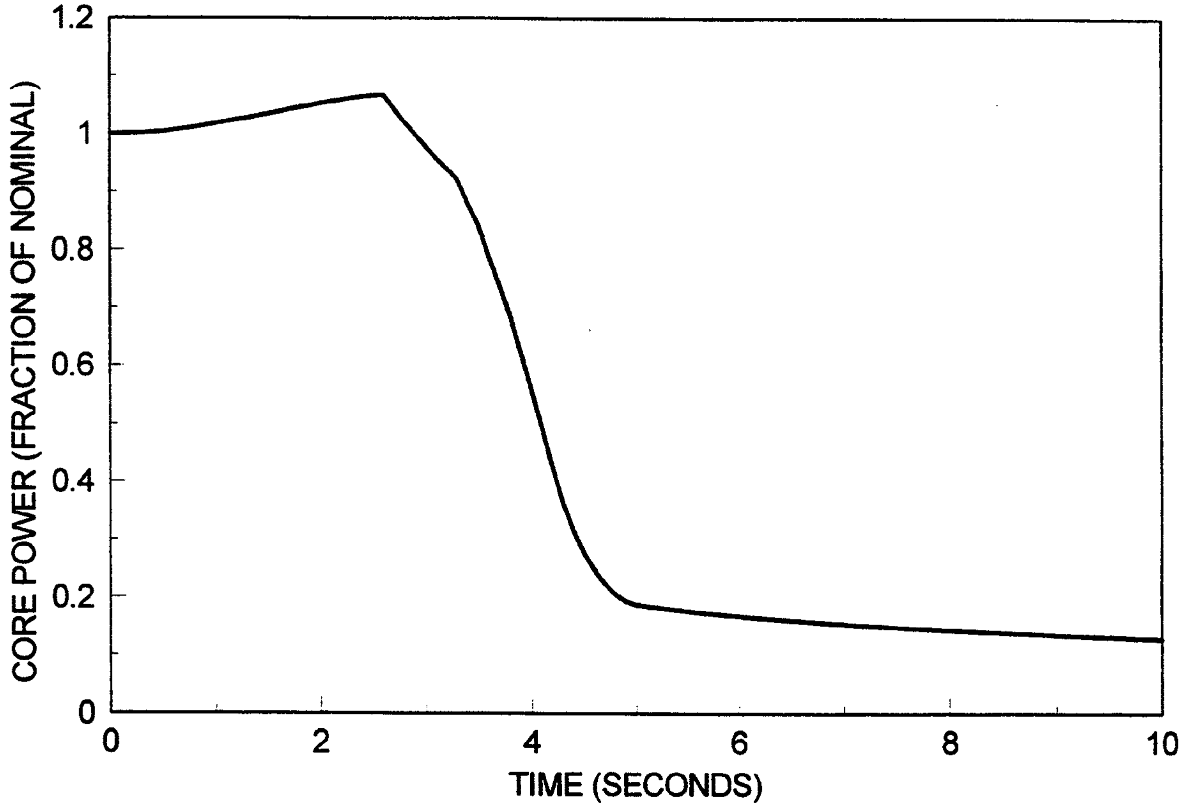

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FIGURE 15.3-9 NORMALIZED CORE POWER TRANSIENT FOR FOUR LOOPS IN OPERATION, FOUR PUMPS COASTING DOWN

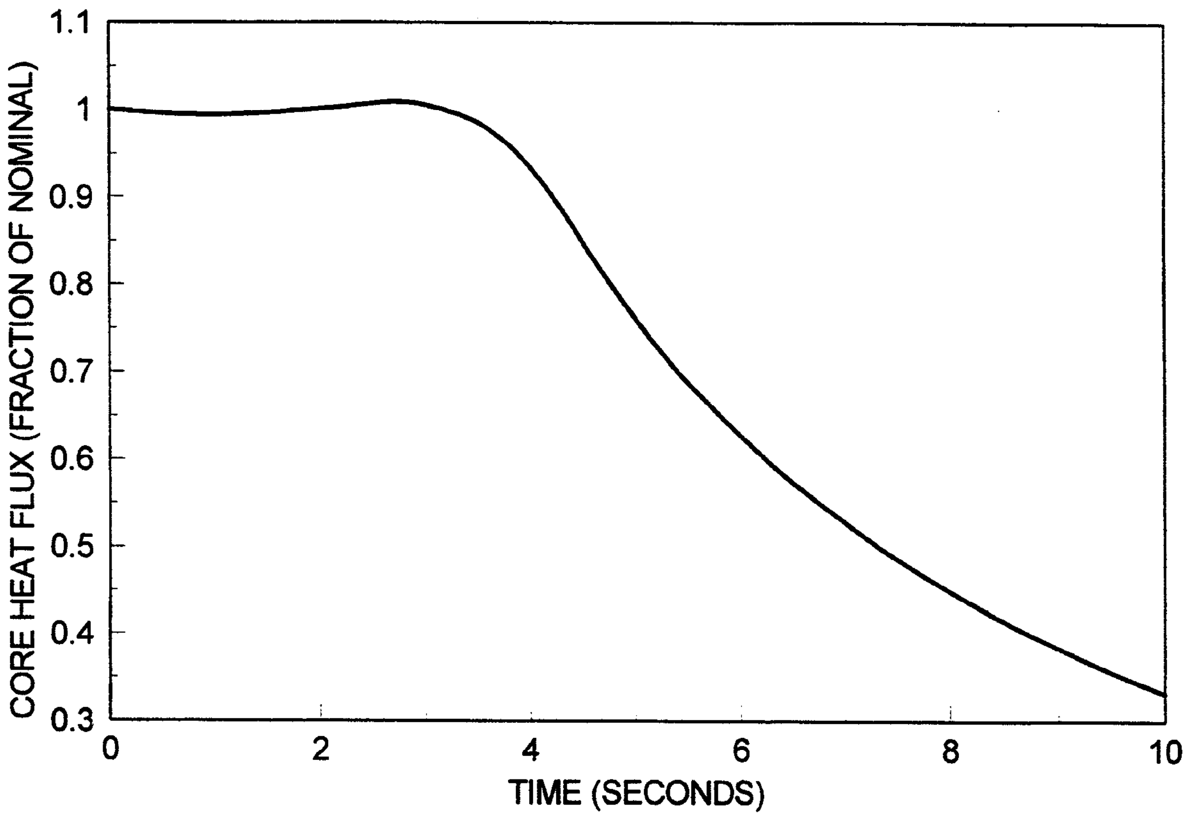

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FIGURE 15.3-10
CORE HEAT FLUX TRANSIENT FOR FOUR LOOPS IN OPERATION, FOUR PUMPS COASTING DOWN

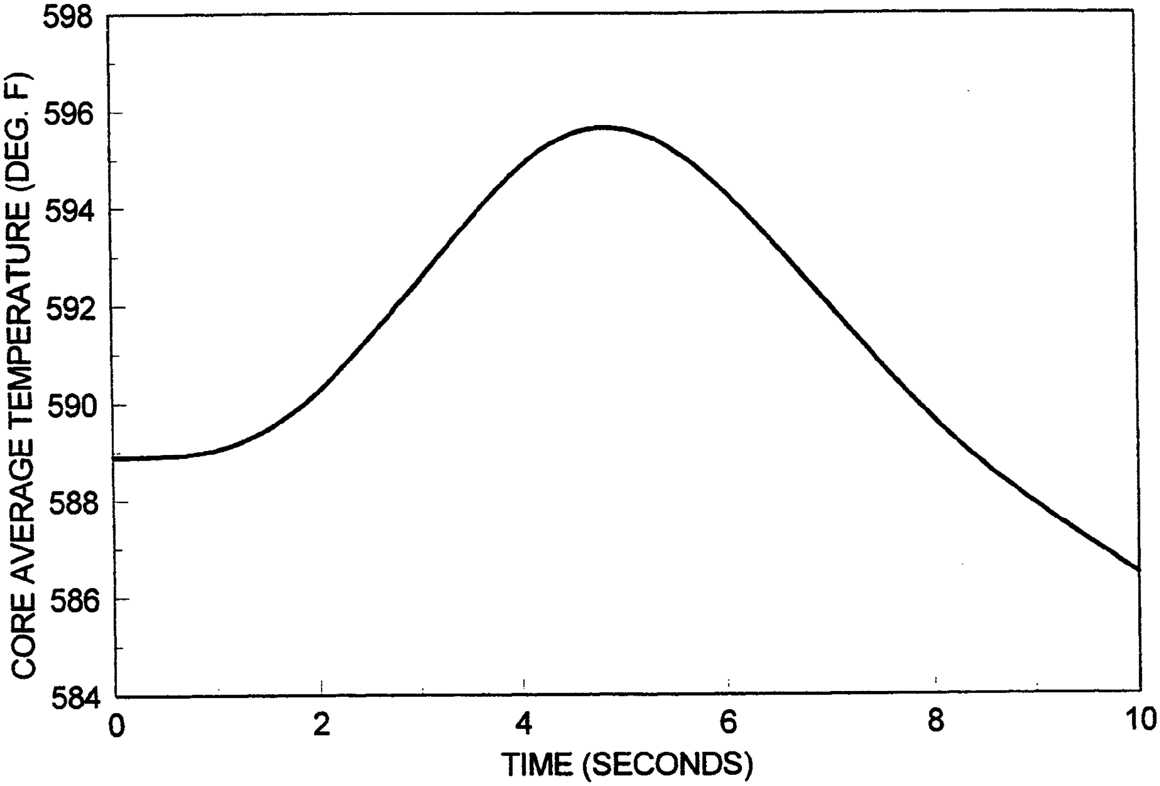

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FIGURE 15.3-11
CORE AVERAGE TEMPERATURE FOR FOUR LOOPS IN OPERATION, FOUR PUMPS COASTING DOWN

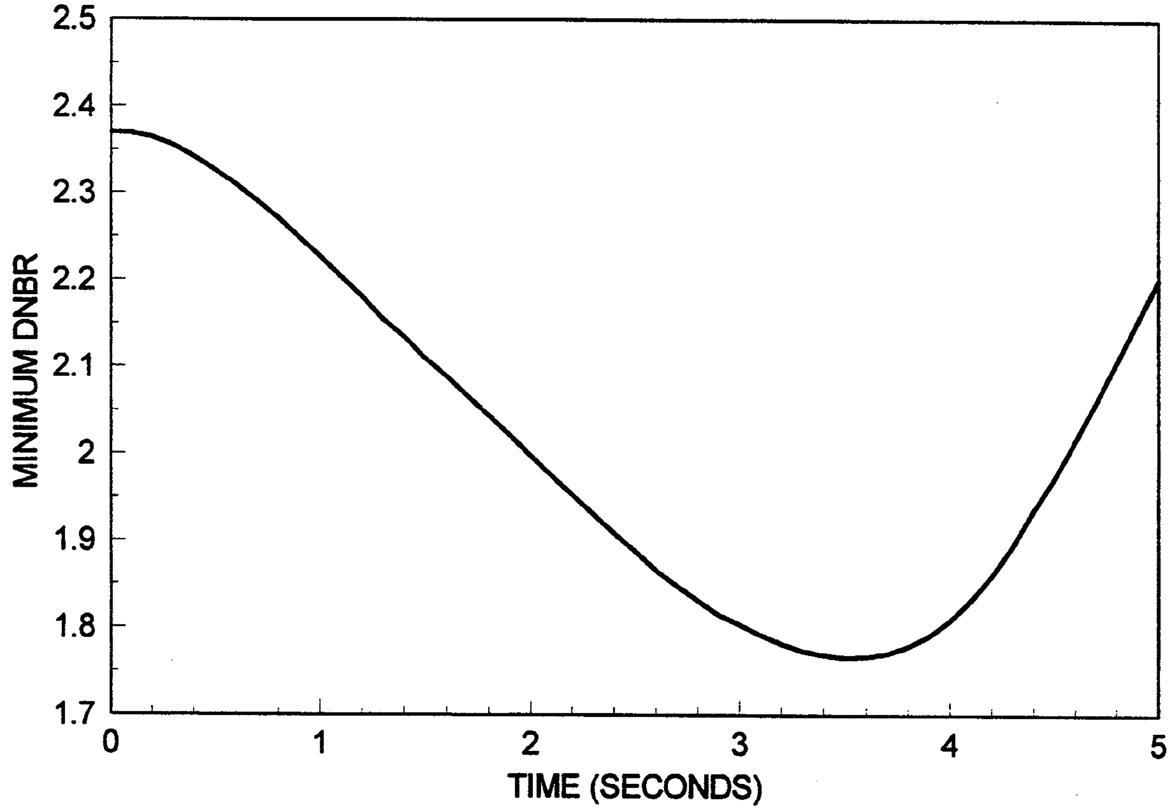

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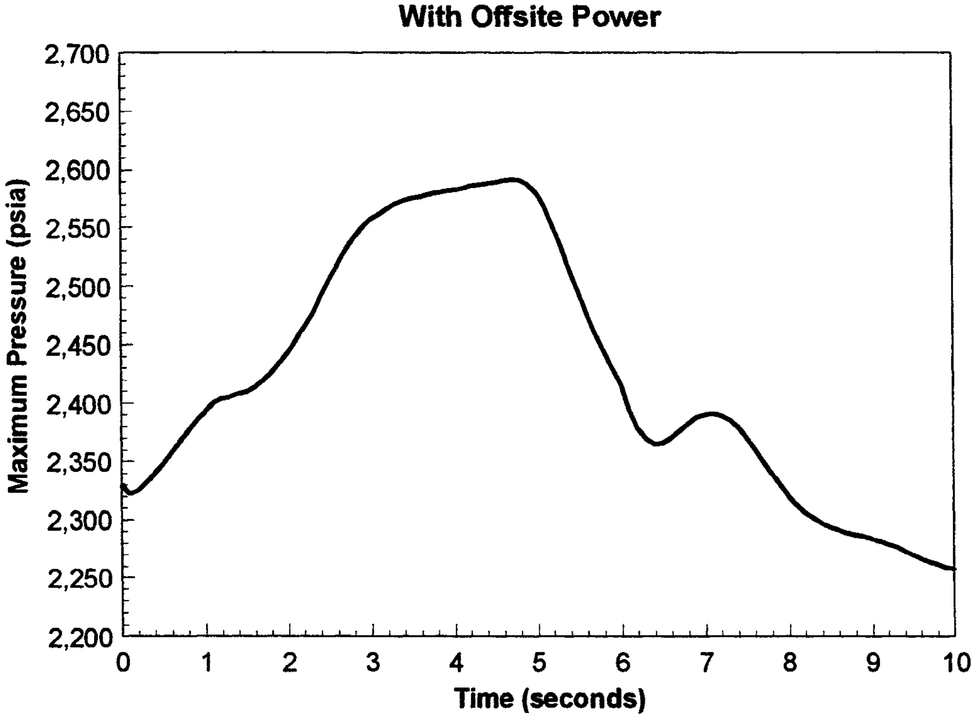
FIGURE 15.3-12 DNBR TRANSIENT FOR FOUR LOOPS IN OPERATION, FOUR PUMPS COASTING DOWN



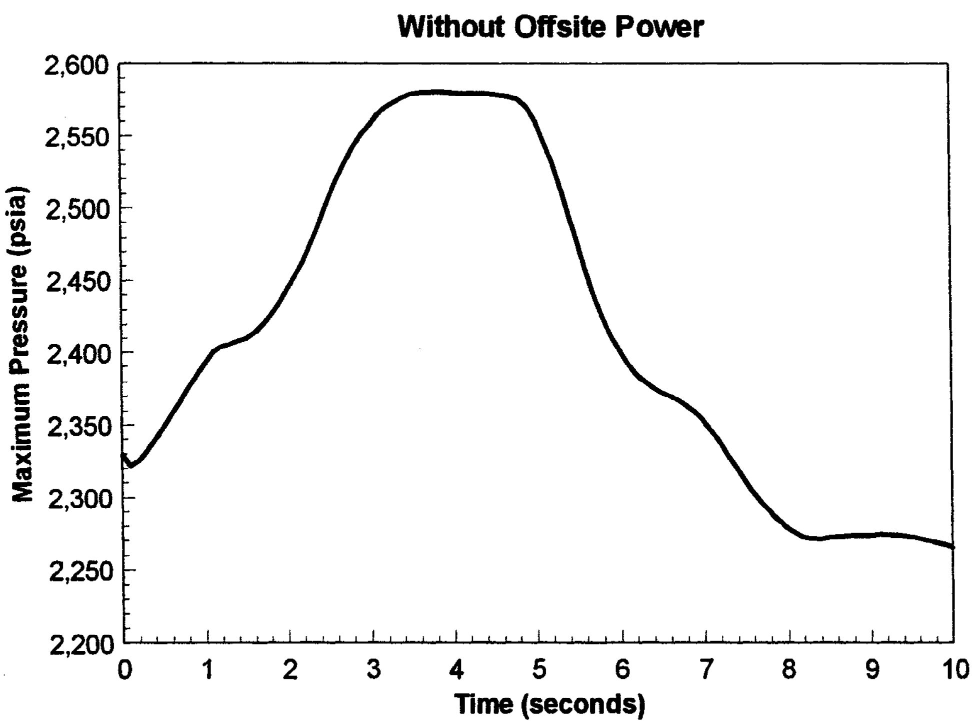


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FIGURE 15.3-13

PEAK RCS PRESSURE TRANSIENT
LOCKED ROTOR WITH AND WITHOUT
OFFSITE POWER

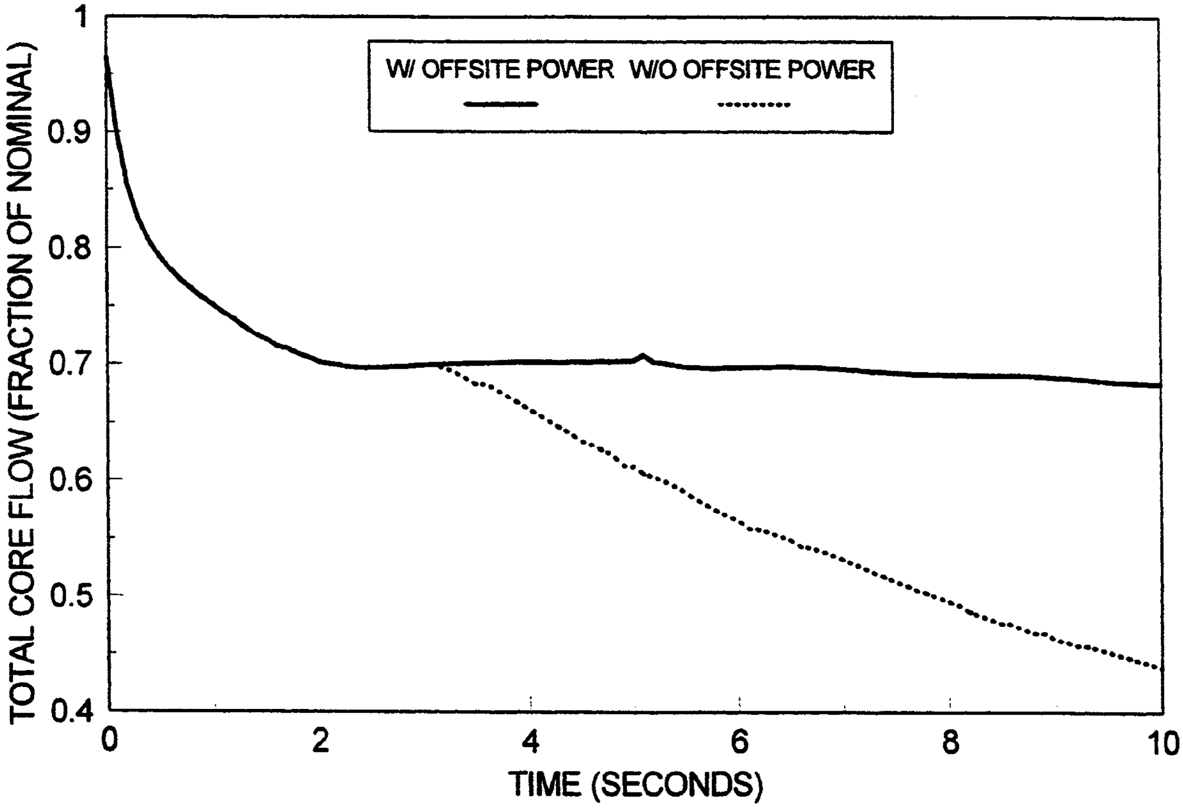

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FIGURE 15.3-14 TOTAL CORE FLOW TRANSIENT LOCKED ROTOR WITH AND WITHOUT OFFSITE POWER

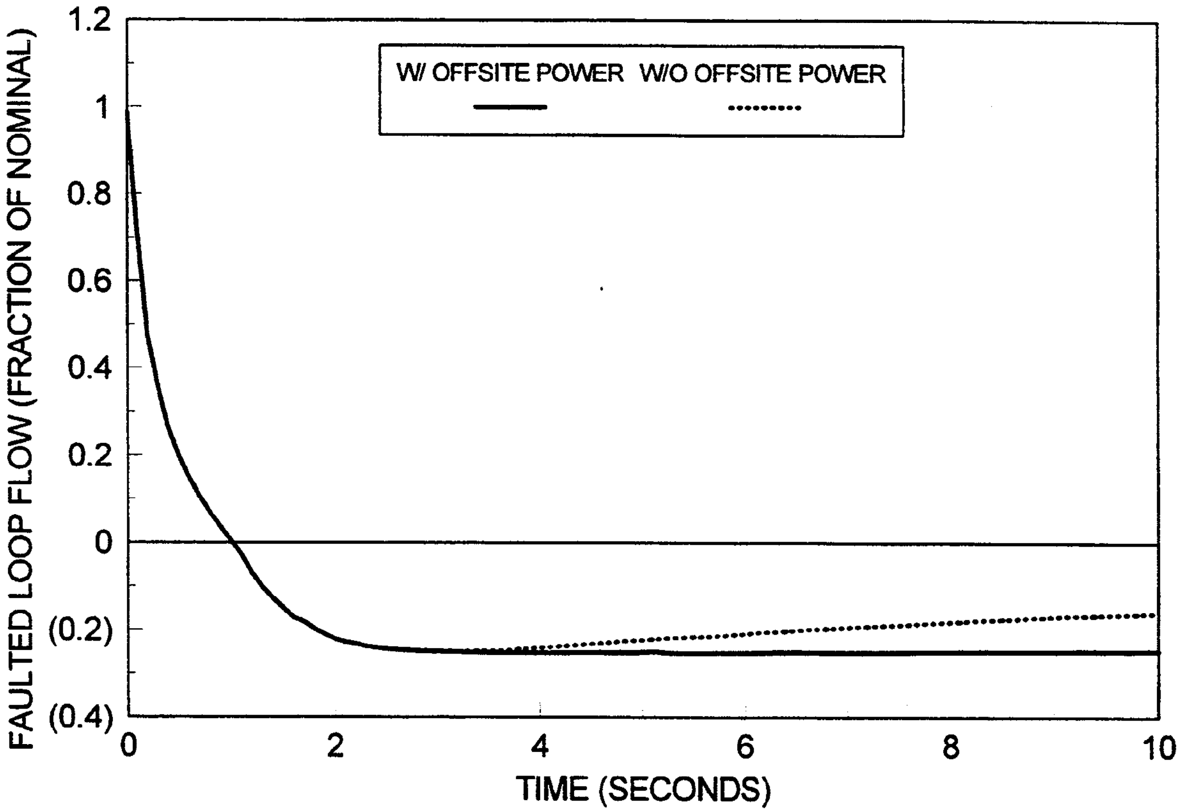

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FIGURE 15.3-15 FAULTED LOOP FLOW TRANSIENT LOCKED ROTOR WITH AND WITHOUT OFFSITE POWER

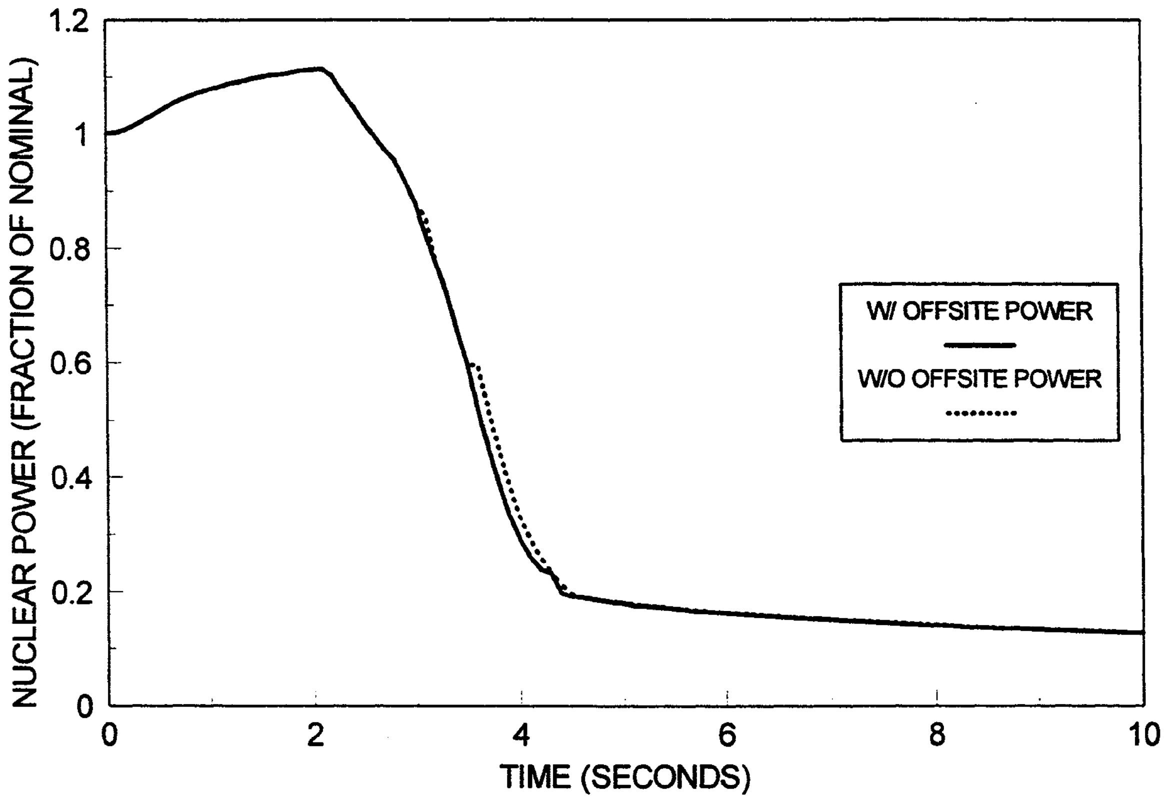

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FIGURE 15.3-16
NORMALIZED POWER TRANSIENT
LOCKED ROTOR WITH AND WITHOUT
OFFSITE POWER

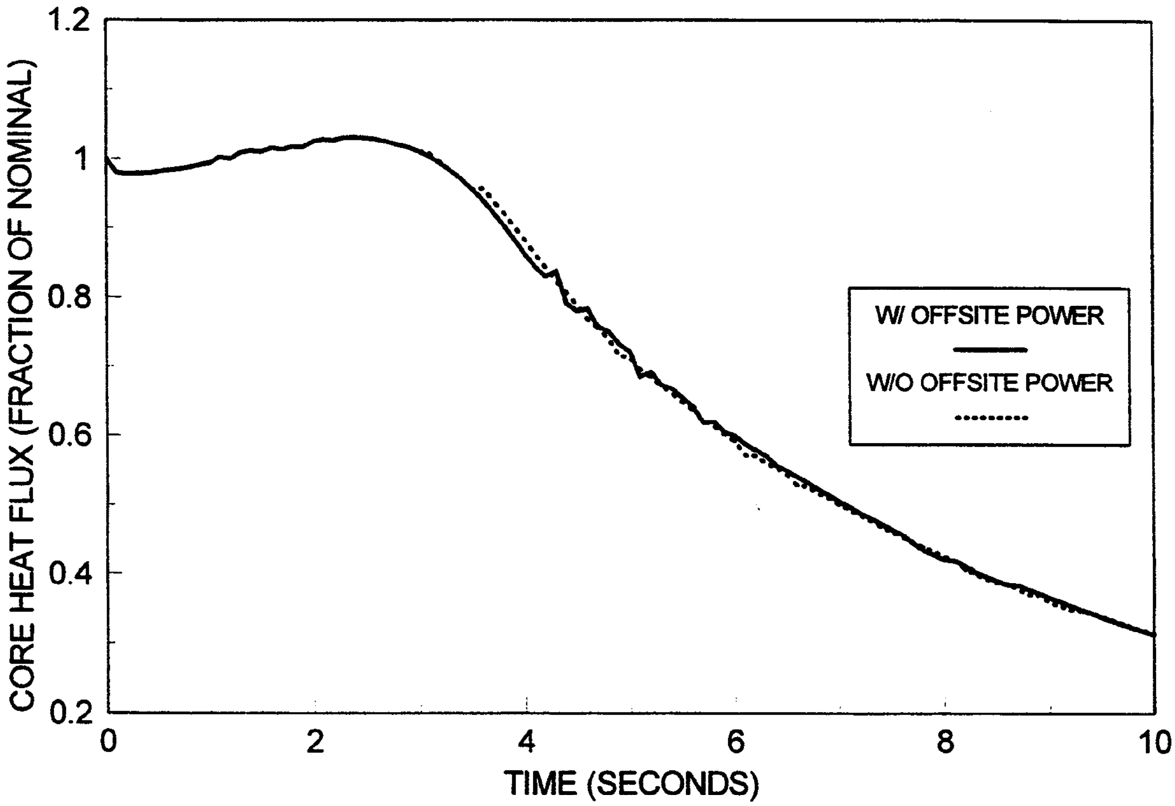

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FIGURE 15.3-17 CORE HEAT FLUX TRANSIENT LOCKED ROTOR WITH AND WITHOUT OFFSITE POWER

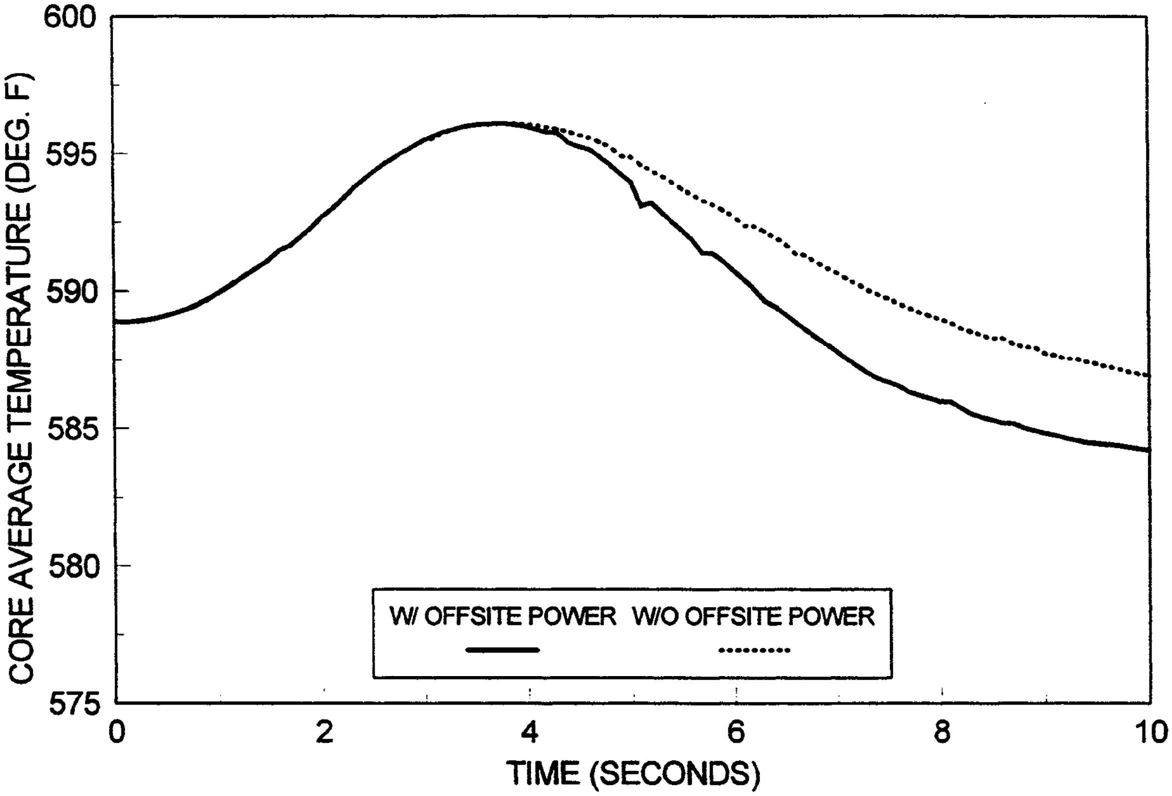

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FIGURE 15.3-18 CORE AVERAGE TEMPERATURE LOCKED ROTOR WITH AND WITHOUT OFFSITE POWER

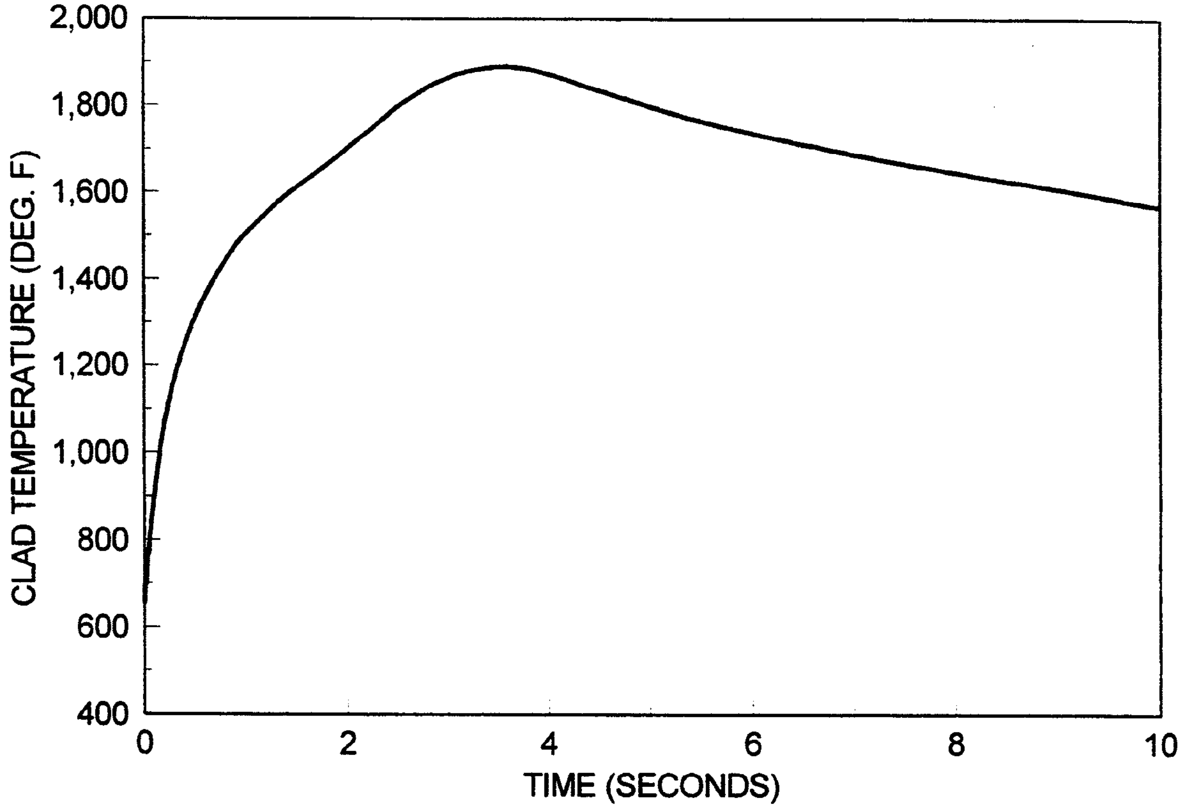

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FIGURE 15.3-19 OUTER CLAD TEMPERATURE FOR LOCKED ROTOR WITHOUT OFFSITE POWER

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

A number of faults have been postulated which could result in reactivity and power distribution anomalies. Reactivity changes could be caused by control rod motion or ejection, boron concentration changes, or addition of cold water to the reactor coolant system (RCS). Power distribution changes could be caused by control rod motion, misalignment, or ejection, or by static means such as fuel assembly mislocation. These events are discussed in this section. Detailed analyses are presented for the most limiting of these events.

Discussions of the following events are presented in this section:

a. Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition

b. Uncontrolled rod cluster control assembly bank withdrawal at power

c. Rod cluster control assembly misalignment

d. Startup of an inactive reactor coolant pump at an incorrect temperature

e. A malfunction or failure of the flow controller in a BWR recirculation loop that results in an increased reactor coolant flow rate (not applicable to Wolf Creek)

f. Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant

g. Inadvertent loading and operation of a fuel assembly in an improper position

h. Spectrum of rod cluster control assembly ejection accidents

Items a, b, d, and f above are considered to be ANS Condition II events; item g, an ANS Condition III event; and item h, an ANS Condition IV event. Item c entails both Conditions II and III events. Section 15.0.1 contains a discussion of ANS classifications.

All of the accidents in this section have been analyzed. It has been determined that the most severe radiological consequences result from the complete rupture of a control rod drive mechanism housing provided in Section 15.4.8. Therefore, radiological consequences are reported only for that limiting case.

15.4.1 UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER STARTUP CONDITION

15.4.1.1 Identification of Causes and Accident Description

A rod cluster control assembly (RCCA) withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCAs, resulting in a power excursion. Such a transient could be caused by a malfunction of the reactor control or rod control systems. This could occur with the reactor subcritical, at hot zero power or at power. The "at power" case is discussed in Section 15.4.2.

Although the reactor is normally brought to power from a subcritical condition by means of RCCA withdrawal, initial startup procedures with a clean core call for boron dilution. The maximum rate of reactivity increase in the case of boron dilution is less than that assumed in this analysis (see Section 15.4.6).

The RCCA drive mechanisms are wired into preselected bank configurations which are not altered during reactor life. These circuits prevent the RCCAs from being automatically withdrawn in other than their respective banks. Power supplied to the banks is controlled so that no more than two banks can be withdrawn at the same time and in their proper withdrawal sequence. The RCCA drive mechanisms are of the magnetic latch type, and coil actuation is sequenced to provide variable speed travel. The maximum reactivity insertion rate analyzed in the detailed plant analysis is that occurring with the simultaneous withdrawal of the combination of two sequential control banks having the maximum combined worth at maximum speed.

This event is classified as an ANS Condition II incident (a fault of moderate frequency), as defined in Section 15.0.1.

The neutron flux response to a continuous reactivity insertion is characterized by a very fast rise terminated by the reactivity feedback effect of the negative Doppler coefficient. This self-limitation of the power excursion is of primary importance, since it limits the power to a tolerable level during the delay time for protective action. Should a continuous RCCA withdrawal accident occur, the transient will be terminated by the following automatic features of the reactor protection system:

a. Source range high neutron flux reactor trip

This trip function is actuated when either of two independent source range channels indicates a neutron flux level above a preselected manually adjustable setpoint. It may be manually bypassed only after an intermediate range flux channel indicates a flux level above a specified level. It is automatically reinstated when both intermediate range channels indicate a flux level below a specified level.

b. Intermediate range high neutron flux reactor trip

This trip function is actuated when either of two independent intermediate range channels indicates a flux level above a preselected manually adjustable setpoint. It may be manually bypassed only after two out of the four power range channels are reading above approximately 10 percent of full power, and is automatically reinstated when three out of the four channels indicate a power level below this value.

c. Power range high neutron flux reactor trip (low setting)

This trip function is actuated when two out of the four power range channels indicate a power level above approximately 25 percent of full power. It may be manually bypassed when two out of the four power range channels indicate a power level above approximately 10 percent of full power, and is automatically reinstated only after three out of the four channels indicate a power level below this value.

d. Power range high neutron flux reactor trip (high setting)

This trip function is actuated when two out of the four power range channels indicate a power level above a preset setpoint. It is always active.

e. High nuclear flux rate reactor trip

This trip function is actuated when the positive rate of change of neutron flux on two out of four nuclear power range channels indicates a rate above the preset setpoint. It is always active.

In addition, control rod stops on high intermediate range flux level (one out of two) and high power range flux level (one out of four) serve to discontinue rod withdrawal and prevent the need to actuate the intermediate range flux level trip and the power range flux level trip, respectively.

15.4.1.2 Analysis of Effects and Consequences

Method of Analysis

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in three stages: first an average core nuclear power transient calculation, then an average core heat transfer calculation, and finally the departure from nucleate boiling ratio (DNBR) calculation. The average core nuclear calculation is performed, using spatial neutron kinetics methods, TWINKLE (Ref. 1), to determine the average power generation with time, including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. In the second stage, the average heat flux and temperature transients are determined by performing a fuel rod transient heat transfer calculation in FACTRAN (Ref. 2). VIPRE-01 (Ref 14) is used to evaluate the core thermal limits to determine DNBR.

Plant characteristics and initial conditions are discussed in Section 15.0.3. To give conservative results for a startup accident, the following assumptions are made:

a. Since the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler coefficient, conservatively low values are used. This does not correlate to Figure 15.0-2 since the TWINKLE computer code, on which the neutronics analysis is based, is a diffusion theory code rather than a point-kinetics approximation.

b. Contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time between the fuel and the moderator is much longer than the neutron flux response time. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. A conservative value is used in the analysis to yield the maximum peak heat flux.

c. The reactor is assumed to be at hot zero power (557 F average coolant temperature). This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a larger fuel-water heat transfer coefficient, larger specific heats, and a less negative (smaller absolute magnitude) Doppler coefficient, all of which tend to reduce the Doppler feedback effect, thereby increasing the neutron flux peak. The initial effective multiplication factor (k_{eff}) is assumed to be 1.0, since this results in the worst nuclear power transient.

d. Reactor trip is assumed to be initiated by power range high neutron flux (low setting). The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and RCCA release, is taken into account. A 10-percent error increase is assumed for the power range flux trip setpoint, raising it from the nominal value of 25 percent to 35 percent. Since the rise in the neutron flux is so rapid, the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition, the reactor trip insertion characteristic is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. See Section 15.0.5 for RCCA insertion characteristics.

e. The maximum positive reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the combination of the two sequential control banks having the greatest combined worth at maximum speed (45 inches/minute). Control rod drive mechanism design is discussed in Section 4.6.

f. The most limiting axial and radial power shapes, associated with having the two highest combined worth banks in their high worth position, are assumed in the DNB analysis.

g. The initial power level is assumed to be below the power level expected for any shutdown condition (10^{-9} of nominal power). The combination of highest reactivity insertion rate and lowest initial power produces the highest peak heat flux.

h. Two reactor coolant pumps are assumed to be in operation.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-16.

Results

Figures 15.4-1 through 15.4-3 show the transient behavior for the uncontrolled RCCA bank withdrawal accident, with the accident terminated by reactor trip at 35 percent of nominal power. The reactivity insertion rate used is greater than that calculated for the two highest worth sequential control banks, both assumed to be in their highest incremental worth region.

Figure 15.4-1 shows the average neutron flux transient.

The energy release and the fuel temperature increases are relatively small. The thermal flux response, of interest for DNB considerations, is shown on Figure 15.4-2. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux much less than the full power nominal value. There is a large margin to DNB during the transient, since the rod surface heat flux remains below the design value, and there is a high degree of subcooling at all times in the core. Figure 15.4-3 shows the response of the hot spot average fuel and inner cladding temperature. The average fuel temperature increases to a value lower than the nominal full power value. The minimum DNBR at all times remains above the limiting value.

The calculated sequence of events for this accident is shown in Table 15.4-1. With the reactor tripped, the plant returns to a stable condition. The plant may subsequentlÿ be cooled down further by following normal plant shutdown procedures.

15.4.1.3 Conclusions

In the event of an RCCA withdrawal accident from the subcritical condition, the core and the RCS are not adversely affected, since the combination of thermal power and the coolant temperature result in a DNBR greater than the limiting value. The DNBR design basis is described in Section 4.4; applicable acceptance criteria have been met.

15.4.2 UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL AT POWER

15.4.2.1 Identification of Causes and Accident Description

Uncontrolled RCCA bank withdrawal at power results in an increase in the core heat flux. Since the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in DNB. Therefore, to avert damage to the fuel clad the reactor protection system is designed to terminate any such transient before the DNBR falls below the safety analysis limit.

This event is classified as an ANS Condition II incident (a fault of moderate frequency), as defined in Section 15.0.1.

The automatic features of the reactor protection system which prevent core damage following the postulated accident include the following:

a. Power range neutron flux instrumentation actuates a reactor trip if two out of four channels exceed any overpower setpoint.

b. A reactor trip actuated if any two out of four ΔT channels exceed the overtemperature ΔT setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature, and pressure to protect against DNB.

c. A reactor trip actuated if any two out of four DT channels exceed the overpower DT setpoint. This setpoint is automatically varied with axial power imbalance to ensure that the allowable heat generation rate (kW/ft) is not exceeded.

d. A high pressurizer pressure reactor trip actuated from any two out of four pressure channels which are set at a fixed point. This set pressure is less than the set pressure for the pressurizer safety valves.

e. A high pressurizer water level reactor trip actuated from any two out of three level channels when the reactor power is above approximately 10 percent (Permissive 7).

f. A reactor trip actuated from any two out of four steam generator low-low level channels falling below the trip setpoint.

In addition to the above listed reactor trips, there are the following RCCA withdrawal blocks:

a. High neutron flux (one out of four power range)

b. Overpower ΔT (two out of four)

c. Overtemperature ΔT (two out of four)

The manner in which the combination of overpower and overtemperature ΔT trips provide protection over the full range of RCS conditions is described in Chapter 7.0. Figure 15.0-1 presents allowable reactor coolant loop average temperature and ΔT for the design power distribution and flow as a function of primary coolant pressure. The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT 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 utility of this diagram is in the fact that the limit imposed by any given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the safety analysis limit. All points below and to the left of a DNB line for a given pressure have a DNBR greater than the safety analysis limit. The diagram shows that DNB is prevented 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 set-point); low pressure (fixed setpoint); and overpower and overtemperature ΔT (variable setpoints).

15.4.2.2 Analysis of Effects and Consequences

Method of Analysis

The Rod Withdrawal at Power RWAP transients are analyzed by employing the detailed digital computer program RETRAN-02 (Ref. 13). RETRAN-02 has been found acceptable by the NRC for use as a licensing basis safety analysis code. RETRAN-02 is a thermal-hydraulic systems analysis code employing a one-dimensional, homogeneous equilibrium mixture thermal-hydraulic model for the RCS, a point kinetics model for the reactor core, special component and auxiliary models (ex., pumps, temperature transport, non-equilibrium pressurizer) and control system models. The code computes pertinent plant transient information including core power level, RCS pressure and temperature. VIPRE-01 (Ref. 14) is used to evaluate the core thermal limits to determine DNBR. RETRAN-02 generated state points are used as VIPRE-01 boundary conditions to perform a Statistical Core Design (SCD) DNB analysis.

Plant characteristics and initial conditions are discussed in Section 15.0.3. To obtain conservative results for an uncontrolled RCCA bank withdrawal at power accident, the following assumptions are made:

a. Initial conditions of nominal core power, reactor coolant temperatures, and reactor coolant pressure are assumed to perform an SCD DNB analysis (see section 15.0.3).

   Initial conditions of maximum core power, reactor coolant temperature, and reactor coolant pressure are assumed to perform an RCS overpressurition analysis (see section 15.0.3).

b. Reactivity coefficients - two cases are analyzed:

   1. Minimum reactivity feedback

      A most positive moderator coefficient of reactivity is assumed, corresponding to the beginning of core life. A variable Doppler power coefficient with core power is used in the analysis. A conservatively small (in absolute magnitude) value is assumed.

   2. Maximum reactivity feedback

      A conservatively large positive moderator density coefficient and a large (in absolute magnitude) negative Doppler power coefficient are assumed.

c. The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118 percent of nominal full power. The ΔT trips include all adverse instrumentation and setpoint errors; the delays for trip actuation are assumed to be the maximum values.

d. The RCCA trip insertion characteristic is based on the assumption that the highest worth assembly is stuck in its fully withdrawn position.

e. A range of reactivity insertion rates is examined. The maximum positive reactivity insertion rate is greater than that for the simultaneous withdrawal of the combination of the two control banks, having the maximum combined worth at maximum speed.

The effect of RCCA movement on the axial core power distribution is accounted for by causing a decrease in overtemperature ΔT trip setpoint proportional to a decrease in margin to DNB.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-16.

Results

Figures 15.4-4 through 15.4-6 show the transient response for a slow RCCA withdrawal incident starting from full power. Reactor trip on high neutron flux occurs shortly after the start of the accident. Since this is rapid with respect to the thermal time constants of the plant, small changes in Tavg and pressure result, and margin to DNB is maintained. The design basis for DNBR is described in Section 4.4.

Figure 15.4-7 shows the minimum DNBR as a function of reactivity insertion rate from intial full power operation for minimum and maximum reactivity feedback. It can be seen that two reactor trip channels provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and overtemperature ΔT channels.

Figures 15.4-8 and 15.4-9 show the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents starting at 60- and 10-percent power, respectively. The results are similar to the 100-percent power case, except as the initial power is decreased, the range over which the overtemperature ΔT trip is effective is increased. In neither case does the DNBR outside thermal limits.

The shape of the curves of minimum DNBR versus reactivity insertion rate in the referenced figures is due both to reactor core and coolant system transient response and to protection system action in initiating a reactor trip.

Referring to Figures 15.4-7 and 15.4-8, for example, it is noted that:

a. For high reactivity insertion rates, reactor trip is initiated by the high neutron flux trip for both reactivity feedback cases. The neutron flux level in the core rises rapidly for these insertion rates while core heat flux and coolant system temperature lag behind due to the thermal capacity of the fuel and coolant system fluid. Thus, the reactor is tripped prior to significant increase in heat flux or water temperature with resultant high minimum DNBRs during the transient. As reactivity insertion rate decreases, core heat flux and coolant temperatures can remain more nearly in equilibrium with the neutron flux; minimum DNBR during the transient thus decreases with decreasing insertion rate.

b. The overtemperature ΔT reactor trip circuit initiates a reactor trip when measured coolant loop ΔT exceeds a setpoint based on measured RCS average temperature and pressure. This trip circuit is described in detail in Chapter 7.0; however, it is important in this context to note that the average temperature contribution to the circuit is lead-lag compensated to decrease the effect of the thermal capacity of the RCS in response to power increases.

c. With further decrease in reactivity insertion rate, the overtemperature ΔT and high neutron flux trips become equally effective in terminating the transient.

   The effectiveness of the overtemperature ΔT trip increases at the lower in sertion rates (in terms of increased minimum DNBR) due to the fact that with lower insertion rates the power increase rate is slower, the rate of rise of average coolant temperature is slower, and the system lags and delays become less significant.

Figures 15.4-7, 15.4-8, and 15.4-9 illustrate minimum DNBRs calculated for minimum and maximum reactivity feedback.

Since the RCCA withdrawal at power incident is an overpower transient, the fuel temperatures rise during the transient until after reactor trip occurs. For high reactivity insertion rates, the overpower transient is fast with respect to the fuel rod thermal time constant, and the core heat flux lags behind the neutron flux response. Due to this lag, the peak core heat flux does not exceed 118 percent of its nominal value (i.e., the high neutron flux trip setpoint assumed in the analysis). Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel temperature will still remain below the fuel melting temperature.

For slow reactivity insertion rates, the core heat flux remains more nearly in equilibrium with the neutron flux. The overpower transient is terminated by the overtemperature ΔT reactor trip before a DNB condition is reached. The peak heat flux again is maintained below 118 percent of its nominal value. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak clad centerline temperature will remain below the fuel melting temperature.

The reactor is tripped sufficiently fast during the RCCA bank withdrawal at power transient so that the ability of the primary coolant to remove heat from the fuel rods is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

The calculated peak RCS pressure is approximately 2680 psia, which is less than 110% of the design pressure (2750 psia).

The calculated sequence of events for this accident is shown on Table 15.4-1. With the reactor tripped, the plant eventually returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

15.4.2.3 Conclusions

The high neutron flux and overtemperature ΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates, i.e., the minimum value of DNBR is always larger than the safety analysis limit.

15.4.3 ROD CLUSTER CONTROL ASSEMBLY MISOPERATION (System Malfunction or Operator Error)

15.4.3.1 Identification of Causes and Accident Description

Rod cluster control assembly (RCCA) misoperation accidents include:

a. One or more dropped RCCAs within the same group, or a dropped RCCA bank

b. Statically misaligned RCCA

c. Withdrawal of a single RCCA

Each RCCA has a position indicator channel which displays the position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by a rod at bottom signal, which actuates a local alarm and a control room annunciator. Group demand position is also indicated.

RCCAs are always moved in preselected banks, and the banks are always moved in the same preselected sequence. Each bank of RCCAs is divided into groups of four or five. The RCCAs comprising a group operate in parallel through multiplexing thyristors. The groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation of the stationary gripper, movable gripper, and lift coils of a mechanism) is required to withdraw the RCCA attached to the mechanism. Since the stationary gripper, movable gripper, and lift coils associated with the four RCCAs of a rod group are driven in parallel, any single failure which would cause rod withdrawal would affect a minimum of one group. Mechanical failures are in the direction of insertion or immobility.

The dropped RCCA assemblies, dropped RCCA assembly bank, and statically misaligned RCCA assembly events are classified as ANS Condition II incidents (incidents of moderate frequency), as defined in Section 15.0.1. The single RCCA withdrawal incident is classified as an ANS Condition III event, as discussed below.

No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single RCCA from the inserted bank at full power operation. The operator could withdraw a single RCCA in the control bank since this feature is necessary in order to retrieve an assembly should one be accidentally dropped. The event analyzed must result from multiple wiring failures [probability for single random failure is on the order of 10^{-4}/year (refer to Section 7.7.2.2)] or multiple significant operator errors and subsequent and repeated operator disregard of event indication. The probability of such a combination of conditions is considered low. The limiting consequences of such errors or failures may include slight fuel damage.

Thus, consistent with the philosophy and format of ANSI N18.2, the event is classified as a Condition III event. By definition "Condition III occurrences include incidents, any one of which may occur during the lifetime of a particular plant," and "shall not cause more than a small fraction of fuel elements in the reactor to be damaged . . ."

This classification is in accordance with GDC-25 which states, "The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any <u>single</u> malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control <u>rods</u>." (Emphasis has been added). It has been shown that single failures resulting in RCCA bank withdrawals do not violate specified fuel design limits. Moreover, no single malfunction can result in the withdrawal of a single RCCA. Thus, it is concluded that the criterion established for the single rod withdrawal at power is appropriate and in accordance with GDC-25.

A dropped RCCA or RCCA bank is detected by:

a. Sudden drop in the core power level as seen by the nuclear instrumentation system

b. Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples

c. Rod at bottom signal

d. Rod deviation alarm (control rods only)

e. Rod position indication

Misaligned RCCAs are detected by:

a. Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples

b. Rod deviation alarm (control rods only)

c. Rod position indication

The resolution of the rod position indicator channel is +12 steps (+7.5 inches). Deviation of any RCCA from its group by twice this distance (+24 steps or +15.0 inches) will not cause power distributions worse than the design limits. The deviation alarm (control rods only) alerts the operator to rod deviation with respect to the group position in excess of 12 steps. If the rod deviation alarm is not operable, the operator is required to take action required by the Technical Requirements Manual.

If one or more rod position indicator channels should be out of service, detailed operating instructions shall be followed to assure the alignment of the nonindicated RCCAs. The operator is also required to take action, as required by the Technical Specifications.

In the extremely unlikely event of simultaneous electrical failures which could result in single RCCA withdrawal, rod deviation and rod control urgent failure would both be displayed on the plant annunciators, and the rod position indicators would indicate the relative positions of the assemblies in the bank. The urgent failure alarm also inhibits automatic rod motion in the group in which it occurs. Withdrawal of a single RCCA by operator action, whether deliberate or by a combination of errors, would result in activation of the same alarm and the same visual indications. Withdrawal of a single RCCA results in both positive reactivity insertion tending to increase core power and an increase in local power density in the core area associated with the RCCA. Automatic protection for this event is provided by the overtemperature ΔT reactor trip, although due to the increase in local power density it is not possible in all cases to provide assurance that the core safety limits will not be violated.

Major plant systems and equipment available for mitigation of transient and accident conditions which may be required to function to mitigate the effects of the various control rod misoperations and discussed in Section 15.0.8 and listed in 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figures 15.0-17 & 18.

15.4.3.2 Analysis of Effects and Consequences

15.4.3.2.1 Dropped RCCAs, Dropped RCCA Bank, and Statically Misaligned RCCA

Method of Analysis

a. One or more dropped RCCAs from the same group or bank:

The purpose of this calculation is to demonstrate that, for all dropped RCCA worths considered, the post-drop minimum DNBR will be greater than the design limit. Credit is taken for high negative flux rate trip in that no combination of dropped RCCAs is considered which has a worth greater than 400 pcm. For evaluation of the dropped RCCA event, various combinations of dropped RCCAs (patterns) are considered. The initial conditions prior to RCCA drop are assumed to be: full power, RCCA control system on automatic, bank D inserted to its full-power limit, and all other RCCAs full out.

Calculations are performed with the code DRPROD (Reference 15). Plant and cycle specific values of several conditions are verified to be within appropriate ranges for use of the Westinghouse generic systems transient analysis. Generic systems transient statepoints programmed into the DRPROD code cover the ranges of reactivity change mechanisms (RCCA drop, control bank withdrawal, MTC) such that all appropriate combinations of these parameters throughout cycle life are considered.

Maximum reactivity insertion due to bank D withdrawal from HFP RIL is determined at 150 MWD/T and EOL (HFP equilibrium xenon) by ANC calculation (Reference 16). DRPROD code uses these values and linear interpolation to determine maximum reactivity insertion due to bank D withdrawal at burnups between 150 MWD/T and EOL. HFP MTC values are determined at BOL, EOL, and at the burnup for which MTC is at a maximum (least negative) for the cycle; these are determined by ANC calculation. These reactivity and MTC values are input to DRPROD. Other plant-specific and cycle-specific input to DRPROD calculations as outlined in Reference 15 includes HFP moderator inlet temperature and core average moderator temperature, nuclear uncertainty on $F_{\Delta H}$, dropped rod limit lines and design $F_{\Delta H}$. Dropped rod limit lines show, as a function of RCS pressure and inlet temperature, the power at which minimum DNBR equals the design limit.

b. Statically misaligned RCCA:

Cases of fully misaligned RCCAs are covered by the analyses described above in Part a. Consequences of cases of partially misaligned RCCAs are bounded by the analyses for Part a.

Results

a. One of more dropped RCCAs from the same group or bank:

Due to negative flux rate trip protection, the minimum dropped RCCA worth which will cause reactor trip is 400 pcm (Reference 15). Dropped RCCAs with worths greater than this value will result in reactor trip and need not be included in the power distribution analysis.

Output from DRPROD includes a table of $F_{\Delta H}$ versus burnup; these $F_{\Delta H}$ are the values calculated to be necessary as pre-drop $F_{\Delta H}$ if the post-drop transient is to produce a minimum DNBR which is less than the design limit. DRPROD code is run and output is checked to confirm that, for each of the analysized patterns of dropped RCCAs, the precondition $F_{\Delta H}$ which would be required in order for the dropped RCCA event with bank D withdrawal to violate minimum DNBR, is greater than the design limit. This demonstrates that the DNB design basis is not violated in any dropped RCCA event since necessary preconditions would not exist.

Dropped RCCA transient analysis parameters are presented in Figures 15.4-10 and 15.4-11a.

b. Statically misaligned RCCA

The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which one RCCA is fully inserted or where bank D is fully inserted with one RCCA fully withdrawn. Multiple independent alarms, including a bank insertion limit alarm, alert the operator well before the postulated conditions are approached. The bank can be inserted to its insertion limit with any one assembly fully withdrawn without the DNBR falling below the limit value.

The insertion limits in the Core Operating Limits Report (COLR) may vary from time to time, depending on a number of limiting criteria. The full power insertion limits on control bank D must be chosen to be above that position which meets minimum DNBR and peaking factors. The full power insertion limit is usually dictated by other criteria. Detailed results will vary from cycle to cycle, depending on fuel arrangements.

For this RCCA misalignment with bank D inserted to its full power insertion limit and one RCCA fully withdrawn, DNBR does not fall below the limit value. This case is analyzed assuming the initial reactor power, pressure, and RCS temperature are at their nominal values including uncertainties (as given in Subsection 15.0.3.2), but with the increased radial peaking factor associated with the misaligned RCCA.

For RCCA misalignments with one RCCA fully inserted, the DNBR does not fall below the limit value. This case is analyzed assuming that the initial reactor power, pressure, and RCS temperatures are at their nominal values, including uncertainties, but with the increased radial peaking factor associated with the misaligned RCCA.

DNB does not occur for the RCCA misalignment incident, and thus, the ability of the primary coolant to remove heat from the fuel is not reduced. The peak fuel temperature corresponds to a linear heat generation rate based on the radial peaking factor penalty associated with the misaligned RCCA and the design axial power distribution. The resulting linear heat generation is well below that which would cause fuel melting.

Following the identification of an RCCA group misalignment condition by the operator, the operator is required to take action as required by the plant Technical Specifications and operating instructions.

15.4.3.2.2 Single RCCA Withdrawal

Method of Analysis

Power distributions within the core are calculated using the computer code ANC. Analysis of the redistribution of power which occurs with the withdrawal of a single RCCA determines the fraction of pins for which DNBR falls below the limit value. The case of the worst rod withdrawn from bank D inserted at the insertion limit, with the reactor initially at full power, is analyzed. The purpose of this calculation is to confirm that the number of fuel rods that go through DNB is less than the safety analysis limit of 5%.

The calculations is performed at the burnup step which has the highest peak $F_{\Delta H}$. Power distributions are generated for all combinations of bank D inserted less one bank D RCCA out using xenon reconstruction to skew the axial offset to upper allowable limit. This determines the bank D RCCA which produces the highest peak assembly powers when stuck out of the core.

Results

ANC analysis is done at the burnup step and with the stuck bank D RCCA as mentioned above. If the highest peak $F_{\Delta H}$ from ANC results, multiplied by a calculational uncertainly factor of 1.08, is less than the $F_{\Delta H}$ design limit, then all rods meet the DNB criteria. If this is not the case, ANC analysis is repeated with the generation of a fuel census including the uncertainly factor of 1.08. From this census the percentage of fuel rods with $F_{\Delta H}$ exceeding the design limit is determined.

15.4.3.3 Conclusions

For cases of dropped RCCAs or dropped banks for which the reactor is tripped by the power range negative neutron flux rate trip, there is no reduction in the margin to core thermal limits, and consequently, the DNB design basis is met. For all cases that do not result in reactor trip, it is shown that the DNBR remains greater than the limit value and, therefore, the DNB design is met.

For all cases of any RCCA fully inserted, or bank D inserted to its rod insertion limits with any single RCCA in that bank fully withdrawn (static misalignment), the DNBR remains greater than the limit value.

For the case of the accidental withdrawal of a single RCCA, with the reactor in the automatic or manual control mode and initially operating at full power with bank D at the insertion limit, an upper bound of the number of fuel rods experiencing DNB is 5 percent of the total fuel rods in the core.

15.4.4 STARTUP OF AN INACTIVE REACTOR COOLANT PUMP AT AN INCORRECT TEMPERATURE

15.4.4.1 Identification of Causes and Accident Description

If the plant is operating with one pump out of service, there is reverse flow through the inactive loop due to the pressure difference across the reactor vessel. The cold leg temperature in an inactive loop is identical to the cold leg temperature of the active loops (the reactor core inlet temperature). If the reactor is operated at power, and assuming the secondary side of the steam generator in the inactive loop is not isolated, there is a temperature drop across the steam generator in the inactive loop and, with the reverse flow, the hot leg temperature of the inactive loop is lower than the reactor core inlet temperature.

Administrative procedures require that the unit be brought to a load of less than 10 percent of full power prior to starting the pump in an inactive loop in order to bring the inactive loop hot leg temperature closer to the core inlet temperature. Starting of an idle reactor coolant pump without bringing the inactive loop hot leg temperature close to the core inlet temperature would result in the injection of cold water into the core, which would cause a reactivity insertion and subsequent power increase.

This event is classified as an ANS Condition II incident (a fault of moderate frequency) as defined in Section 15.0.1.

Should the startup of an inactive reactor coolant pump accidentally occur, the ensuing transient is terminated automatically by a reactor trip on low coolant loop flow when the power range neutron flux (two out of four channels) exceeds the P-8 setpoint, which has been previously reset for three-loop operation.

15.4.4.2 Analysis of Effects and Consequences

Method of Analysis

This transient is analyzed by using three digital computer codes. The LOFTRAN code (Ref. 3) is used to calculate the loop and core flow, nuclear power and core pressure, and temperature transients following the startup of an idle pump. VIPRE-01 (Ref 14) is used to evaluate the core thermal limits to determine DNBR. LOFTRAN generated state points are used as VIPRE-01 boundary conditions to perform the DNB analysis.

Plant characteristics and initial conditions are discussed in Section 15.0.3. To obtain conservative results for the startup of an inactive pump accident, the following assumptions are made:

a. Initial conditions of maximum core power and reactor coolant average temperatures and minimum reactor coolant pressure resulting in minimum initial margin to DNB. These values are consistent with maximum steady state power level allowed with three loops in operation. The high initial power gives the greatest temperature difference between the core inlet temperature and the inactive loop hot leg temperature.

b. Following initiation of startup of the idle pump, the inactive loop flow reverses and accelerates to its nominal full flow value in approximately 27 seconds.

c. A conservatively large negative moderator temperature coefficient.

d. A least negative Doppler only power coefficient (see Figure 15.0-2).

e. The initial reactor coolant loop flows are at the appropriate values for one pump out of service.

f. The reactor trip is assumed to occur on low coolant loop flow when the power range neutron flux exceeds the P-8 setpoint. The P-8 setpoint is conservatively assumed to be 84 percent of rated power which corresponds to the nominal setpoint plus 9 percent for nuclear instrumentation errors.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0-8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-19.

Results

The results following the startup of an idle pump with the above listed assumptions are shown in Figures 15.4-12 through 15.4-16. As shown in these curves, during the first part of the transient, the increase in core flow with cooler water results in an increase in nuclear power and a decrease in core average temperature. The minimum DNBR during the transient is considerably greater than the safety analysis limit. See Section 4.4 for a description of the DNBR design basis.

Reactivity addition for the inactive loop startup accident is due to the decrease in core water temperature. During the transient, this decrease is due both to the increase in reactor coolant flow and, as the inactive loop flow reverses, to the colder water entering the core from the hot leg side (colder temperature side prior to the start of the transient) of the steam generator in the inactive loop. Thus, the reactivity insertion rate for this transient changes with time. The resultant core nuclear power transient, computed with consideration of both moderator and Doppler reactivity feedback effects, is shown on Figure 15.4-12.

The calculated sequence of events for this accident is shown on Table 15.4-1. The transient results illustrated in Figures 15.4-12 through 15.4-16 indicate that a stabilized plant condition, with the reactor tripped, is rapidly attained. Plant cooldown may subsequently be achieved by following normal shutdown procedures.

15.4.4.3 Conclusions

The transient results show that the core is not adversely affected. There is a considerable margin to the limiting DNBR; thus, the DNB design basis, as described in Section 4.4, is met.

15.4.5 A MALFUNCTION OR FAILURE OF THE FLOW CONTROLLER IN A BWR LOOP THAT RESULTS IN AN INCREASED REACTOR COOLANT FLOW RATE

This section is not applicable to Wolf Creek.

15.4.6 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT RESULTS IN A DECREASE IN THE BORON CONCENTRATION IN THE REACTOR COOLANT

15.4.6.1 Identification of Causes and Accident Description

One of the two principal means of positive reactivity insertion to the core is the addition of unborated, primary grade water from the demineralized and reactor makeup water system (RMWS) into the

RCS through the reactor makeup portion of the chemical and volume control system (CVCS). Boron dilution with these systems is a manually initiated operation under strict administrative controls requiring close operator surveillance with procedures limiting the rate and duration of the dilution. A boric acid blend system is available to allow the operator to match the makeup's boron concentration to that of the RCS during normal charging.

The means of causing an inadvertent boron dilution are the opening of the primary water makeup control valve and failure of the blend system, either by controller or mechanical failure. The CVCS and RMWS are designed to limit, even under various postulated failure modes, the potential rate of dilution to values which, with indication by alarms and instrumentation, will allow sufficient time for automatic or operator response (depending on the mode of operation) to terminate the dilution. An inadvertent dilution from the RMWS may be terminated by closing the primary water makeup control valve. All expected sources of dilution may be terminated by closing isolation valves in the CVCS, BG-LCV-112B and C. The lost shutdown margin (SDM) may be regained by the opening of isolation valves to the RWST, BN-LCV-112D and E, thus allowing the addition of borated water to the RCS.

Generally, to dilute, the operator must perform two distinct actions:

a. Switch control of the makeup from the automatic makeup mode to the dilute mode

b. Depress the start button

Failure to carry out either of the above actions prevents initiation of dilution. Also, during normal operation the operator may add borated water to the RCS by blending boric acid from the boric acid storage tanks with primary grade water. This requires the operator to determine the concentration of the addition and to set the blended flow rate and the boric acid flow rate. The makeup controller will then limit the sum of the boric acid flow rate and primary grade water flow rate to the blended flow rate, i.e., the controller determines the primary grade water flow rate after the start button is depressed.

The status of the RCS makeup is continuously available to the operator by:

a. Indication of the boric acid and blended flow rates

b. CVCS and RMWS pump status lights

c. Deviation alarms, if the boric acid or blended flow rates deviate by more than 10 percent from the preset values

d. High charging flow rate alarm

e. Volume control tank high and high-high water level and high pressure alarms

f. Letdown divert valve position alarm

g. Source range neutron flux - when reactor is subcritical

   1. High flux at shutdown alarm

   2. Indicated source range neutron flux count rates

   3. Audible source range neutron flux count rate

   4. Source range neutron flux - doubling alarm

h. When the reactor is critical

   1. Axial flux difference alarm (reactor power $\geqq$50 percent RTP)

   2. Control rod insertion limit low and low-low alarms

   3. Overtemperature ΔT alarm (at power)

   4. Overtemperature ΔT turbine runback (at power)

   5. Overtemperature ΔT reactor trip

   6. Power range neutron flux - high, both high and low setpoint reactor trips

This event is classified as an ANS Condition II incident (a fault of moderate frequency) as defined in Section 15.0.1.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-20.

15.4.6.2 Analysis of Effects and Consequences

To cover all phases of plant operation, boron dilution during Cold Shutdown, Hot Shutdown, Hot Standby, Start-up, and Power modes of operation is considered in this analysis. Conservative values for necessary parameters were used, i.e., high RCS critical boron concentrations, high boron worths, minimum shutdown margins, and lower than actual RCS volumes. These assumptions result in conservative determinations of the time available for operator or system response after detection of a dilution transient in progress.

Dilution During Refueling

An uncontrolled boron dilution transient cannot occur during this mode of operation. Inadvertent dilution is prevented by administrative controls which isolate the RCS from the potential source of unborated water. Valves V-178 and V-601 (or V-602) in the CVCS will be locked closed during refueling operations. These valves block all flow paths that could allow unborated makeup water to reach the RCS. Any makeup which is required during refueling will be borated water supplied from the RWST by the RHR pumps or the CCPs if using the CCP to flood up the Refueling Pool.

Dilution During Cold Shutdown, Hot Shutdown and Hot Standby

The Technical Specifications require the reactor to be shutdown to a SDM value as specified in the COLR when the unit is operating in Mode 5, Cold Shutdown. When in Modes 4 or 3, (Hot Shutdown or Hot Standby), the Technical Specifications require the plant to be shutdown to a SDM value as specified in the COLR. The following conditions are assumed for the analysis of the inadvertent boron dilution event while in these operating modes:

a. An assumed dilution flow rate of 157.5 gpm is used. This value corresponds to the high charging flow rate alarm plus a 5% allowance for uncertainties.

b. A minimum RCS water volume of 8725 ft^3 is assumed. This volume is a conservative estimate of the active volume of the RCS when at least one reactor coolant pump is in operation.

c. If no reactor coolant pump is in operation, all dilution sources are isolated or under administrative control.

Dilution During Start-up

In this mode, the plant is being taken from one long-term mode of operation, Hot Standby, to another, Power. The plant is maintained in the Start-up mode only for the purpose of start-up testing at the beginning of each cycle. During this mode of operation, the plant is in manual control, i.e., T_{avg}/rod control is in manual. All normal actions required to change power level, either up or down, require operator initiation. The COLR and Technical Specifications require an available trip reactivity of 1.3 percent ΔK/K (in MODE 2 with K_{eff} < 1.0) and four reactor coolant pumps operating. Other conditions assumed are:

a. Dilution flow is the maximum capacity of two centrifugal charging pumps with the RCS at 2250 psia (approximately 245 gpm).

b. A minimum RCS water volume of 9965 ft^3. This is a conservative estimate of the active RCS volume, minus the pressurizer volume.

Dilution During Full Power Operation

The plant may be operated at power two ways: automatic T_{avg}/rod control and under operator control. The COLR specifies an available trip reactivity of 1.3 percent ΔK/K and four reactor coolant pumps operating. With the plant at power and the RCS at pressure, the dilution rate is limited by the capacity of the centrifugal charging pumps (analysis is performed assuming two charging pumps are in operation even though normal operation is with one pump). Conditions assumed for this mode are:

a. Dilution flow from two centrifugal charging pumps is at the maximum at an RCS pressure of 2250 psia (approximately 245 gpm) when the reactor is in manual control. When in automatic control, the dilution flow is the maximum letdown flow (approximately 120 gpm).

b. A minimum RCS water volume of 9965 ft^3. This is a conservative estimate of the active RCS volume, minus the pressurizer volume.

15.4.6.3 Conclusions

Dilution During Refueling

Dilution during this mode has been precluded through administrative control of valves in the possible dilution flow paths (see Section 15.4.6.2).

Dilution During Cold Shutdown, Hot Shutdown or Hot Standby

In Modes 3, 4 or 5, the reactor operators are relied upon to detect and recover from a inadvertent boron dilution event. Numerous alarms from the CVCS and RMWS and in the Nuclear Instrumentation System are available to provide assistance to the reactor operator in the detection of an inadvertent boron dilution event. In the analyses of the event initiated from Modes 3, 4 or 5, the VCT high water level alarm (at 70% span) is credited as the initial alarm. The time required to perform the VCT/RWST valve swap-over and to purge the charging lines of dilute water are included in the calculation of the time available to the reactor operators to initiate corrective actions. In Modes 3, 4 and 5, analyses have demonstrated that the reactor operators have at least 15 minutes in which to initiate actions to terminate the dilution and initiate boration of the RCS.

There may exist other dilution events not obviously bounded by the analysis scenario described above. For example, during small dilution flow rate situations, the time required to fill the VCT to the high VCT water level setpoint may exceed the time required to dilute the RCS to the critical condition. However, the time intervals involved in this case are relatively long and alternate alarms, such as the concentrated boric acid flow and total makeup flow deviation alarms, would alert the reactor operator to a potential inadvertent boron dilution. For these relatively slow transients, an alternate event acceptance criterion is applied. This criterion requires that the time between the start of the event and the complete loss of shutdown margin (inadvertent criticality) be at least 30 minutes.

Dilution During Startup

This mode of operation is a transitory mode to go to power and is the operational mode in which the operator intentionally dilutes and withdraws control rods to take the plant critical. During this mode, the reactor is in manual rod control with the operator required to maintain a very high awareness of the plant status. For a normal approach to criticality, the operator must manually initiate a limited dilution and subsequently manually withdraw the control rods, a process that takes several hours. The plant Technical Specifications require that the operator determine the estimated critical position of the control rods prior to approaching criticality thus assuring that the reactor does not go critical with the control rods below the insertion limits. Once critical, the power escalation must be sufficiently slow to allow the operator to manually block the Source Range reactor trip after receiving P-6 from the Intermediate Range. Too fast a power escalation (due to an unknown dilution) would result in reaching P-6 unexpectedly, leaving insufficient time to manually block the Source Range reactor trip. Failure to perform this manual action results in a reactor trip and immediate shutdown of the reactor.

For Mode 2 - Startup, the reactor is assumed to trip when the flux level reaches the source range high flux trip setpoint. In the event of an inadvertent dilution, no credit is taken for operator action blocking the source range trip (permissive P-6). This reactor trip also functions as an alarm indicating a boron dilution transient is under way, as the control rods are not moving. The control rods are assumed to be at the insertion limits, minimizing the available shutdown margin; it is assumed that the RCS boron concentration at the time of reactor trip corresponds to the predicted critical boron concentration for rods at the insertion limits. Note: Due to the selection of boron concentration assumptions, the time available from the alarm is calculated to be identical to the total time from event initiation to loss of shutdown margin, i.e., the time of alarm will coincide with the analysis initial condition assumption. After reactor trip there is at least 15 minutes for operator action prior to return to criticality. The required operator action is the opening of valves BN-LCV-112D and E to initiate boration and the closing of valves BG-LCV-112B and C to terminate dilution.

Dilution During Full Power Operation

With the reactor under manual rod control and no operator action taken to terminate the transient, the power and temperature rise will cause the reactor to reach the Overtemperature ΔT trip setpoint resulting in a reactor trip. After reactor trip there is at least 60 minutes for operator action prior to return to criticality. The required operator action is the opening of valves BN-LCV-112D and E and the closing of valves BG-LCV-112B and C. The boron dilution transient in this case is essentially equivalent to an uncontrolled rod withdraw at power. A reactor trip occurs when either the Hi Neutron Flux or the Overtemperature ΔT setpoint is reached. The maximum reactivity insertion rate for a boron dilution transient is conservatively estimated to be within the range of insertion rates analyzed for uncontrolled rod withdraw at power.

It should be noted that prior to reaching the Overtemperature ΔT reactor trip the operator will have received an alarm on Overtemperature ΔT and an Overtemperature ΔT turbine runback. With the reactor in automatic rod control, the pressurizer level controller will limit the dilution flow rate to the maximum letdown rate, approximately 120 gpm. If a dilution rate in excess of the letdown rate is present, the pressurizer level controller will throttle charging flow down to match the letdown rate.

Thus, with the reactor in automatic rod control, a boron dilution will result in a power and temperature increase such that the rod controller will attempt to compensate by slow insertion of the control rods. This action by the controller will result in at least three alarms to the operator:

a. rod insertion limit - low level alarm.

b. rod insertion limit - low-low level alarm if insertion continued after (a) above, and

c. axial flux difference alarm (ΔT outside of the target band).

Given the many alarms, indications, and the inherent slow process of dilution at power, the operator has sufficient time for action. For example, the operator has at least 60 minutes from the rod insertion limit low-low alarm until shutdown margin is lost at beginning-of-life. The time would be significantly longer at end-of-life, due to the low initial boron concentration, when shutdown margin is a concern.

The above results demonstrate that in all modes of operation an inadvertent boron dilution is precluded, or sufficient time is available for operator action to terminate the transient. Following termination of the dilution flow and initiation of boration, the reactor is in a stable condition with the operator regaining the required shutdown margin.

15.4.7 INADVERTENT LOADING AND OPERATION OF A FUEL ASSEMBLY IN IMPROPER POSITION

15.4.7.1 Identification of Causes and Accident Description

Fuel and core loading errors that can arise from the inadvertent loading of one or more fuel assemblies into improper positions, loading a fuel rod during manufacture with one or more pellets of the wrong enrichment, or the loading of a full fuel assembly during manufacture with pellets of the wrong enrichment will lead to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment. Also included among possible core loading errors is the inadvertent loading of one or more fuel assemblies requiring burnable poison rods into a new core without burnable poison rods.

Any error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes which are more peaked than those calculated with the correct enrichments. There is a 5-percent uncertainty margin included in the design value of power peaking factor assumed in the analysis of Condition I and Condition II transients. The in-core system of moveable flux detectors, which is used to verify power shapes at the start of life, is capable of revealing any assembly enrichment error or loading error which causes power shapes to be peaked in excess of the design value.

To reduce the probability of core loading errors, each fuel assembly is marked with an dentification number and loaded in accordance with a core loading diagram. After the core is completely loaded, the identification number and position of each fuel assembly is recorded and compared to the design pattern.

The power distortion due to any combination of misplaced fuel assemblies would significantly raise peaking factors and would be readily observable with in-core flux monitors. In addition to the flux monitors, thermocouples are located at the outlet of about one-quarter of the fuel assemblies in the core. There is a high probability that these thermocouples would also indicate any abnormally high coolant enthalpy rise. In-core flux measurements are taken during the startup subsequent to every refueling operation.

This event is classified as an ANS Condition III incident (an infrequent fault), as defined in Section 15.0.1.

15.4.7.2 Analysis of Effects and Consequences

Method of Analysis

Steady state power distributions in the x-y plane of the initial core were calculated using the TURTLE code (Ref. 4) based on macroscopic cross sections calculated by the LEOPARD code (Ref. 5). A discrete representation was used wherein each individual fuel rod was described by a mesh interval. Current methodology applies the ANC code (Ref 16) to calculate power distributions; however, the TURTLE/LEOPARD analysis of the initial core remains valid.

For each core loading error case analyzed, for the initial core, the percent deviations from detector readings for a normally loaded core are shown at all in-core detector locations (see Figures 15.4-17 to 15.4-21, inclusive).

Results, Initial Core

The enrichments for cycle 1 at Wolf Creek were 2.10 (Region 1), 2.60 (Region 2), and 3.10 (Region 3) weight percent. The following core loading error cases were analyzed:

Case A:

Case in which a Region 1 assembly is interchanged with a Region 3 assembly. The particular case considered was the interchange to two adjacent assemblies near the periphery of the core (see Figure 15.4-17).

Case B:

Case in which a Region 1 assembly is interchanged with a neighboring Region 2 fuel assembly. Two analyses have been performed for this case (see Figures 15.4-18 and 15.4-19).

In Case B-1, the interchange is assumed to take place with the burnable poison rods transferred with the Region 2 assembly mistakenly loaded into Region 1.

In Case B-2, the interchange is assumed to take place closer to core center and with burnable poison rods located in the correct Region 2 position, but in a Region 1 assembly mistakenly loaded in the Region 2 position.

Case C:

Enrichment error: Case in which a Region 2 fuel assembly is loaded in the core central position (see Figure 15.4-20).

Case D:

Case in which a Region 2 fuel assembly instead of a Region 1 assembly is loaded near the core periphery (see Figure 15.4-21).

15.4.7.3 Conclusions

Fuel assembly enrichment errors are prevented by administrative procedures implemented in fabrication.

In the event that a single pin or pellet has a higher enrichment than the nominal value, the consequences in terms of reduced DNBR and increased fuel and clad temperatures will be limited to the incorrectly loaded pin or pins and perhaps the immediately adjacent pins.

Fuel assembly loading errors are prevented by administrative procedures implemented during core loading. In the unlikely event that a loading error occurs, the resulting power distribution effects will either be readily detected by the in-core moveable detector system or will cause a sufficiently small perturbation to be acceptable within the uncertainties allowed between nominal and design power shapes.

15.4.8 SPECTRUM OF ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENTS

15.4.8.1 Identification of Causes and Accident Description

This accident is defined as the mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of an RCCA and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion and system depressurization together with an adverse core power distribution, possibly leading to localized fuel rod damage.

15.4.8.1.1 Design Precautions and Protection

Certain features in the WCGS pressurized water reactor are intended to preclude the possibility of a rod ejection accident, or to limit the consequences if the accident were to occur. These include a sound, conservative, mechanical design of the rod housings, together with a thorough quality control (testing) program during assembly, and a nuclear design which lessens the potential ejection worth of RCCAs and minimizes the number of assemblies inserted at high power levels.

Mechanical Design

The mechanical design is discussed in Section 4.6. Mechanical design and quality control procedures intended to preclude the possibility of an RCCA drive mechanism housing failure are listed below:

a. Each control rod drive mechanism housing is completely assembled and shop tested at 4,100 psi.

b. The mechanism housings are individually hydrotested after they are attached to the head adapters in the reactor vessel head and checked during the hydrotest of the completed RCS.

c. Stress levels in the mechanism are not affected by anticipated system transients at power, or by the thermal movement of the coolant loops. Moments induced by the design earthquake can be accepted within the allowable primary working stress range specified by the ASME Code, Section III, for Class 1 components.

d. The latch mechanism housing and rode travel housing are each a single length of forged Type 304 stainless steel. This material exhibits excellent notch toughness at all temperatures which are encountered.

A significant margin of strength in the elastic range, together with the large energy absorption capability in the plastic range, gives additional assurance that gross failure of the housing will not occur. The joints between the latch mechanism housing and head adapter, and between the latch mechanism housing and rod travel housing, are threaded joints reinforced by canopy-type rod welds which are subject to periodic inspections.

Nuclear Design

Even if a rupture of an RCCA drive mechanism housing is postulated, the operation utilizing chemical shim is such that the severity of an ejected RCCA is inherently limited. In general, the reactor is operated with the RCCAs inserted only far enough to permit load follow. Reactivity changes caused by core depletion and xenon transients are compensated for by boron changes. Further, the location and grouping of control RCCA banks are selected during the nuclear design to lessen the severity of an RCCA ejection accident. Therefore, should an RCCA be ejected from its normal position during full power operation, only a minor reactivity excursion, at worst, could be expected to occur.

However, it may be occasionally desirable to operate with larger-than-normal insertions. For this reason, a rod insertion limit is defined as a function of power level. Operation with the RCCAs above this limit guarantees adequate shutdown capability and acceptable power distribution. The position of all RCCAs is continuously indicated in the control room. An alarm will occur if a bank of RCCAs approaches its insertion limit or if one RCCA deviates from its bank. Operating instructions require boration at a low level alarm and emergency boration at the low-low alarm.

Reactor Protection

The reactor protection in the event of a rod ejection accident has been described in Reference 6. The protection for this accident is provided by high neutron flux trip (high and low setting) and high rate of neutron flux increase trip. These protection functions are described in detail in Section 7.2.

Effects on Adjacent Housings

Disregarding the remote possibility of the occurrence of an RCCA mechanism housing failure, investigations have shown that failure of a housing due to either longitudinal or circumferential cracking would not cause damage to adjacent housings. The control rod drive mechanism is described in Section 3.9(N).4.

Effects of Rod Travel Housing Longitudinal Failures

If a longitudinal failure of the rod travel housing should occur, the region of the position indicator assembly opposite the break would be stressed by the reactor coolant pressure of 2,250 psia. The most probable leakage path would be provided by the radial deformation of the position indicator coil assembly, resulting in the growth of axial flow passages between the rod travel housing and the hollow tube along which the coil assemblies are mounted.

If a failure of the position indicator coil assembly should occur, the resulting free radial jet from the failed housing could cause it to bend and contact adjacent rod housings. If the adjacent housings were on the periphery, they might bend outward from their bases. The housing material is quite ductile; plastic hinging without cracking would be expected. Housings adjacent to a failed housing, in locations other than the periphery, would not be bent because of the rigidity of multiple adjacent housings.

Effect of Rod Travel Housing Circumferential Failures

If circumferential failure of a rod travel housing should occur, the separated section of the housing would be ejected vertically because the driving force is vertical and the position indicator coil assembly and the drive shaft would tend to guide the separated section upwards during its travel. Travel is limited by the missile shield, thereby limiting the projectile acceleration. When the projectile reached the missile shield, it would dissipate its kinetic energy without penetrating the shield. The water jet from the break would continue to push the separated section against the missile shield.

If the separated section of the rod travel housing were short enough to clear the break when fully ejected, it would rebound after impact with the missile shield. The top end plates of the position indicator coil assemblies would prevent the separated section from directly hitting the rod travel housing of a second drive mechanism. Even if a direct hit by the rebounding separated section were to occur, the low kinetic energy of the rebounding projectile would not be expected to cause significant damage (sufficient to cause failure of an adjacent housing).

Possible Consequences

From the above discussion, the probability of damage to an adjacent housing must be considered remote. However, even if damage is postulated, it would not be expected to lead to a more severe transient since RCCAs are inserted in the core in symmetric patterns, and control rods immediately adjacent to worst ejected rods are not in the core when the reactor is critical. Damage to an adjacent housing could, at worst, cause that RCCA not to fall on receiving a trip signal; however, this is already taken into account in the analysis by assuming a stuck rod adjacent to the ejected rod.

Summary

The considerations given above lead to the conclusion that failure of a control rod housing, due either to longitudinal or circumferential cracking, would not cause damage to adjacent housings that would increase severity of the initial accident.

15.4.8.1.2 Limiting Criteria

This event is classified as an ANS Condition IV incident. See Section 15.0.1 for a discussion of ANS classifications. Due to the extremely low probability of an RCCA ejection accident, some fuel damage would be considered an acceptable consequence.

Comprehensive studies of the threshold of fuel failure and of the threshold of significant conversion of the fuel thermal energy to mechanical energy have been carried out as part of the SPERT project by the Idaho Nuclear Corporation (Ref. 7).

Extensive tests of UO_2 zirconium clad fuel rods representative of those in pressurized water reactor type cores have demonstrated failure thresholds in the range of 240 to 257 cal/gm. However, other rods of a slightly different design have exhibited failures as low as 225 cal/gm. These results differ significantly from the

TREAT (Ref. 8) results, which indicated a failure threshold of 280 cal/gm. Limited results have indicated that this threshold decreases by about 10 percent with fuel burnup. The clad failure mechanism appears to be melting for zero burnup rods and brittle fracture for irradiated rods. Also important is the conversion ratio of thermal to mechanical energy. This ratio becomes marginally detectable above 300 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods; catastrophic failure (large fuel dispersal, large pressure rise), even for irradiated rods, did not occur below 300 cal/gm.

In view of the above experimental results, criteria are applied to ensure that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are (Ref. 12):

a. Average fuel pellet enthalpy at the hot spot is below 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel.

b. Peak reactor coolant pressure is less than that which could cause stresses to exceed the faulted condition stress limits.

c. Fuel melting is limited to less than 10 percent of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of criterion a above.

d. Average clad temperature at the hot spot is below the temperature at which clad embrittlement may be expected (3000°F).

15.4.8.2 Analysis of Effects and Consequences

15.4.8.2.1 Method of Analysis

The calculation of the RCCA ejection transient is performed in two stages, first an average core channel calculation and then a hot region calculation. The average core calculation is performed, using spatial neutron kinetics methods, to determine the average power generation with time, including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients at the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is conservatively assumed to persist throughout the transient.

A detailed discussion of the method of analysis can be found in Reference 6 and Reference 12.

Average Core Analysis

The spatial kinetics computer code, TWINKLE (Ref. 1), is used for the average core transient analysis. This code uses cross sections generated by LEOPARD (Ref. 5) to solve the two group neutron diffusion theory kinetic equation in one, two, or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 2,000 spatial points. The computer code includes a detailed multiregion, transient fuel-clad coolant heat transfer model for calculation of pointwise Doppler and moderator feedback effects. In this analysis, the code is used as a one-dimensional axial kinetics code, since it allows a more realistic representation of the spatial effects of axial moderator feedback and RCCA movement. However, since the radial dimension is missing, it is still necessary to employ very conservative methods (described below) of calculating the ejected rod worth and hot channel factor. Further description of TWINKLE appears in Section 15.0.11.

Hot Spot Analysis

In the hot spot analysis, the initial heat flux is equal to the nominal value times the design hot channel factor. During the transient, the heat flux hot channel factor is linearly increased to the transient value in 0.1 second, the time for full ejection of the rod. Therefore, the assumption is made that the hot spot before and after ejection is coincident. This is very conservative since the peak after ejection will occur in or adjacent to the assembly with the ejected rod, and prior to ejection the power in this region will necessarily be depressed.

The hot spot analysis is performed using the detailed fuel and clad transient heat transfer computer code, FACTRAN (Ref. 2). This computer code calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod, and the heat flux at the surface of the rod, using as input the nuclear power versus time and the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A parabolic radial power distribution is used within the fuel rod.

FACTRAN uses the Dittus-Boelter (Ref. 17) or Jens-Lottes (Ref. 18) correlation to determine the film heat transfer coefficient before DNB, and the Bishop-Sandburg-Tong correlation (Ref. 9) to determine the film boiling coefficient after DNB. The Bishop-Sandburg-Tong correlation is conservatively used, assuming zero bulk fluid quality. The DNBR is not calculated; instead the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient can be calculated by the code; however, it is adjusted in order to force the full power steady state temperature distribution to agree with the fuel heat transfer design codes. Further description of FACTRAN appears in Section 15.0.11.

System Overpressure Analysis

Because safety limits for fuel damage specified earlier are not exceeded, there is little likelihood of fuel dispersal into the coolant. The pressure surge may, therefore, be calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant.

The pressure surge is calculated by first performing the fuel heat transfer calculation to determine the average and hot spot heat flux versus time. Using this heat flux data, a THINC calculation is conducted to determine the volume surge. Finally, the volume surge is simulated in a plant transient computer code. This code calculates the pressure transient, taking into account fluid transport in the RCS and heat transfer to the steam generators. No credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing. The overpressure analysis is a generic analysis provided in Reference 6. The results of the generic analysis remain applicable to WCGS.

15.4.8.2.2 Calculation of Basic Parameters

Input parameters for the analysis are conservatively selected on the basis of values calculated for this type of core. The more important parameters are discussed below. Table 15.4-2 presents the parameters used in this analysis.

Ejected Rod Worths and Hot Channel Factors

The values for ejected rod worths and hot channel factors are calculated, using either three-dimensional static methods or by a synthesis method employing one-dimensional and two-dimensional calculations. Standard nuclear design codes are used in the analysis. No credit is taken for the flux flattening effects of reactivity feedback. The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. Adverse xenon distributions are considered in the calculation.

Appropriate margins are added to the ejected rod worth and hot channel factors to account for any calculational uncertainties, including an allowance for nuclear power peaking due to densification.

Power distributions before and after ejection for a "worst case" can be found in Reference 6. During initial plant startup physics testing, ejected rod worths and power distributions are measured in the zero and full power configurations and compared to values used in the analysis. It has been found that the ejected rod worth and power peaking factors are consistently overpredicted in the analysis.

Reactivity Feedback Weighting Factors

The largest temperature increases, and hence the largest reactivity feedbacks, occur in channels where the power is higher than average. Since the weight of a region is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple channel analysis. Physics calculations have been carried out for temperature changes with a flat temperature distribution and with a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers which, when applied to single-channel feedbacks, correct them to effective whole core feedbacks for the appropriate flux shape. In this analysis, since a one-dimensional (axial) spatial kinetics method is employed, axial weighting is not necessary if the initial condition is made to match the ejected rod configuration. In addition, no weighting is applied to the moderator feedback. A conservative radial weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time, accounting for the missing spatial dimension. These weighting factors have also been shown to be conservative, compared to three-dimensional analysis (Ref. 6).

Moderator and Doppler Coefficient

The critical boron concentrations at the beginning-of-life and end-of-life are adjusted in the nuclear code in order to obtain moderator density coefficient curves which are conservative, compared to actual design conditions for the plant. As discussed above, no weighting factor is applied to these results.

The Doppler reactivity defect is determined as a function of power level, using a one-dimensional steady state computer code with a Doppler weighting factor of 1.0. The Doppler defect used does not correlate to Figure 15.0-2 since the TWINKLE computer code, on which the neutronic analysis is based, is a diffusion-theory code rather than a point-kinetics approximation. The Doppler weighting factor will increase under accident conditions, as discussed above.

Delayed Neutron Fraction, β

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values no less than 0.70 percent at beginning-of-life and 0.50 percent at end-of-life for the first cycle. The accident is sensitive to if the ejected rod worth is equal to or greater than as in zero power transients. In order to allow for future cycles, conservative estimates of β of 0.49 percent at beginning of cycle and 0.44 percent at end of cycle were used in the analysis.

Trip Reactivity Insertion

The trip reactivity insertion assumed is given in Table 15.4-2 and includes the effect of one stuck RCCA. These values are reduced by the ejected rod reactivity. The shutdown reactivity was simulated by dropping a rod of the required worth into the core. The start of rod motion occurred 0.5 seconds after the high neutron flux trip point was reached. This delay is assumed to consist of 0.2 second for the instrument channel to produce a signal, 0.15 second for the trip breaker to open, and 0.15 seconds for the coil to release the rods. A curve of trip rod insertion versus time was used which assumed that insertion to the dashpot does not occur until 3.3 seconds after the start of fall. The choice of such a conservative insertion rate means that there is over 1 second after the trip point is reached before significant shutdown reactivity is inserted into the core. This is particularly important conservatism for hot full power accidents.

The minimum design shutdown margin available for this plant at hot zero power (HZP) may be reached only at end-of-life in the equilibrium cycle. This value includes an allowance for the worst stuck rod and adverse xenon distribution, conservative Doppler and moderator defects, and an allowance for calculational uncertainties. Physics calculations for this plant have shown that the effect of two stuck RCCAs (one of which is the worst ejected rod) is to reduce the shutdown by about an additional 1 percent Δk. Therefore, following a reactor trip resulting from an RCCA ejection accident, the reactor will be subcritical when the core returns to HZP.

Depressurization calculations have been performed for a typical four-loop plant, assuming the maximum possible size break (2.75-inch diameter) located in the reactor pressure vessel head. The results show a rapid pressure drop and a decrease in system water mass due to the break. The safety injection system is actuated on low pressurizer pressure within 1 minute after the break. The RCS pressure continues to drop and reaches saturation (1,200 psi) in about 2 to 3 minutes. Due to the large thermal inertia of primary and secondary systems, there has been no significant decrease in the RCS temperature below no-load by this time, and the depressurization itself has caused an increase in shutdown margin by about 0.2 percent Δk due to the pressure coefficient. The cooldown transient could not absorb the available shutdown margin until more than 10 minutes after the break. The addition of borated safety injection flow starting 1 minute after the break is sufficient to ensure that the core remains subcritical during the cooldown.

Reactor Protection

As discussed in Section 15.4.8.1.1, reactor protection for a rod ejection is provided by high neutron flux trip (high and low setting) and high rate of neutron flux increase trip. These protection functions are part of the reactor trip system. No single failure of the reactor trip system will negate the protection functions required for the rod ejection accident, or adversely affect the consequences of the accident.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-21.

Results

Cases are presented for both beginning and end-of-life at zero and full power.

a. Beginning of cycle, full power

   Control bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor were conservatively calculated to be 0.23 percent Δk and 5.7, respectively. The peak hotspot clad average temperature reached 2,401°F. The peak hot spot fuel center temperature reached melting at 4,900 °F. However, melting was restricted to less than 10 percent of the pellet.

b. Beginning of cycle, zero power

   For this condition, control bank D was assumed to be fully inserted, and banks B and C were at their insertion limits. The worst ejected rod is located in control bank D and has a worth of 0.77 percent Δk and a hot channel factor of 10.7. The peak hotspot clad average temperature reached 2,954 °F. The peak hot spot fuel center temperature was 4,611 °F.

c. End of cycle, full power

   Control bank D was assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors were conservatively calculated to be 0.250 percent Δk and 6.3, respectively. The peak hotspot clad average temperature reached 2,325 °F. The peak hotspot fuel center temperature reached melting at 4,800°F. However, melting was restricted to less than 10 percent of the pellet.

d. End of cycle, zero power

The ejected rod worth and hot channel factor for this case were obtained, assuming control bank D to be fully inserted with banks B and C at their insertion limits. The results were 0.86 percent Δk and 13.0, respectively The peak hotspot clad average temperature reached 2,967 °F.The peak fuel center temperature was 4,220 °F.

A summary of the cases presented above is given in Table 15.4-2. The nuclear power and hot spot fuel and clad temperature transients for the worst cases (beginning-of-life, full power and beginning-of-life, zero power) are presented in Figures 15.4-22 through 15.4-25.

The calculated sequence of events for the worst case rod ejection accidents, as shown in Figures 15.4-22 through 15.4-25, is presented in Table 15.4-1. For all cases, reactor trip occurs very early in the transient, after which the nuclear power excursion is terminated. As discussed previously in Section 15.4.8.2.2, the reactor will remain subcritical following reactor trip.

The ejection of an RCCA constitutes a break in the RCS, located in the reactor pressure vessel head. The effects and consequences of loss-of-coolant accidents are discussed in Section 15.6.5. Following the RCCA ejection, the operator would follow the same emergency instructions as for any other loss-of-coolant accident to recover from the event.

Fission Product Release

It is assumed that fission products are released from the gaps of all rods experiencing DNB. In all cases considered, less than 10 percent of the rods experienced the DNB. Although limited fuel melting at the hot spot was predicted for the full power cases, in practice melting is not expected since the analysis conservatively assumed that the hot spots before and after ejection were coincident. Parameters assumed in the radiological consequences analysis are summarized in Table 15.4-3.

Pressure Surge

A detailed calculation of the pressure surge for an ejection worth of one dollar at beginning-of-life, hot full power, indicates that the peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits (Ref. 6). Since the severity of the present analysis does not exceed the "worst case" analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the RCS.

Lattice Deformations

A large temperature gradient will exist in the region of the hot spot. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a differential expansion tending to bow the midpoint of the rods toward the hotter side of the rod. Calculations have indicated that this bowing would result in a negative reactivity effect at the hot spot since Westinghouse cores are under-moderated, and bowing will tend to increase the under-moderation at the hot spot. In practice, no significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced. Boiling in the hot spot region would produce a net flow of coolant away from that region. However, the heat from the fuel is released to the water relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator to fuel ratio, and a large reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

15.4.8.3 Radiological Consequences

15.4.8.3.1 Method of Analysis

15.4.8.3.1.1 Physical Model

Prior to the accident, it is assumed that the plant has been operating with simultaneous fuel defects and steam generator tube leakage for a time sufficient to establish equilibrium levels of activity in the reactor coolant and secondary systems.

The RCCA ejection results in reactivity being inserted to the core which causes the local power to rise. In a conservative analysis, it is assumed that partial cladding failure and fuel melting occurs. The fuel pellet and gap activities are assumed to be immediately and uniformly released within the reactor coolant.

Two release paths to the environment exist which are analyzed separately and conservatively, as if all the activity is available for release from each path.

The activity released to the containment from the reactor coolant through the ruptured control rod mechanism pressure housing is assumed to be mixed instantaneously throughout the containment and is available for leakage to the atmosphere. The only removal processes considered in the containment are iodine plateout, radioactive decay, and leakage from the containment.

The model for the activity available for release to the atmosphere from the relief valves assumes that the release consists of the activity in the secondary system plus that fraction of the activity leaking from the reactor coolant through the steam generator tubes. The leakage of reactor coolant to the secondary side of the steam generator continues until the pressures in the reactor coolant and secondary systems equalize.

Primary and secondary pressures are equalized at 1100 seconds following the accident, thus terminating primary to secondary leakage in the steam generators. Refer to Figures 15.4-26 and 15.4-27.

Thereafter, no mass transfer from the reactor coolant system to the secondary system due to the steam generator tube leakage is assumed. Thus, in the case of coincident loss of offsite power, activity is released to the atmosphere from a steam dump through the relief valves.

15.4.8.3.1.2 Assumptions and Conditions

The major assumptions and parameters used in the analysis are itemized in Tables 15.4-3 and 15A-1 and summarized below. The assumptions are consistent with Regulatory Guide 1.77.

The assumption used to determine the initial concentrations of isotopes in the reactor coolant and secondary coolant prior to the accident are as follows:

a. The reactor coolant iodine activity is based on the dose equivalent of 1.0 μCi/gm of I-131.

b. The noble gas and iodine activity in the reactor coolant are based on 1-percent failed fuel.

c. The secondary coolant activity is based on the dose equivalent of 0.1 μCi/gm of I-131.

The following conditions are used to calculate the activity released and the offsite doses following a RCCA ejection accident.

a. 10 percent of the fuel rod gap activity, except for Kr-85 and I-131, which are 30 percent and 12 percent respectively, is additionally released to the reactor coolant.

b. 0.25 percent of the fuel is assumed to melt.

c. Following the incident until primary and secondary side pressures equalize, secondary steam is released to the environment. The total quantity of steam released is given in Table 15.4.3.

d. The 1-gpm primary-to-secondary leak to the unaffected steam generators.

e. All noble gas activity in the reactor coolant which is transported to the secondary system via the primary-to-secondary leakage is assumed to be immediately released to the environment.

f. Fission products released from the fuel-cladding gap of the damaged fuel rods are assumed to be instantaneously and homogeneously mixed with the reactor coolant.

g. The iodine activity present in the primary to secondary leakage is assumed to mix homogeneously with the iodine activity initially present in the steam generators.

h. A partition factor of 0.1 between the vapor and liquid phases for radioiodine in the steam generators is utilized to determine iodine releases to the environment via steam venting from the steam generators.

i. The activity released from the steam generators is immediately vented to the environment.

j. The containment is assumed to leak at 0.2 volume percent/day during the first 24 hours immediately following the accident and 0.1 volume percent/day thereafter.

k. No credit is taken for radioactive decay or ground deposition during radioactivity transport to offsite location.

l. Short-term accident atmospheric dispersion factors corresponding to ground level releases, breathing rates, and dose conversion factors are given in Table 15A-2, and 15A-4, respectively.

m. Offsite power is assumed lost.

15.4.8.3.1.3 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 the onsite meteorological measurement programs described in Section 2.3 and are provided in Table 15A-2.

c. The thyroid inhalation and total-body immersion doses to a receptor at the exclusion area boundary or outer boundary of the low-population zone were analyzed, using the models described in Appendix 15A.

15.4.8.3.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

The leakage pathways are:

a. Direct steam dump to the atmosphere through the secondary system relief valves for the secondary steam

b. Primary-to-secondary steam generator tube leakage and subsequent steam dump to the atmosphere through the secondary system relief valves

c. The resultant activity released to the containment is assumed available for leakage directly to the environment.

Table 15.4-3 shows the total curies released

15.4.8.3.2 Identification of Uncertainties and Conservative Elements in the Analysis

a. Reactor coolant and secondary coolant activities of 1 percent failed fuel and 0.1 μCi/gm I-131 dose equivalent, respectively, are many times greater than assumed for normal operation conditions.

b. A 1-gpm steam generator primary-to-secondary leakage, which is significantly greater than that anticipated during normal operation, is assumed.

c. The coincident loss of offsite power with the occurrence of a RCCA ejection accident is a highly conservative assumption. In the event of the availability of offsite station power, the condenser steam dump valves will open, permitting steam dump to the condenser. Thus there is no direct release via that path to the environment.

d. It is assumed that 50 percent of the iodines released to the containment atmosphere is adsorbed (i.e. plate out) onto the internal surfaces of the containment or adheres to internal components. However, it is estimated that the removal of airborne iodines by various physical phenomena such as adsorption, adherence, and settling could reduce the resultant doses by a factor of 3 to 10.

e. The activity released to the containment atmosphere is assumed to leak to the environment at the containment leakage rate of 0.2-volume percent/day for the first 24 hours and 0.1-volume percent/day thereafter. The initial containment leakage rate is based on the peak calculated internal containment pressure anticipated after a LOCA. The pressures associated with a RCCA ejection accident are considerably lower than that calculated for a LOCA. The pressure inside the containment also decreases considerably with time, with an expected decrease in leakage rates. Taking into account that the containment leak rate is a function of pressure, the resultant doses could be reduced by a factor of 5 to 10 (Ref. 10).

f. 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 meteorological conditions assumed will 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.

15.4.8.3.3 Conclusions

15.4.8.3.3.1 Filter Loadings

The only ESF filtration system considered in the analysis which limits the consequences of the RCCA ejection accident is the control room filtration system. Activity loadings on the control room charcoal filter are based on the flow rate through the filter, the concentration of activity at the filter inlet, and the filter efficiency.

The activity in the control room filter as a function of time has been evaluated for the loss-of-coolant accident, Section 15.6.5. Since the control room filters are capable of accommodating the potential design-basis loss-of-coolant accident fission product iodine loadings, more than adequate design margin is available with respect to postulated RCCA ejection accident releases.

15.4.8.3.3.2 Doses to Receptor at the Exclusion Area Boundary and Low-Population Zone Outer Boundary

The potential radiological consequences resulting from the occurrence of a postulated RCCA ejection accident have been conservatively analyzed, using assumptions and models described in previous sections.

The total-body doses due to immersion from direct radiation and the thyroid dose due to inhalation have been analyzed for the 0-2 hour dose at the exclusion area boundary and for the duration of the accident at the low-population zone outer boundary. The results are listed in Table 15.4-4. The resultant doses are well within the guideline values of 10 CFR 100.

15.4.8.4 Conclusions

Even on a conservative basis, the analyses indicate that the described fuel and clad limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the RCS. The analyses have demonstrated that upper limit in fission product release as a result of a number of fuel rods entering DNB amounts to 10 percent.

The RCS integrated break flow to containment following a rod ejection accident is shown in Figure 15.4-28.

15.4.9 REFERENCES

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TABLE 15.4-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	Time (sec)
Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	Initiation of uncontrolled rod withdrawal from 10^{-9} of nominal power	0.0
	Power range high neutron flux low setpoint reached	10.36
	Peak nuclear power occurs	10.50
	Rods begin to fall into core	10.86
	Minimum DNBR occurs	12.6
	Peak average clad temperature occurs	12.50
	Peak average fuel temperaure occurs	12.80
	Peak heat flux occurs	12.90
Uncontrolled RCCA bank withdrawal at power	Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (5 pcm/sec, 100% Power, BOL))	0.0
	Overtemperature DT reactor trip signal initiated	45.0
	Maximum Pressurizer Pressure	47.3
	Maximum Pressurizer Level	47.3

Accident	Event	Time (sec)
Startup of inac tive reactor coolant loop at an incorrect temperature		
	Initiation of pump startup	1.0
	Power reaches P-8 trip setpoint	12.8
	Rods begin to drop	12.9
	Minimum DNBR occurs	12.9
Rod cluster control assembly ejection accident		
1. Beginning-of-life, full power		
	Initiation of rod ejection	0.0
	Power range high neutron flux setpoint reached	0.05
	Peak nuclear power occurs	0.135
	Rods begin to fall into core	0.55
	Peak clad temperature occurs	2.41
	Peak heat flux occurs	2.42
	Peak fuel centerline temperature	3.07
2. End-of-life, zero power		
	Initiation of rod ejection	0.0
	Power range high neutron flux low setpoint reached	0.14
	Peak nuclear power occurs	0.17
	Rods begin to fall into core	0.64
	Peak heat flux occurs	0.96
	Peak clad temperature occurs	1.00
	Peak centerline fuel temperature	2.96

TABLE 15.4-2

PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT

Time in Life	HZP (Beginning)	HFP (Beginning)	HZP (End)	HFP (End)
Power level, %	0	102	0	102
Initial average coolant temperature, F	557.0	594.9	557.0	594.9
Ejected rod worth, %Dk	0.77	0.23	0.86	0.25
Delayed neutron fraction, %	0.49	0.49	0.44	0.44
Feedback reactivity weighting	2.398	1.30	3.55	1.30
Trip reactivity, %Dk	1.3	4.0	1.3	4.0
F_Q before rod ejection		2.32		2.32
F_Q after rod ejection	10.7	5.7	13.0	6.3
Number of operational pumps	2	4	2	4
Fuel Melt	0.0	9.3	0.0	9.0
Maximum fuel center temperature, F	4611	4900	4220	4800
Maximum clad average temperature, F	2954	2401	2967	2325
Maximum fuel stored energy, Btu/lbm	311	332	295	319

TABLE 15.4-3

PARAMETERS USED IN EVALUATING THE RCCA EJECTION ACCIDENT

I. Source Data

a.	Core power level, MWT	3565
b.	Burnup, full power days	1000
c.	Core inventories	Table 15A-3
d.	Steam generator tube leakage, gpm	1
e.	Reactor coolant	Based on 1-percent failed fuel, provided in Table 11.1-5
f.	Secondary system activity	Based on 1-percent failed fuel, 4 times the values provided in Table 11.1-4
g.	Extent of core damage	10 percent of fuel rods experience cladding failure; 0.25 percent of fuel experiences melting
h.	Activity released to reactor coolant, percent	
	1. Cladding failure	
	(a) Noble gas gap activity	100
	(b) Iodine gap activity	100
	2. Fuel melting	
	(a) Noble gas gap activity	100
	(b) Iodine fuel activity	50
i.	Iodine carryover factor for steam generators	0.1
j.	Reactor coolant mass, lbs	4.94E + 5
k.	Steam generator mass, lbs/steam generator	1.04E + 5

II. Atmospheric Dispersion Factors — Table 15A-2

III. Activity Release Data

a.	Containment volume, ft^3	2.5E + 6
b.	Containment leak rate, volume percent/day	
	1. 0-24 hours	0.20
	2. 1-30 days	0.10
c.	Percent of containment leakage that is unfiltered	100
d.	Plateout of iodine within containment, percent	50
e.	Offsite power	Lost
f.	Steam release from relief valves, lbs	48,600
g.	Duration of release from relief valves, sec	140

h. Activity released to the environment via steam release and containment release

	Steam Generator Release (Ci)	Containment Release (Ci)	
Isotope	0-2hr	0-2hr	0-30 days
I-131	5.369E+00	1.040E+02	7.275E+03
I-132	6.546E+00	9.655E+01	2.139E+02
I-133	9.393E+00	1.769E+02	2.142E+03
I-134	9.937E+00	1.012E+02	1.275E+02
I-135	8.703E+00	1.554E+02	8.184E+02
Xe-131m	5.287E-01	2.225E+00	2.028E+02
Xe-133m	3.038E+00	1.265E+01	3.124E+02
Xe-133	9.881E+01	4.146E+02	2.087E+04
Xe-135m	1.768E+01	1.469E+01	1.476E+01
Xe-135	2.325E+01	9.114E+01	5.940E+02
Xe-137	6.881E+01	1.640E+01	1.640E+01
Xe-138	7.650E+01	5.812E+01	5.829E+01
Kr-83m	6.080E+00	1.810E+01	3.400E+01
Kr-85m	1.316E+01	4.788E+01	1.774E+02
Kr-85	1.392E+00	5.874E+00	1.079E+03
Kr-87	2.521E+01	6.547E+01	9.863E+01
Kr-88	3.580E+01	1.201E+02	3.105E+02
Kr-89	3.454E+01	7.095E+00	7.095E+00

TABLE 15.4-4

RADIOLOGICAL CONSEQUENCES OF A ROD-EJECTION ACCIDENT

	Wolf Creek Dose (rem)
Exclusion Area Boundary (0-2 hr)	
Thyroid	1.17E+1
Whole body	5.85E-2
Low Population Zone Outer Boundary (duration)	
Thyroid	1.44E+1
Whole body	2.34E-2

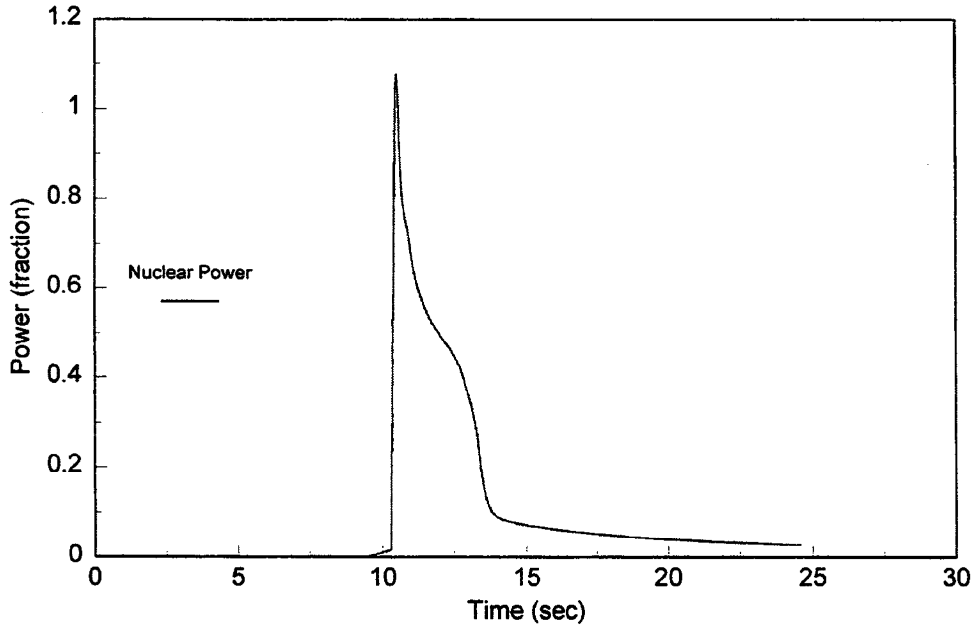

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FIGURE 15.4-1
Core Average Neutron Flux Transient for Uncontrolled Rod Withdrawal from a Subcritical Condition

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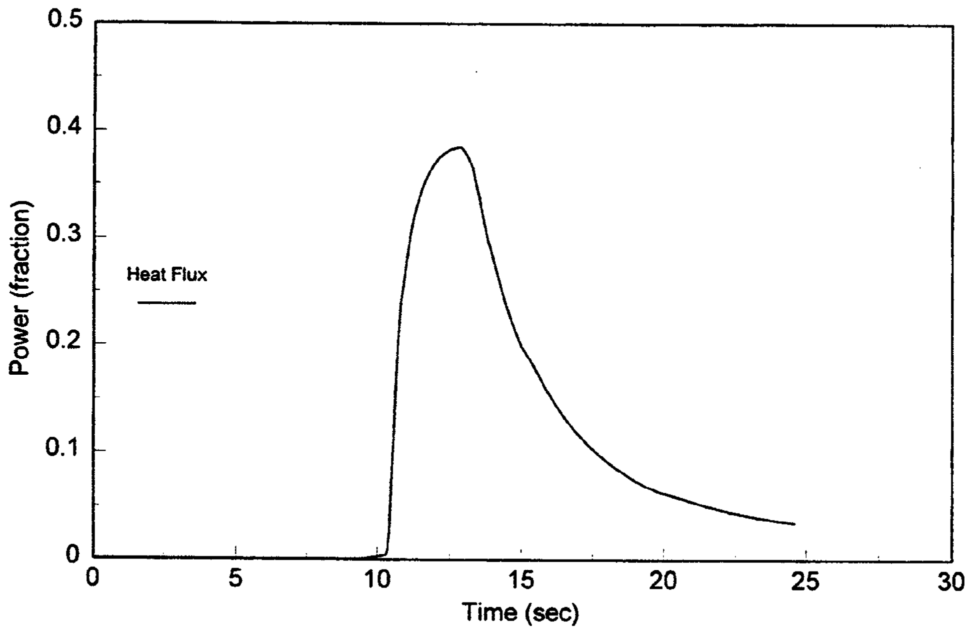

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FIGURE 15.4-2

Thermal Flux Transient for Uncontrolled Rod Withdrawal from a Subcritical Condition

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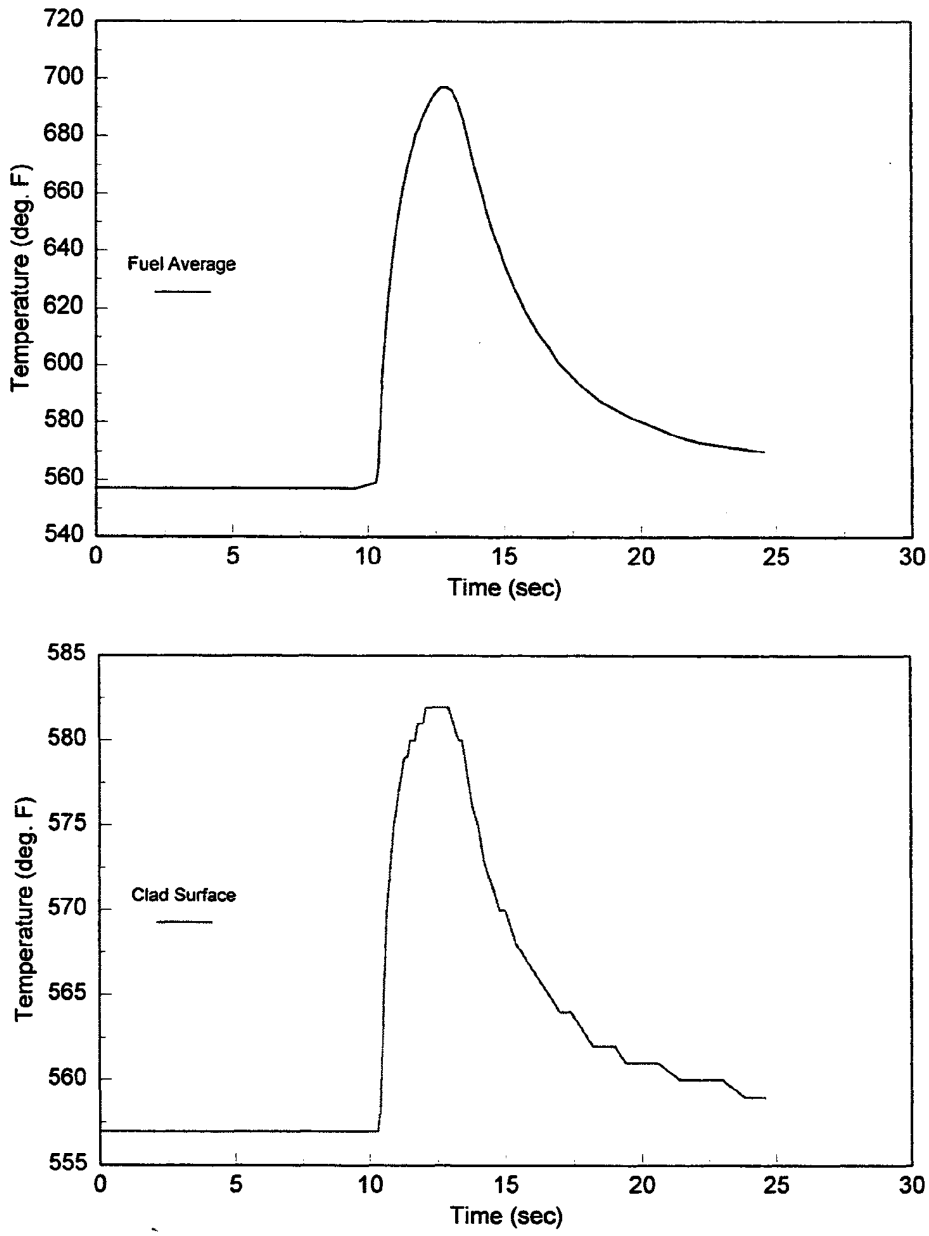

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FIGURE 15.4-3 Fuel and Clad Temperature Transient for Uncontrolled Rod Withdrawal from a Subcritical Condition

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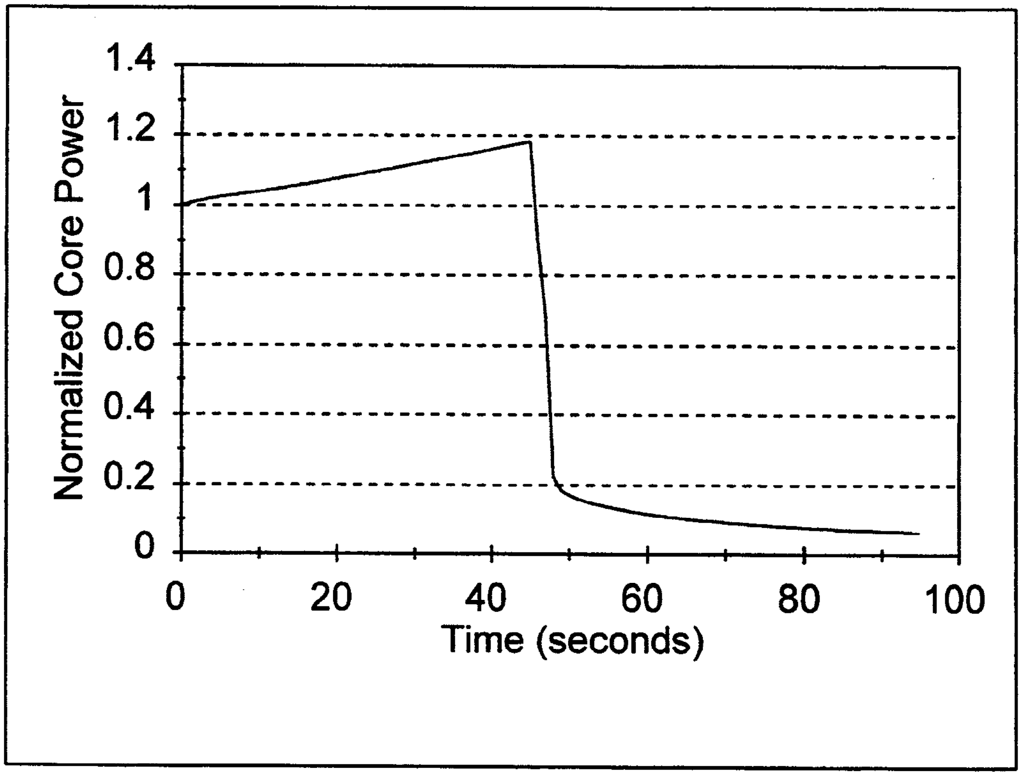

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FIGURE 15.4-4

Uncontrolled RCCA Bank Withdrawal from Full Power with Minimum Reactivity Feedback (5 pcm/sec Withdrawal Rate)

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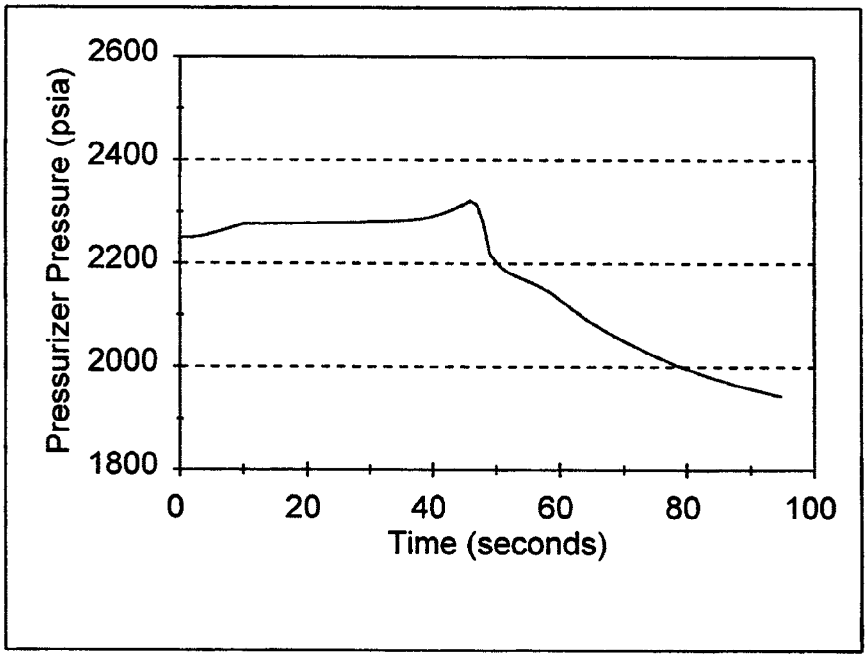

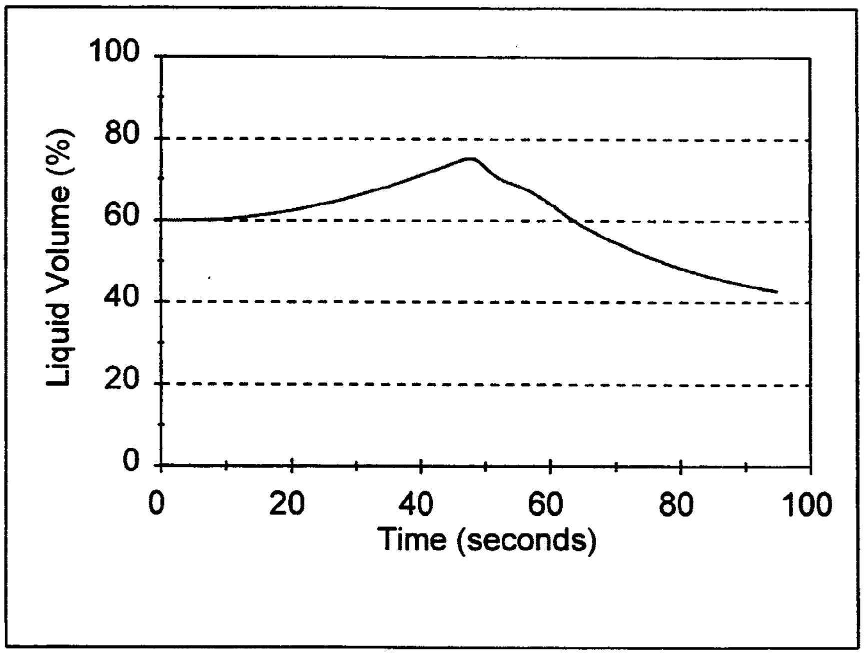

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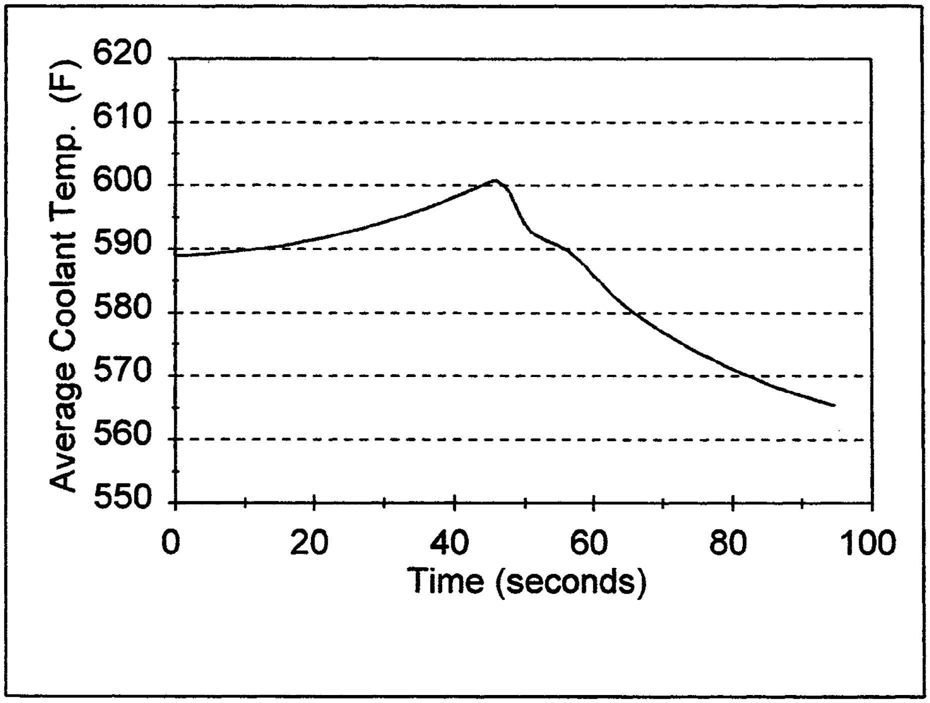
FIGURE 15.4-5

Uncontrolled RCCA Bank Withdrawal from Full Power with Minimum Reactivity Feedback (5 pcm/sec Withdrawal Rate)



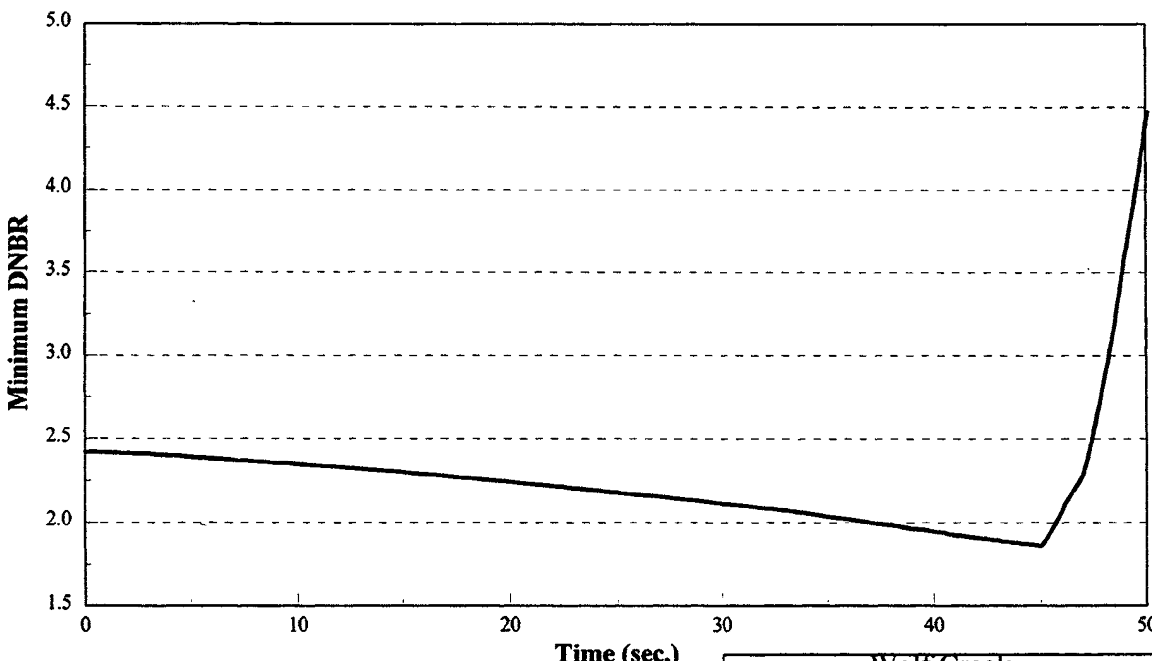


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Figure 15.4-6
Withdrawal From Full Power with Beginning-of-Life Reactivity (5 pcm/sec Withdrawal Rate)

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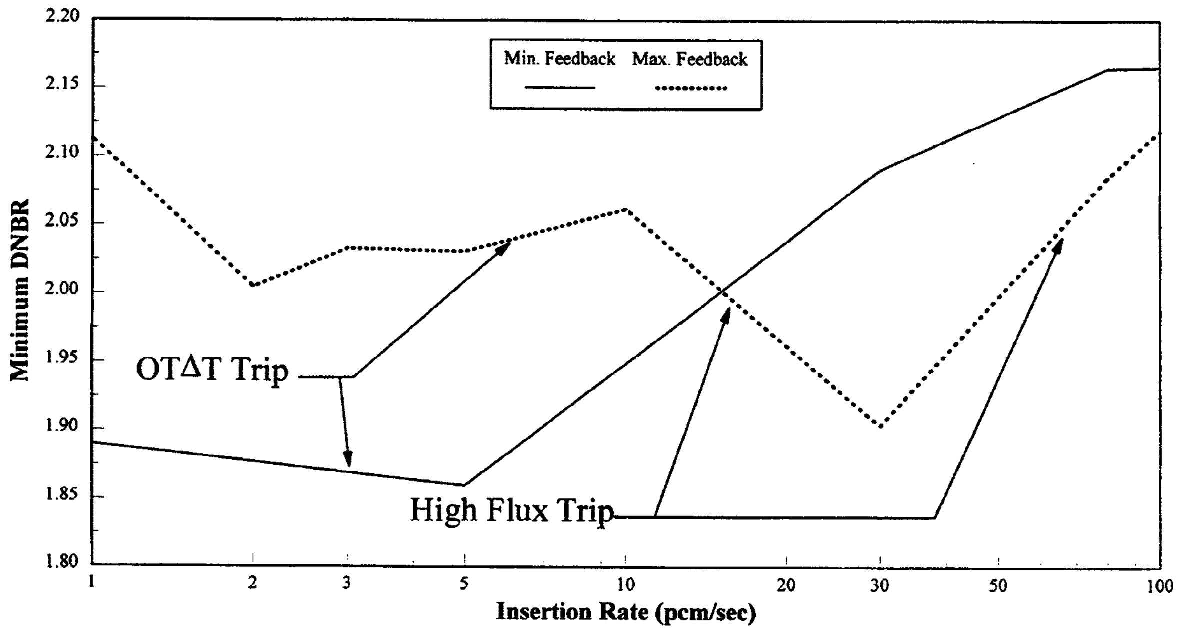

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Figure 15.4-7
Minimum DNBR Vs. Reactivity Insertion Rate, Rod Withdrawal from 100% Power

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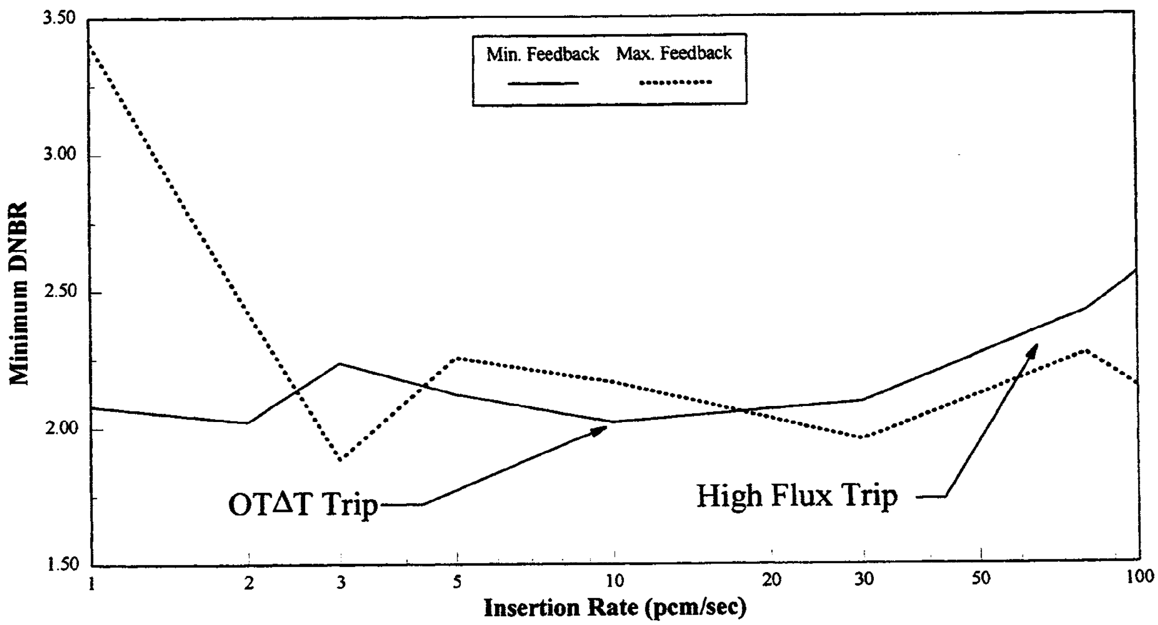

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Figure 15.4-8
Minimum DNBR Vs. Reactivity Insertion Rate, Rod Withdrawal from 60% Power

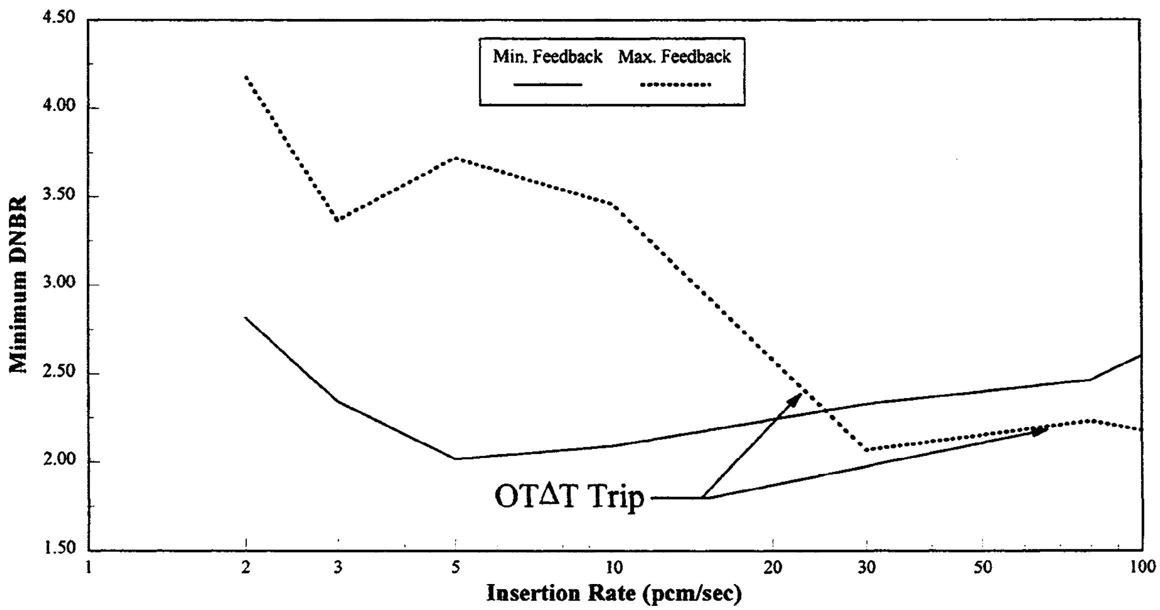

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Figure 15.4-9
Minimum DNBR Vs. Reactivity Insertion Rate, Rod Withdrawal from 10% Power

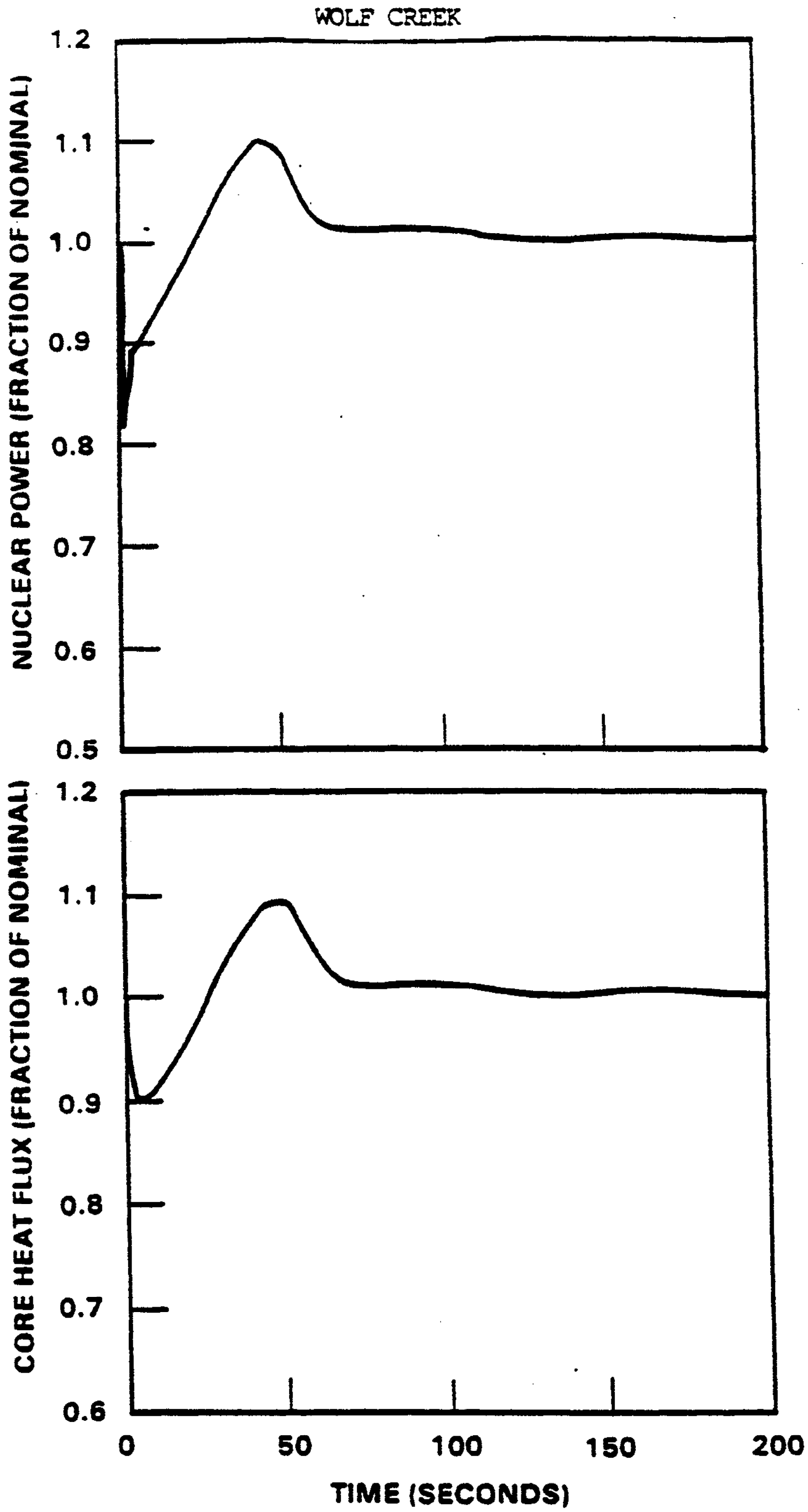

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FIGURE 15.4-10
NUCLEAR POWER TRANSIENT AND CORE HEAT FLUX TRANSIENT FOR DROPPED ROD CLUSTER CONTROL ASSEMBLY

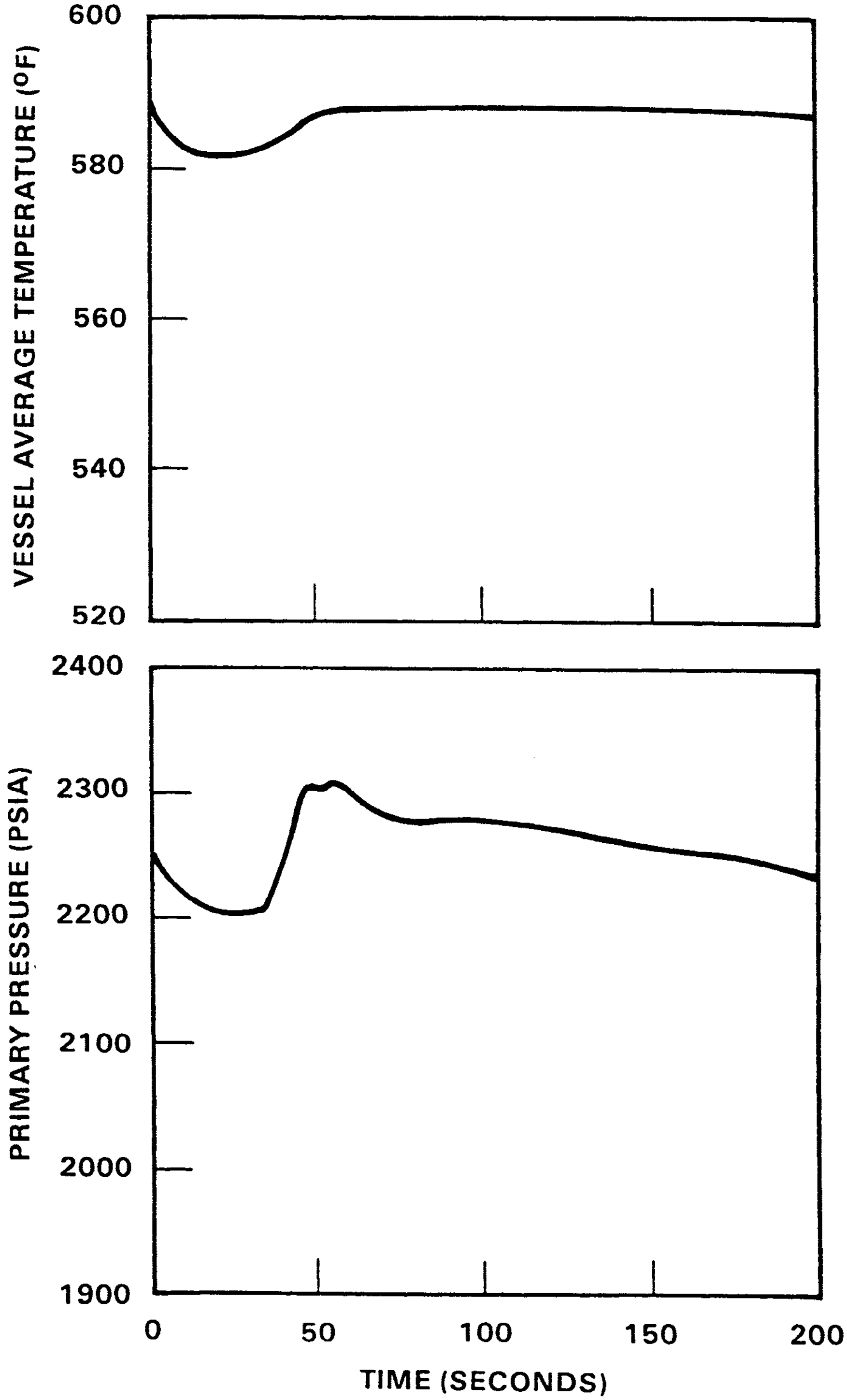

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FIGURE 15.4-11a
PRESSURIZER PRESSURE TRANSIENT AND CORE AVERAGE TEMPERATURE TRANSIENT FOR DROPPED ROD CLUSTER CONTROL ASSEMBLY

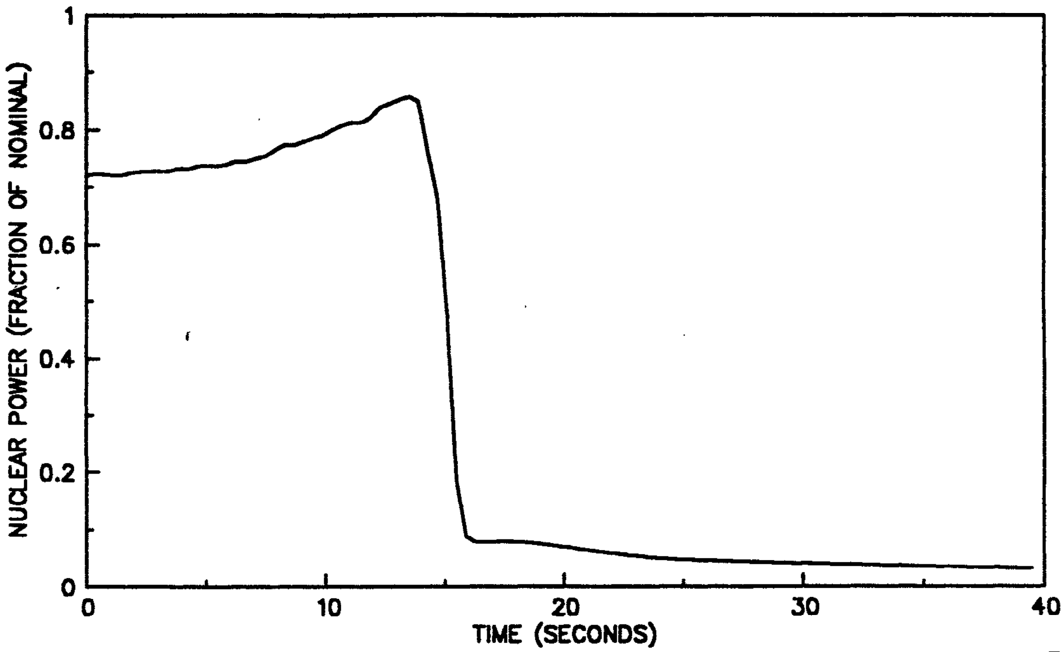

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FIGURE 15.4-12 NUCLEAR POWER TRANSIENT FOR STARTUP OF AN INACTIVE REACTOR COOLANT LOOP

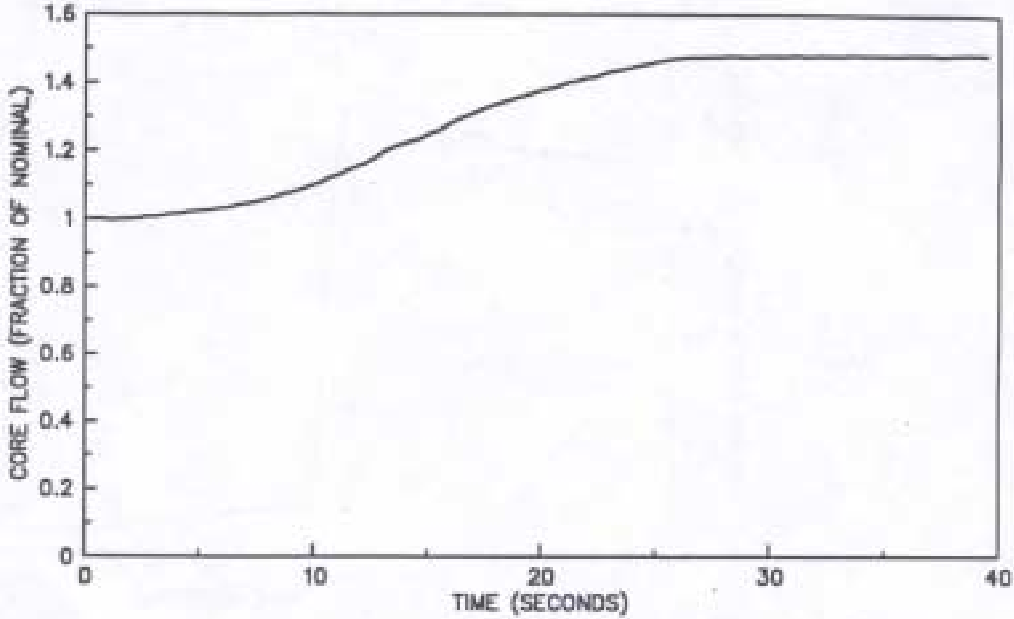

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FIGURE 15.4-13

CORE FLOW TRANSIENT FOR
STARTUP OF AN INACTIVE REACTOR
COOLANT LOOP

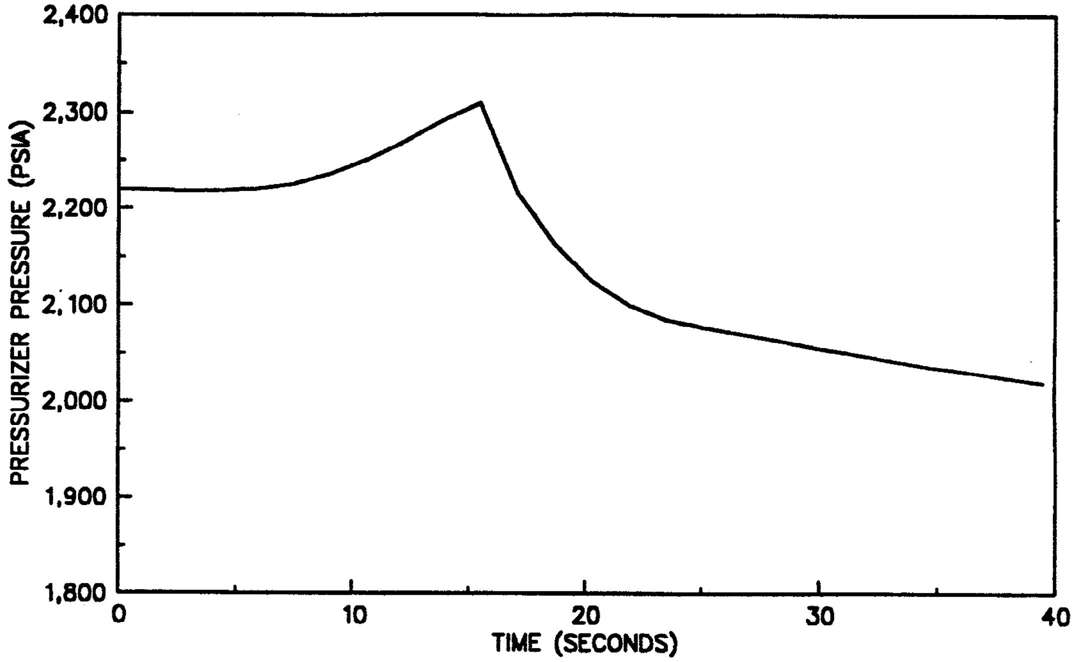

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FIGURE 15.4-14 PRESSURIZER PRESSURE TRANSIENT FOR STARTUP OF AN INACTIVE REACTOR COOLANT LOOP

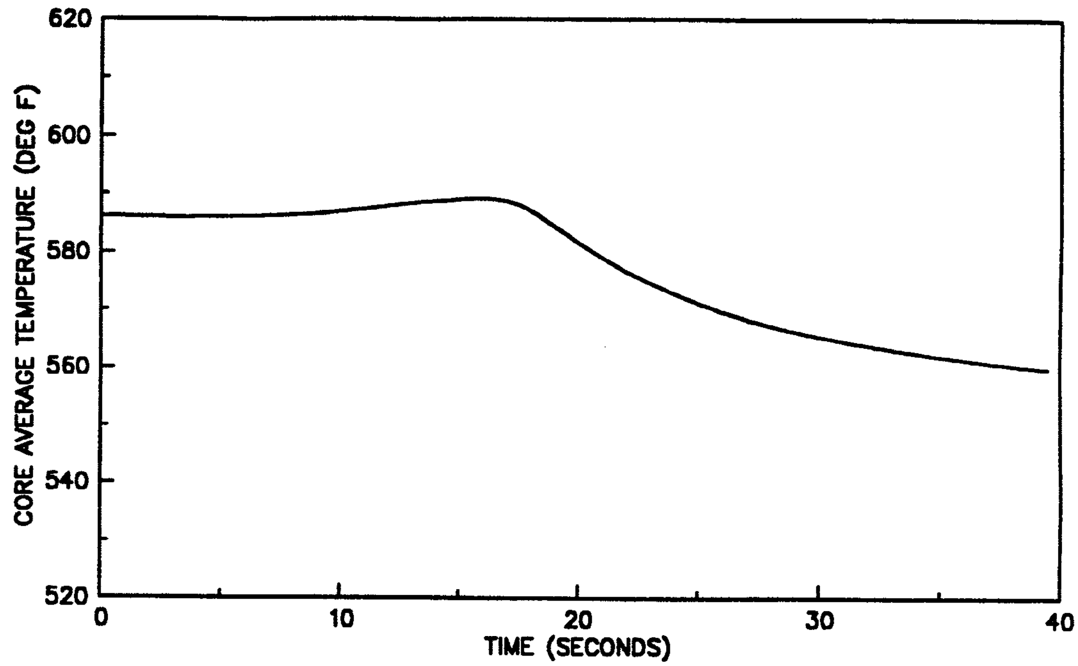

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FIGURE 15.4-15

CORE AVERAGE TEMPERATURE TRANSIENT FOR STARTUP OF AN INACTIVE REACTOR COOLANT LOOP

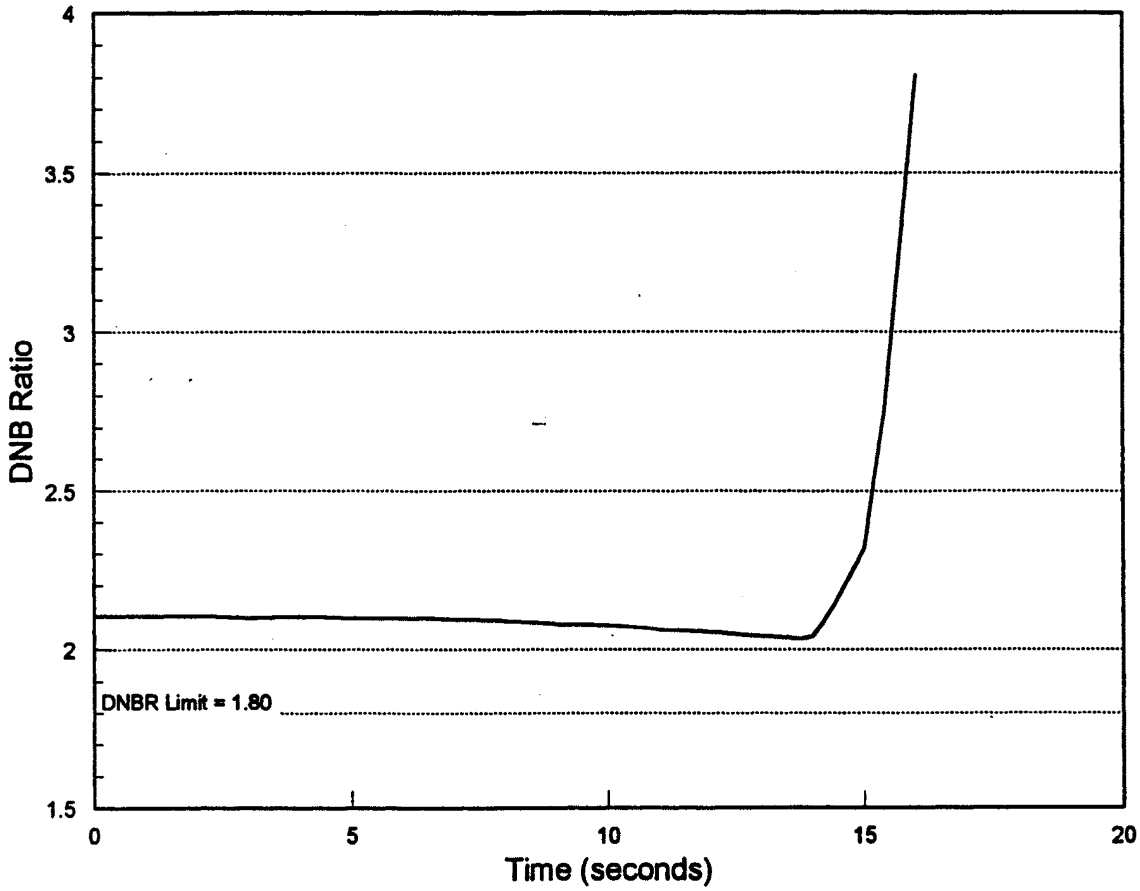

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FIGURE 15.4-16

DNBR TRANSIENT FOR STARTUP OF AN INACTIVE REACTOR COOLANT LOOP

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-8.9			-7.4					
2			-5.6			-9.1		-8.5							
3								-8.2		-6.8		-4.1		0.2	
4		-7.9	-8.2					-7.7							
5					-8.4				-6.0		-3.8		-1.8		
6	-8.5		-8.4			-7.4		-5.5						-0.3	
7				-7.7			-5.0			-1.2			-1.0		
8	-7.7		-7.3		-5.9		-3.2			1.5		3.2	3.4	3.6	
9		-6.9							2.7		5.9				6.0
10					-3.4		0.7					10.6			
11	-5.3				-1.8			5.9			17.1				11.4
12						1.3			12.3			24.6			
13			0.1		0.7			7.7						23.6	
14			2.5				4.7			11.1		17.6			
15					2.1			6.5							

CASE A

Rev. 7

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FIGURE 15.4-17
REPRESENTATIVE PERCENT CHANGE IN LOCAL ASSEMBLY AVERAGE POWER FOR AN INTERCHANGE BETWEEN A REGION 1 AND A REGION 3 ASSEMBLY

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-4.8			-4.5					
2			-0.4			-4.8		-4.8							
3								-4.8		-4.4		-2.6		1.4	
4		-2.8	-3.1					-4.5							
5					-3.9				-4.3		-4.6		-1.5		
6	-3.8		-3.8			-3.6		-2.9						0.5	
7				-3.6			-2.0			-2.3			2.2		
8	-3.5		-3.4		-2.6		-0.7			11.4		11.3	5.8	4.4	
9		-3.2							5.2		16.7				5.4
10					-1.7		0.5					8.8			
11	-2.2				-1.0			2.2			6.9				6.6
12						0.0			2.9			6.5			
13			1.2		0.0			1.6						10.3	
14			3.2				0.8			3.2		6.0			
15					0.3				1.5						

CASE B-1

Rev. 7

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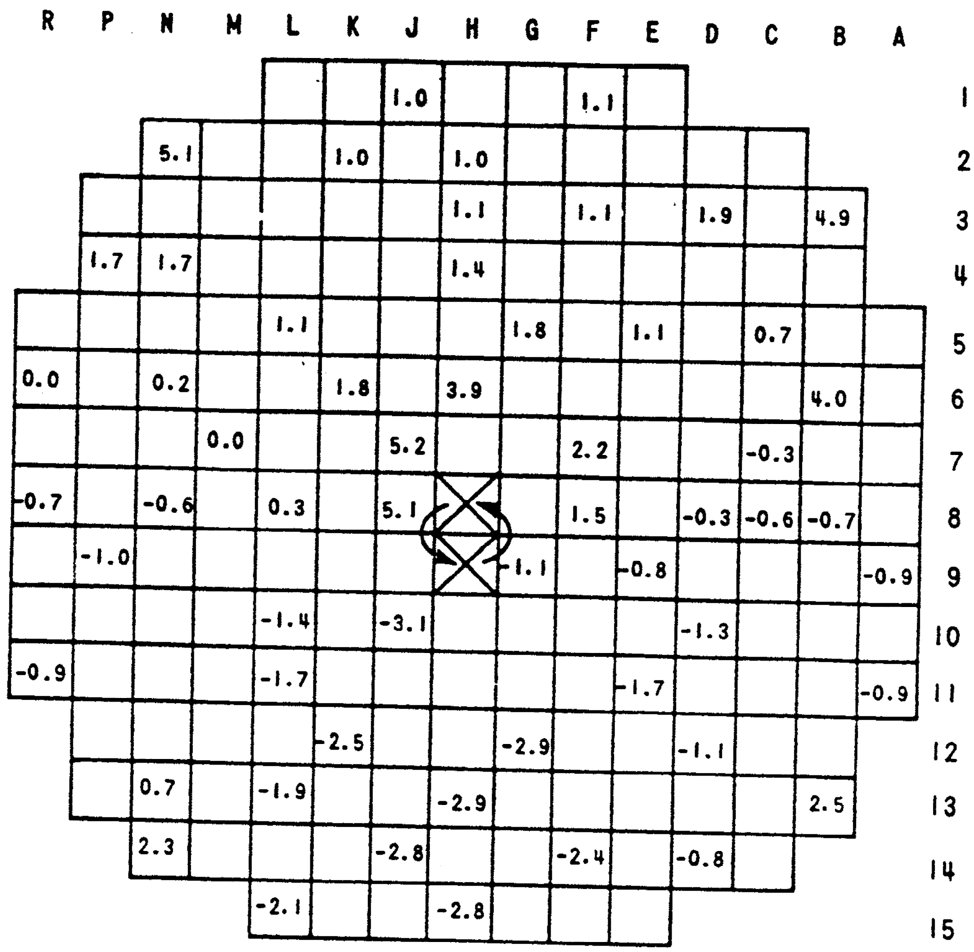
FIGURE 15.4-18
REPRESENTATIVE PERCENT CHANGE IN LOCAL ASSEMBLY AVERAGE POWER FOR AN INTERCHANGE BETWEEN A REGION 1 AND A REGION 2 ASSEMBLY WITH THE BURNABLE POISON RODS BEING RETAINED BY THE REGION 2 ASSEMBLY



CASE B-2

Rev. 7

WOLF CREEK
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FIGURE 15.4-19
REPRESENTATIVE PERCENT CHANGE IN LOCAL ASSEMBLY AVERAGE POWER FOR AN INTERCHANGE BETWEEN A REGION 1 AND A REGION 2 ASSEMBLY WITH THE BURNABLE POISON RODS BEING TRANSFERRED TO THE REGION 1 ASSEMBLY

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	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-2.2			-2.1					
2			2.0			-2.0		-2.1							
3								-1.5		-1.6		-1.0		2.0	
4		-0.9	-1.0					-0.4							
5					-0.4				1.2		-0.5		-1.4		
6	-2.1		-1.6			2.3		5.7						-2.0	
7				-3.2			9.7			4.4			-1.7		
8	-2.3		-1.6		1.8		13.6	X		5.6		-0.4	-1.6	-2.1	
9		-2.2							9.7		1.1				-2.2
10					0.3		4.5					-0.9			
11	-1.9				-0.4			1.8			-0.5				-1.9
12						-0.9			-0.6			-1.1			
13			0.4		-1.4			-1.5						2.0	
14			2.0				-2.1			-2.0		-0.9			
15					-1.9			-2.2							

CASE C

Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.4-20
REPRESENTATIVE PERCENT CHANGE IN LOCAL ASSEMBLY AVERAGE POWER FOR AN ENRICHMENT ERROR (A REGION 2 ASSEMBLY LOADED INTO THE CORE CENTRAL POSITION)

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-11			-14					
2			0.4			-9.2		-12							
3								-12		-14		-15		-13	
4		3.2	1.2					-11							
5					-1.5				-12		-15		-16		
6	9.8		7.1			-4.6		-8.0						-16	
7				9.2			-2.3			-12			-14		
8	20.0		17.8		10.8		0.8			-10		-14	-15	-16	
9		27.2							-5.5		-11				-15
10					20.7		5.8					-12			
11	42.0		X		23.6			1.9			-8.6				-13
12						14.0			-1.7			-8.9			
13			38.6		20.4			2.8						-7.0	
14			35.9				7.0			-3.3		-6.3			
15					15.3			2.9							

CASE D

Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.4-21
REPRESENTATIVE PERCENT CHANGE IN LOCAL ASSEMBLY AVERAGE POWER FOR LOADING A REGION 2 ASSEMBLY INTO A REGION 1 POSITION NEAR THE CORE PERIPHERY

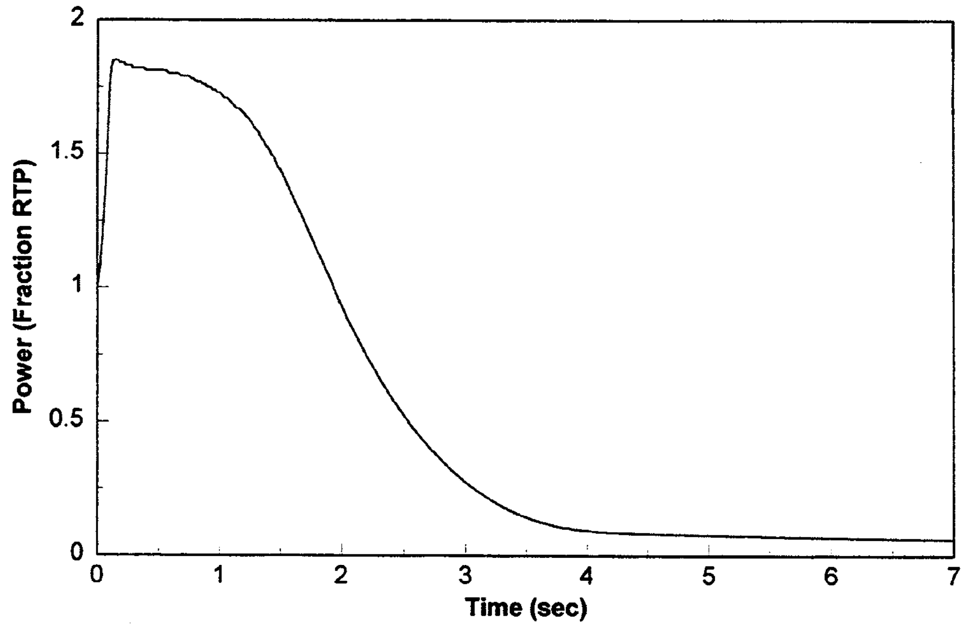

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FIGURE 15.4-22
Nuclear Power Transient, BOL, HFP, Rod Ejection Accident

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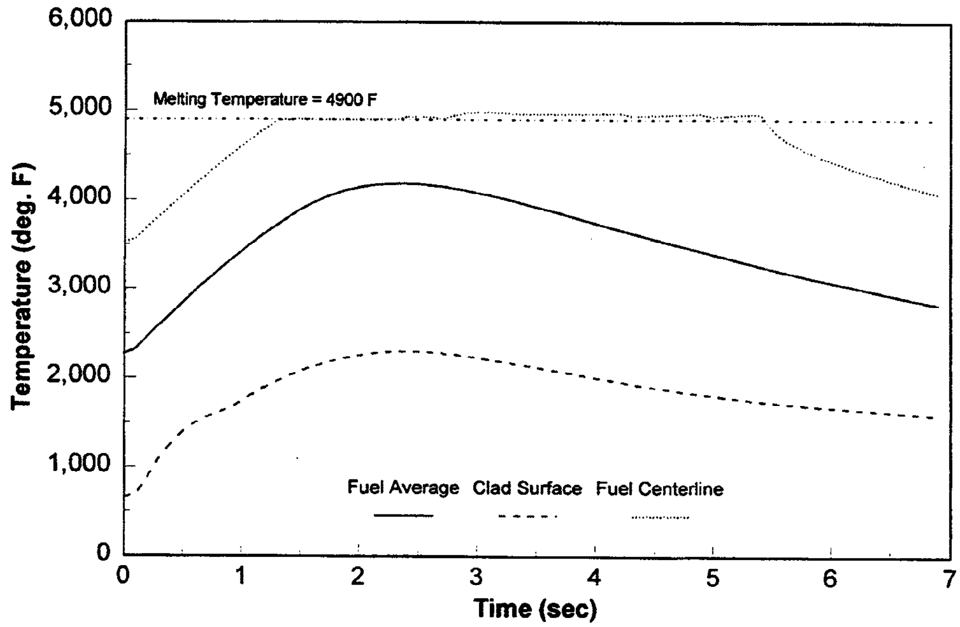

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FIGURE 15.4-23

Fuel Centerline, Fuel Average, and Cladding Surface Temperature Transients, BOL, HFP, Rod Ejection Accident

Rev. 10

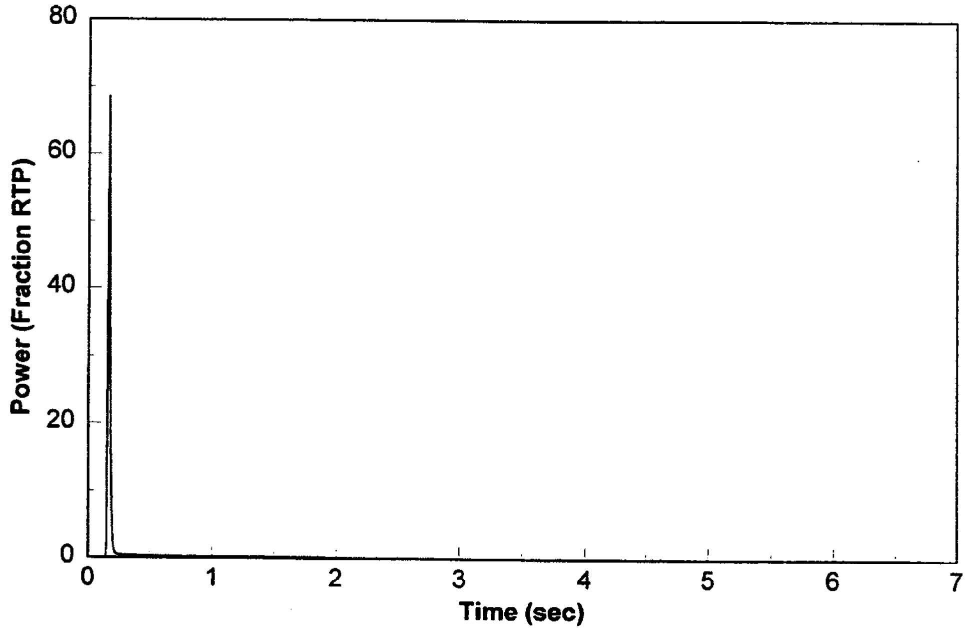

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FIGURE 15.4-24

Nuclear Power Transient, EOL, HZP, Rod Ejection Accident

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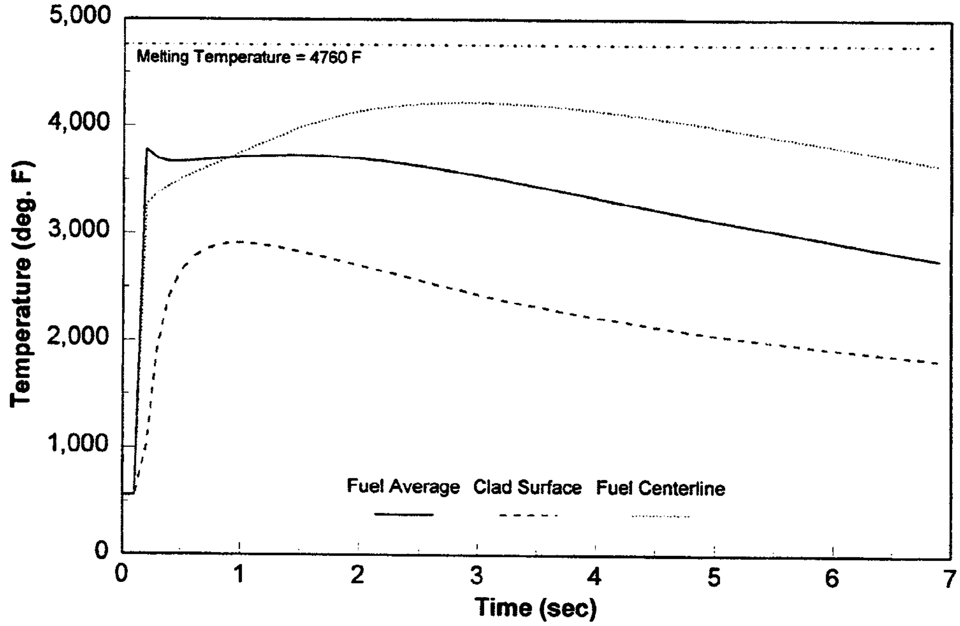

WOLF CREEK

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FIGURE 15.4-25

Fuel Centerline, Fuel Average, and Cladding Surface Temperature Transients, EOL, HZP, Rod Ejection Accident

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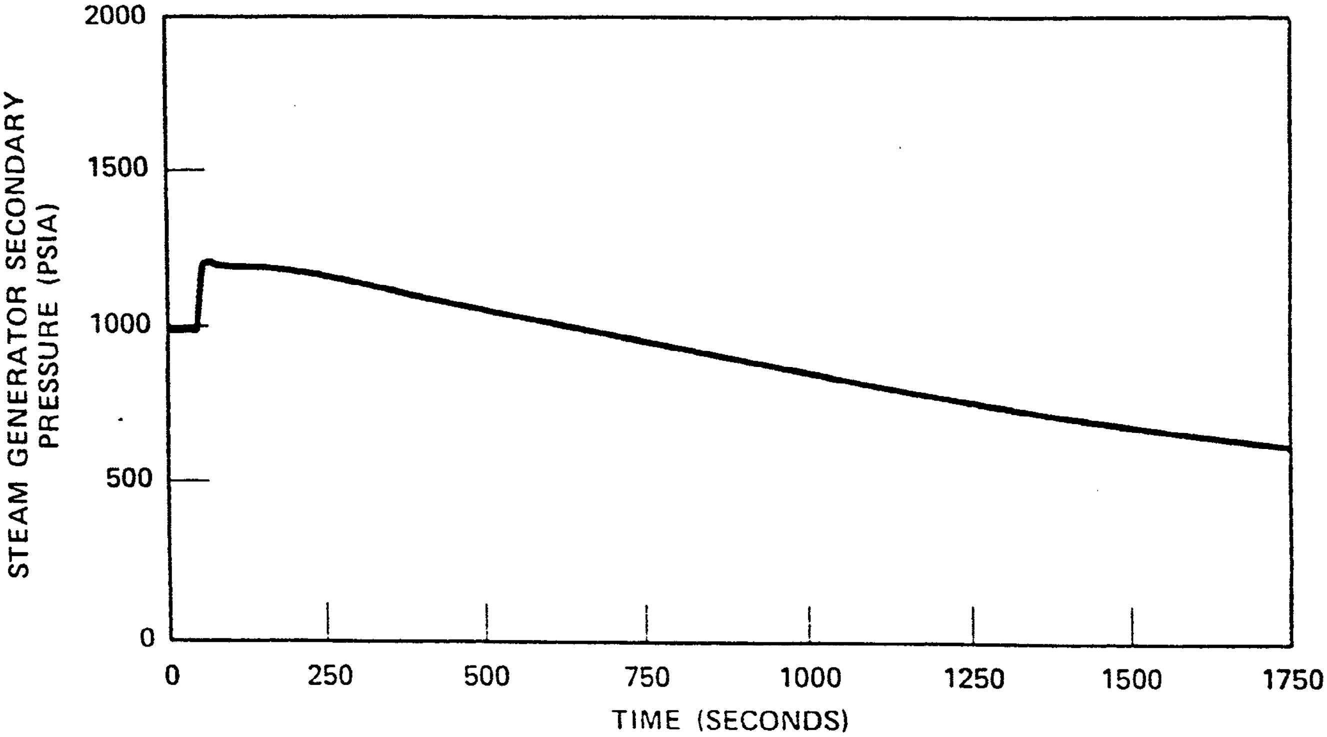

Rev. 12

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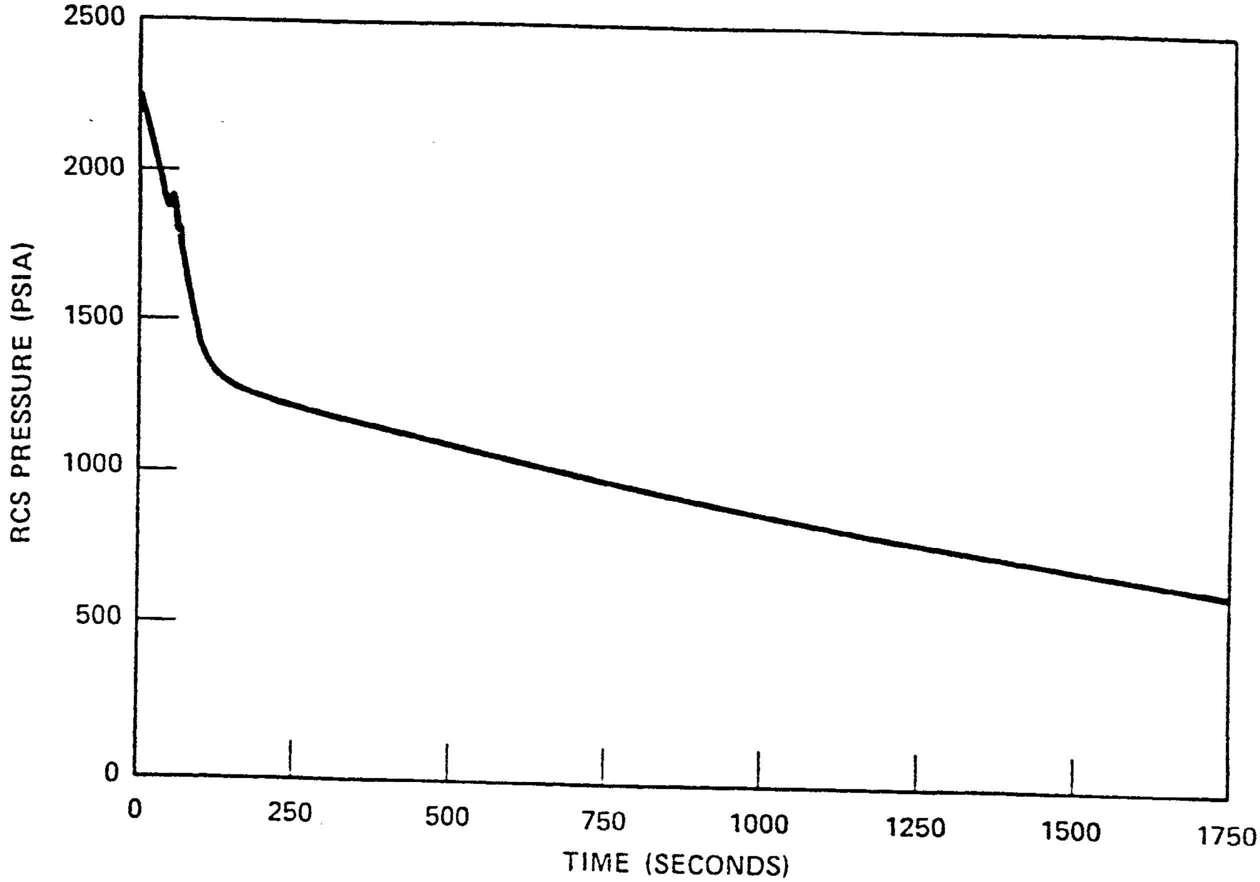
Figure 15.4-26

STEAM GENERATOR SECONDARY PRESSURE TRANSIENT FOR ROD EJECTION ANALYSIS



Rev. 7

WOLF CREEK
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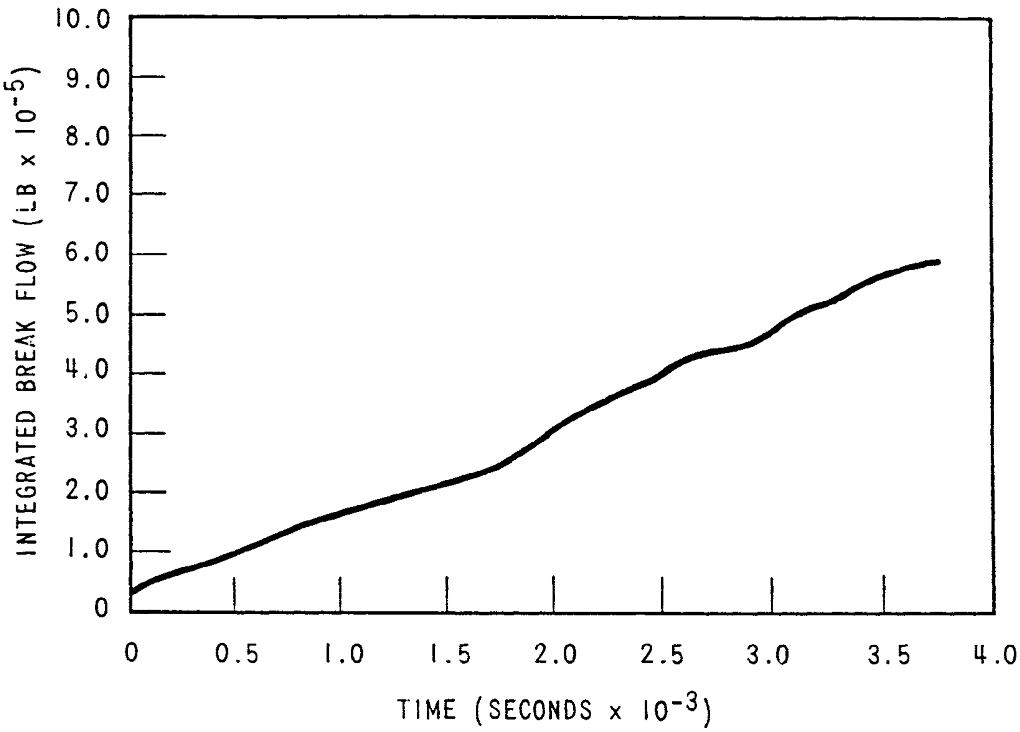
FIGURE 15.4-27

RCS PRESSURE TRANSIENT FOR
ROD EJECTION ANALYSIS



Rev. 7

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FIGURE 15.4-28 REACTOR COOLANT SYSTEM INTEGRATED BREAK FLOW FOLLOWING A ROD EJECTION ACCIDENT

15.5 INCREASE IN REACTOR COOLANT INVENTORY

Discussion and analysis of the following events are presented in this section:

a. Inadvertent operation of the emergency core cooling system during power operation.

b. Chemical and volume control system malfunction that increases reactor coolant inventory.

c. A number of BWR transients. (Not applicable to WCGS).

These events, considered to be ANS Condition II, cause an increase in reactor coolant inventory. Section 15.0.1 contains a discussion of ANS classifications.

15.5.1 INADVERTENT OPERATION OF THE EMERGENCY CORE COOLING SYSTEM DURING POWER OPERATION

15.5.1.1 Identification of Causes and Accident Description

Spurious emergency core cooling system (ECCS) operation at power could be caused by operator error or a false electrical actuation signal. A spurious signal may originate from any of the safety injection actuation channels, as described in Section 7.3.

Following the actuation signal, the suction of the charging pumps is diverted from the volume control tank to the refueling water storage tank. The valves isolating the boron injection tank from the centrifugal charging pumps and the valves isolating the boron injection tank from the injection header then automatically open. The centrifugal charging pumps then inject RWST boric acid solution into the cold leg of each loop. The safety injection pumps also start automatically but provide no flow when the reactor coolant system (RCS) is at normal pressure. The passive, accumulator safety injection system and the low head, residual heat removal system also provide no flow at normal RCS pressure.

A safety injection signal (SIS) normally results in a reactor trip followed by a turbine trip. However, it cannot be assumed that any single fault that actuates ECCS will also produce a reactor trip. If a reactor trip is generated by the spurious SIS, the operator should determine if the spurious signal was transient or steady state in nature. The operator must also determine if the SIS should be blocked. For a spurious occurrence, the operator would terminate ECCS and maintain the plant in the hot shutdown condition. If the ECCS actuation instrumentation must be repaired, subsequent plant operation would be in accordance with the Technical Specifications.

If the reactor protection system does not produce an immediate trip as a result of the spurious SIS, the reactor experiences a negative reactivity excursion due to the injected boron, causing a decrease in reactor power. The power mismatch causes a drop in Tavg and consequent coolant shrinkage. The pressurizer pressure and water level decrease. Load will decrease due to the effect of reduced steam pressure on load after the turbine throttle valve is fully open. If automatic rod control is used, these effects will be lessened until the rods have moved out of the core. The transient is eventually terminated by the reactor protection system low pressurizer pressure trip or by manual reactor trip.

The time to trip is affected by initial operating conditions, including core burnup history, which affects initial boron concentration, rate of change of boron concentration, Doppler, and moderator coefficients.

Recovery from the no reactor trip case is made in the same manner as described for the case where the SIS results directly in a reactor trip. The only difference is the lower Tavg and RCS pressure associated with the power mismatch during the transient. The time at which reactor trip occurs is of no concern for this transient. At lower loads, coolant contraction will be slower, resulting in a longer time to trip.

15.5.1.2 Analysis of Effects and Consequences

Method of Analysis

The spurious operation of the ECCS is analyzed by employing the detailed digital computer program RETRAN02 (Ref. 3). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and the effect of the SIS. The program computes pertinent plant variables, including temperatures, pressures, and power level.

Based on its expected frequency of occurrence, the inadvertent ECCS actuation at power is considered to be a Condition II event, a fault of moderate frequency. The specific criteria established for Condition II events include the following:

a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values,
b. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs,
c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

With respect to overpressurization, the inadvertent ECCS actuation at power event is bounded by the Loss of Load/Turbine trip events, in which assumptions are made to conservatively calculate the RCS and main steam system pressure transients. For the inadvertent ECCS actuation at power event, turbine trip occurs following reactor trip, whereas for the Loss of Load/Turbine Trip event, the turbine trip is the initiating fault. Therefore, the primary to secondary power mismatch and resultant RCS and main steam system heatup and pressurization transient are always more severe for the Loss of Load/Turbine Trip event. For this reason, it is not necessary to calculate the maximum RCS or main steam system pressure for the inadvertent ECCS actuation at power.

Based on historical precedence, the inadvertent ECCS actuation at power event does not lead to a serious challenge of the DNB design basis. The decrease in core power and RCS average temperature more than offset the decrease in RCS pressure such that the minimum calculated DNBR occurs at the start of the transient. As such, no explicit analysis of the event has been performed to DNB concerns.

The primary concern that results from an inadvertent actuation of the ECCS at power is that associated with pressurizer overfill. The pressurizer water volume increases as a result of the safety injection flow. This may eventually lead to filling of the pressurizer and subsequent water relief through the safety or relief valves. The passing of liquid through the pressurizer safety or relief valves could result in rupture of the pressurizer relief tank rupture disks, spilling radioactive coolant into the containment building, thereby escalating a Condition II event to a Condition III or IV event. Historically, Criterion c has been shown to be met by demonstrating that the pressurizer will not become water-solid. To prevent the pressurizer from becoming water-solid, operator action is ultimately required to terminate safety injection or mitigate the consequences of this event. Therefore, the event is analyzed to demonstrate that sufficient time is available for the appropriate operator actions to be taken to prevent filling the pressurizer.

A bounding pressurizer overfill transient is presented representing maximum reactivity feedback. For calculational simplicity, zero injection line purge volume was assumed in this analysis, thus the boration transient begins immediately when the appropriate valves are opened. Plant characteristics and initial conditions are further discussed in Section 15.0.3.

The assumptions are as follows:

a. Initial operating conditions

   Initial conditions with maximum uncertainties on power (+2%, vessel average temperature (-6.5°F), and pressurizer pressure (-30 psi) are assumed. The lower initial temperature results in a higher RCS coolant mass, causing a more severe pressurizer water volume transient.

b. Reactivity feedback

   A large (absolute value) negative moderator temperature coefficient and a most-negative Doppler power coefficient are assumed.

c. Reactor control

   The reactor was assumed to trip at the time of the SI signal. Thus, the reactor control mode is of no consequence.

d. Pressurizer heaters

   Pressurizer heaters are assumed to be inoperable. This assumption yields a higher rate of pressure drop.

e. Boron Injection

   At the initiation of the event, two ECCS centrifugal charging pumps (CCPs) and a normal charging pump (NCP) inject RWST boron solution into the cold leg of each loop.

f. Turbine load
The reactor and turbine both trip at the time of SI actuation with the turbine load dropping to zero simultaneously with the steam flow.

g. ECCS injection flowrates
110% of the maximum safeguards ECCS flow is delivered from two CCPs, plus flow from a NCP used for normal charging prior to the occurrence of SI signal. All letdown is assumed isolated at the initiation of the event.

h. Reactor Trip
An immediate reactor trip on the initiating SI signal to maximize pressurizer inventory is assumed.

i. Operator Action to terminate SI
The operator would take necessary action to terminate the safety injection flow within 8 minutes following event initiation. The reduced operator action time is required to preclude the possibility of a pressurizer water-solid condition and is supported by plant operations through use of an emergency operating procedure.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-22. No single active failure in any of these systems or equipment will adversely offset the consequences of the accident.

Results

The calculated sequence of events is shown on Table 15.5-1.

Figures 15.5-1 through 15.5-3 show the transient response to inadvertent operation of ECCS during power operation.

Reactor trip occurs at event initiation followed by a rapid cooldown of the RCS. Coolant contraction due to the decrease in reactor power and RCS temperature results in a short-term reduction in pressurizer pressure and water level. The temperature drop is mainly caused by the decrease in power but is also affected by the addition of the cold SI flow. The combination of the RCS heatup, due to residual heat generation, and ECCS injected flow causes the pressure and level transients to rapidly turn around. Pressurizer water level then increases steadily and reaches the peak water volume of 1763.5 ft^3 shortly after the ECCS flow is terminated. The pressurizer does not reach a water-solid condition, and hence, no water relief occurs through the pressurizer safety or relief valves.

Recovery from this accident is discussed in Section 15.5.1.1.

15.5.1.3 Conclusions

Results of the analysis show that the spurious ECCS operation with immediate reactor trip meets all acceptance criteria for an ANS Condition II event.

If the reactor does not trip immediately, the low pressurizer pressure reactor trip will be actuated. This trips the turbine and prevents excess cooldown, thereby expediting recovery from the incident.

15.5.2 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT INCREASES REACTOR COOLANT INVENTORY

15.5.2.1 Identification of Causes and Accident Description

Increases in reactor coolant inventory caused by the chemical and volume control system may be postulated to result from operator error or a false electrical signal. Transients examined in this

section are characterized by increasing pressurizer level, increasing pressurizer pressure, and constant boron concentration. The transients analyzed in this section are done to demonstrate that there is adequate time for the operator to take corrective action to prevent filling the pressurizer. An increase in reactor coolant inventory, which results from the addition of cold, unborated water to the RCS, is analyzed in Section 15.4.6, CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT RESULTS IN A DECREASE IN BORON CONCENTRATION IN THE REACTOR COOLANT. An increase in reactor coolant inventory which results from the injection of highly borated water into the RCS is analyzed in Section 15.5.1, INADVERTENT OPERATION OF THE EMERGENCY CORE COOLING SYSTEM DURING POWER OPERATION.

Transients postulated as a result of operator error or failure of the charging pump controller which increase primary side inventory will be automatically terminated by a high pressurizer level reactor trip before the pressurizer can be filled, thus these are not the worst cases.

The most limiting case would result if charging was in automatic control and the pressurizer level channel being used for charging control failed in a low direction. This would cause maximum charging flow to be delivered to the RCS and letdown flow would be isolated. The worst single failure for this event would be another pressurizer level channel failing in an as is condition or a low condition. This will defeat the reactor trip on two out of three high pressurizer level channels. To prevent filling the pressurizer the operator must be relied upon to terminate charging.

15.5.2.2 Analysis of Effects and Consequences

Method of Analysis

The charging malfunction is analyzed by employing the detailed digital computer program LOFTRAN (Ref. 1). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and the effect of the SIS. The program computes pertinent plant variables, including temperatures, pressures, and power level.

Four cases were analyzed;

a. Minimum reactivity feedback with automatic pressurizer spray

b. Minimum reactivity feedback without automatic pressurizer spray

c. Maximum reactivity feedback with automatic pressurizer spray

d. Maximum reactivity feedback without automatic pressurizer spray

The assumptions incorporated in the analyses were as follows;

a. Initial Operating Conditions

   The initial reactor power, RCS temperature, RCS pressure, and pressurizer level are assumed at their nominal values consistent with steady state full power operation to perform an SCD analysis.

b. Reactivity Coefficients

   1. Minimum Reactivity Feedback Case

      A most positive moderator temperature coefficient and a least negative Doppler-only power coefficient.

   2. Maximum Reactivity Feedback Case

      A conservatively large negative moderator temperature coefficient and a most negative Doppler-only power coefficient.

c. Reactor Control

   A conservative analysis is performed assuming the reactor is in manual control.

d. Charging System

   Maximum charging system flow based on RCS back pressure from one centrifugal charging pump is delivered to the RCS. The charging flow is assumed to have the same boron concentration as the RCS.

e. Reactor Trip

   The transient is initiated by the pressurizer level channel which is used for control purposes failing low. As a worst single failure, another pressurizer level channel fails low, defeating the two out of three high pressurizer level trip. Reactor trip on low pressurizer pressure is modeled in the cases presented herein.

Results

Figures 15.5-4 through 15.5-11 show the transient response due to the charging system malfunction. In the analyzed cases which model maximum reactivity feedback, core power and RCS average temperature remain relatively constant. In the cases which model minimum reactivity feedback, the core power initially decreases resulting in a reduction in RCS average temperature and pressure. A reactor trip ultimately occurs on low pressurizer pressure.

Cases where the pressurizer spray is inoperable show the pressurizer level increases at a relatively constant rate (following reactor trip for the minimum feedback case). This is because the pressurizer pressure initially rises very quickly to the pressure at which the relief valves open and remains there.

Cases where the pressurizer spray is operable show the pressurizer level increases with varying rates. Spray actuation tends to keep the pressurizer pressure lower for several minutes, which allows the charging pumps to deliver more flow. Eventually, pressurizer pressure does increase enough to open the relief valves.

Times at which the operator would receive alarms are listed in Table 15.5-1.

15.5.2.3 Conclusions

Results show none of the operating conditions during the transient approach core limits. The high pressurizer level trip has been defeated by failures, however, reactor trip on low pressurizer pressure occurs in some analyzed cases. The transient is ultimately terminated by the plant operators by isolating the ECCS injection and, thereby, terminating the pressurizer in-surge. The sequence of events presented in Table 15.5-1 show that the operators have sufficient time to take action necessary to prevent a pressurizer water-solid condition.

15.5.3 A NUMBER OF BWR TRANSIENTS

This section is not applicable to WCGS.

15.5.4 REFERENCES

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

2. Stewart, C. W., et. al.,"VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores," Battelle, Pacific Northwest Laboratories, EPRI NP-2511-CCM-A, August 1989.

3. McFadden, J. H., et. Al., "RETRAN-02 - A Program for Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," EPRI NP-1850-CCM-A, October 1984.

4. WCAP-11397-P-A, "Revised Thermal Design Procedure," A. J. Fiedland, et al., April 1989.

TABLE 15.5-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN AN INCREASE IN REACTOR COOLANT INVENTORY

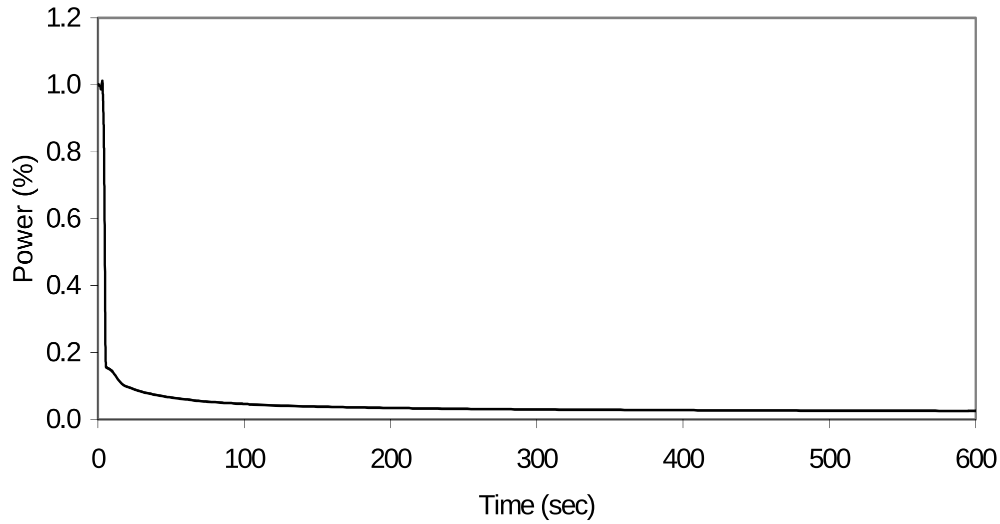
Accident	Event	Time (sec)
Inadvertent actuation of The ECCS during power Operation	ECCS injection begins	0.0
	Control rod motion begins (actuated on SI signal)	2.0
	Peak Pressurizer pressure occurs	17.5
	AFW flow delivered to four steam generators	392
	Safety injection flow terminated (operator action)	480
	Peak pressurizer water volume occurs	~600
Chemical and volume control system malfunction, minimum reactivity feedback, without pressurizer spray	Two pressurizer level channels fail low	0.0
	Maximum charging flow from one centrifugal charging pump is begun	0.0
	Letdown is isolated	0.0
	Lo-lo pressurizer level alarm	0.0
	Pressurizer relief valve setpoint reached	358.
	Hi pressurizer level alarm from the one working level channel	1388.
	Pressurizer fills	1572.

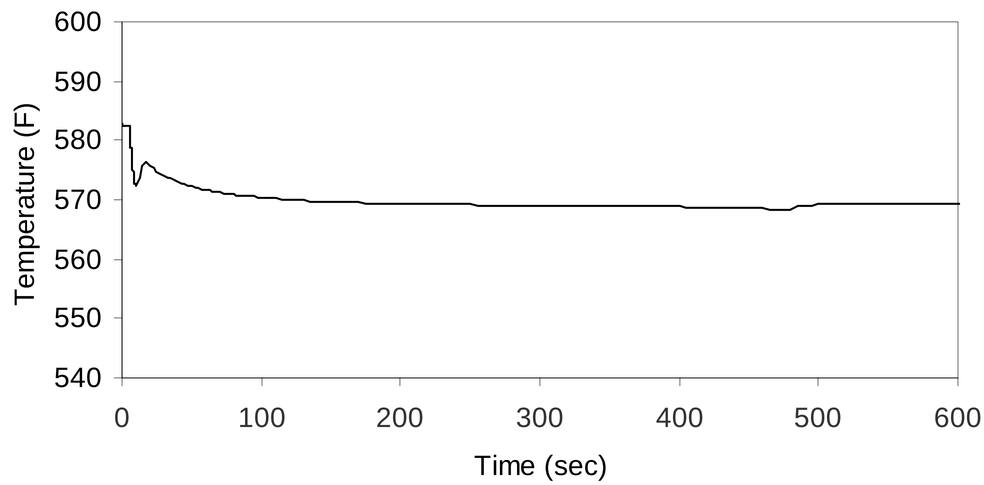
TABLE 15.5-1 (Sheet 2)

Accident	Event	Time (sec)
Chemical and volume control system malfunction, minimum reactivity feed-back, with pressurizer spray	Two pressurizer level channels fail low	0.0
	Maximum charging flow from one centrifugal charging pump is begun	0.0
	Letdown is isolated	0.0
	Lo-lo pressurizer level alarm	0.0
	Hi pressurizer level alarm from the one working level channel	1490.
	Pressurizer relief valve setpoint reached	1712.
Chemical and volume control system malfunction, maximum reactivity feed-back, without pressurizer spray	Two pressurizer level channels fail low	0.0
	Maximum charging flow from one centrifugal charging pump is begun	0.0
	Letdown is isolated	0.0
	Lo-lo pressurizer level alarm	0.0
	Pressurizer relief valve setpoint reached	77.
	Hi pressurizer level alarm from the one working level channel	1468.
	Pressurizer fills	1766.

TABLE 15.5-1 (Sheet 3)

Accident	Event	Time (sec)
Chemical and volume control system mal-function, maximum reac-tivity feed-back, with pressurizer spray	Two pressurizer level channels fail low	0.0
	Maximum charging flow from one centrifugal charging pump is begun	0.0
	Letdown is isolated	0.0
	Lo-lo pressurizer level alarm	0.0
	Hi pressurizer level alarm from the one working level channel	916.
	Pressurizer relief valve setpoint reached	1400.
	Pressurizer fills	1704.





Rev. 17

WOLF CREEK
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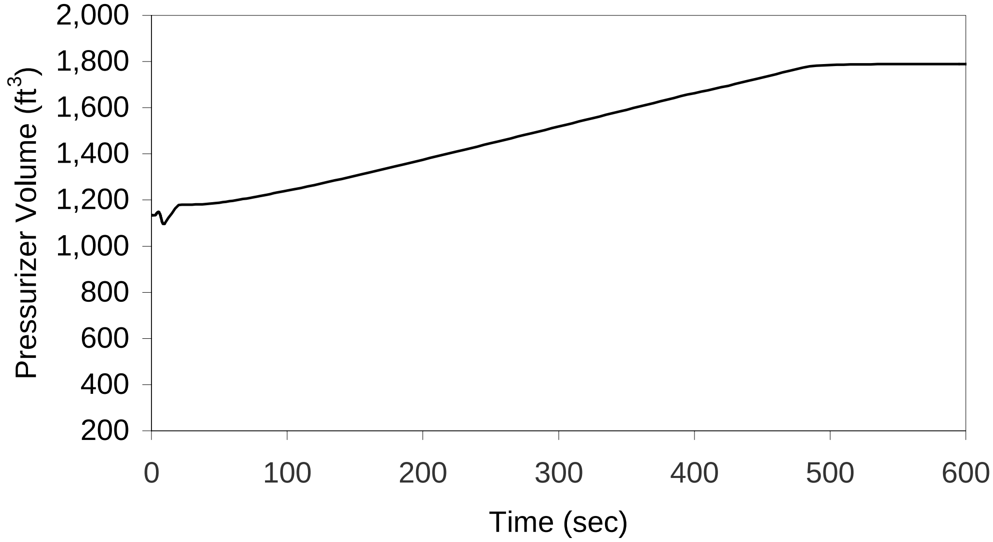
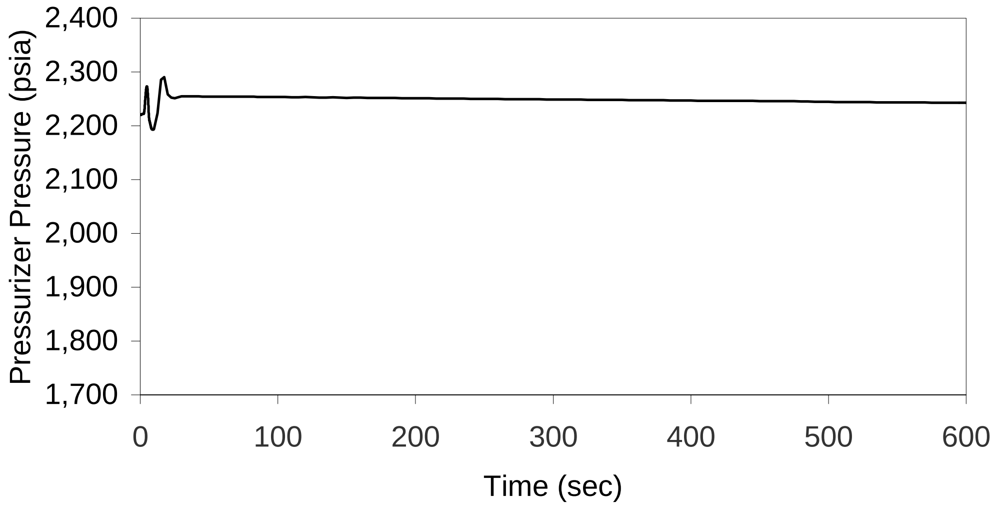
FIGURE 15.5-1

INADVERTENT OPERATION OF ECCS
DURING POWER OPERATION





Rev. 17

WOLF CREEK
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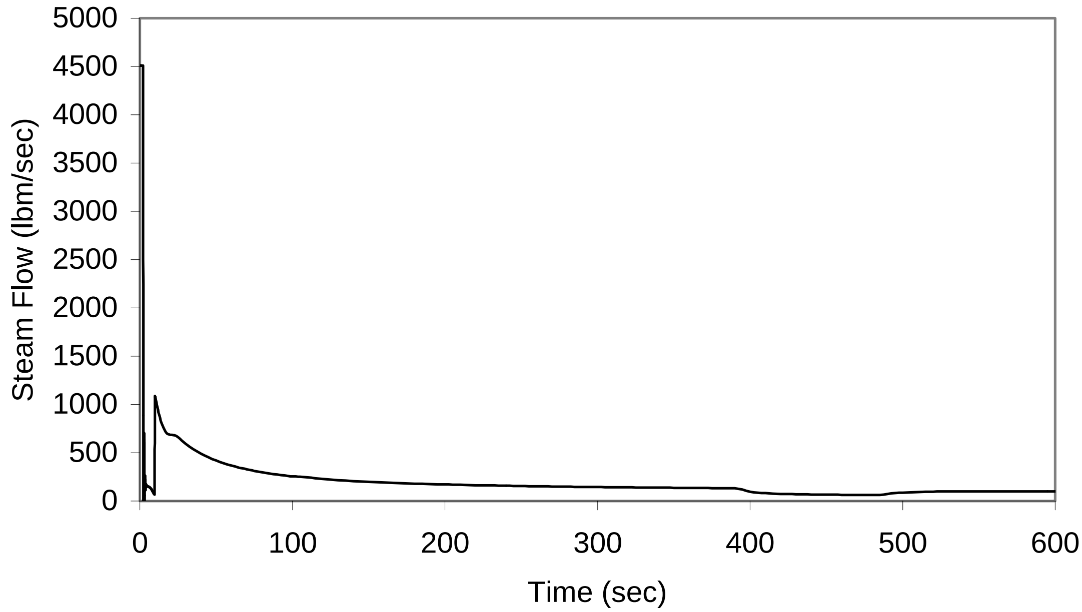
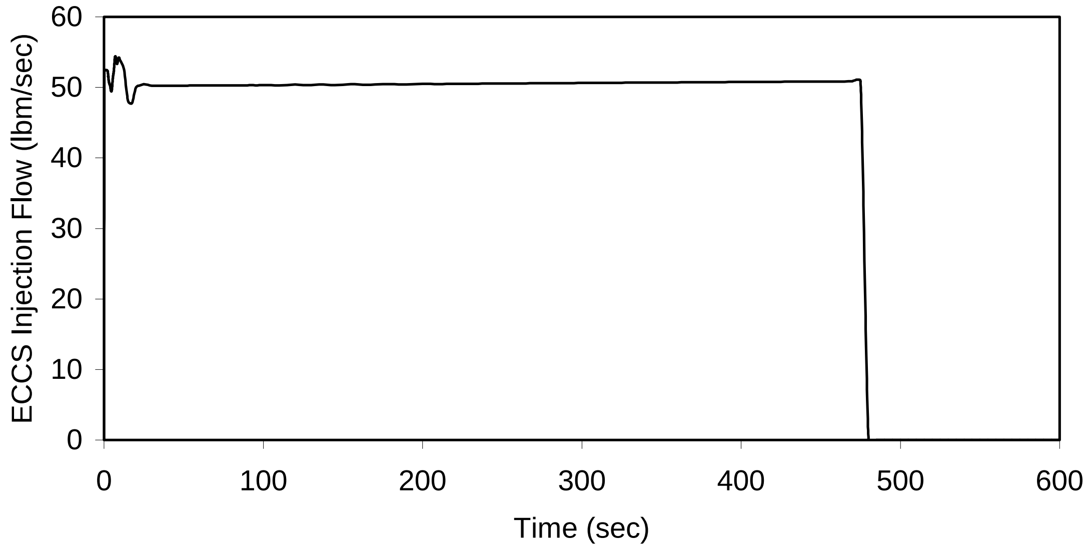
FIGURE 15.5-2

INADVERTENT OPERATION OF ECCS
DURING POWER OPERATION



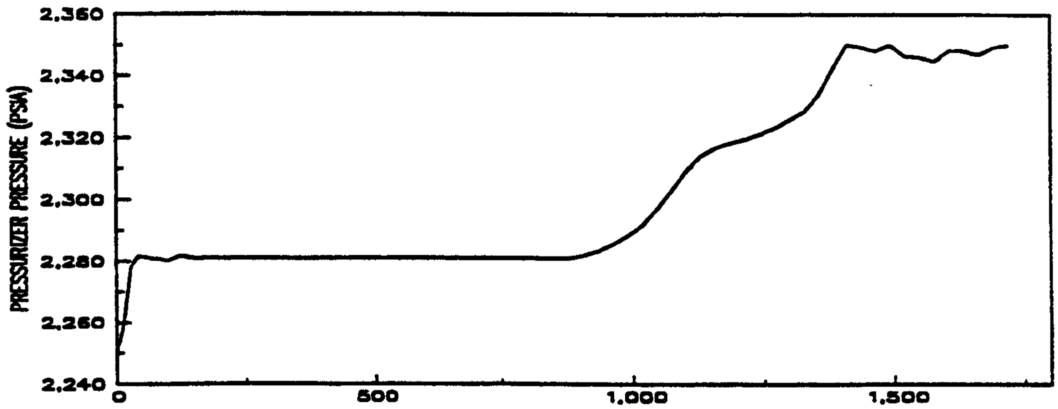


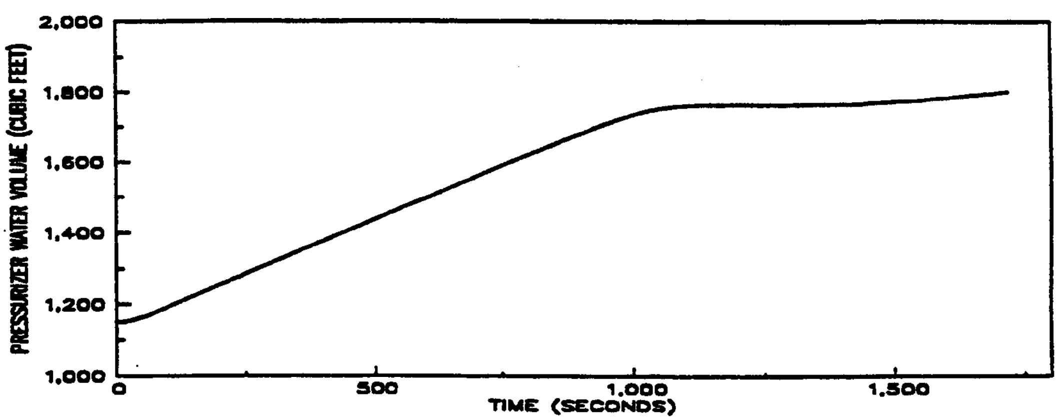
Rev. 17

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.5-3

INADVERTENT OPERATION OF ECCS
DURING POWER OPERATION

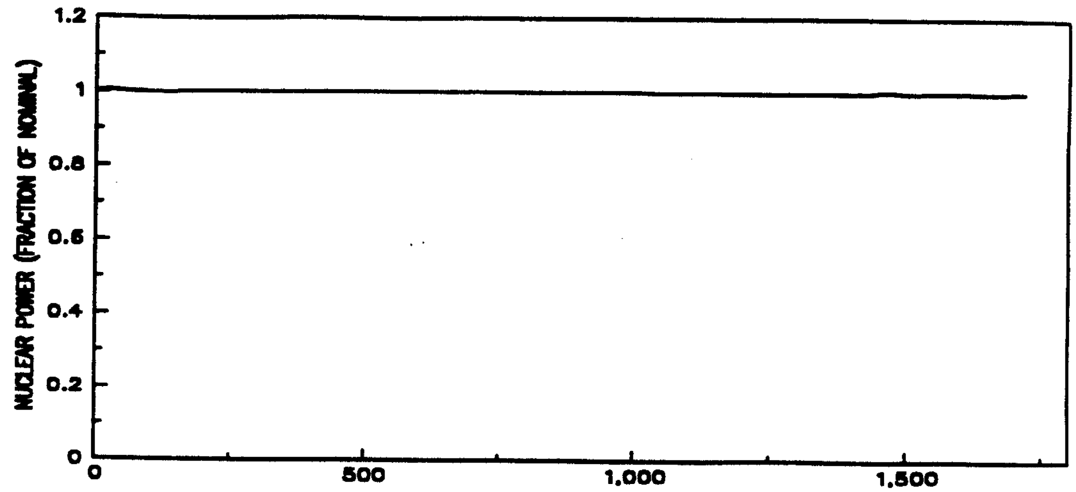


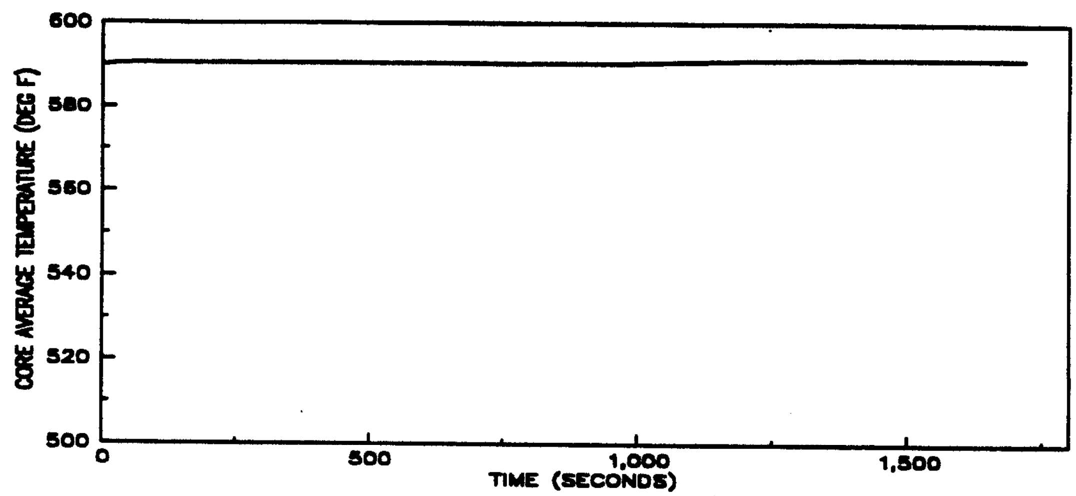


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.5-4
CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION MAXIMUM REACTIVITY FEEDBACK WITH PRESSURIZER SPRAY

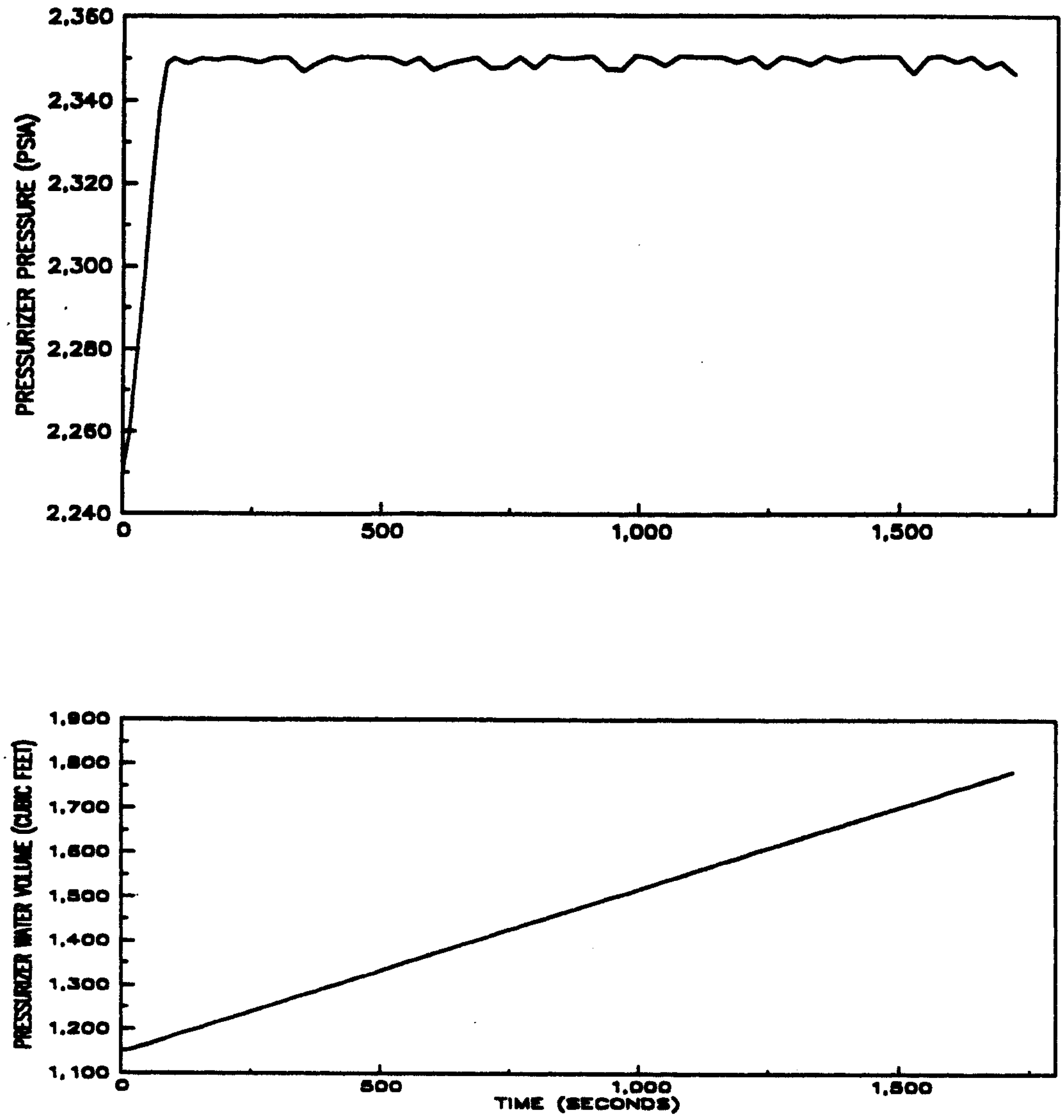




Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

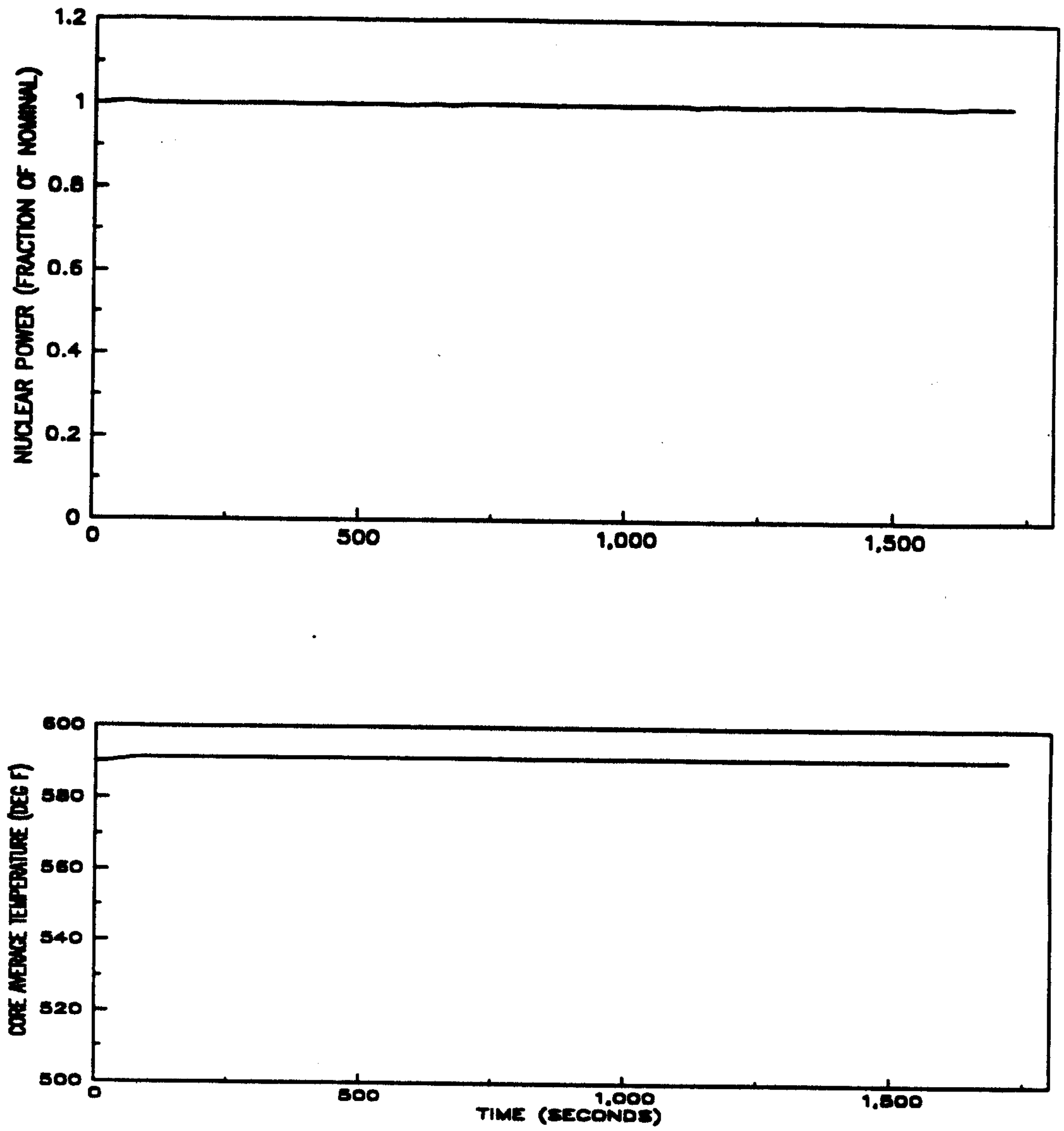
FIGURE 15.5-5
CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION MAXIMUM REACTIVITY FEEDBACK WITH PRESSURIZER SPRAY



Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

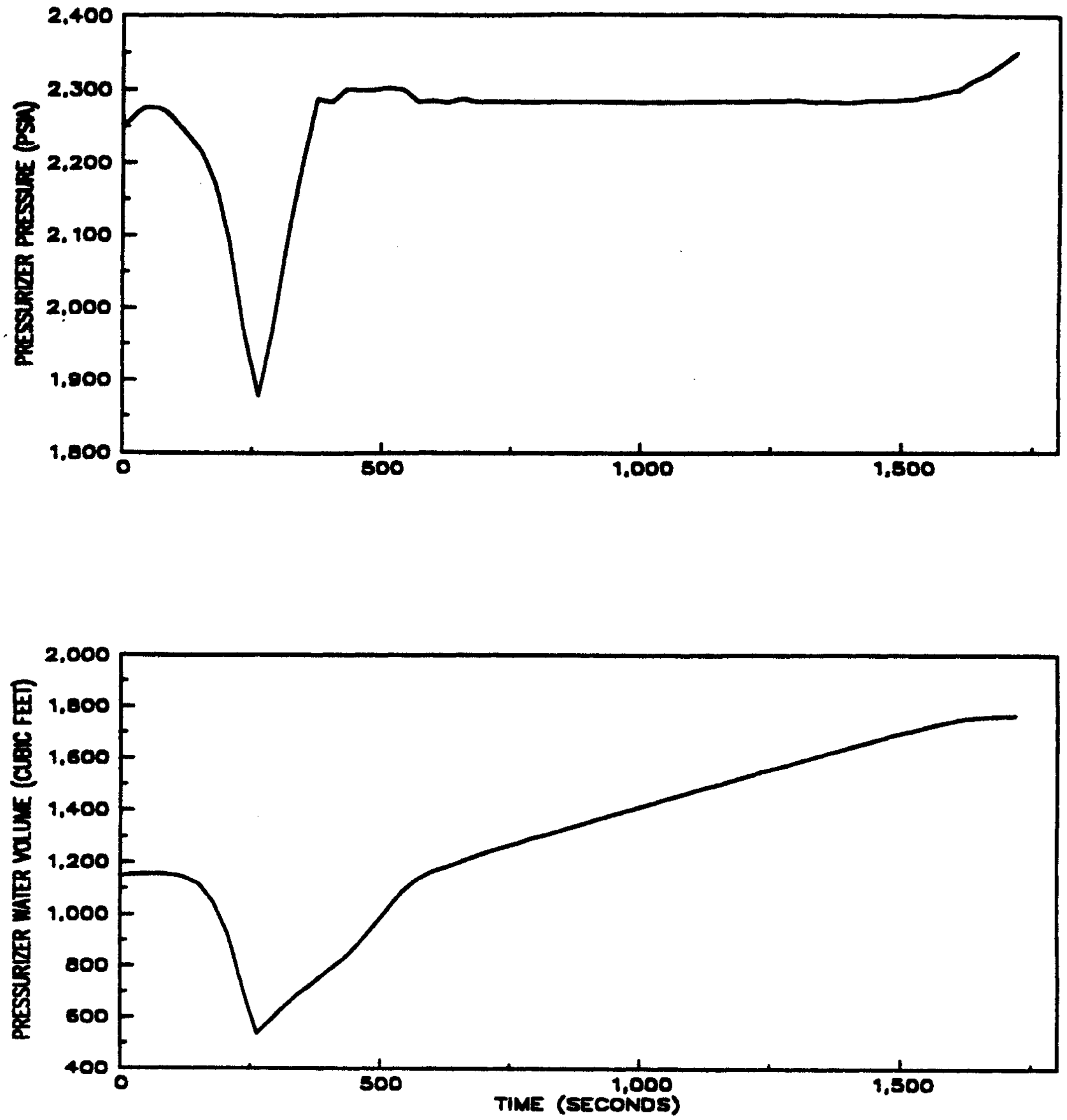
FIGURE 15.5-6
CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION MAXIMUM REACTIVITY FEEDBACK WITHOUT PRESSURIZER SPRAY



Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

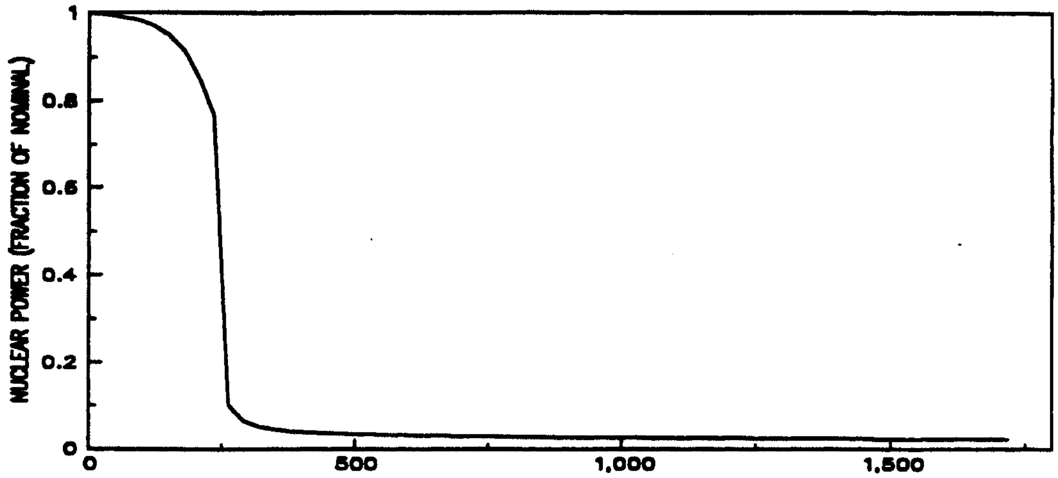
FIGURE 15.5-7
CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION MAXIMUM REACTIVITY FEEDBACK WITHOUT PRESSURIZER SPRAY

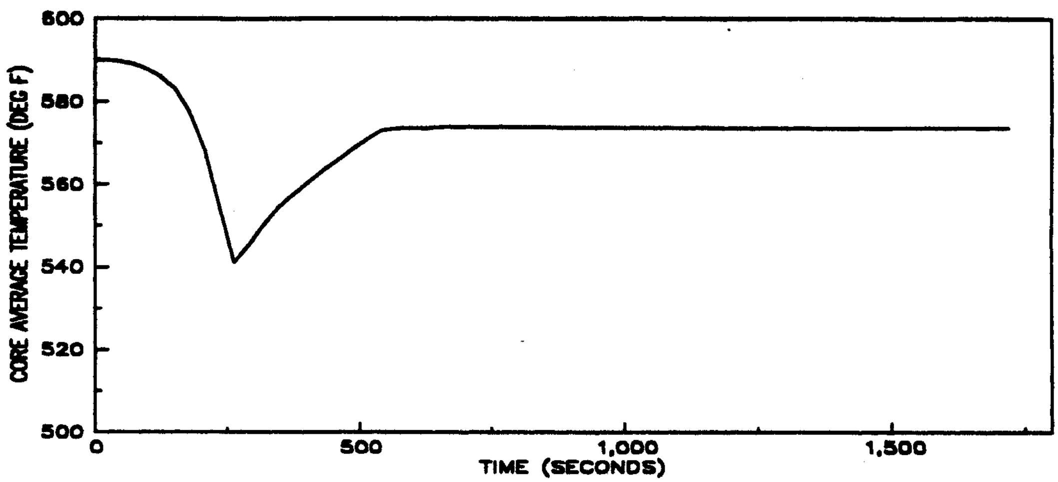


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.5-8
CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION MINIMUM REACTIVITY FEEDBACK WITH PRESSURIZER SPRAY

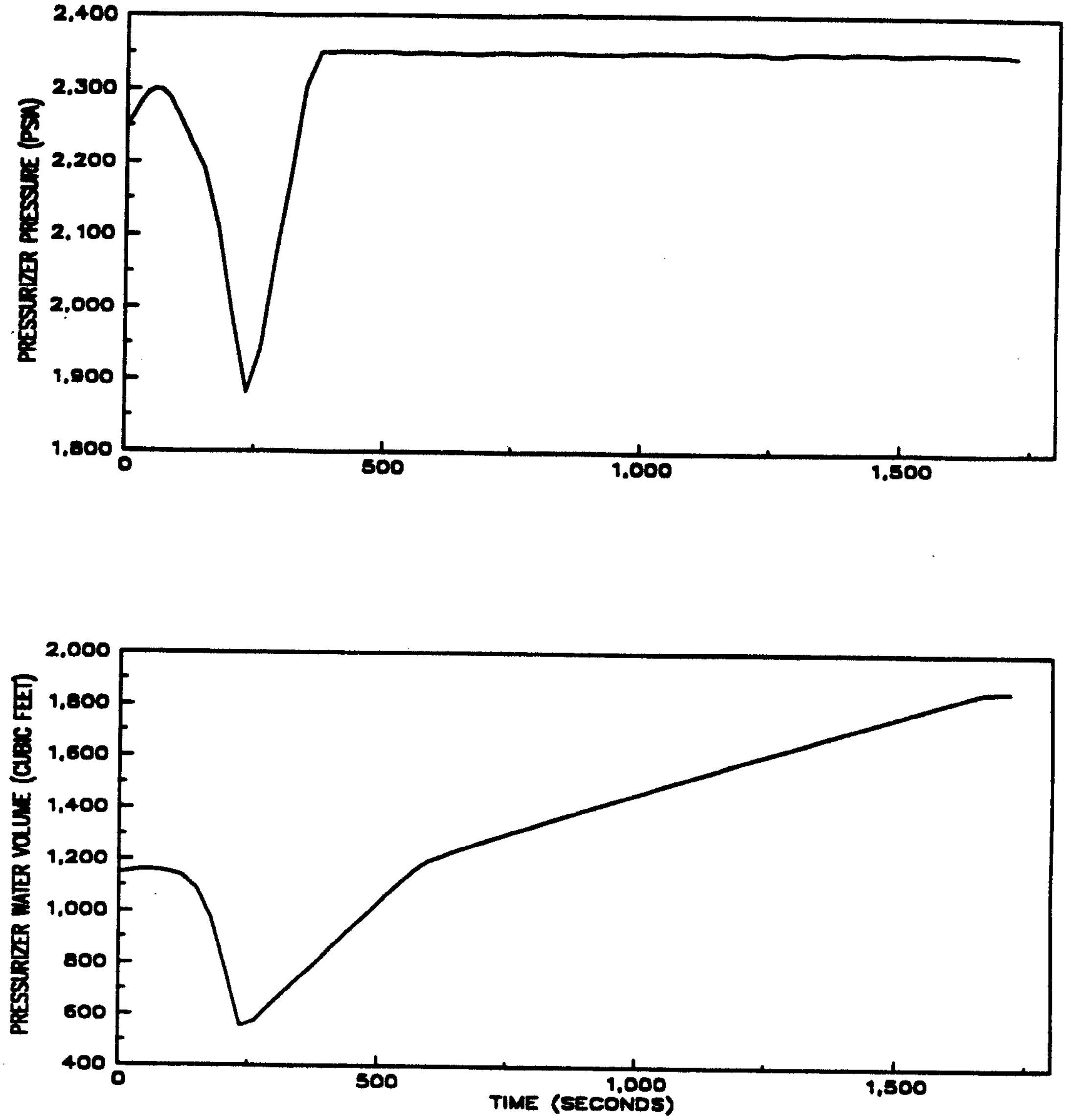




Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

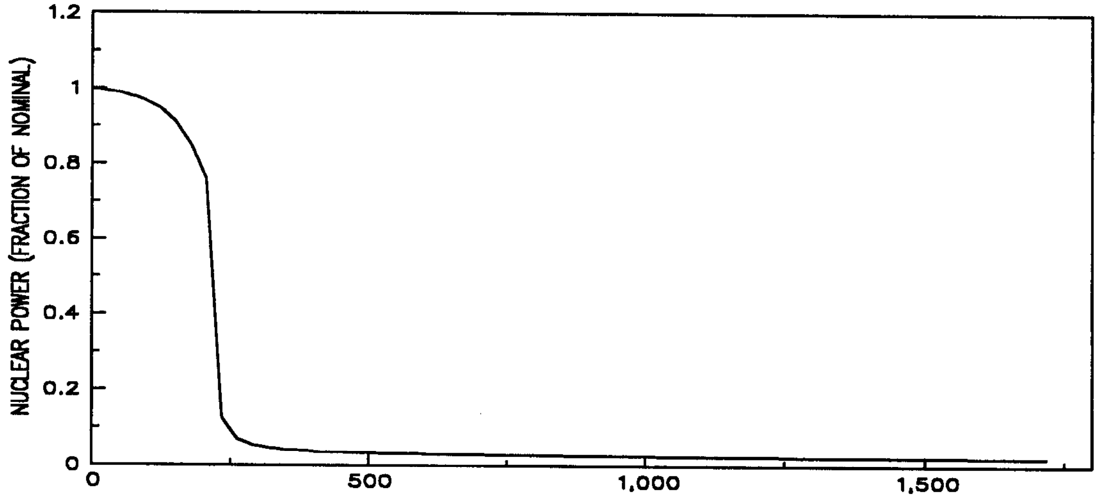
FIGURE 15.5-9
CHEMICAL AND VOLUME CONTROL
SYSTEM MALFUNCTION MINIMUM
REACTIVITY FEEDBACK WITH
PRESSURIZER SPRAY

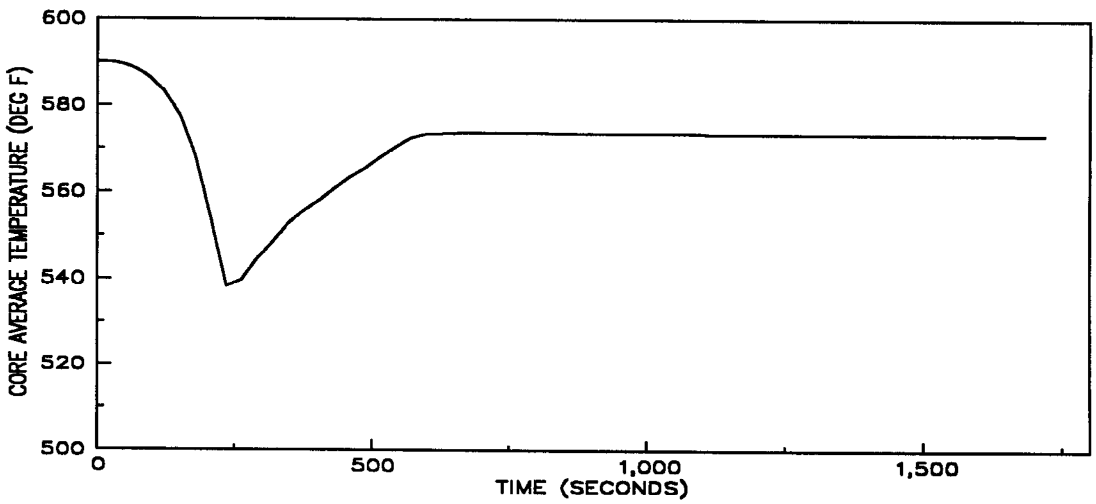


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.5-10
CHEMICAL AND VOLUME CONTROL
SYSTEM MALFUNCTION MINIMUM
REACTIVITY FEEDBACK WITHOUT
PRESSURIZER SPRAY





Rev 7

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FIGURE 15.5-11 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION MINIMUM REACTIVITY FEEDBACK WITHOUT PRESSURIZER SPRAY

15.6 DECREASE IN REACTOR COOLANT INVENTORY

Events which result in a decrease in reactor coolant inventory, as discussed in this section, are as follows:

a. Inadvertent opening of a pressurizer safety or relief valve

b. Break in instrument line or other lines from the reactor coolant pressure boundary that penetrate the containment

c. Steam generator tube rupture (SGTR)

d. Spectrum of boiling water reactor (BWR) steam system piping failures outside of the containment (Not applicable to WCGS)

e. Loss-of-coolant accident (LOCA) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary

f. A number of BWR transients (Not applicable to WCGS)

All of the applicable accidents in this category have been analyzed. It has been determined that the most severe radiological consequences will result from the major LOCA of Section 15.6.5. Therefore, the LOCA is the design basis accident. The LOCA CVCS letdown line break outside the containment and the SGTR accident have been analyzed radiologically. All other accidents in this section are bounded by these accidents.

15.6.1 INADVERTENT OPENING OF A PRESSURIZER SAFETY OR RELIEF VALVE

15.6.1.1 Identification of Causes and Accident Description

An accidental depressurization of the reactor coolant system (RCS) could occur as a result of an inadvertent opening of a pressurizer relief or safety valve. Since a safety valve is sized to relieve approximately twice the steam flow rate of a relief valve, and will therefore allow a much more rapid depressurization upon opening, the most severe core conditions resulting from an accidental depressurization of the RCS are associated with an inadvertent opening of a pressurizer safety valve. Initially, the event results in a rapidly decreasing RCS pressure until this pressure reaches a value corresponding to the hot leg saturation pressure. At this time, the rate of pressure decrease is reduced considerably. The pressure continues to decrease throughout the transient. The effect of the pressure decrease would be to decrease power via the moderator density feedback, but the reactor control system (if in the automatic mode) functions to maintain the power essentially constant throughout the initial stage of the transient. The average coolant temperature decreases slowly, but the pressurizer level increases until reactor trip.

The reactor may be tripped by the following reactor protection system signals:

a. Overtemperature ΔT
b. Pressurizer low pressure

An inadvertent opening of a pressurizer safety or relief valve is classified as an ANS Condition II event, a fault of moderate frequency. See Section 15.0.1 for a discussion of Condition II events.

15.6.1.2 Analysis of Effects and Consequences

Method of Analysis
The inadvertent opening of a pressurizer safety valve transient is analyzed by employing the detailed digital computer program RETRAN-02 (Ref 3 in section 15.6.3.5). RETRAN-02 has been found acceptable by the NRC for use as a licensing basis safety analysis code. RETRAN-02 is a thermal-hydraulic systems analysis code employing a one-dimensional, homogeneous equilibrium mixture thermal-hydraulic model for the RCS, a point kinetics model for the reactor core, special component and auxiliary models (ex., pumps, temperature transport, non-equilibrium pressurizer) and control system models. The code computes pertinent plant transient information including core power level, RCS pressure and temperature. VIPRE-01 (Ref 5 in section 15.6.3.5) is used to evaluate the core thermal limits to determine DNBR. RETRAN-02 generated state points are used as VIPRE-01 boundary conditions to perform a Statistical Core Design (SCD) DNB analysis.

Plant characteristics and initial conditions are discussed in Section 15.0.3. In order to give conservative results in calculating the departure from nucleate boiling ratio (DNBR) during the transient, the following assumptions are made:

a. Initial conditions of nominal core power, reactor coolant temperatures, and reactor coolant pressure are assumed to perform an SCD DNB analysis (see Section 15.0.3).
b. A most positive moderator temperature coefficient of reactivity is assumed. The spatial effect of the void due to local or subcooled boiling, which tends to flatten the core power distribution, is not considered in the analysis with respect to reactivity feedback or core power shape.
c. A large (absolute value) Doppler coefficient of reactivity such that the resultant amount of positive feedback is conservatively high to retard any power decrease due to moderator reactivity feedback.

Major plant systems and equipment available for mitigation of transient and accident conditions which may be required to function to mitigate the effects of RCS depressurization caused by inadvertent safety valve opening are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-23.

Normal reactor control systems are not required to function. The rod control system is assumed to be in the automatic mode in order to hold the core at full power longer and thus delay the trip. This is a worst case assumption; if the reactor were in manual control, a trip could occur earlier on overtemperature DT or low pressurizer pressure. The reactor protection system functions to trip the reactor on the appropriate signal. No single active failure will prevent the reactor protection system from functioning properly.

Results

The system response to an inadvertent opening of a pressurizer safety valve is shown on Figures 15.6-1 and 15.6-2. Figure 15.6-1 illustrates the nuclear power transient following the depressurization. Nuclear power is maintained at the initial value until reactor trip occurs on overtemperature DT. The pressure decay transient and average temperature transient following the accident are given in Figure 15.6-1. Pressure drops more rapidly when core heat generation is reduced via the trip, and then slows once saturation temperature is reached in the hot leg. The DNBR decreases initially, but increases rapidly following the trip, as shown in Figure 15.6-2. The DNBR remains above the safety analysis limit throughout the transient. The DNBR design basis is described in Section 4.4.

The calculated sequence of events for the inadvertent opening of a pressurizer safety or relief valve incident is shown on Table 15.6-1.

15.6.1.3 Conclusions

The results of the analysis show that the pressurizer low pressure and the overtemperature DT reactor protection system signals provide adequate protection against the RCS depressurization event.

15.6.2 BREAK IN INSTRUMENT LINE OR OTHER LINES FROM REACTOR COOLANT PRESSURE BOUNDARY THAT PENETRATE CONTAINMENT

There are no instrument lines connected to the RCS that penetrate the containment. There are, however, the grab sample lines from the hot legs of reactor coolant loops 1 and 3, from the steam and liquid space of the pressurizer, and from the 3-inch chemical and volume control system letdown line penetrating the containment. The grab sample lines are provided with normal closed isolation valves on both sides of the containment wall and are designed in accordance with the requirements of GDC-55.

The most severe pipe rupture with regard to radioactivity release during normal plant operation is rupture of the chemical and volume control system letdown line at a point outside of the containment. For such a break, the reactor coolant letdown flow would have passed sequentially from the cold leg and through the regenerative heat exchanger and letdown orifices. The letdown orifice reduces the letdown line pressure from 2,235 psig to less than 600 psig outside containment during normal plant operation when letdown flow is maintained at 120 gpm. Increase in flow will occur due to a rupture of the letdown line downstream of the orifices. It has been determined that the occurrence of a complete severance of the letdown line would result in a loss of reactor coolant at the rate of 141 gpm.

Since the reactor makeup water transfer pumps are designed to deliver 120 gpm to the boric acid blending tee, the capability of the reactor makeup system can not maintain VCT level. The imbalance of the VCT outflow and inflow would eventually result in water level dropping to VCT level Lo/Refueling Water Sequence setpoint (5%) and the suction of the charging pump would automatically be shifted from the VCT to the RWST. In addition, the calculated releases rate is beyond the capacity of a single charging pump, which is capable of delivering 150 gpm flow to the RCS under normal operating conditions, if the RCP seal leakoff and any identified leakage are accounted for. The control room operators would be alerted of the failure by a high charging flow alarm and/or continuous VCT makeup and a slowly decreasing VCT and pressurizer level. The high charging flow alarm procedure in conjunction with the plant off-normal procedure for high RCS leakage would require letdown isolation inside the containment which would terminate the coolant loss.

15.6.2.1 Radiological Consequences

15.6.2.1.1 Method of Analysis

15.6.2.1.1.1 Physical Model

The volatile fractions of the spilled reactor coolant are assumed to be available for immediate release to the environment.

15.6.2.1.1.2 Assumptions and Conditions

The major assumptions and parameters used in the analysis are provided in Table 15.6-2 and summarized below:

a. The reactor coolant iodine activity is based on the dose equivalent of 1.0 μCi/gm of I-131.

b. The noble gas activity in the reactor coolant is based on 1-percent failed fuel.

c. A total of 70900 pounds of reactor coolant is spilled onto the floor of the auxiliary building. (based on doubling the maximum flowrate of 141 gpm to account for backflow over a thirty minute release, followed by a ten second valve closure period)

d. All of the noble gases in the spilled reactor coolant are released to the environment.

e. Ten percent of the spill is assumed to flash. All of the iodine activity in the flashed fraction of the spill is assumed to be released.

f. No credit is taken for mixing and holdup of the releases within the auxiliary building, nor are the auxiliary building normal exhaust filters credited with reducing the release. That is, the release is modeled as being direct to the environment.

15.6.2.1.1.3 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 the onsite meteorological measurement programs described in Section 2.3 and provided in Table 15A-2.

c. The thyroid inhalation and total body immersion doses to a receptor at the exclusion area boundary or outer boundary of the low-population zone were analyzed, using the models described in Appendix 15A.

15.6.2.1.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

The reactor coolant spilled in the auxiliary building will collect in the floor drain sumps. From there, it will be pumped to the radwaste treatment system. Therefore, the only release paths that present a radiological hazard involve the volatile fraction of spilled coolant.

Normally, gases released in the auxiliary building mix with the building atmosphere and are gradually exhausted through the filtered building ventilation system. The charcoal filters normally remove a very large fraction of the airborne iodine in the building atmosphere. However, the ventilation system is not designed to mitigate the consequences of an accident (e.g., it might not survive an earthquake more severe than the operating-basis earthquake), nor can the possibility of unplanned leakages from the auxiliary building be eliminated; hence, no credit is taken for these effects reducing the released activity.

The evaporated radionuclides are assumed to be available immediately to the outside atmosphere. This activity is tabulated in Table 15.6-2.

15.6.2.1.2 Identification of Uncertainties and Conservatisms in the Analysis

The principal uncertainties in the calculation of doses following a letdown line rupture arise from the unknown extent of reactor coolant contamination by radionuclides, the quantity of coolant spilled, the fraction of the spilled activity that escapes the auxiliary building, and the environmental conditions at the time. Each of these uncertainties is treated by taking worst-case or extremely conservative assumptions.

The extent of coolant contamination assumed greatly exceeds the levels expected in practice. The rupture is postulated in a seismic Category I, ASME Section III, Class 2 piping system. It is assumed that the leak goes undetected for 30 minutes. It is expected that considerable holdup and filtration occurs in the auxiliary building, but no credit is assumed.

The purpose of all these conservatisms is to place an upper bound on doses.

15.6.2.1.3 Conclusions

15.6.2.1.3.1 Filter Loadings

No filter is credited with the collection of radionuclides in this accident analysis. The buildup on these filters (auxiliary building and control building charcoal filters) that may be expected due to the adsorption of some of the iodine is very small compared with the design capacity of these filters.

15.6.2.1.3.2 Dose to Receptor at the Exclusion Area Boundary and Low-Population Zone Outer Boundary

The radiological consequences resulting from the occurrence of a postulated letdown line rupture have been conservatively analyzed, using assumptions and models described in previous sections.

The thyroid inhalation total-body immersion doses have been analyzed for the 0-2 hour dose at the exclusion area boundary and for the duration of the accident at the low-population zone outer boundary. The results are listed in Table 15.6-3. The resultant doses are within a small fraction of the guideline values of 10 CFR 100.

15.6.3 STEAM GENERATOR TUBE RUPTURE (SGTR)

Two SGTR scenarios have been identified which result in the most limiting radionuclide releases to the environment. Detailed analyses are presented for the following two scenarios:

a. SGTR with postulated failure of the faulted steam generator Auxiliary Feedwater (AFW) flow control valve.

b. SGTR with postulated stuck-open Atmospheric Relief Valve (ARV) for the faulted steam generator.

In the failed-controller case, auxiliary feedwater flow is maximized in order to increase probability for faulted steam generator overfill and subsequent water relief from its safety valve. The radioactive releases are maximized by assuming that the safety valve is stuck-open following water relief with an effective flow area equal to 5% of the total safety valve flow area (Reference 2).

In the stuck-open ARV scenario, the discharge of contaminated secondary fluid is maximized by assuming the faulted steam generator ARV stuck-open for 20 minutes (Reference 1).

The above cases are considered to be ANS Condition IV events, a limiting fault (USAR Section 15.0.1). It has been determined that the most severe radiological consequences will result from the forced steam generator overfill scenario with a stuck-open safety valve (Reference 2). Therefore, the radiological consequences are reported for that limiting case only.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-24.

15.6.3.1 STEAM GENERATOR TUBE RUPTURE WITH FAILURE OF FAULTED STEAM GENERATOR AFW CONTROL VALVE

15.6.3.1.1 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 a limited number of defective fuel rods. The accident leads to an increase in the contamination of the secondary system due to the leakage of radioactive coolant from the RCS. Loss of off-site power is assumed to occur coincident with reactor trip. Discharge of activity to the atmosphere takes place via the steam generator safety and/or atmospheric relief valves.

In view of the fact that the steam generator tube material is Inconel-600 and is a highly ductile material, the assumption of a complete severance is conservative. The more probable mode of tube failure would be one or more minor leaks of undetermined origin. Activity in the steam and power conversion system is subject to continual surveillance, and an accumulation of minor leaks which exceed the limits established in the Technical Specifications is not permitted during the unit operation.

Following the occurrence of the SG tube rupture, the primary to secondary leakage causes the pressurizer level and the RCS pressure to decrease. As the RCS pressure continues to decrease, automatic reactor trip occurs on low pressurizer pressure or over-temperature delta-T(OTΔT) signal. Because of the assumed loss of offsite power, the steam dump system will not be available, and the secondary side pressure increases rapidly after reactor trip until the steam generator ARVs and/or SV lift to dissipate the energy. After reactor trip, the RCS pressure continues to decrease and the safety injection is automatically initiated on low pressurizer pressure signal. Due to the assumed loss of offsite power at the reactor trip, normal feedwater flow is terminated and the AFW is initiated.

The analysis assumes failure of the AFW control valve on the discharge side of the motor-driven AFW pump feeding the ruptured steam generator. It is assumed that this valve fails in the wide-open position to maximize the flow to the ruptured steam generator. Failure of this valve coupled with the contribution from the turbine-driven AFW pump has a greater potential for overfilling the ruptured steam generator.

The operator is expected to determine that a SGTR has occurred and to identify and isolate the ruptured steam generator on a restricted time scale to minimize contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the ruptured steam generator. The recovery procedure then can be carried out on a time scale to ensure that break flow to the secondary system is terminated.

Consideration of the indications provided at the control board, together with the magnitude of the break flow, leads to the conclusion that the accident diagnostics and isolation procedure can be completed so that pressure equalization between primary and secondary side of the ruptured steam generator can be achieved to stop the break flow.

Assuming normal operation of the various plant control systems, the following sequence of events is initiated by a design basis tube rupture:

a. Pressurizer low pressure and low level alarms are activated 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 ruptured steam generator is reduced due to the additional break flow being supplied to that loop.

b. Decrease in RCS pressure (Figure 15.6-3a) due to continued loss of reactor coolant inventory leads to a reactor trip signal generated by low pressurizer pressure or over-temperature Delta T. It is conservatively postulated that loss of offsite power (LOOP) occurs when the reactor trips.

The LOOP signal initiates auxiliary feedwater addition. Resultant plant cooldown (Figures 15.6-3b and 15.6-3c) following reactor trip leads to a rapid change of pressurizer level (Figure 15.6-3f), and the safety injection signal, initiated by low pressurizer pressure, follows soon after the reactor trip. With the loss of offsite power occurring at the reactor trip, normal feedwater is terminated and the auxiliary feedwater is being supplied to the steam generators.

c. The steam generator blowdown liquid monitor alarm and/or the condenser air discharge radiation monitor alarm will activate, indicating a sharp increase in radioactivity in the secondary system. The alarms automatically cause termination of steam generator blowdown.

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 a coincident loss of offsite power, as assumed in the transient presented in this section, the steam dump valves would automatically close to protect the condenser. The steam generator pressure (Figure 15.6-3d) would rapidly increase, resulting in steam discharge to the atmosphere through the steam generator safety/atmospheric relief valves. In Figure 15.6-3g, the steam flow is constant initially until reactor trip, followed by turbine trip, which results in a large decrease in flow, but a rapid increase in steam pressure to the safety/relief valve setpoints.

e. Following reactor trip, the continued action of auxiliary feedwater supply and borated safety injection flow (supplied from the refueling water storage tank) provide a heat sink which absorbs the decay heat.

f. Safety injection flow results in increasing the pressurizer water level (Figure 15.6-3f); the rate of which depends upon the amount of operating auxiliary equipment.

g. Following water relief through the ruptured steam generator safety valve, the ruptured steam generator pressure is uncontrollably decreasing (Figure 15.6-3d) and the operator is directed to enter the Emergency Operating Procedure EMG C-31.

15.6.3.1.2 Analysis of Effects and Consequences

Method of Analysis

Mass and energy balance calculations are performed using RETRAN (Reference 3) to determine primary-to-secondary mass release and the amount of steam vented from each of the steam generators from the occurrence of the tube rupture to the second RCS depressurization following the switchover from the Emergency Operating Procedure EMG E-3 to EMG C-31.

In estimating the mass transfer from the RCS through the broken tube, the following assumptions are made:

a. Reactor trip occurs automatically as a result of low pressurizer or overtemperature Delta T. Loss of offsite power occurs at reactor trip.

b. Auxiliary feedwater flow rate is allowed to vary with the fluctuation in the faulted steam generator pressure. This results in higher AFW flow when the faulted steam generator pressure decreases. Six minutes following the initiation of the safety injection signal, the AFW from the turbine driven AFW pump to the ruptured steam generator is terminated by closing the un-failed AL-HV valve. Eighteen minutes following the initiation of the safety injection signal, the AFW from the motor-driven AFW pump to the ruptured steam generator is terminated by locally closing the failed AL-HV valve. AFW flow to the intact steam generators maintains the narrow range level between 4% and 50% as indicated in the Emergency Operating Procedure EMG E-3.

c. Cooldown of the RCS is initiated at 30 minutes following the initiation of the safety injection signal. It is assumed that steam is released through the remaining three operable steam generator ARVs in the intact loops until the RCS temperature of the core exit thermal couples corresponds to the ruptured SG pressure as listed in the EMG E-3 procedure. Note that the analysis assumes that the operators will continue to maintain the plant at that temperature for the duration of the transient, consistent with the EMG E-3 procedure. Technical Specification LCO 3.7.1.6 requires that all four steam generator ARVs shall be operable. With one of the required ARVs unavailable due to its association with the ruptured steam generator, the remaining three ARVs are available to ensure that subcooling can be achieved for the RCS.

d. Following termination of the RCS cooldown, the RCS is depressurized by opening a pressurizer PORV to assure an adequate coolant inventory prior to terminating SI flow. Primary depressurization is initiated at 5 minutes following the termination of the RCS cooldown and continues until the RCS pressure is less than the ruptured steam generator pressure.

e. Following the depressurization, termination of SI is delayed to ensure enough liquid enters the ruptured steam generator steamline to force the safety valve open and cause water relief. It is assumed that 5 minutes following the termination of the RCS depressurization that the safety injection flow is reduced to just one centrifugal charging pump. At 15 minutes following the termination of the RCS depressurization, the one CCP is throttled back to 100 gpm and at 30 minutes following termination, letdown is initiated such that the net flow due to SI and letdown is zero.

f. As the pressure in the ruptured steam generator uncontrollably decreases following water relief, the operators are directed to switch to EMG C-31 procedure. The first major step in EMG C-31 is to initiate a second RCS cooldown. It is assumed that the operators will be able to perform the second cooldown 10 minutes after water relief. This cooldown process is continued to the end of the analysis using the two intact steam generator atmospheric relief valves.

The above assumptions are conservatively made to increase the probability for faulted steam generator overfill and to maximize the radioactive releases to the atmosphere.

Key Recovery Sequence

The recovery sequence to be followed consists of the following major operator actions:

a. Identification of the faulted steam generator

b. Isolation of the faulted steam generator

c. Assuring subcooling of the RCS fluid to approximately 50°F below the faulted steam generator temperature

d. Depressurization of the RCS to a value equal to the faulted steam generator pressure

e. Subsequent termination of safety injection flow

f. Further cooldown and depressurization of the RCS to conditions suitable for RHR initiation

Results

In Table 15.6-1, the sequence of event are presented. These events include postulated operator response times and normal plant response to the normal plant setpoints. Primary and secondary system parameters are plotted as a function of time in Figures 15.6-3a through 15.6-3j.

As depicted in Figure 15.6-3j, the steam generator overfilling occurs at approximately 1800 seconds. Water relief through the ruptured steam generator safety valve occurs at approximately 2830 seconds.

15.6.3.2 STEAM GENERATOR TUBE RUPTURE WITH POSTULATED STUCK-OPEN ATMOSPHERIC RELIEF VALVE

15.6.3.2.1 Identification of Causes and Accident Description

The accident description for this SGTR is identical to that discussed in Section 15.6.3.1.1 with the exception that the ARV for the faulted steam generator is assumed to remain open for 20 minutes following initial secondary pressure relief, shortly after reactor trip.

In this SGTR scenario, the operator is expected to determine that a SGTR has occurred, to recognize the failure of the ARV, to dispatch personnel to manually isolate the ARV, and then to isolate the affected steam generator. In this accident the failure of the ARV allows a longer period for radionuclide release than otherwise expected.

The accident involves the complete severance of a single steam generator tube with Loss of Offsite Power coincident with reactor trip. For a discussion of normal operation of plant control systems to a design basis SGTR refer to Section 15.6.3.1.1.

15.6.3.2.2 Analysis of Effects and Consequences

Methods of Analysis

Mass and energy balance calculations are performed using RETRAN (Reference 3) to determine primary-to-secondary mass release and to determine the amount of steam vented from each of the steam generators from the occurrence of the tube rupture until termination of SI. Supplementary mass and energy calculations are performed for the period from termination of SI until initiation of RHR cooling.

In estimating the mass transfer from the RCS through the broken tube, the following assumptions are made:

a. Reactor trip occurs automatically as a result of low pressurizer pressure or overtemperature Delta T. Loss of offsite power occurs at reactor trip.

b. As pressures rise on the secondary side, the steam generator atmospheric relief valves (ARVs) open to release excess secondary pressure. Although the ARVs in the unaffected steam generator close within 7 minutes, the ARV for the faulted steam generator is assumed to remain open and steam release to continue for 20 minutes until the ARV block valve is manually closed.

c. AFW is initially delivered at a rate of 250 gpm to each steam generator. AFW is maintained to assure that narrow range level in each steam generator exceeds 15%.

d. Following ruptured steam generator isolation, cooldown is initiated when the narrow range in the ruptured steam generator is greater than 10% and its pressure exceeds 630 psia. Cooldown continues until RCS temperature is reduced to 50°F less than the ruptured steam generator saturation temperature.

e. Reactor coolant system depressurization is initiated three minutes after completion of cooldown. This timing is consistent with observed simulator exercises.

f. After primary side depressurization is completed and SI termination criteria are met, a three minute time delay is assumed prior to SI termination.

g. Following SI termination, the operators equalize pressure in the RCS and faulted SG in 5 minutes. During this time break flow in the faulted SG continues. After pressures are equalized, it is conservatively assumed that the transition to cold shutdown is made utilizing steam release to the atmosphere from the faulted SG.

<u>Key Recovery Sequence</u>

The recovery sequence to be followed consists of the following major operator actions:

a. Identification of the faulted steam generator

b. Identification and manual closure of the stuck-open atmospheric relief valve

c. Isolation of the faulted steam generator

d. Assuring subcooling of the RCS fluid to approximately 50°F below the faulted steam generator temperature

e. Depressurization of the RCS to a value equal to the faulted steam generator pressure

f. Subsequent termination of safety injection flow

g. Further cooldown and depressurization of the RCS to conditions suitable for RHR initiation.

Results

In Table 15.6-1, the sequence of events are presented. These events include postulated operator response times and normal plant responses to the normal plant setpoints. Primary and secondary system parameters are plotted as a function of time in Figures 15.6-3k through 15.6-3t.

15.6.3.3 Radiological Consequences

15.6.3 3.1 Method of Analysis

15.6.3.3.1.1 Physical Model

The evaluation of the radiological consequences of a postulated steam generator tube rupture (SGTR) utilizes the results of the RETRAN analyses to calculate releases of radioactive iodines and noble gases to the atmosphere to the time of RHR cut-in conditions.

Concentrations of radioactivity in the RCS water and in the faulted and intact steam generators are calculated utilizing release rates from the fuel, calculated mass flows and conventionally used partitioning coefficients between the liquid and steam phases. These radioactivity concentrations and the calculated releases of mass to the atmosphere yield the released activity. Dose rates are calculated using atmospheric dispersion coefficients, breathing rates, and other aspects of conventional dose rate calculations.

15.6.3.3.1.2 Assumptions and Conditions

The major assumptions and parameters assumed in the analysis are itemized in Table 15.6-4 and 15A-1 and are summarized below.

a. The assumed reactor coolant iodine activity is determined for the following two cases:

Case 1 - An initial reactor coolant iodine activity equal to the dose equivalent of 60.0 μCi/gm of I-131 due to a pre-accident iodine spike caused by RCS transients prior to the SGTR.

Case 2 - An initial reactor coolant iodine activity equal to the dose equivalent of 1.0 μCi/gm of I-131 with an iodine spike that increases the escape rate from the fuel into the coolant by a factor of 500 immediately after the accident. This increased escape rate is assumed for the duration of the accident.

b. The noble gas activity in the reactor coolant is based on 1-percent failed fuel as provided in Table 11.1-5.

c. The initial secondary coolant activity is based on the dose equivalent of 0.1 mCi/gm of I-131.

The following assumptions and parameters are used to calculate the activity released and the offsite dose following an SGTR:

a. The amount of discharge or reactor coolant in the secondary system as a function of time is as calculated by RETRAN analysis. The analysis yields 138,417 pounds of reactor coolant transferred to the secondary side of the faulted steam generator.

b. It is assumed that all of the iodine in the fraction of reactor coolant that flashes to steam upon reaching the secondary side is released to the steam phase. No credit is taken for scrubbing.

c. A 3-gpm primary-to-secondary leak is assumed to occur to the unaffected steam generators, through the accident sequence.

d. All noble gas activity in the reactor coolant that is transported to the secondary system via the tube rupture and the primary-to-secondary leakage is released to the atmosphere.

e. The iodine partition fraction between the liquid and steam in the steam generator is assumed to be 0.01.

f. The steam releases from the steam generators to the atmosphere are as calculated by RETRAN analysis and given in Table 15.6-4. The total faulted feedwater flows to all steam generators are also listed in Table 15.6-4.

g. Radioactivity releases to the atmosphere are based on the concentrations of radioactivity in the steam phase times the calculated amounts of steam release.

h. No additional radioactivity releases occur after the initiation of RHR system cooling.

i. Radioactive decay prior to the release of activity is considered. No decay during transit or ground deposition is considered.

j. Short-term accident atmospheric dispersion factors, breathing rates, and dose conversion factors are provided in Tables 15A-2, 15A-1, and 15A-4, respectively.

The total activity released is provided in Tables 15.6-4.

15.6.3.3.1.3 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 Reference 1 and 2.
b. The atmospheric dispersion factors used in the analysis were calculated based on the onsite meteorological measurements program, as described in USAR Section 2.3, and are provided in Table 15A-2.
c. The thyroid inhalation 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.

15.6.3.3.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

The activity released from the faulted steam generator, is released directly to the environment by the atmospheric relief valves. The intact steam generators discharge steam and entrained activity via the safety and atmospheric relief valves until the time that initiation of the RHR system can be accomplished. In addition, the steam release via the exhaust stack of the TDAFW pumps is also considered in the SGTR dose consequences shown in Table 15.6-4. Since the activity is released directly to the environment with no credit for plateout or fall out, the results of the analysis are based on the most direct leakage pathway available. Therefore, the resulting radiological consequences represent a conservative estimate of the potential integrated dose to the postulated SGTR.

15.6.3.3.2 Identification of Uncertainties and Conservatisms in the Analysis

a. Reactor coolant activities based on extreme iodine spiking effects are conservatively high.

b. The assumed 3-gpm steam generator primary-to-secondary leakage is greater than that anticipated during normal operation.

c. Tube rupture of the steam generator is assumed to be a double-ended severance of a single steam generator tube. This is a conservative assumption, since the steam generator tubes are constructed of highly ductile materials. The more probable mode of tube failure is one or more minor leaks of undetermined origin. Activity in the secondary steam system is subject to continual surveillance, and the accumulation of activity from minor leaks that exceeds the limits established in the technical specifications would lead to reactor shutdown. Therefore, it is unlikely that the total amount of activity considered available for release in this analysis would ever be realized.

d. The coincident loss of offsite power with the occurrence of an SGTR is a conservative assumption. In the event of the availability of offsite power, the condenser dump valves will open, permitting steam dump to the condenser. This will reduce the amount of steam and entrained activity discharged directly to the environment from the unaffected steam generators.

e. The radiological consequences have been based on a worst-case scenario, i.e., forced steam generator overfill with a stuck-open safety valve.

f. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is unlikely that meteorological conditions assumed will 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 steam generator tube rupture is the control room filtration system. Activity loadings on the control room charcoal filter are based on flow rate through the filter, concentration of activity at the filter inlet, and filter efficiency.

Activity in the control room filter as a function of time has been evaluated for the LOCA, Section 15.6.5. Since the control room filters are capable of accommodating the potential design-basis LOCA fission product iodine loadings, more than adequate design margin is available with respect to postulated SGTR accident releases.

15.6.3.3.3.2 Doses to Receptor at the Exclusion Area Boundary

The total-body dose due to immersion and the thyroid dose due to the inhalation have been analyzed for the 0-2 hour period at the exclusion area boundary and the duration of the accident at the low-population zone outer boundary. The results are listed in Table 15.6-5. As can be seen from this table, the calculated radiological consequence of a postulated steam generator tube rupture accident does not exceed: (1) a small fraction (≤ 10 percent) of the exposure limits set forth in 10 CFR Part 100, for the case of an iodine spike that results from the accident, and (2) the exposure limits set forth in 10 CFR part 100, for the case of the pre-accident spike.

15.6.3.4 <u>Conclusions</u>

A steam generator tube rupture will cause no subsequent damage to the RCS or the reactor core. An orderly recovery from the accident can be completed, even assuming simultaneously loss of offsite power.

15.6.3.5 REFERENCES

1. Letter SLNRC 86-1, Petrick, N. A., SNUPPS to Denton, H. R. (NRC), "Steam Generator Single-Tube Rupture Analysis for SNUPPS Plants - Callaway and Wolf Creek", dated January 8,1986.

2. Letter WM 87-0145, Withers, B. D. WCNOC to US NRC, "Response to RAI Regarding the SGTR Overfill Case Analysis", dated May 15, 1987.

3. McFadden, J. H., et al, "RETRAN-02 - A Program for Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems", EPRI-NP-1850-CCM-A, October 1984.

4. Letter from NRC to Withers, B. D., "Safety Evaluation Report for the Wolf Creek Generating Station Steam Generator Tube Rupture Analysis", TAC No. 57363, dated May 7, 1991.

5. Stewart, C. W., et. al., "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores," Battelle, Pacific Northwest Laboratories, EPRI NP-2511-CCM-A, August 1989.

15.6.4 SPECTRUM OF BWR STEAM SYSTEM PIPING FAILURES OUTSIDE OF CONTAINMENT

This section is not applicable to WCGS.

15.6.5 LOSS-OF-COOLANT ACCIDENTS RESULTING FROM A SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

15.6.5.1 Identification of Causes and Frequency Classification

A loss-of-coolant accident (LOCA) is the result of a pipe rupture of the reactor coolant system (RCS) pressure boundary. For the analysis reported here, a small break is defined as a rupture of the RCS piping with a cross-sectional area less than $1.0ft^2$, in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. A small break LOCA is classified as an ANS Condition III event (an infrequent fault), as defined in Section 15.0. A major break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than $1.0ft^2$. This event is considered an ANS Condition IV event, a limiting fault, in that it is not expected to occur during the life of a plant but is postulated as a conservative design basis.

The Acceptance Criteria for the LOCA are described in 10 CFR 50.46 as follows:

A. The calculated peak fuel element clad temperature shall not exceed the requirement of 2200°F.

B. The amount of fuel element cladding that reacts chemically with water or steam to generator hydrogen, shall not exceed 1 percent of the total amount of Zircaloy in the fuel cladding.

C. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

D. The core remains amenable to cooling during and after the break.

E. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

These criteria were established to provide a significant margin in emergency core cooling system (ECCS) performance following a LOCA. WASH-1400 (USNRC 1975) Reference 2) presents a study in regards to the probability of occurrence of RCS pipe ruptures.

In all cases, small breaks (Less than 1.0 $ft.^2$) yield results with more margin to the Acceptance Criteria Limits than large breaks.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-25.

15.6.5.2 Sequence of Events and Systems Operations

Should a major pipe break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low pressure trip setpoint is reached. A safety injection signal is generated when the appropriate setpoint (high containment pressure or low pressurizer pressure) is reached. These countermeasures limit the consequences of the accident in two ways:

A. Reactor trip and borated water injection supplement void formation in causing rapid reduction of power to the residual level corresponding to fission product decay heat. No credit is taken in the LOCA analysis for the boron content of the injection water. However, an average RCS/sump mixed boron concentration is calculated and verified on a cycle-specific basis to ensure that the post-LOCA core remains subcritical. In addition, the insertion of control rods to shut down the reactor is neglected in the large break analysis, but is credited in the post-LOCA subcriticality evaluation. Details on the method for assuring post-LOCA subcriticality is discussed in Section 15.6.5.3.4.

B. Injection of borated water provides for heat transfer from the core and prevents excessive clad temperatures.

Description of the Large Break LOCA Transient

The sequence of events for the large break LOCA transient is depicted in Figure 15.6-4.

Before the break occurs, the unit is in an equilibrium condition; i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from fission product decay, hot internals, and the vessel continues to transfer to the reactor coolant. At the beginning of the blowdown phase, the entire RCS contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break develops, the time to departure from nucleate boiling is calculated, consistent with Appendix K of 10 CFR 50. Thereafter, the core heat transfer is unstable, with both nucleate boiling and film boiling occurring. As the core becomes uncovered, both transition boiling and forced convection are considered as the dominant core heat transfer mechanisms. Heat transfer due to radiation is also considered.

Turbine trip on a reactor trip signal is assumed in the analysis but is not explicitly modeled. The steam generators are isolated at reactor trip and become a heat source early in the transient due to the rapid energy loss of the primary. Further, the delays associated with the AFW system prevent the secondary from having a significant effect on the transient. As such, no credit for the secondary system engineered safety features (i.e., main steam safety valves, AFW, etc.) is taken for the large break LOCA.

When the RCS depressurizes to 600 psia, the accumulators begin to inject borated water into the reactor coolant loops. The conservative assumption is made that all of the accumulator water injected during the bypass period is subtracted from the RCS after the bypass period terminates (called end-of-bypass). End-of-bypass (EOB) occurs when the expulsion or entrainment mechanisms responsible for the bypassing are calculated not to be effective. This conservatism is again consistent with Appendix K of 10 CFR 50. Since loss of offsite power (LOOP) is assumed, the reactor coolant pumps are assumed to trip at the inception of the accident. The effects of pump coastdown are included in the blowdown analysis.

The blowdown phase of the transient ends when the RCS pressure (initially assumed at 2300 psia) falls to a value approaching that of the containment atmosphere. Prior to, or at the end of, the blowdown, termination of bypass occurs and refill of the reactor vessel lower plenum begins. Refill is completed when emergency core cooling water has filled the lower plenum of the reactor vessel to the bottom of the active fuel region (BOC time).

The reflood phase of the transient is defined as the time period lasting from BOC recovery until the reactor vessel has been filled with water to the extent that the core temperature rise has been terminated. From the latter stage of blowdown and then the beginning of reflood, the accumulator tanks rapidly discharge borated cooling water into the RCS, thus contributing to the filling of the reactor vessel downcomer. The downcomer head provides the driving force required for the reflooding of the reactor core. The RHR (low head), safety injection, and high head centrifugal charging pumps aid in the filling of the downcomer and subsequently supply water to maintain a full downcomer and complete the reflooding process.

Continued operation of the ECCS pumps supplies water during long-term cooling. Core temperatures have been reduced to long-term steady state levels associated with dissipation of residual heat generation. After the water level of the refueling water storage tank (RWST) reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching to the cold leg recirculation phase of operation. Spilled borated water is drawn from the engineered safety features (ESF) containment sump by the RHR (low head) pumps and returned to the RCS cold legs. The containment spray pumps are manually aligned to the containment sump and continue to operate to further reduce containment pressure and temperature. Approximately 10.0 hours after initiation of the LOCA, the ECCS is realigned to supply water to the RCS hot legs in order to control the boric acid concentration in the reactor vessel.

Description of Small-Break LOCA Transient

As contrasted with the large break, the blowdown phase of the small break occurs over a longer time period. Thus, for a small-break LOCA there are only three characteristic stages, i.e., a gradual blowdown in which the decrease in water level is checked, core recovery, and long-term recirculation.

15.6.5.3 Core and System Performance

15.6.5.3.1 Mathematical Model

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10 CFR 50.

Large Break LOCA Evaluation Model

Because of the distinct phenomena associated with the accident, the analysis of a large break LOCA transient is divided into three phases: (1) blowdown, (2) refill, and (3) reflood. Further, there are three distinct transients analyzed in each phase, including the thermal-hydraulic transient in the RCS, the pressure and temperature transient within the containment, and the fuel and clad temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of interrelated computer codes has been developed for the analysis of the LOCA.

A description of the various aspects of the LOCA analysis methodology is given by Bordelon, Massie, and Zordan (1974) (Reference 3). This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the Acceptance Criteria. The SATAN-VI, WREFLOOD, BASH and LOCBART codes, which are used in the LOCA analysis, are described in detail by Bordelon et al. (1974) (Reference 4); Kelly et al. (1974) (Reference 5); Young et al. (1987) (Reference 6); and Bordelon et al. (1974) (Reference 3). Code modification are specified in References 8, 9, 10, and 15. These codes assess the core heat transfer geometry and determine if the core remains amenable to cooling through, and subsequent to, the blowdown, refill, and reflood phases of the LOCA. The code interfaces are described in Figure 15.6-5.

SATAN-VI calculates the thermal-hydraulic transient, including the RCS pressure, enthalpy, density, and the mass and energy flow rates in the RCS, as well as steam generator energy transfer between the primary and secondary systems as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator water mass and internal pressure and also the break mass and energy flow rates that are assumed to be vented to the containment during blowdown. At the end of the blowdown, the mass and energy release rates during blowdown are transferred to the COCO code, detailed in Reference 16, for use in determination of the containment pressure response during the first phase of the LOCA. Additional SATAN-VI output data from the end-of-blowdown, including the core inlet flow rate and enthalpy, the core pressure, and the core power decay transient, are input to the LOCBART code.

At the end of the blowdown, information from SATAN-VI on the state of the system is transferred to the WREFLOOD code which calculates the time to bottom of core (BOC) recovery, RCS conditions at BOC and mass and energy release from the break during the reflood phase of the LOCA. Since the mass flow rate to the containment depends upon the core flooding rate and the local core pressure, which is a function of the containment back pressure, the WREFLOOD and COCO codes are interactively linked. The BOC conditions calculated by WREFLOOD and the containment pressure transient calculated by COCO are used as input to the BASH code. Data from both the SATAN-VI and WREFLOOD codes out to BOC are input to the LOCBART code which calculates core average conditions at BOC for use by the BASH code.

The BASH code provides a realistic thermal-hydraulic simulation of the reactor core and RCS during the reflood phase of a large break LOCA. Instantaneous values of the accumulator conditions and safety injection flow at the time of completion of lower plenum refill are provided to BASH by WREFLOOD. Figure 15.6-5 illustrates how BASH has been substituted for WREFLOOD in calculating transient values of core inlet flow, enthalpy , and pressure for the detailed fuel rod model, LOCBART. A detailed description of the BASH code is available in Reference 6. The BASH code provides a sophisticated treatment of steam/water flow phenomena in the reactor coolant system during core reflood. The

BART code has been coupled with a loop model to form the BASH code. The loop model determines the loop flows and pressure drops in response to the calculated core exit flow determined by BART. The updated core inlet flow calculated by the loop model is used by BART to calculate a new entrainment rate to be fed into the loop code. This process of transferring data between BART, the loop code, and back to BART forms the calculation process for analyzing the reflood transient. This coupling of the BART code with a loop code produces a dynamic flooding transient, which reflects the close coupling between core thermal-hydraulics and loop behavior.

The cladding heat-up transient is calculated by LOCBART which is a combination of the LOCTA code with BART, a more detailed description of the LOCBART code can be found in References 6 and 9. During reflood, the LOCBART code provides a significant improvement in the prediction of fuel rod behavior. In LOCBART the empirical FLECHT correlation has been replaced by the BART code. BART employs rigorous mechanistic models to generate heat transfer coefficients appropriate to the actual flow and heat transfer regimes experienced by the fuel rods.

The NOTRUMP and LOCTA-IV computer codes are used in the analysis of LOCA due to small breaks in the RCS. The NOTRUMP computer code is a state-of-the-art, one-dimensional general network code consisting of a number of advanced features. Among these features are the calculation of thermal nonequilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flow limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. The NOTRUMP small break LOCA emergency core cooling system (ECCS) evaluation model was developed to determine the RCS response to design basis small break LOCAs and to address the NRC concerns expressed in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plant." In NOTRUMP, the RCS is nodalized into volumes interconnected by flowpaths. the broken loop is modeled explicitly with the intact loops lumped into a second loop. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum applied throughout the system. A detailed description of NOTRUMP is given in Reference 11 and 12.

The use of NOTRUMP in the analysis involves, among other things, the representation of the reactor core as heated control volumes with an associated bubble rise model to permit a transient mixture height calculation. The multinode capability of the program enables an explicit and detailed spatial representation of various system components. In particular, it enables a proper calculation of the behavior of the loop seal during a loss-of-coolant transient.

Cladding thermal analyses are performed with the LOCTA-IV (Reference 7) code which uses the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core, and mixture height history from the NOTRUMP hydraulic calculations as input.

Figure 15.6-47 presents the hot rod power shape utilized to perform the small-break analysis. This power shape was chosen because it provides a conservative distribution of power versus core height, and also local power is maximized in the upper regions of the reactor core. This power shape is skewed to the top of the core with the peak local power occurring at about the 10-foot core elevation.

This is limiting for the small-break analysis, because of the core uncovery process for small breaks. As the core uncovers, the cladding in the upper elevation of the core heats up and is sensitive to the local power at that elevation. The cladding temperatures in the lower elevation of the core, below the two-phase mixture height, remain low. The peak clad temperature occurs above 10 feet.

Schematic representation of the computer code interface is given in Figure 15.6-6.

The small-break analysis was performed with the approved Westinghouse ECCS Small Break Evaluation Model with NOTRUMP (References 7, 11, and 12).

15.6.5.3.2 Input Parameters and Initial Conditions

Table 15.6-9 lists important input parameters and initial conditions used in the analysis.

The analysis presented in this section was performed with a reactor vessel upper head temperature equal to the RCS cold leg temperature and an upflow barrel-baffle configuration.

The bases used to select the numerical values that are input parameters to the analysis have been conservatively determined from sensitivity studies (References 17, 18, and 19). In addition, the requirements of Appendix K regarding specific model features were met by selecting models which provide a significant overall conservatism in the analysis. The assumptions made pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA occurs and include such items as the core-peaking factors, the containment pressure, and the performance of the ECCS. Decay heat generator, and the performance of the ECCS. Decay heat generated throughout the transient is also conservatively calculated.

Small Break Specific Input Parameters and Initial Conditions

Additional analyses were performed to determine the direction of conservatism for two parameters, the reactor coolant system average temperature and the auxiliary feedwater flowrate. The limiting conditions which resulted are reported as input parameters in Table 15.6-9 (the reactor coolant system average temperature is the average of the vessel inlet/outlet temperatures) and were the conditions assumed in the ECCS analyses.

ECCS flow rate to the RCS as a function of the system pressure is used as part of the input. The ECCS was assumed to be delivering to the RCS 39-seconds after the generation of a safety injection signal. For these analyses, the ECCS delivery considers pumped injection flow which is depicted in Figure 15.6-46 as a function of RCS pressure. This figure represents injection flow from the centrifugal charging (CCP) and safety injection (SI) pumps. The 39-second delay includes time required for sensor response, diesel startup, and loading of the CCP and SI pumps onto the emergency buses. The effect of flow from the RHR pumps is not considered here since their shutoff head is lower than RCS pressure during the portion of the transient considered here. Also, minimum safeguards Emergency Core Cooling System capability and operability have been assumed in this analysis.

15.6.5.3.3 Results

Large Break LOCA Results

Based on the results of the LOCA sensitivity studies [Westinghouse 1974 (Reference 17); Salvatori 1974 (Reference 18); Johnson, Massie, and Thompson 1975 (Reference 19)], the limiting large break was found to be the double-ended cold leg guillotine (DECLG) break. Therefore, only the DECLG break was considered in the large break ECCS performance analysis.

Calculations were performed for a range of Moody break discharge coefficients (Cp) A limiting PCT of 1916.0°F was calculated for the break with C_D=0.4 case analyzed for V5H fuel with IFMs at reduced T_{AVG} and minimum safeguards assumptions. This is less than the acceptance criteria limit of 2200°F. The maximum local metal-water reaction at the end of the transient is 3.64%. At this point, the clad temperature excursion has been reversed and the metal-water reaction rate begins to decrease at a rate sufficient to conclude that the embrittlement limit of 17 percent, as required by 10 CFR 50.46, will not be exceeded. The total core metal-water reaction is less than 1.0 percent for all breaks analyzed, which is less than the 1.0 percent criterion of 10 CFR 50.46. The clad temperature transient is terminated at a time when the core geometry is still amenable to core cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided. PCTs of 1828.4°F and 1627.4°F were calculated for the breaks with C_D=0.6 and C_D=0.8, respectively.

A summary of results appears in Tables 15.6-10 and 15.6-11. Figures 15.6-7 through 15.6-30 present the results of the specific analysis for the following key parameters of interest:

- peak clad temperature
- core pressure
- core and downcomer collapsed liquid water levels during reflood
- heat transfer coefficient at the hot spot on the hot rod
- fluid temperature at the hot spot on the hot rod
- fluid quality

For the limiting break, the following additional transient parameters are presented:

- core inlet flow velocity during reflood
- core power transient
- core flow rate during blowdown
- break mass flow rate
- break energy release rate
- accumulator flow rate
- mass velocity at the hot spot

For the C_D=0.6 break, pumped ECCS flow for the minimum and maximum safeguards cases is provided. The maximum safeguards case is discussed below.

The containment pressure transient resulting from the limiting large break LOCA is presented in Section 6.2.1.5.

A maximum safeguards case is typically performed for the limiting break since the additional SI can result in higher calculated PCTs. This occurs if minimum safeguards is insufficient to condense all the steam at the top of the downcomer. In this instance, maximum safeguards results in additional condensation and a subsequent pressure drop at the top of the downcomer. The reduction in driving head will adversely affect the core flooding rate. For Wolf Creek, however, the large break LOCA is characterized by downcomer underfill, as described below, with the C_D=0.4 break being the most severely affected. As such, additional SI would merely result in filling the downcomer sooner which increases the driving head and results in lower calculated PCTs. Therefore, based on the degree of downcomer underfill and the difference in calculated PCTs, the C_D=0.6 case was analyzed for maximum safeguards. This case would more likely be adversely affected by maximum safeguards than the C_D=0.4 break because of the degree of downcomer underfill and because it is only 87.6°F less limiting. While the C_D=0.8 case exhibits the least downcomer underfill, it is less limiting than the C_D=0.4 break by 288.6°F. Figure 15.6-24a and 24b show the pumped SI flow for minimum and maximum safeguards assumptions (note: the blowdown portion of the minimum safeguards case is the same as the maximum safeguards case since no pumped safety injection occurs prior to EOB). As expected, the additional SI filled the downcomer sooner resulting in a better flooding rate and, subsequently, a lower resultant PCT of 1739.4°F.

Prior to performing the standard break spectrum calculations, analyses for a range of RCS operating temperatures corresponding to T_{AVG} at nominal (588.4°F) and reduced (570.7°F) temperatures were performed in order to determine the limiting operating conditions. Calculations for both nominal and reduced T_{AVG} were performed for the limiting Moody break discharge coefficient (C_D=0.4). Reduced T_{AVG} was determined to result in higher calculated PCTs for Wolf Creek. The increased RCS mass at the lower temperature resulted in a longer EOB. As required by Appendix K to 10 CFR 50, no credit is taken for injected water prior to EOB in refilling the vessel. as such, a longer EOB results in less accumulator volume remaining to refill the lower plenum and downcomer. Typically, sufficient accumulator volume is available at EOB to fill or nearly fill the downcomer . For Wolf Creek, however, the accumulators empty with only several feet of water in the downcomer and pumped SI is required to fill the downcomer. This is referred to as downcomer underfill. The delay in downcomer filling reduces the driving head available to reflood the core which results in significant adverse effects on the core flooding rate.

Small-Break Results

As noted previously, the calculated peak clad temperature resulting from a small-break LOCA is less than that calculated for a large break. Based on the results of the LOCA sensitivity studies (Reference 17) the limiting small break was found to be less than a 10-inch-diameter rupture of the RCS cold leg. Therefore, a range of small-break analyses is presented which establishes the limiting break size. The results of these analyses are summarized in Table 15.6-10 and 15.6-12. Figures 15.6-31 through 15.6-48 present the principal parameters of interest for the small-break ECCS analyses. For all cases analyzed, the following transient parameters are presented:

a. RCS pressure

b. Core mixture height

c. Hot spot clad temperature

For the limiting break analyzed, the following additional transient parameters are presented:

a. Core steam flow rate

b. Core heat transfer coefficient

c. Hot spot fluid temperature

The maximum calculated peak clad temperature for all small breaks analyzed is 1510°F. The results of the analyses given in Table 15.6-12 are well below all Acceptance Criteria limits of 10 CFR 50.46, and in all cases are not limiting when compared to the results presented for large breaks.

15.6.5.3.4 Post-LOCA Long-Term Core Cooling/Subcriticality

10CFR50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" paragraph (b) item (5) sets forth the requirements for post-LOCA long-term core cooling. To satisfy the requirements, the core is maintained in a shutdown state solely by the soluble boron contained in the ECCS water after a LOCA because credit for the shutdown provided by the control rods was not taken for cold leg breaks ≥ 3.0 ft^2, or hot leg breaks ≥ 1.0 ft^2. Since safety injection flow is drawn from the sump following switchover from the RWST, the containment sump post-LOCA boron concentration must be higher than the boron concentration required to ensure subcritical conditions.

To determine if the requirements for post-LOCA long-term core cooling subcriticality are met, a calculation is performed for each reload to determine the boron concentration required to keep the core subcritical (K_{eff} < 1.0) and the mixed mean boron concentration (MMBC) of the post-LOCA sump water. This calculation, documented in the cycle-specific Reload Safety Analysis Checklist (RSAC), confirms that the post-LOCA sump MMBC exceeds the core critical boron concentration, thereby ensuring the reload core remains subcritical. Note: The post-LOCA long-term core cooling critical boron concentration is determined at the most reactive time in life, assuming an all rods out (ARO) no Xenon condition and a post-LOCA fluid temperature range of 68-212°F. All sources of water that may eventually reside in the containment sump at cold leg recirculation switchover time and their respective pre-accident boron concentrations are considered.

Westinghouse has identified a potential safety issue concerning core recriticality following a large break cold leg break LOCA (Reference 22). The potential safety issue is that during hot leg switchover the core will be flushed with a diluted sump solution, which may cause the core to return to criticality. The sump solution would become diluted as boron accumulates in the core during the core leg recirculation phase due to core boiling. The accumulation of boron in the core prevents the boron from being displaced to the sump which leads to a diluted sump solution.

However, a generic assessment (Reference 22) concludes that for any given plant and fuel cycle, the boron worth of the inserted control rods plus the equivalent boron worth due to the presence of Xenon at the time of hot leg switchover, would offset any reasonable calculation of sump dilution. As documented in WCAP-15704 (Reference 23), it was demonstrated that control rods will insert following a licensing basis cold leg LOCA for 3-loop and 4-loop Westinghouse plant designs. Thus, the negative reactivity credit associated with control rods can be applied when evaluating recriticality at the time of switchover to hot leg ECCS recirculation.

Consequently, post-LOCA subcriticality will continue to be confirmed using sump boron calculations that do not consider sump dilution, but also use the conservative assumptions of ARO and no Xenon at the most reactive time in life, with an assumed post-LOCA core/fluid temperature in the range of 68-212°F.

15.6.5.4 Radiological Consequences

15.6.5.4.1 Method of Analysis

15.6.5.4.1.1 Containment Leakage Contribution

PHYSICAL MODEL - Following a postulated double-ended rupture of a reactor coolant pipe with subsequent blowdown, the ECCS limits the clad temperature to well below the melting point and ensures that the reactor core remains intact and in a coolable geometry, minimizing the release of fission products to the containment. However, to demonstrate that the operation of a nuclear power plant does not represent any undue radiological hazard to the general public, a hypothetical accident involving a significant release of fission products to the containment is evaluated.

It is assumed that 100 percent of the noble gases and 50 percent of the iodine equilibrium core saturation fission product inventory is immediately released to the containment atmosphere. Of the iodine released to the containment, 50 percent is assumed to plateout onto the internal surfaces of the containment or adhere to internal components. The remaining iodine and the noble gas activity are assumed to be immediately available for leakage from the containment.

Once the gaseous fission product activity is released to the containment atmosphere, it is subject to various mechanisms of removal which operate simultaneously to reduce the amount of activity in the containment. The removal mechanisms include radioactive decay, containment sprays, and containment leakage. For the noble gas fission products, the only removal processes considered in the containment are radioactive decay and containment leakage.

a. Radioactive Decay - Credit for radioactive decay for fission product concentrations located within the containment is assumed throughout the course of the accident. Once the activity is released to the environment, no credit for radioactive decay or deposition is taken.

b. Containment Sprays - The containment spray system is designed to absorb airborne iodine fission products within the containment atmosphere. To enhance the iodine-removal capability of the containment sprays, sodium hydroxide is added to the spray solution. The spray effectiveness for the removal of iodine is dependent on the iodine chemical form.

c. Containment Leakage - The containment leaks at a rate of 0.2 volume percent/day as incorporated as a Technical Specification requirement at peak calculated internal containment pressure for the first 24 hours and at 50 percent of this leak rate for the remaining duration of the accident. The containment leakage is assumed to be directly to the environment.

ASSUMPTIONS AND CONDITIONS - The major assumptions and parameters assumed in the analysis are itemized in Tables 15A-1 and 15.6-6.

In the evaluation of a LOCA, all the fission product release assumptions of Regulatory Guide 1.4 have been followed. The following specific assumptions were used in the analysis. Table 15.6-7 provides a comparison of the analysis to the requirements of Regulatory Guide 1.4.

a. The reactor core equilibrium noble gas and iodine inventories are based on long-term operation at the ultimate core power level of 3,565 MWt.

b. One hundred percent of the core equilibrium radioactive noble gas inventory is immediately available for leakage from the containment.

c. Twenty-five percent of the core equilibrium radioactive iodine inventory is immediately available for leakage from the containment.

d. Of the iodine fission product inventory released to the containment, 91 percent is in the form of elemental iodine, 5 percent is in the form of particulate iodine, and 4 percent is in the form of organic iodine.

e. Credit for iodine removal by the containment spray system is taken, starting at time zero and continuing until a decontamination factor of 100 for the elemental and particulate species has been achieved.

f. The following iodine removal constants for the containment spray system are assumed in the analysis:

Elemental iodine	10.0 hr^{-1}
Organic iodine	0.0 hr^{-1}
Particulate iodine	0.45 hr^{-1}

g. The following parameters were used in the two-region spray model:

Fraction of containment sprayed - 0.85
Fraction of containment unsprayed - 0.15
Mixing rate (cfm) between sprayed and unsprayed regions - 85,000

Section 6.5 contains a detailed analysis of the sprayed and unsprayed volumes and includes an explanation of the mixing rate between the sprayed and unsprayed regions.

h. The containment is assumed to leak at 0.2 volume percent/day during the first 24 hours immediately following the accident and 0.1 volume percent/day thereafter.

i. The containment leakage is assumed to be direct unfiltered to the environment.

j. The ESF filters are 90 percent efficient in the removal of all species of iodine.

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 Section 15A.2.

b. The atmospheric dispersion factors used in the analysis were calculated, based on the onsite meteorological measurements program described in Section 2.3 and are provided in Table 15A-2.

c. The thyroid inhalation total-body immersion doses to a receptor exposed at the exclusion area boundary and the outer boundary of the low population zone were analyzed, using the models described in Sections 15A.2.4 and 15A.2.5, respectively.

d. Buildup of activity in the control room and the integrated doses to the control room personnel are analyzed, based on models described in Section 15A.3.

IDENTIFICATION OF LEAKAGE PATHWAYS AND RESULTANT LEAKAGE ACTIVITY - For evaluating the radiological consequences of a postulated LOCA, the resultant activity released to the containment atmosphere is assumed to leak directly to the environment.

No credit is taken for ground deposition or radioactive decay during transit to the exclusion area boundary or LPZ outer boundary.

15.6.5.4.1.2 Radioactive Releases Due to Leakage from ECCS and Containment Spray Recirculation Lines

Subsequent to the injection phase of ESF system operation, the water in the containment recirculation sumps is recirculated by the residual heat removal, centrifugal charging and safety injection pumps, and the containment spray pumps. Due to the operation of the ECCS and the containment spray system, most of the radioiodine released from the core would be contained in the containment sump. It is conservatively assumed that a leakage rate of 2 gpm from the ECCS and containment spray recirculation lines exists for the duration of the LOCA. This leakage would occur inside the containment as well as inside the auxiliary building. For this analysis, all the leakage is assumed to occur inside the auxiliary building. Only trace quantities of radioiodine are expected to be airborne within the auxiliary building due to the temperature and pH level of the recirculated water. However, 10 percent of the radioiodine in the leaked water is assumed to become airborne. This airborne iodine is assumed to be released immediately to the environment from the unit vent, via the safety grade filters associated with the auxiliary building emergency exhaust system. No credit is taken for holdup or mixing in the auxiliary building; however, mixing and holdup in the containment sumps are included in the determination of radioactive material releases and radioiodine removal through radioactive decay for this leakage pathway.

Radiological Consequences of ECCS/CS Recirculation Line Leakage - The assumptions used to calculate the amount of radioiodine released to the environment are given in Table 15.6-6. The dose models are presented in Section 15.A. The offsite doses at the site boundary and LPZ and the doses to control room personnel from this pathway are given in Table 15.6-8.

15.6.5.4.1.3 Radioactive Releases Due to Operation of Containment Mini-Purge System

The containment mini-purge is designed to reduce the containment noble gas concentration. The containment mini-purge system will be operated during power operation if access to the containment is desired. The containment mini-purge isolation valves are automatically closed upon a containment purge isolation signal should a LOCA occur during containment purging when the reactor is at power. The radioactive release via containment mini-purge will exist until the containment isolation signal is received and the valves can be closed. Exhaust from the containment is processed through the containment purge exhaust system filter absorber train prior to discharge through the unit vent.

The maximum time for the purge valve closure is limited to five seconds to assure that the purge valves would be closed before the onset of fuel failures following a LOCA. Therefore, the source terms used in the radiological consequences calculation is based on the fission product activity in the primary coolant with consideration of pre-existing iodine spike. The containment mini-pure system is assumed to be isolated within 5 seconds following the initiation of the accident. The release rate from the containment mini-purge system is assumed at 4680 cfm. Filter efficiency of 90% for the removal of all species of iodine is assumed. Credit for iodine removal by the containment spray system is not assumed in the analysis for activity release via containment mini-purge system.

From the safety analysis perspective, it is acceptable to use either the shutdown purge or mini-purge during refueling operations. This conclusion is based on an assumption used in the fuel handling accident (FHA) involving the radioactive material relief rate. To comply with Reg. Guide 1.25, all of the gap activity in the damaged rods is assumed as a result of a FHA, to be released and escape to the environment over a two-hour time period. The analysis does not assume pathway, only that all the radioactivity is released from containment within a two hour period in some fashion. Thus the operation of a particular purge system, mini or shutdown is not of importance in the analysis. This assumption translates into a very large air exhaust rate. Therefore, the difference between the large volume containment shutdown purge (20,000 cfm) and mini-purge (4,000 cfm) would have no impact on the calculated dose consequences.

15.6.5.4.1.4 Radioactive Releases Due to Leakage from the Containment Sump to the RWST

The leakage pathway is from the containment recirculation sump through ECCS boundary valves back to the RWST, which is vented to the atmosphere. It is assumed that the activity released to the holdup system (in this case, the containment recirculation sump) instantaneously diffuses to uniformly occupy the sump volume. Removal mechanisms from the sump include decay and release (i.e., leakage) to the RWST.

It is assumed that 10% of the radioiodine leaked to the RWST becomes airborne, mixes with the RWST volume, and is released to the environment. Credit is taken for radioactive decay in the RWST. The leakage rate from containment sump to RWST is assumed at 5 gpm for the first 16 hours. At 16 hours following initiation of the LOCA event, the operator was instructed to close BNV0011 per procedure EMG ES-12. Also, after 16 hours, a 2 gpm leakage instead of a 5 gpm leakage was assumed based on the double isolation of ECCS boundary valves and small driving force present. The RWST volume is assumed to be 400,000 gallons. Other assumptions used to calculate the amount of radioiodine released to the environment are the same as in the calculation for ECCS recirculation leakage inside the auxiliary building.

15.6.5.4.2 Identification of Uncertainties and Conservatisms in the Analysis

The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of a LOCA result principally from assumptions made involving the amount of the gaseous fission products available for release to the environment and the meteorology present at the site during the course of the accident. The most significant of these assumptions are:

a. The ECCS is designed to prevent fuel cladding damage that would allow the release of the fission products contained in the fuel to the reactor coolant. Severe degradation of the ECCS (i.e., to the unlikely extent of simultaneous failure of redundant components) would be necessary in order for the release of fission products to occur of the magnitude assumed in the analysis.

b. The release of fission products to the containment is assumed to occur instantaneously.

c. It is assumed that 50 percent of the iodines released to the containment atmosphere is plated-out onto the internal surfaces of the containment or adheres to internal components; however, it is estimated that the removal of airborne iodines by various physical phenomena such as adsorption, adherence, and settling could reduce the resultant doses by a factor of 3 to 10 (Ref. 24).

d. The activity released to the containment atmosphere is assumed to leak to the environment at the containment leakage rate of 0.2-volume percent/day for the first 24 hours and 0.1-volume percent/day thereafter. The initial containment leakage rate is based on the peak calculated internal containment pressure anticipated after a LOCA. The pressure within the containment actually decreases with time. Taking into account that the containment leak rate is a function of pressure, the resultant doses could be reduced by a factor of 5 to 10 (Ref. 24).

e. The meteorological conditions assumed to be present at the site during the course of the accident are based on X/Q values, which are expected to be exceeded 5 percent of the time. This condition results in the poorest values of atmospheric dispersion calculated for the exclusion area boundary and the LPZ outer boundary. Furthermore, no credit has been taken for the transit time required for activity to travel from the point of release to the exclusion area boundary and LPZ outer boundary. Hence, the radiological consequences evaluated under these conditions are conservative.

15.6.5.4.3 Conclusions

15.6.5.4.3.1 Filter Loadings

No recirculating or single-pass filters are used for fission product cleanup and control within the containment following a postulated LOCA. The only ESF filtration systems expected to be operating under post-LOCA conditions are the control room HVAC system and the auxiliary building emergency exhaust filtration system.

Activity loadings on the control room charcoal adsorbers are based on the flowrate through the adsorber, the concentration of activity at the adsorber inlet, and the adsorber efficiency. Based on the radioactive iodine release assumptions previously described, the assumption that 25 percent of the core inventory of isotopes I-127 and I-129 is available for release from the containment atmosphere and the assumption that the charcoal adsorber is 100 percent efficient, the calculated filter loadings are in accordance with Regulatory Guide 1.52, which limits the maximum loading to 2.5 mg of iodine per gram of activated charcoal.

15.6.5.4.3.2 Doses to a Receptor at the Exclusion Area Boundary and Low Population Zone Outer Boundary

The potential radiological consequences resulting from the occurrence of the postulated LOCA have been conservatively analyzed, using assumptions and models described in previous sections.

The total-body dose due to immersion and the thyroid dose due to inhalation have been analyzed for the 0-2 hour dose at the exclusion area boundary and for the duration of the accident at the LPZ outer boundary. The results are listed in Table 15.6-5. The resultant doses are within the guideline values of 10 CFR 100.

15.6.5.4.3.3 Doses to Control Room Personnel

Radiation doses to control room personnel following a postulated LOCA are based on the ventilation, cavity dilution, and dose model discussed in Section 15A.3.

Control room personnel are subject to a total-body dose due to immersion and a thyroid dose due to inhalation. These doses have been analyzed, and are provided in Table 15.6-8. The resultant doses are within the limits established by GDC-19.

15.6.6 A NUMBER OF BWR TRANSIENTS

This section is not applicable to WCGS.

15.6.7 REFERENCES

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TABLE 15.6-1

TIME SEQUENCE OF EVENTS FOR INCIDENT WHICH RESULTS IN A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	Time (sec)
Inadvertent opening of a pressurizer safety valve	Safety valve opens fully	0.01
	Overtemperature DT reactor trip setpoint reached	26.7
	Turbine Trip	29.2
Steam generator tube rupture: Failed-open auxiliary feedwater control valve and safety valve	Tube Rupture Occurs	0.0
	Reactor Trip Signal	138.6
	Reactor Trip (Rod Motion)	140.6
	Auxiliary Feedwater Injection	170.6
	Safety Injection Signal	216.6
	Safety Injection (Pump Start)	231.6
	Terminate Auxiliary Feedwater to Faulted SG by closing unfailed AL-HV valve	696.6
	Terminate Auxiliary Feedwater to Faulted SG by locally closing failed AL-HV valve	1296.6
	Initiate RCS Cooldown	2016.6
	Terminate RCS Cooldown	2404.7
	Water Relief Through SG Safety Valve	2710.0
	Initiate RCS Depressurization	2884.7
	Terminate RCS Depressurization	3047.7
	Terminate Safety Injection (except 1 CCP)	3349.4
	Throttle CCP flow to 100 gpm	3947.7
	Establish Letdown	4847.7

TABLE 15.6-1 (Sheet 2)

Accident	Event	Time (sec)
Steam generator tube rupture: stuck open atmospheric relief valve	Reactor Trip Signal	140.0
	Reactor Trip	142.0
	Faulted SG ARV Open	144.0
	Auxiliary Feedwater Injection	202.0
	Safety Injection Signal	297.0
	Safety Injection	322.0
	Faulted SG ARV Isolated	1344.0
	Initiate RCS Cooldown	2216.0
	Terminate RCS Cooldown	3024.0
	Initiate RCS Depressurization	3204.0
	Terminate RCS Depressurization	3364.0
	Terminate Safety Injection	3544.0
	Pressure Equalization	3844.0

TABLE 15.6-2

PARAMETERS USED IN EVALUATING
THE RADIOLOGICAL CONSEQUENCE OF
THE CVCS LETDOWN LINE RUPTURE OUTSIDE OF CONTAINMENT

I.	Source Data	
	a. Core power level, MWt	3,565
	b. Reactor coolant iodine activity	Dose equivalent of 1.0 μCi/gm of I-131
	c. Reactor coolant noble gas activity	Based on 1-percent failed fuel. See Table 11.1-5.
II.	Atmospheric Dispersion Factors	See Table 15A-2.
III.	Activity Release	
	a. Break flow rate, gpm	282
	b. Duration, secs	1810
	c. Fraction of iodine activity in the spill that is airborne	0.10
	d. Activity released to the environment	

Isotope	0-2 hr (Ci)
I-131	2.35
I-132	2.47
I-133	4.11
I-134	5.38E-1
I-135	2.36
Xe-131m	1.10E+2
Xe-133m	1.72E+2
Xe-133	9.31E+3
Xe-135m	1.22E+1
Xe-135	3.11E+2
Xe-137	2.15
Xe-138	1.60E+1
Kr-83m	1.65E+1
Kr-85m	7.03E+1
Kr-85	3.03E+2
Kr-87	4.23E+1
Kr-88	1.30E+2
Kr-89	1.01

TABLE 15.6-3

RADIOLOGICAL CONSEQUENCES OF A
CVCS LETDOWN LINE BREAK OUTSIDE
OF CONTAINMENT

	Doses (rem)
Exclusion Area Boundary (0-2 hr)	
Thyroid,	2.49E-1
Whole body	2.81E-2
Low Population Zone Outer Boundary (duration)	
Thyroid,	3.31E-2
Whole body	3.75E-3

Table 15.6-4

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE WITH FORCED OVERFILL

I. Source Data

a.	Core power level, MWt	3636.3(1)
b.	Steam generator tube leakage, gpm	3(2)
c.	Reactor coolant iodine activity:	
	1. Case 1	Pre-accident iodine spike per Standard Review Plan 15.6.3. A pre-accident reactor transient has occurred which raised the primary coolant I-131 equivalent concentration to the maximum value permitted by Technical Specifications (60 μCi/gm).
	2. Case 2	Accident initiated iodine spike per Standard Review Plan 15.6.3. An iodine spike occurs concurrently with reactor trip where the release rate from the fuel rods increases to a value 500 time greater than the release rate corresponding to the I-131 equivalent concentration at the equilibrium value (1 μCi/gm) stated in the Technical Specifications.
d.	Reactor coolant noble gas activity	Based on 1% failed fuel as provided in USAR Table 11.1-5.
e.	Secondary system initial activity	Dose equivalent of 0.1 μCi/gm of I-131
f.	Reactor coolant mass, lbs	4.814E+5
g.	Steam generator mass (each), lbs	
	1. Water	9.904E+4
	2. Steam	8.007E+3
h.	Offsite power	Lost at reactor trip

i.	Primary-to-secondary leakage duration	Break flow continues throughout the transient, due to primary-secondary pressure inequalities, although most of the flow is terminated beyond the end of the first depressurization at approximately 6300 seconds(3).

II. Atmospheric Dispersion Factor — The atmospheric dispersion factors used are the same as those listed in USAR Table 15A-2.

III. Activity Release Data

a. Faulted steam generator

1. Reactor coolant discharged to steam generator, lbs — 196,395
2. Flashed reactor coolant — Discussion below

The fraction of the primary-to-secondary leakage that flashes to steam is defined as, X(t), where

$$X(t) = \frac{h_{avg}(t) - h_f(t)}{h_g(t) - h_f(t)}$$

and where $h_g(t)$ = saturated vapor specific enthalpycorresponding to the faulted steamgenerator water temperature,

$h_f(t)$ = saturated liquid specific enthalpy corresponding to the faulted steamgenerator water temperature.

The value $h_{avg}(t)$ is the average specific enthalpy and is defined as:

$$h_{avg} = \frac{[G_{HL}(t)*h_{HL}(t)] + [G_{CL}(t)*h_{CL}(t)]}{G_{HL}(t) + G_{CL}(t)}$$

where $h_{HL}(t)$ = specific enthalpy of the fluid from the hot leg,

$h_{CL}(t)$ = specific enthalpy of the fluid from the cold leg,

Table 15.6-4 (sheet 3)

$G_{HL}(t)$ = leakage flow from the hot leg to the faulted steam generator,

$G_{CL}(t)$ = leakage flow from the cold leg to faulted steam generator.

	3.	Iodine partition factor for flashed fraction of reactor coolant	1.0
	4.	Steam release to atmosphere, lbs	
		0-2 hours	164,982[4]
		2 hour-RHR cut-in conditions	131,760[4]
	5.	Iodine carryover factor for the non-flashed fraction of reactor coolant that mixes with the initial iodine activity in the steam generator	0.01
b.		Intact steam generators	
	1.	Primary-to-secondary leakage, lbs	4467[2]
	2.	Flashed reactor coolant, percent	0
	3.	Feedwater flow rate, lbs	
		0-2 hours	487,300
		2 hours-RHR cut-in conditions	295,200
	4.	Steam release to atmosphere, lbs	
		0-2 hours	298,783
		2-hours-RHR cut-in conditions	63,360
	5.	Iodine carryover factor	0.01
	6.	Transient end time	The WCNOC analysis to force SGTR steam generator overfilling was carried out until RHR cut-in conditions were achieved at 14,400 seconds.

Table 15.6-4 (sheet 4)

Isotope	Case 1 0-2 hours (Ci)	Case 1 0-8 hours (Ci)	Case 2 0-2 hours (Ci)	Case 2 0-8 hours Ci)
I-131	5.4587E+2	5.6795E+2	1.9253E+2	2.0266E+2
I-132	4.3633E+2	4.4638E+2	6.5826E+2	6.8013E+2
I-133	9.2867E+2	9.6346E+2	4.2650E+2	4.4795E+2
I-134	6.2338E+1	6.2951E+1	2.3802E+2	2.4183E+2
I-135	4.8817E+2	5.1558E+2	3.6171E+2	3.7799E+2
Xe-131m	3.3625E+2	3.3973E+2	3.3625E+2	3.3973E+2
Xe-133m	5.2972E+2	5.3522E+2	5.2972E+2	5.3522E+2
Xe-133	2.8588E+4	2.8884E+4	2.8588E+4	2.8884E+4
Xe-135m	6.2648E+1	6.3323E+1	6.2648E+1	6.3323E+1
Xe-135	9.7093E+2	9.8117E+2	9.7093E+2	9.8117E+2
Xe-137	2.3820E+1	2.4066E+1	2.3820E+1	2.4066E+1
Xe-138	8.4750E+1	8.5661E+1	8.4750E+1	8.5661E+1
Kr-83m	5.5337E+1	5.5946E+1	5.5337E+1	5.5946E+1
Kr-85m	2.2408E+2	2.2649E+2	2.2408E+2	2.2649E+2
Kr-85	9.2763E+2	9.3722E+2	9.2763E+2	9.3722E+2
Kr-87	1.4756E+2	1.4920E+2	1.4756E+2	1.4920E+2
Kr-88	4.2368E+2	4.2829E+2	4.2368E+2	4.2829E+2
Kr-89	1.2894E+1	1.3026E+1	1.2894E+1	1.3026E+1

Table 15.6-4 (sheet 5)

Notes:

1. This value is 102% of the guaranteed core thermal power output. This power level was chosen to maximize steam generator overfill, and thus the radiological consequences resulting from the overfill were calculated from the results of this model.

2. A constant 1 gpm primary-to-secondary leakage is assumed in each intact steam generator at a density of 46.4 lbs/ft^3 (557 ^{o}F, 2250 psia) over 14,400 seconds.

3. Steam generator overfill was forced by delaying the termination of safety injection to provide enough liquid in the faulted steam generator steam line to force safety valve opening and liquid relief. The extended SI flow time caused the primary pressure to remain high, thus the primary/secondary pressure remained unequal and allowed for break flow to continue for an extended period of time.

4. The SGTR with Forced Overfill analysis assumed that at liquid relief through the faulted steam generator steam line safety valve, the valve would fail open with an effective flow area of 5%, with consequent continued secondary blowdown until RHR cut-in conditions were reached.

TABLE 15.6-5

RADIOLOGICAL CONSEQUENCES OF A
STEAM GENERATOR TUBE RUPTURE

	Dose (rem)
Case 1	
Exclusion Area Boundary (0-2 hr)	
Thyroid	57.12
Whole body	0.189
Low Population Zone (duration)	
Thyroid	7.92
Whole body	0.026
Case 2	
Exclusion Area Boundary (0-2 hr)	
Thyroid	22.50
Whole body	0.195
Low Population Zone (duration)	
Thyroid	3.13
Whole body	0.027

Case 1 - Pre-accident iodine spike per SRP 15.6.3

Case 2 - Concurrent iodine spike per SRP 15.6.3

TABLE 15.6-6

PARAMETERS USED IN EVALUATING
THE RADIOLOGICAL CONSEQUENCES OF A
LOSS-OF-COOLANT-ACCIDENT

I.	Source Data	
	a. Core power level, MWt	3,565
	b. Burnup, full power days per cycle	510
	c. Percent of core activity initially airborne in the containment	
	1. Noble gas	100
	2. Iodine	25
	d. Percent of core activity in containment sump @ 0.47 hours	
	1. Noble gases	0
	2. Iodine	50
	e. Core inventories	Table 15A-3
	f. Iodine distribution, percent	
	1. Elemental	91
	2. Organic	4
	3. Particulate	5
II.	Atmospheric Dispersion Factors	See Table 15A-2
III.	Activity Release Data	
	a. Containment leak rate, volume percent/day	
	1. 0-24 hours	0.20
	2. 1-30 days	0.10
	b. Percent of containment leakage that is unfiltered	100
	c. Credit for containment sprays	
	1. Spray iodine removal constants (per hour)	
	a. Elemental	10.0
	b. Organic	0.0
	c. Particulate	0.45

2. Maximum iodine decontamination factors for the containment atmosphere

a.	Elemental	100
b.	Organic	0
c.	Particulate	100

3.	Sprayed volume, percent	85
4.	Unsprayed volume, percent	15
5.	Sprayed-unsprayed mixing rate, CFM	85,000
6.	Containment volume, ft^3	2.5E+6

d. Activity released to containment

Isotope	Curies
I-131	2.37E+7
I-132	3.43E+7
I-133	4.88E+7
I-134	5.38E+7
I-135	4.58E+7
Xe-131m	1.01E+6
Xe-133m	6.06E+6
Xe-133	1.95E+8
Xe-135m	3.77E+7
Xe-135	4.70E+7
Xe-137	1.71E+8
Xe-138	1.64E+8
Kr-83m	1.24E+7
Kr-85m	2.67E+7
Kr-85	1.02E+6
Kr-87	5.16E+7
Kr-88	7.28E+7
Kr-89	8.94E+7

e. ECCS recirculation leakage

1.	Leak rate (0.47 hours-30 day), gpm	2.0

TABLE 15.6-6 (Sheet 3)

2.	Iodine inventory in sump @ 0.47 hour, curies	
	I-131	4.72E+7
	I-132	5.94E+7
	I-133	9.60E+7
	I-134	7.41E+7
	I-135	8.71E+7
3.	Sump volume, gal.	460,000
4.	Fraction iodine airborne	0.1
5.	ESF filter efficiency, %	90.0

IV. Control room parameters — Tables 15A-1 and 15A-2

TABLE 15.6-7

DESIGN COMPARISON TO THE REGULATORY POSITIONS OF REGULATORY GUIDE 1.4 "ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS OF COOLANT ACCIDENT FOR PRESSURIZED WATER REACTORS," REVISION 2, JUNE 1974

Regulatory Guide 1.4 Position	Design
1. The assumptions related to the release of radioactive material from the fuel and containment are as follows:	
a. Twenty-five percent of the equilibrium radioactive iodine inventory developed from maximum full power operation of the core should be assumed to be immediately available for leakage from the primary reactor containment. Ninety-one percent of this 25 percent is to be assumed to be in the form of elemental iodine, 5 percent of this 25 percent in the form of particulate iodine, and 4 percent of this 25 percent in the form of organic iodides.	1a. Complies.
b. One hundred percent of equilibrium radioactive noble gas inventory developed from maximum full power operation of the core should be assumed to be immediately available for leakage from the reactor containment.	1b. Complies.
c. The effects of radiological decay during holdup in the containment or other buildings should be taken into account.	1c. Complies. Credit for radioactive decay is taken until the activity is assumed to be released.
d. The reduction in the amount of radioactive material available for leakage to the environment by containment sprays, recirculating filter systems, or other engineered safety features may be taken into account, but the amount of reduction in concentration of radioactive materials should be evaluated on an individual case basis.	1d. Complies. See Table 15.6-6 for reduction taken.

TABLE 15.6-7 (Sheet 2)

Regulatory Guide 1.4 Position	Design
e. The primary reactor containment should be assumed to leak at the leak rate incorporated or to be incorporated as a technical specification requirement at peak accident pressure for the first 24 hours, and at 50 percent of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing.	1e. Complies.
2. Acceptable assumptions for atmospheric diffusion and dose conversion are: a. The 0-8 hour ground level release concentrations may be reduced by a factor ranging from one to a maximum of three (see Figure 1) for additional dispersion produced by the turbulent wake of the reactor building in calculating potential exposures. The volumetric building wake correction, as defined in section 3-3.5.2 of Meteorology and Atomic Energy 1968, should be used only in the 0-8 hour period; it is used with a shape factor of 1/2 and the minimum cross-sectional area of the reactor building only.	2a. Complies. Atmospheric dispersion factors were calculated based on the onsite meteorological measurement programs described in Section 2.3.
b. No correction should be made for depletion of the effluent plume of radioactive iodine due to deposition on the ground, or for the radiological decay of iodine in transit.	2b. Same as 2a above.
c. For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.47×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.75×10^{-4} cubic meters per second. After that until the end of the accident, the breathing rate should be assumed to be 1.75×10^{-4} cubic meters per second. After that until the end of the accident, the rate should be assumed to be 2.32×10^{-4} cubic meters	2c. Complies. See Table 15A-1

TABLE 15.6-7 (Sheet 3)

Regulatory Guide 1.4 Position	Design
per second. (These values were developed from the average daily breathing rate [2 x 10^7 cm3/day] assumed in the report of ICRP, Committee II-1959.)	
d. The iodine dose conversion factors are given in ICRP Publication 2, Report of Committee II, "Permissible Dose for Internal Radiation," 1959.	2d. The dose conversion factors provided in Regulatory Guide 1.109 are used. See Table 15A-4.
e. External whole body doses should be calculated using "Infinite Cloud" assumptions, i.e., the dimensions of the cloud are assumed to be large compared to the distance that the gamma rays and beta particles travel. "Such a cloud would be considered an infinite cloud for a receptor at the center because any additional [gamma and] beta emitting material beyond the cloud dimensions would not alter the flux of [gamma rays and] beta particles to the receptor" (Meteorology and Atomic Energy, Section 7.4.1.1-editorial additions made so that gamma and beta emitting material could be considered). Under these conditions the rate of energy absorption per unit volume is equal to the rate of energy released per unit volume. For an infinite uniform cloud containing curies of beta radioactivity per cubic meter the beta dose in air at the cloud center is: $\beta^{D\infty} = 0.457\ \overline{E}_\beta \chi$ The surface body dose rate from beta emitters in the infinite cloud can be approximated as being one-half this amount (i.e., $\beta^{D\infty} = 0.23\ \overline{E}_\beta \chi$) For gamma emitting material the dose rate in air at the cloud center is: $\gamma^{D\infty} = 0.507 \overline{E}_\gamma x$ From a semi-infinite cloud, the gamma dose rate in air is: $\gamma^{D\infty} = 0.25 \overline{E}_\gamma x$	2e. The dose factors given in Regulatory Guide 1.109, for noble gases; for iodine whole body dose factors with 5cm body tissue attenuation; and for beta-skin dose factors with credit for attenuation in the dead skin layer, are used. See Table 15A-4.

TABLE 15.6-7 (Sheet 4)

Regulatory Guide 1.4 Position	Design
Where $_{\beta}D_{\infty}$ = beta dose rate from an infinite cloud (rad/sec) $_{\gamma}D_{\infty}$ = gamma dose rate from an infinite cloud (rad/sec) $\overline{E}_{\beta}$ = average beta energy per disintegration (Mev/dis) $\overline{E}_{\gamma}$ = average gamma energy per disintegration (Mev/dis) χ = concentration of beta or gamma emitting isotope in the cloud (curie/m^3)	
f. The following specific assumptions are acceptable with respect to the radioactive cloud dose calculations:	2f.1 See response to 2e.
(1) The dose at any distance from the reactor should be calculated based on the maximum concentration in the plume at that distance taking into account specific meteorological, topographical, and other characteristics which may affect the maximum plume concentration. These site related characteristics must be evaluated on an individual case basis. In the case of beta radiation, the receptor is assumed to be exposed to an infinite cloud at the maximum ground level concentration at that distance from the reactor. In the case of gamma radiation, the receptor is assumed to be exposed to only one-half the cloud owing to the presence of the ground. The maximum cloud concentration always should be assumed to be at ground level.	
(2) The appropriate average beta and gamma energies emitted per disintegration, as given in the Table of Isotopes, Sixth Edition, by C. M. Lederer, J. M. Hollander, I. Perlman;	2f.2 See response to 2e.

TABLE 15.6-7 (Sheet 5)

Regulatory Guide 1.4 Position	Design
University of California, Berkeley; Lawrence Radiation Laboratory; should be used.	
g. The atmospheric diffusion model should be as follows:	
(1) The basic equation for atmospheric diffusion from a ground level point source is: $\chi/Q = \pi\frac{1}{u\sigma_y}$ Where χ = the short term average centerline value of the ground level concentration (curie/meter^3) Q = amount of material released (curie/sec) u = windspeed (meter/sec) σ_y = the horizontal standard deviation of the plume (meters) [See Figure V-1, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteororological Observations for Estimating Atmospheric Dispersion," F.A. Gifford, Jr.] σ_z = the vertical standard deviation of the plume (meters) [See Figure V-2, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F.A. Gifford, Jr.]	2g.1 Short-term accident atmospheric dispersion factors were calculated based on onsite meteorological measurement programs described in Section 2.3. These factors are for ground level releases and are based on Regulatory Guide 1.145 methodology and represent the worst of the 5 percent site meteorology and the 0.5 percent worst sector meteorology.
(2) For time periods of greater than 8 hours the plume should be assumed to meander and spread uniformly over a 22.5° sector. The resultant equation is: $\chi/Q = \frac{2.032}{\sigma_z u x}$	2g.2 See response to 2g.1 above.

TABLE 15.6-7 (Sheet 6)

Regulatory Guide 1.4 Position	Design
Where x = distance from point of release to the receptor; other variables are as given in g(1).	
(3) The atmospheric diffusion model[2] for ground level releases is based on the information in the following table.	2g.3 See response to 2g.1 above.

Time Following Accident	Atmospheric Conditions
0-8 hours	Pasquill Type F, windspeed 1 meter/sec, uniform direction
8-24 hours	Pasquill Type F, windspeed 1 meter/sec, variable direction within a 22.5° sector
1-4 days	(a) 40% Pasquill Type D, windspeed 3 meter/sec (b) 60% Pasquill Type F, windspeed 2 meter/sec (c) wind direction variable within a 22.5° sector
4-30 days	(a) 33.3% Pasquill Type C, windspeed 3 meter/sec (b) 33.3% Pasquill Type D, windspeed 3 meter/sec (c) 33.3% Pasquill Type F, windspeed 2 meter/sec (d) Wind direction 33.3% frequency in a 22.5° sector

Regulatory Guide 1.4 Position	Design
(4) Figures 2A and 2B give the ground level release atmospheric diffusion factors based on the parameters given in g(3).	2g.4 See response to 2g.1 above.

TABLE 15.6-8
Radiological Consequences of a
Loss-of-Coolant-Accident

		Total Reported Doses (rem)	Regulatory Limits (rem)
I.	Exclusion Area Boundary (0-2 hr)		
	Thyroid	88.82	300
	Whole body	1.53	25
II.	Low Population Zone Outer Boundary(0-30 day)		
	Thyroid	105.87	300
	Whole body	0.57	25
III.	Control Room (0-30 day)		
	Thyroid	17.97	30
	Whole body	0.20	5
	Beta-skin	3.27	30

TABLE 15.6-9
INPUT PARAMETERS USED IN THE ECCS ANALYSIS

License Core Power[1] (MWth)	3565
Peak Linear Power [1] (KW/ft)	14.225
Total Peaking Factor [F_Q^T]	2.50
Axial Peaking Factor [F_z]	1.5151
Hot Channel Enthalpy Rise Factor[$F_{\Delta H}$]	1.65
Maximum ASsembly Average Power [P_{HA}]	1.469
Power Shape	
Large Break	Chopped Cosine
Small Break	Figure 15.6-47
Fuel Type/Assembly Array	VANTAGE 5H with IFMs (17X17)
Accumulator Water Volume (ft^3/accumulator)	850
Accumulator Tank Volume (ft^3/accumulator)	1350
Accumulator Gas Pressure, Minimum (psia)	600
Safety Injection Pumped Flow	
Large Break	Table 15.6-13
Small Break	Table 15.6-14
Containment Parameters	See Section 6.2.1.5
Initial Loop Flow (gpm/loop)	93200
Vessel Average Temperature (F)	
Large Break (Reduced T_{AVG})	570.7
Small Break (Nominal T_{AVG})	588.4
Reactor Coolant Pressure[2] (psia)	2300
Steam Pressure (psia)	
Large Break (Reduced T_{AVG})	802.3
Small Break (Nominal T_{AVG})	944.5
Steam Generator Tube Plugging Level (%)	10
AFW Flow Rate (gpm/steam generator)	210

[1] Two percent is added to this power to account for calorimetric uncertainty. Reactor coolant pump heat is not modeled in LOCA Analysis
[2] This pressure includes 50 psi for measurement uncertainty.

WOLF CREEK

TABLE 15.6-10

TIME SEQUENCE OF EVENTS FOR LOSS-OF-COOLANT ACCIDENTS

Accident	Event	Time (sec)
Large Break LOCA		
a. DECLG C_D-0.4	Start	0.0
	Reactor Trip Signal	0.718
	Safety Injection Signal	2.210
	Accumulator Injection Begins	19.756
	Pumped Safety Injection Begins	41.210
	End-of-Bypass	42.430
	End-of-Blowdown	42.430
	Bottom of Core Recovery	59.363
	Accumulator Empty	62.557
b. DECLG C_D-0.6 (Minimum SI)	Start	0.0
	Reactor Trip Signal	0.706
	Safety Injection Signal	1.800
	Accumulator Injection Begins	14.503
	Pumped Safety Injection Begins	40.800
	End-of-Bypass	32.380
	End-of-Blowdown	32.380
	Bottom of Core Recovery	47.659
	Accumulator Empty	55.300
c. DECLG C_D-0.6 (Minimum SI)	Start	0.0
	Reactor Trip Signal	0.706
	Safety Injection Signal	1.800
	Accumulator Injection Begins	14.503
	Pumped Safety Injection Begins	40.800
	End-of-Bypass	32.380
	End-of-Blowdown	32.380
	Bottom of Core Recovery	47.489
	Accumulator Empty	55.577
d. DECLG C_D-0.8	Start	0.0
	Reactor Trip Signal	0.698
	Safety Injection Signal	1.580
	Accumulator Injection Begins	12.005
	Pumped Safety Injection Begins	40.580
	End-of-Bypass	27.818
	End-of-Blowdown	27.818
	Bottom of Core Recovery	42.362
	Accumulator Empty	51.562

WOLF CREEK

TABLE 15.6-10 (Sheet 2)

Accident	Event	Time (sec)
Small Break LOCA		
a. 2 inch	Start	0.0
	Reactor trip signal	84.07
	Top of core uncovered	2074
	Accumulator injection begins	N/A
	Peak clad temperature occurs	3610
	Top of core covered	5726
b. 3 inch	Start	0.0
	Reactor trip signal	28.48
	Top of core uncovered	936
	Accumulator injection begins	3331
	Peak clad temperature occurs	1446
	Top of core covered	2887
c. 4 inch	Start	0.0
	Reactor trip signal	15.9
	Top of core uncovered	623
	Accumulator injection begins	894
	Peak clad temperature occurs	948
	Top of core covered	1313
d. 6 inch	Start	0.0
	Reactor trip signal	9.37
	Top of core uncovered	216
	Accumulator injection begins	370
	Peak clad temperature occurs	423
	Top of core covered	450

TABLE 15.6-11

LARGE BREAK LOCA RESULTS
FUEL CLADDING DATA

Results	DECLG C_D-0.4	DECLG C_D-0.6	DECLG C_D-0.8	DECLG C_D-0.6 MAXIMUM SI
Peak Clad Temperature (°F)	1916.0	1828.4	1627.4	1739.4
Peak Clad Temperature Location (ft)	8.00	8.75	7.00	8.75
Peak Clad Temperature Time (sec)	161.4	205.5	91.1	182.3
Local Zr/H_2O Reaction*, Maximum (%)	3.64	2.67	1.28	1.91
Local Zr/H_2O Reaction Location (ft)	6.25	8.00	7.00	8.75
Total Zr/H_2O Reaction (%)	<1.0	<1.0	<1.0	<1.0
Hot Rod Burst Time (sec)	71.8	47.8	49.1	47.6
Hot Rod Burst Location (ft)	6.25	6.00	6.00	6.00
Hot Assembly Channel Blockage (%)	39.78	46.24	N/A	45.89

* Total thickness at transient termination. Reaction rate decreasing as core continues to quench.

TABLE 15.6-12

SMALL BREAK LOCA RESULTS FUEL CLADDING DATA

Results	2 inch	3 inch	4 inch	6 inch
Peak Clad Temperature (°F)	1434	1510	1398	1164
Peak Clad Temperature Location (ft)	11.75	11.75	11.5	11.25
Peak Clad Temperature Time (sec)	3610	1446	948	423
Local Zr/H_2O Reaction*, Maximum (%)	0.494	0.939	0.193	0.0441
Local Zr/H_2O Reaction Location (ft)	11.75	11.75	11.5	11.25
Total Zr/H_2O Reaction (%)	<1.0	<1.0	<1.0	<1.0
Hot Rod Burst Time (sec)	No Burst	No Burst	No Burst	No Burst
Hot Rod Burst Location (ft)	N/A	N/A	N/A	N/A

TABLE 15.6-13

SAFETY INJECTION PUMPED FLOW
ASSUMED FOR LARGE BREAK LOCAs

MINIMUM SAFEGUARDS: One line spills to 0 psig containment backpressure

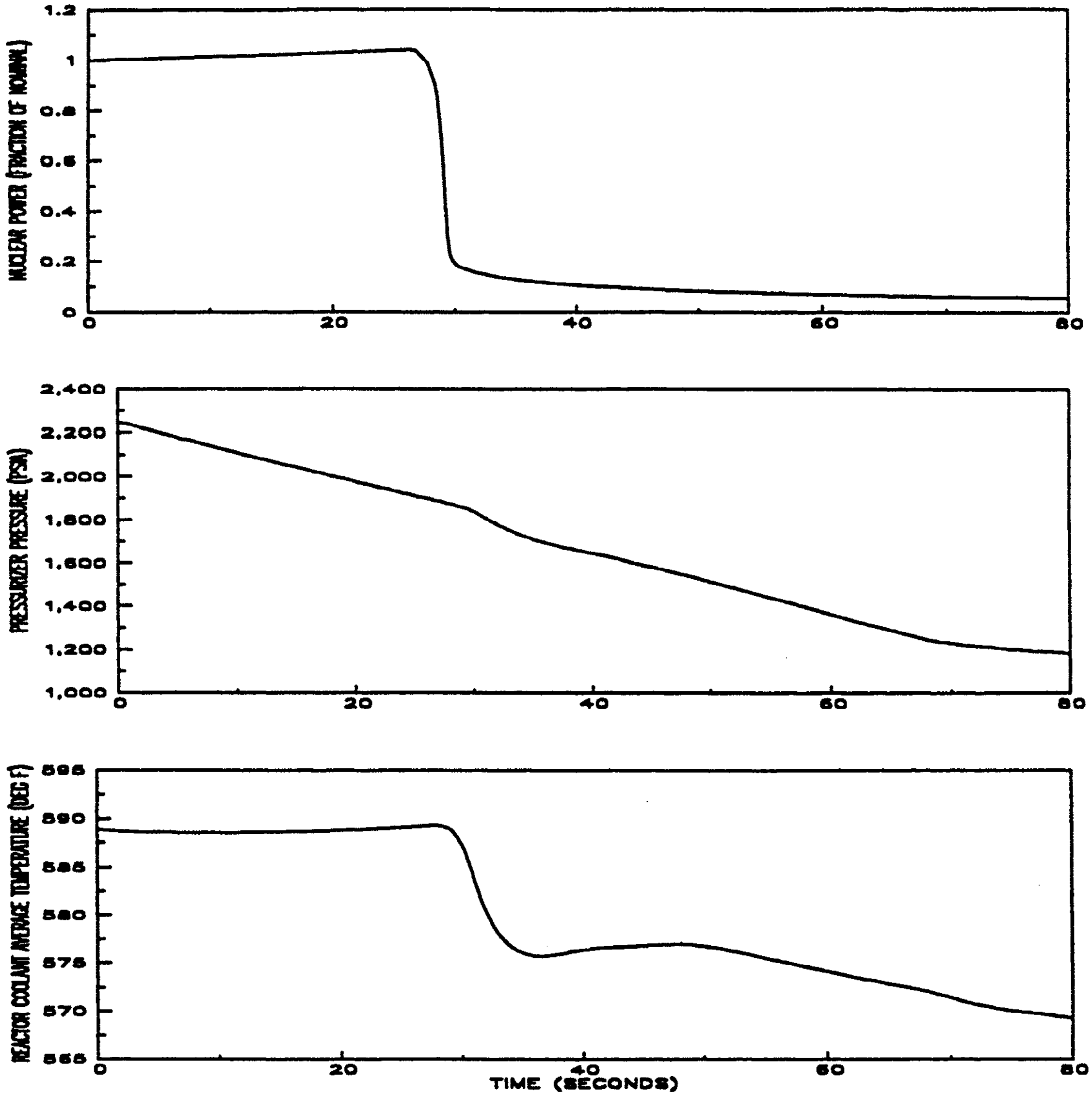
RCS PRESSURE (PSIG)	CCP (GPM)	SIP (gpm)	RHR (gpm)
0	326.85	444.82	2834.19
20	na	na	2579.31
40	na	na	2310.43
60	na	na	2023.36
80	na	na	1712.00
100	310.35	424.60	1366.47
120	na	na	967.94
140	na	na	464.71
150	na	na	186.62
155	na	na	7.15
160	na	na	0
200	293.88	403.66	0
300	277.39	381.93	0
400	261.61	359.28	0
500	244.07	335.54	0
600	227.15	310.53	0
700	209.99	283.99	0
800	192.52	255.56	0
900	174.67	224.71	0
1000	156.36	190.63	0
1100	137.49	151.88	0
1200	117.95	105.54	0
1300	97.56	43.25	0
1400	76.11	0	0
1500	53.27	0	0
1600	28.54	0	0
1800	0	0	0
1900	0	0	0
2000	0	0	0

TABLE 15.6-14

SAFETY INJECTION PUMPED FLOW
ASSUMED FOR SMALL BREAK LOCAs

MINIMUM SAFEGUARDS: One CCP line spills to 0 psig containment backpressure and one SIP line spills to RCS pressure

RCS PRESSURE (psig)	CCP (gpm)	SIP (gpm)
0	326.85	444.82
100	310.35	429.33
200	293.88	413.26
300	277.39	396.55
400	261.61	379.11
500	244.07	360.82
600	227.15	341.54
700	209.99	321.09
800	192.52	299.20
900	174.67	275.52
1000	156.36	249.49
1100	137.49	220.21
1200	117.95	185.98
1300	97.56	142.84
1400	76.11	72.76
1420	na	33.99
1440	na	0.00
1500	53.27	0.00
1600	28.54	0
1800	0	0
1900	0	0
2000	0	0



Rev. 12

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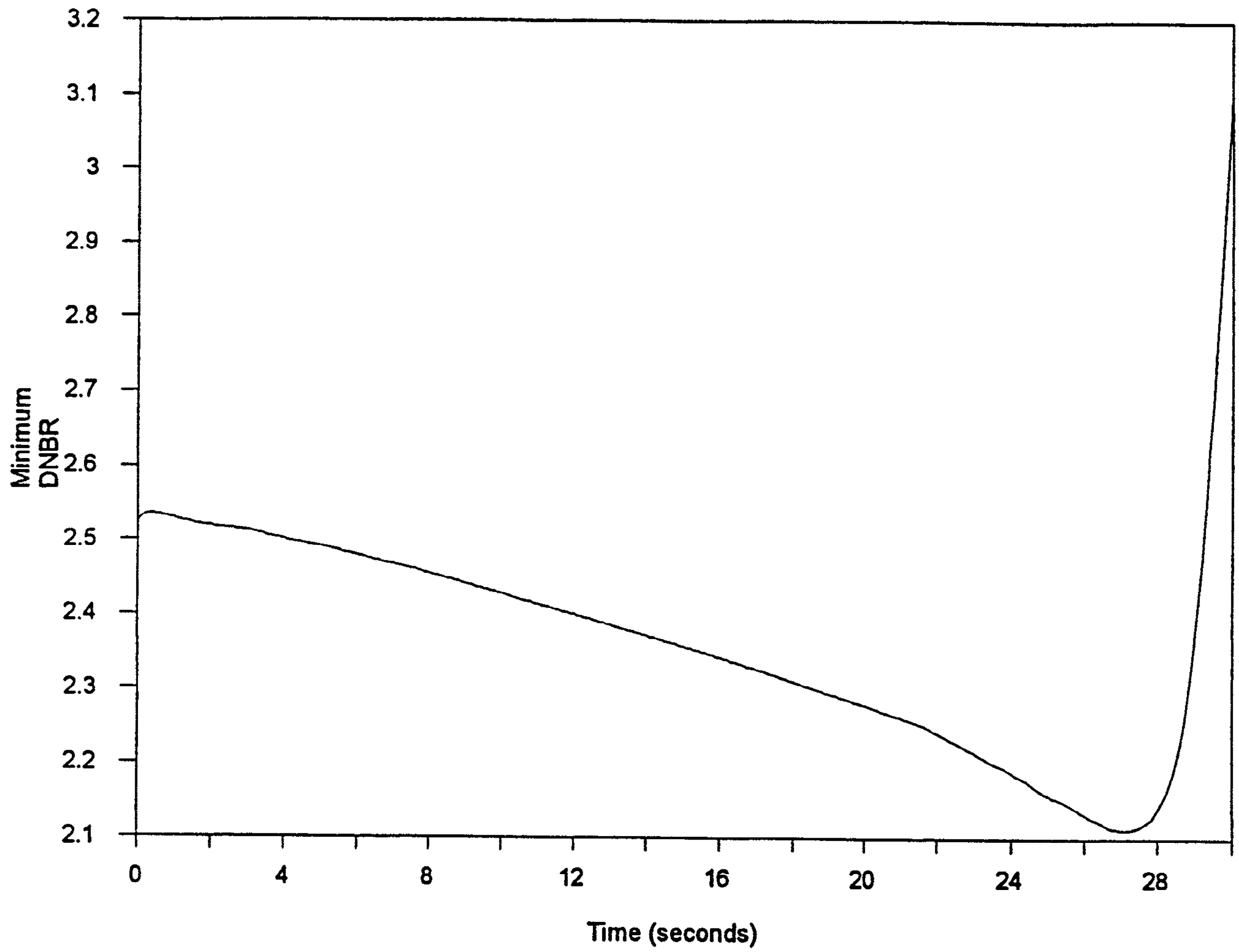
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-1
NUCLEAR POWER, PRESSURIZER PRESSURE AND REACTOR COOLANT AVERAGE TEMPERATURE FOR INADVERTENT OPENING OF A PRESSURIZER SAFETY VALVE



Rev. 12

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FIGURE 15.6-2

DNBR TRANSIENT FOR INADVERTENT OPENING OF A PRESSURIZER SAFETY VALVE

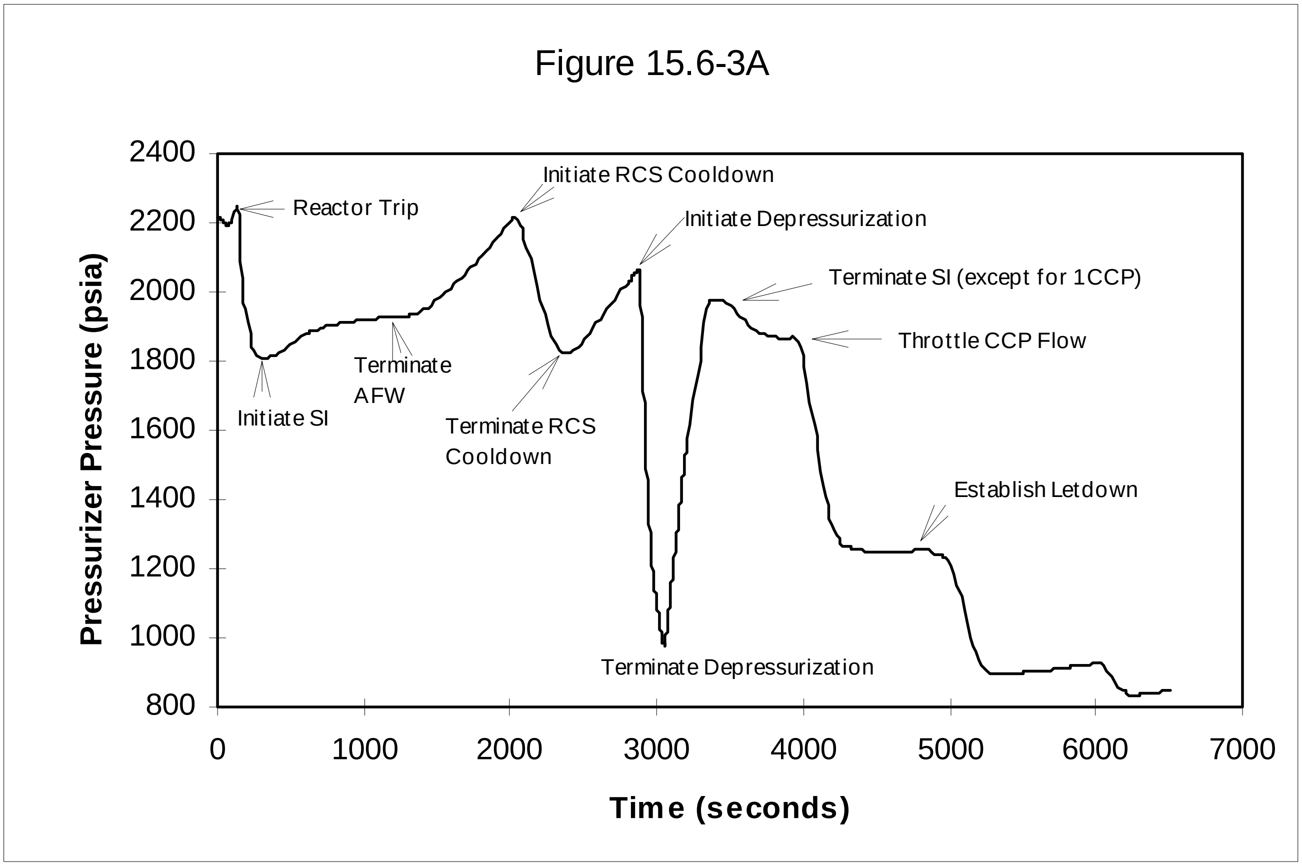

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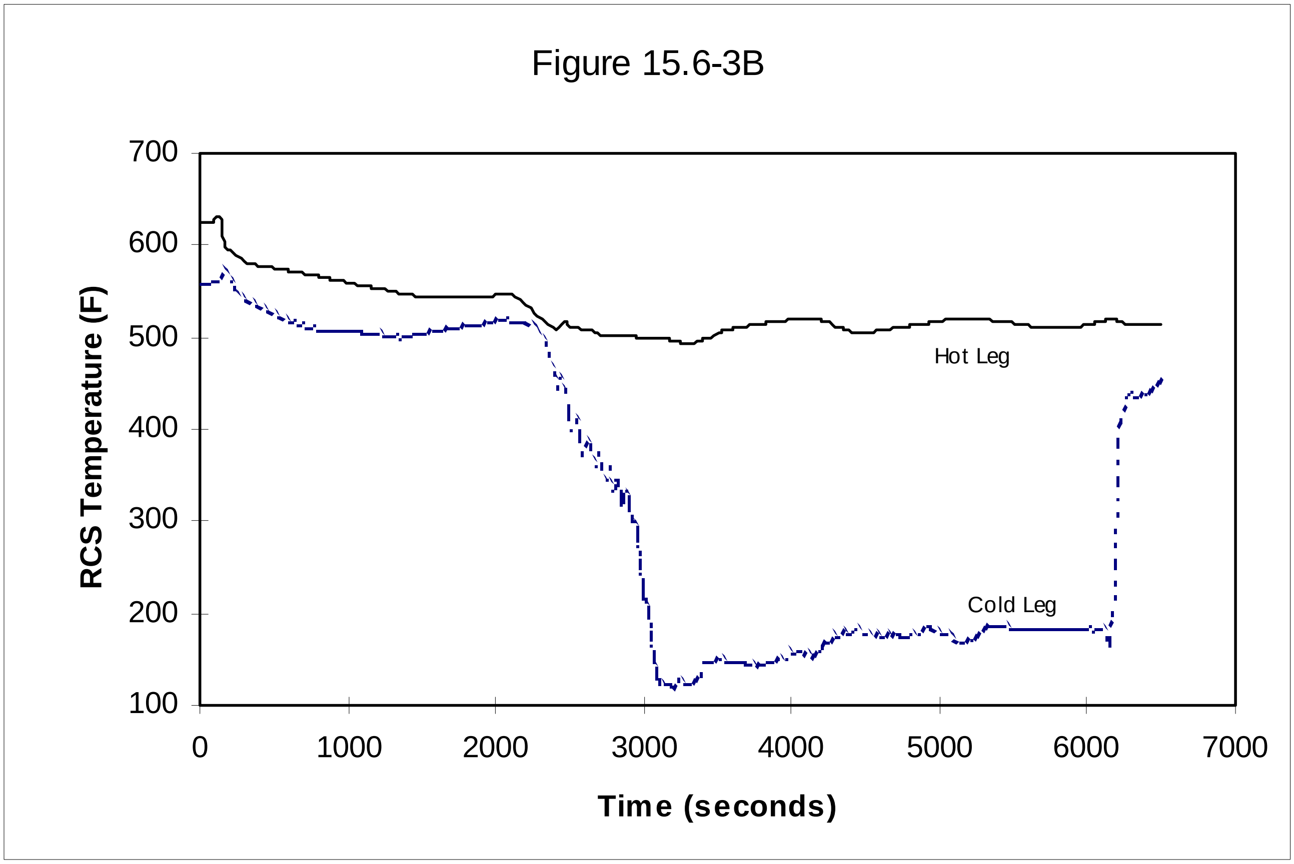
FIGURE 15.6-3a

PRESSURIZER PRESSURE TRANSIENT
SGTR FORCED WITH STUCK-OPEN
STEAM GENERATOR SAFETY VALVE



Rev. 16

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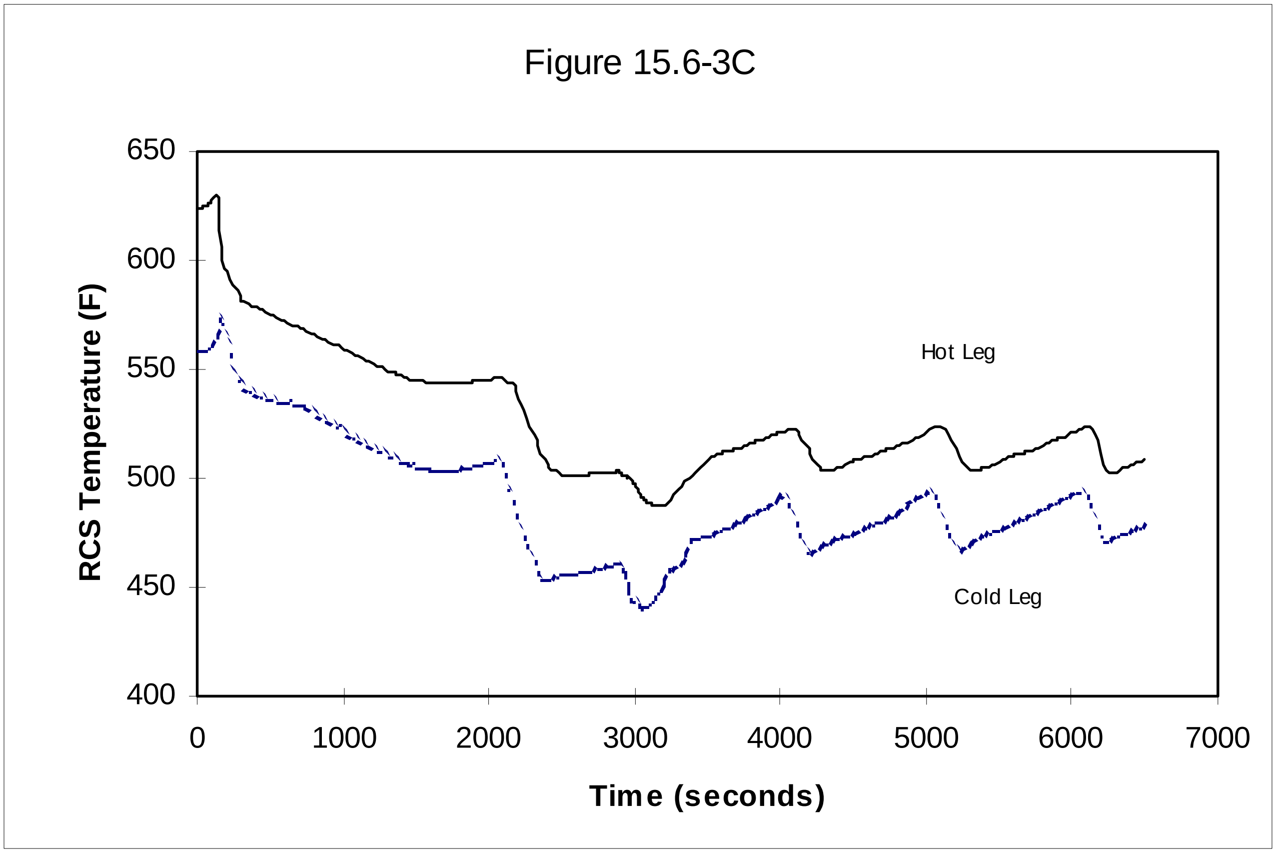
FIGURE 15.6-3b FAULTED LOOP RCS TEMPERATURE TRANSIENTS SGTR FORCED OVERFILL WITH STUCK-OPEN STEAM GENERATOR SAFETY VALVE



Rev. 16

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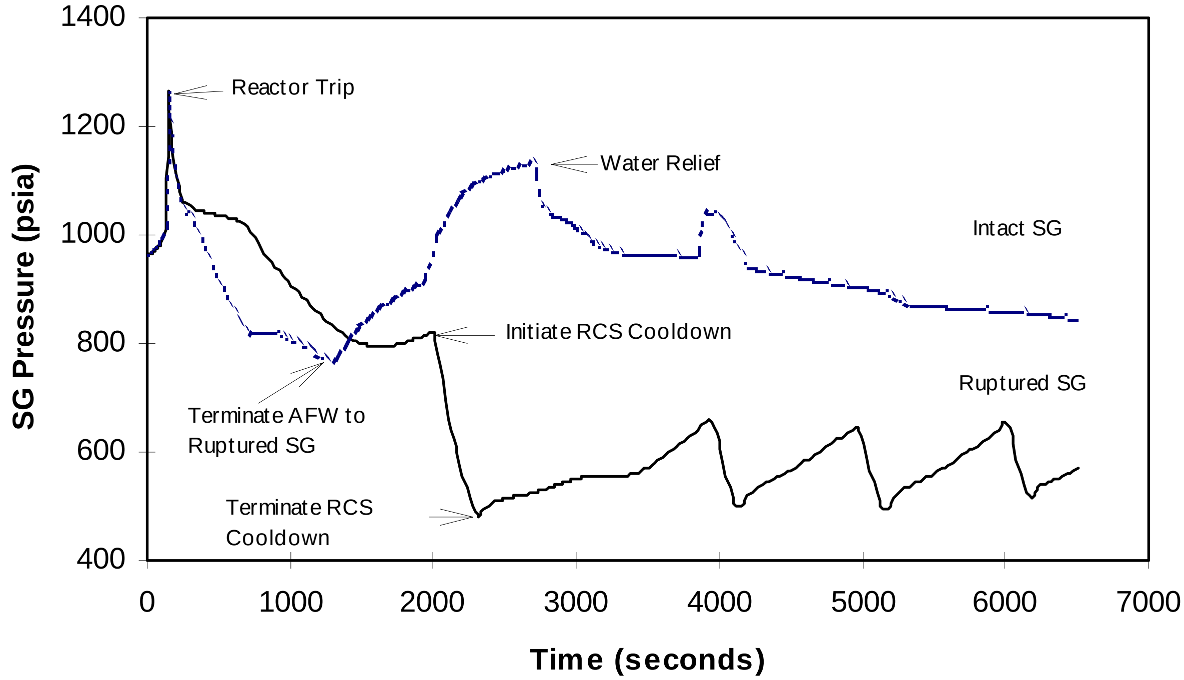
FIGURE 15.6-3c

INTACT LOOP RCS TEMPERATURE TRANSIENTS
SGTR FORCED OVERFILL WITH STUCK-OPEN STEAM GENERATOR SAFETY VALVE

Figure 15.6-3D



Rev. 16

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FIGURE 15.6-3d

STEAM GENERATOR PRESSURE TRANSIENTS
SGTR FORCED OVERFILL WITH STUCK-OPEN STEAM GENERATOR SAFETY VALVE

Figure 15.6-3E

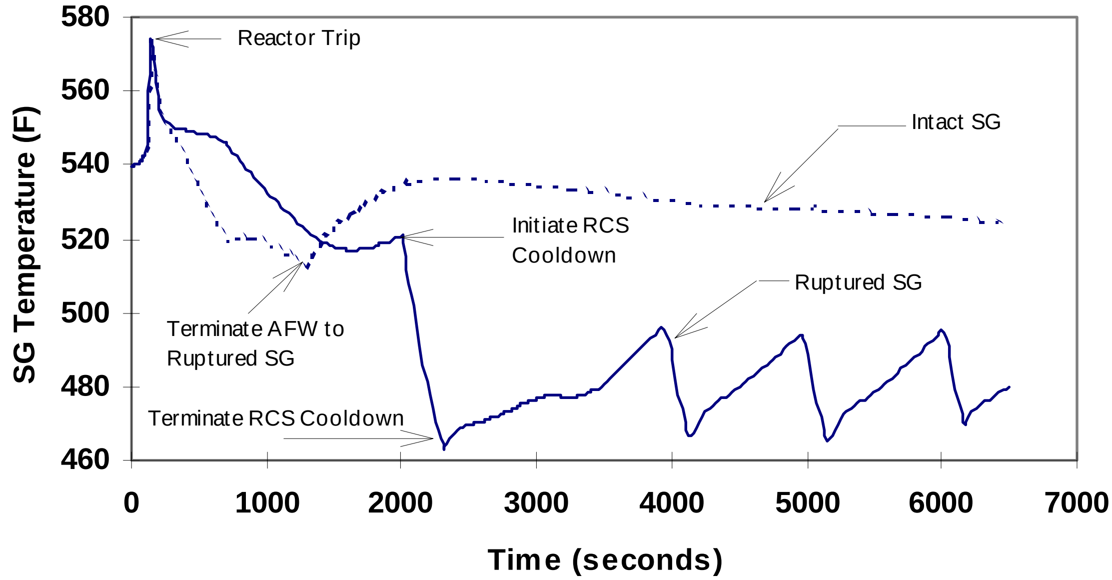

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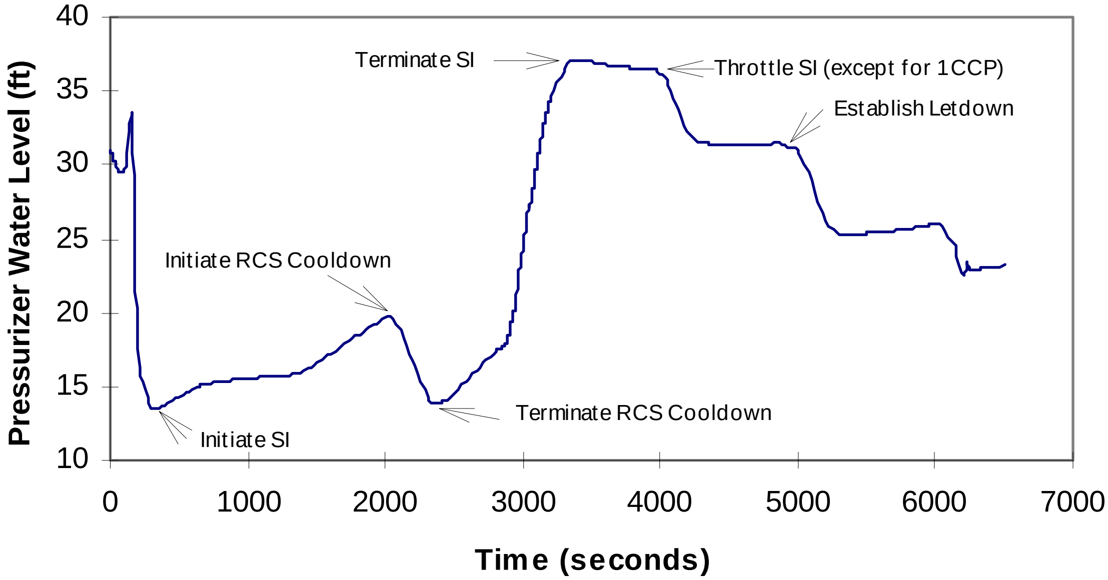
FIGURE 15.6-3e

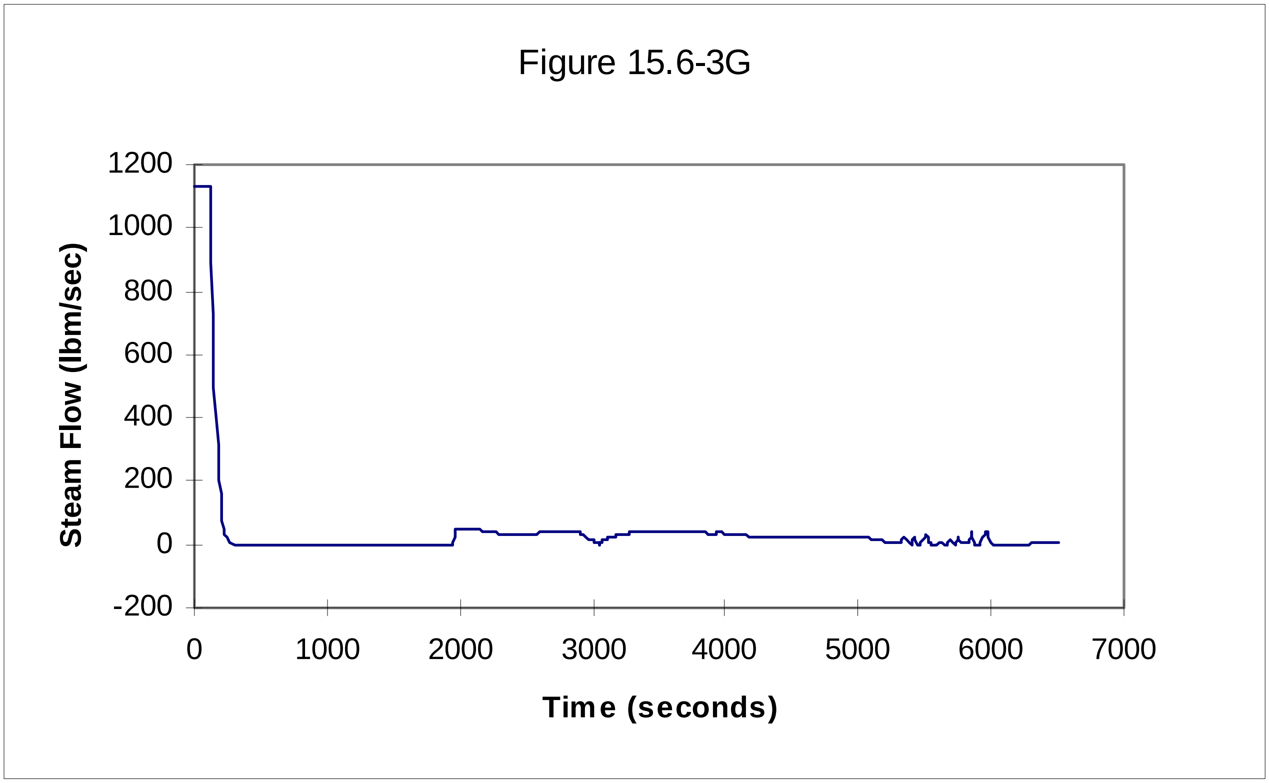
STEAM GENERATOR TEMPERATURE TRANSIENTS
SGTR FORCED OVERFILL WITH STUCK-OPEN STEAM GENERATOR SAFETY VALVE

Figure 15.6-3F



Rev. 16

WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.6-3f PRESSURIZER WATER LEVEL TRANSIENT SGTR FORCED OVERFILL WITH STUCK-OPEN STEAM GENERATOR SAFETY VALVE



Rev. 16

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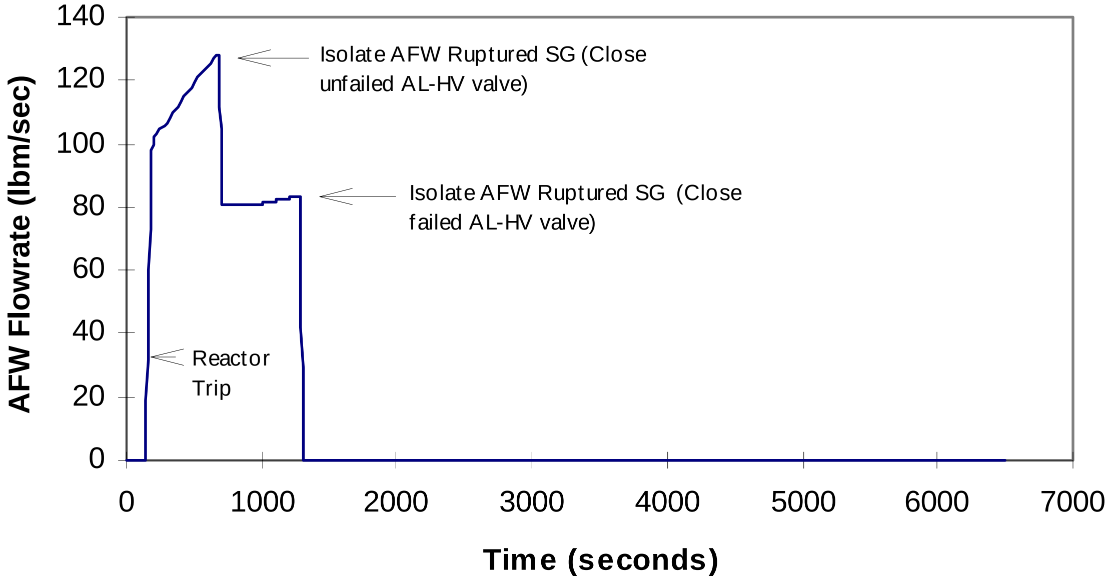
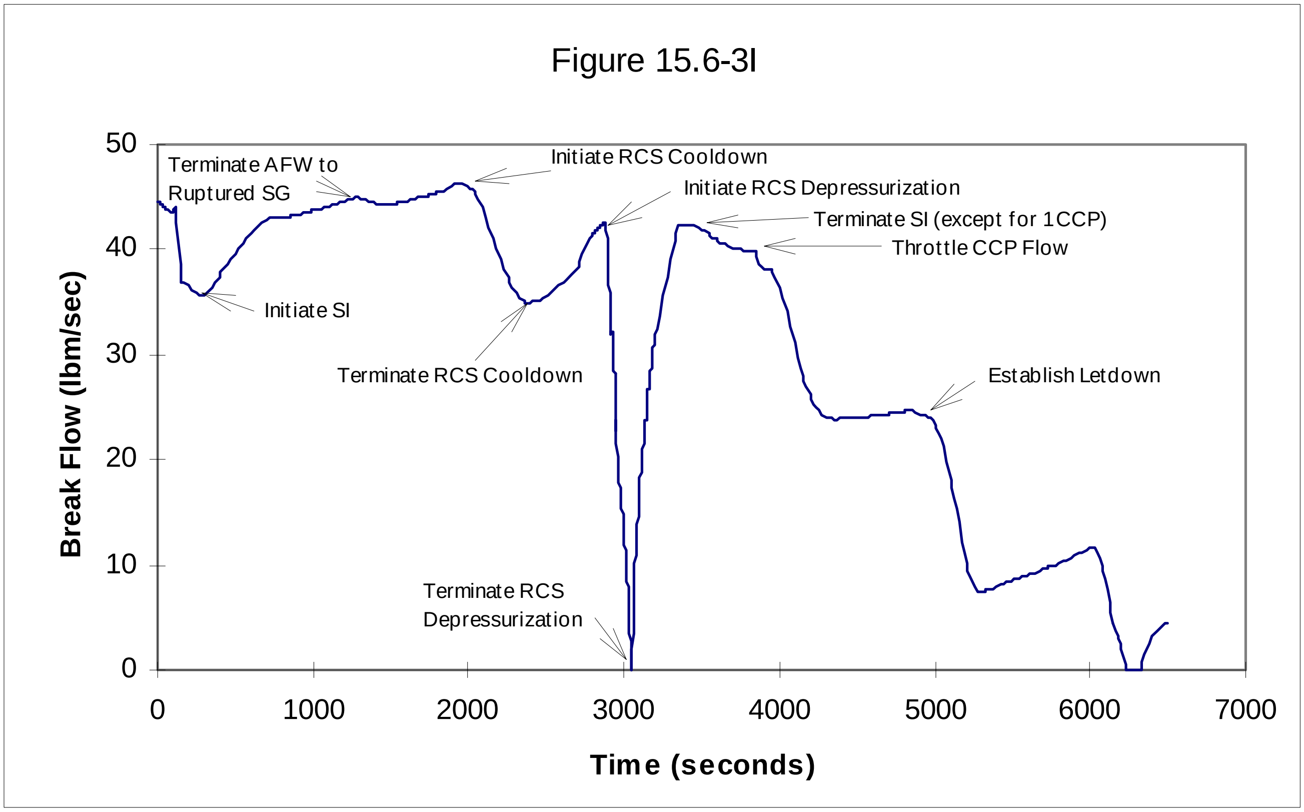
FIGURE 15.6-3g FAULTED STEAM GENERATOR STEAM FLOW TRANSIENT SGTR FORCED OVERFILL WITH STUCK-OPEN STEAM GENERATOR SAFETY VALVE

Figure 15.6-3H



Rev. 16

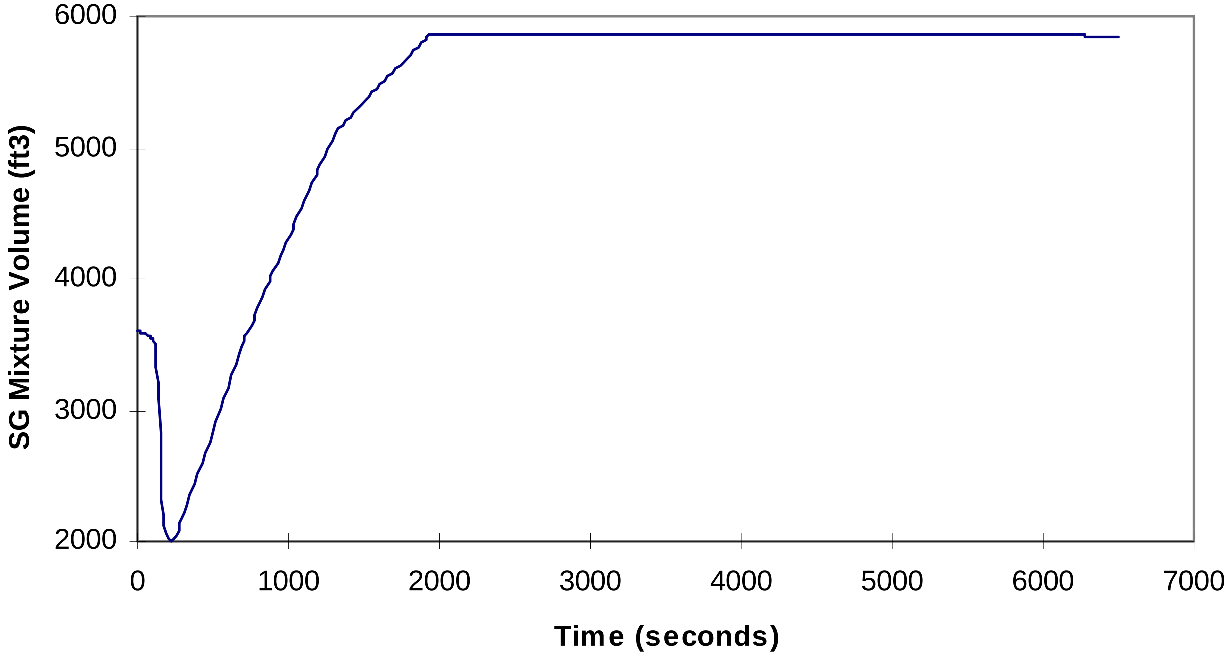
WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.6-3h FAULTED STEAM GENERATOR AFW FLOW TRANSIENT SGTR FORCED OVERFILL WITH STUCK-OPEN STEAM GENERATOR SAFETY VALVE



Rev. 16

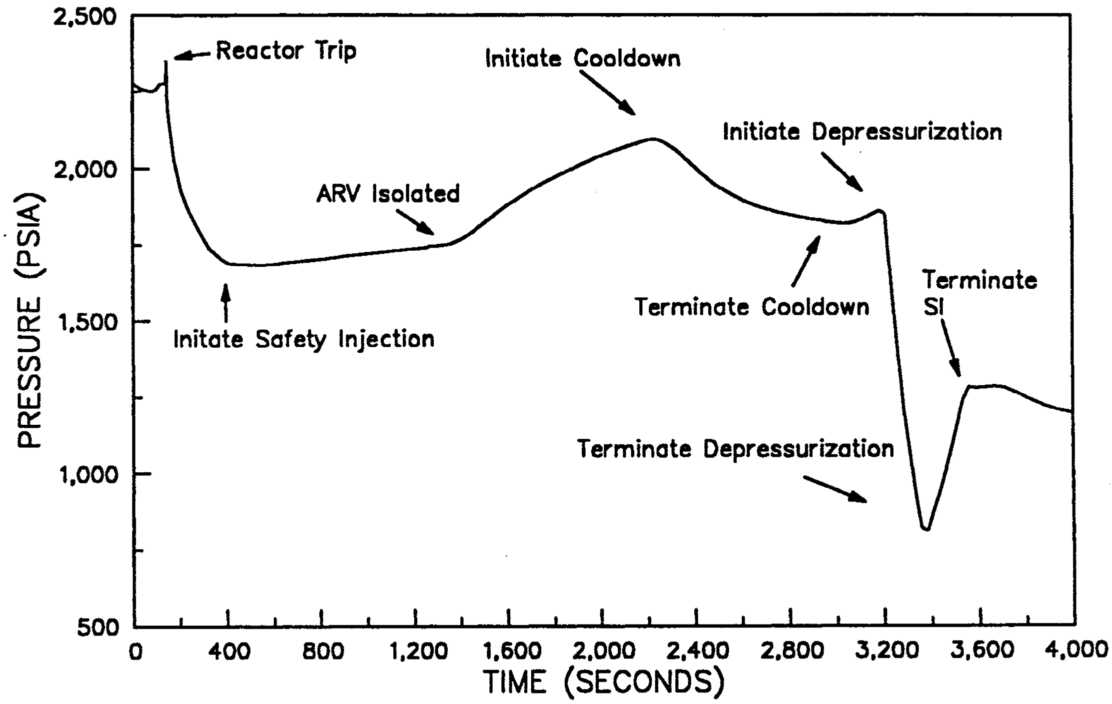
WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.6-3i FAULTED STEAM GENERATOR BREAK FLOW TRANSIENT SGTR FORCED OVERFILL WITH STUCK-OPEN STEAM GENERATOR SAFETY VALVE

Figure 15.6-3J



Rev. 16

WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.6-3j FAULTED STEAM GENERATOR MIXTURE VOLUME TRANSIENT SGTR FORCED OVERFILL WITH STUCK-OPEN STEAM GENERATOR SAFETY VALVE



Rev. 7

WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.6-3k PRESSURIZER PRESSURE TRANSIENT SGTR W/ STUCK-OPEN ATMOSPHERIC RELIEF VALVE

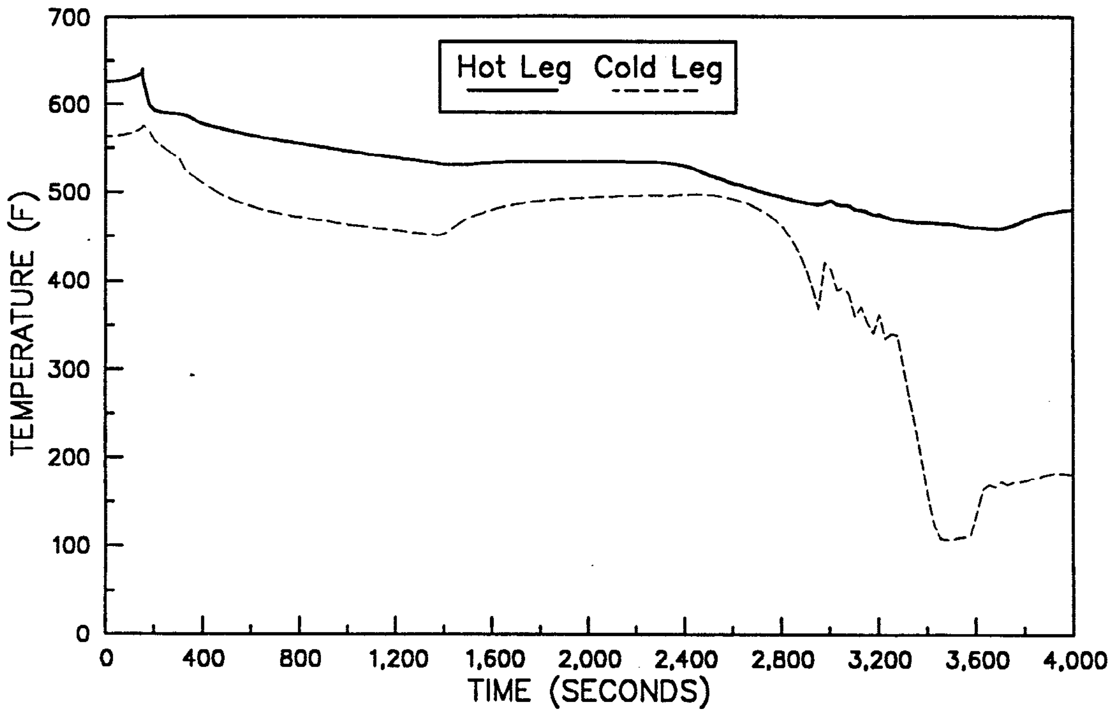


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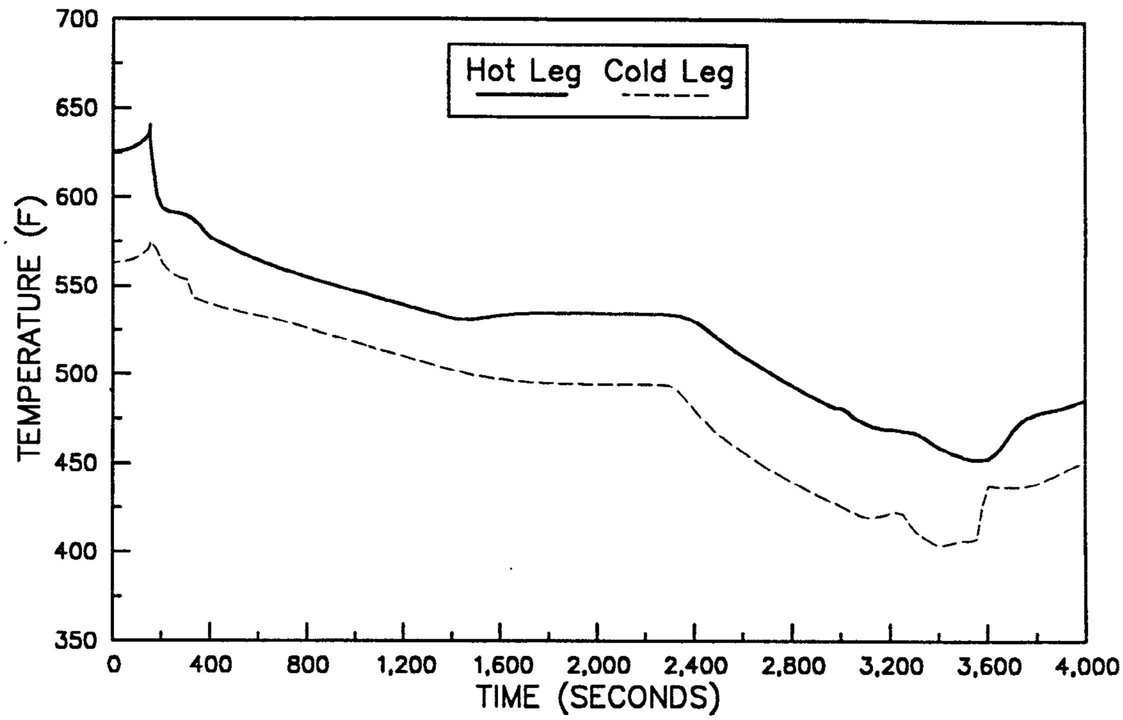
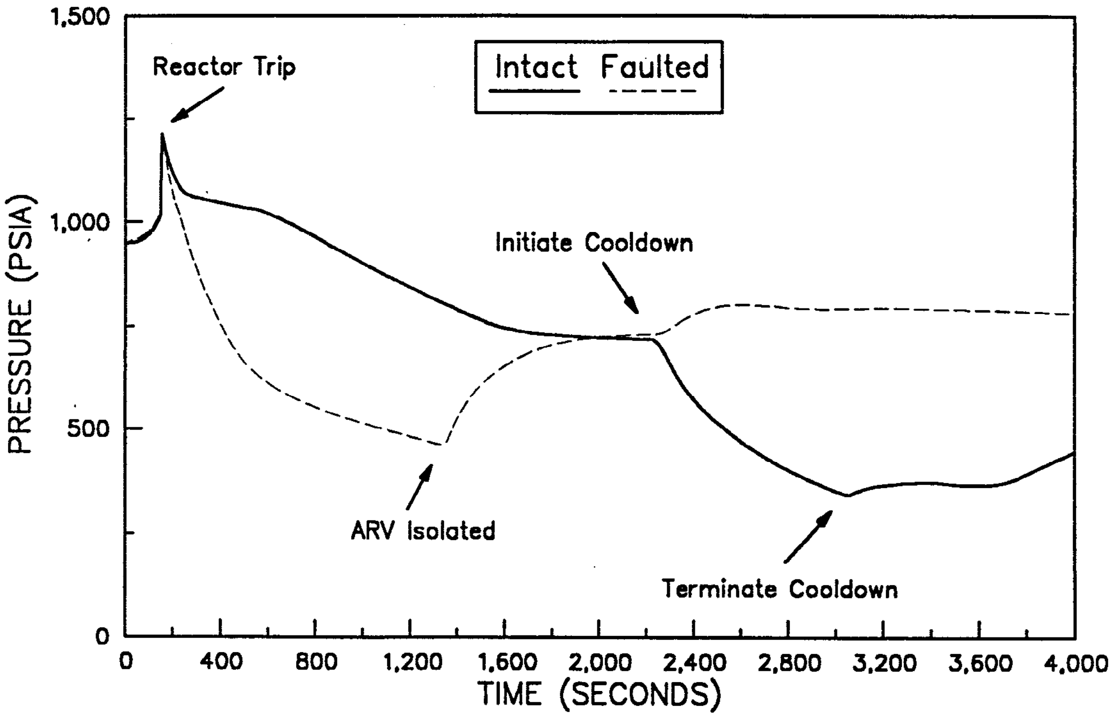
FIGURE 15.6-31 FAULTED LOOP RCS TEMPERATURE SGTR W/ STUCK-OPEN ATMOSPHERIC RELIEF VALVE



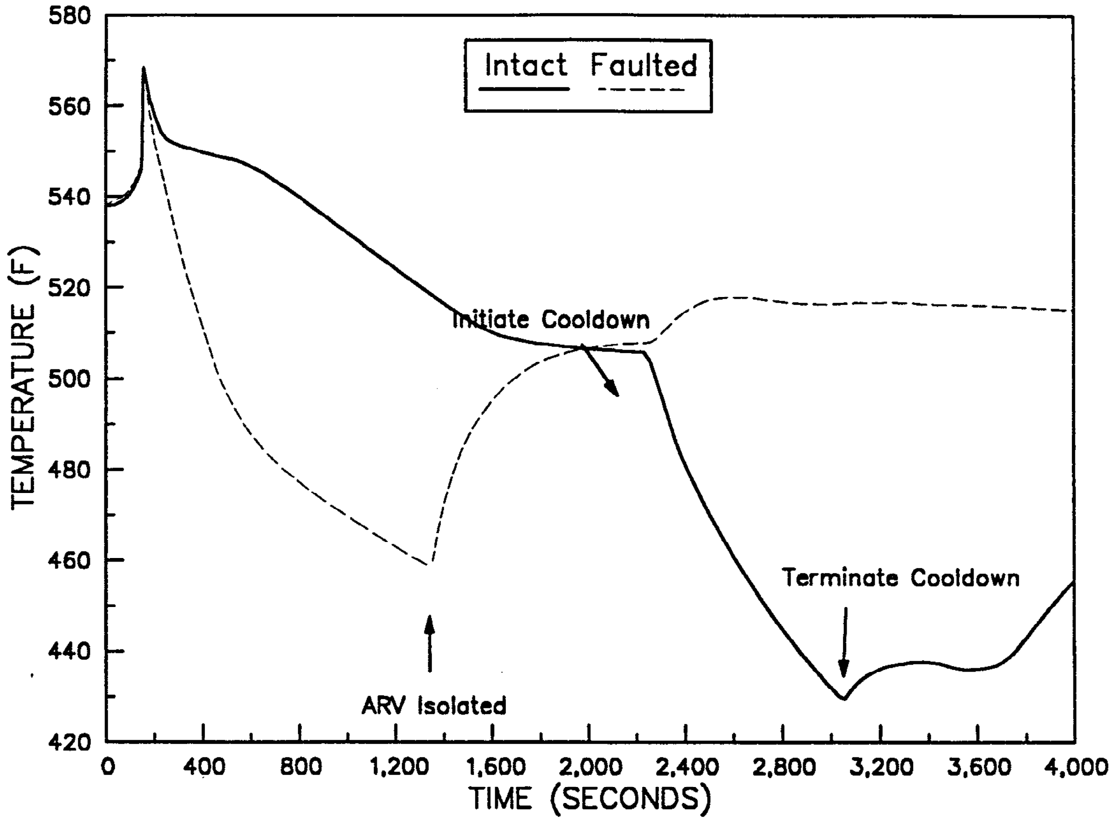
Rev. 7

WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.6-3m INTACT LOOP RCS TEMPERATURE TRANSIENT SGTR W/ STUCK-OPEN ATMOSPHERIC RELIEF VALVE



Rev. 7

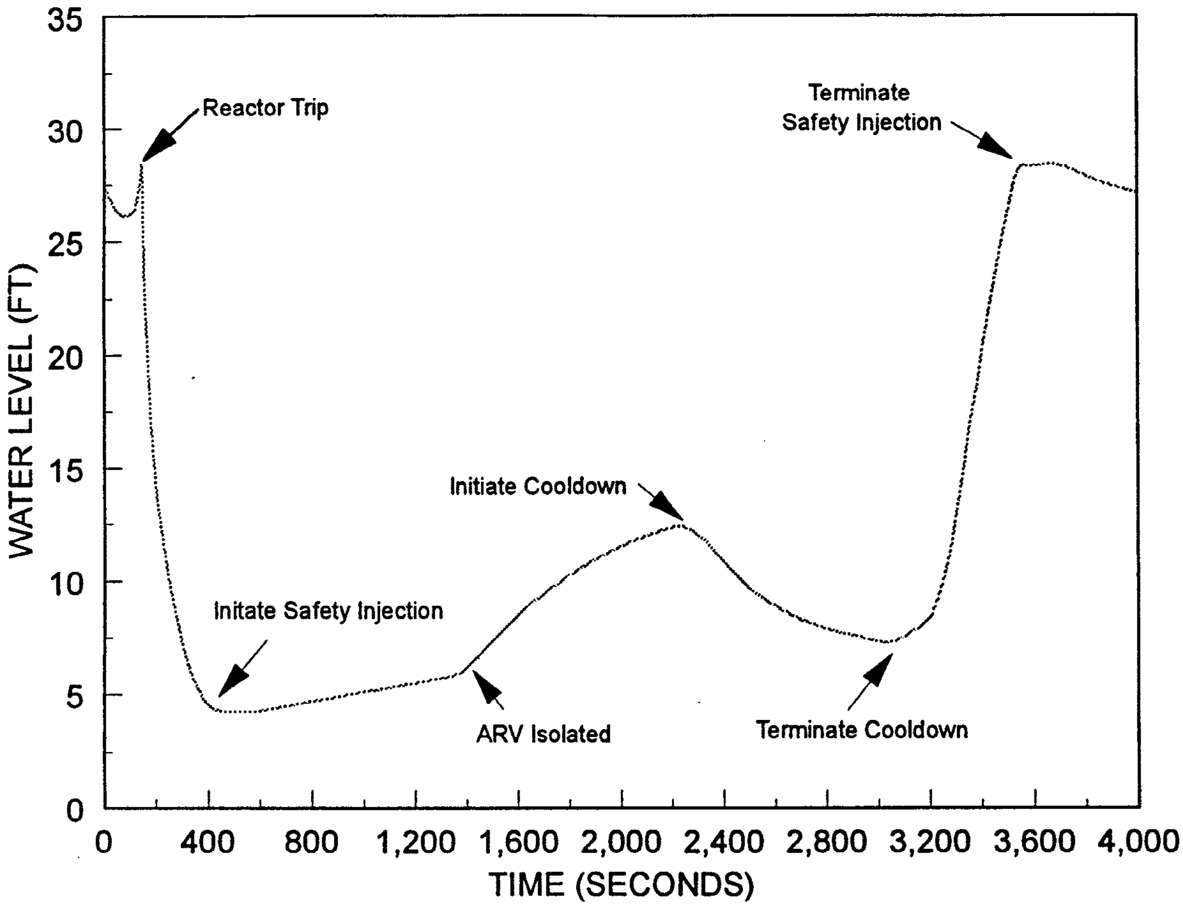
WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.6-3n STEAM GENERATOR PRESSURE TRANSIENT SGTR W/ STUCK-OPEN ATMOSPHERIC RELIEF VALVE



Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-3o
STEAM GENERATOR TEMPERATURE
TRANSIENT
SGTR W/ STUCK-OPEN
ATMOSPHERIC RELIEF VALVE



Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-3p
PRESSURIZER WATER LEVEL TRANSIENT
SGTR W/ STUCK-OPEN
ATMOSPHERIC RELIEF VALVE

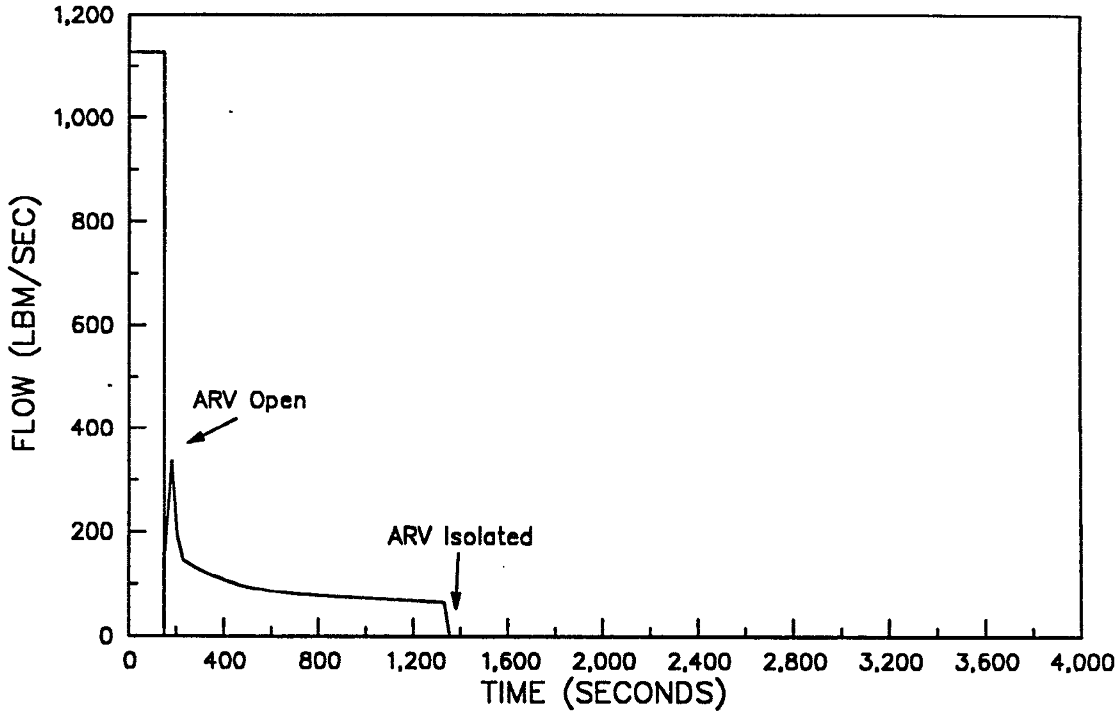


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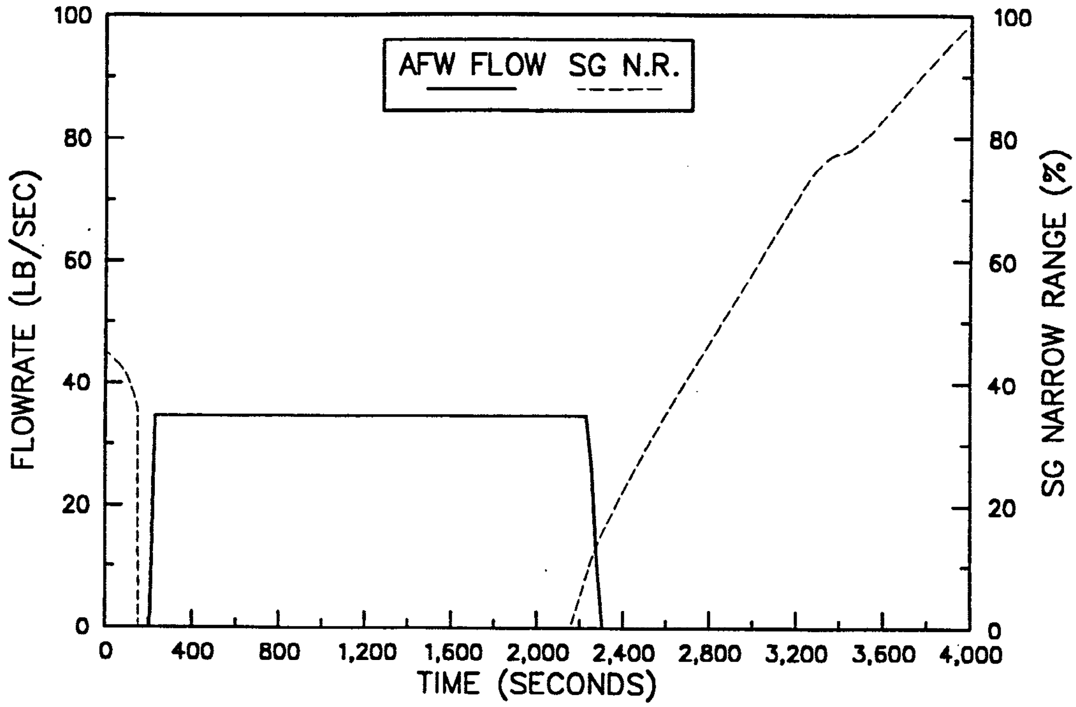
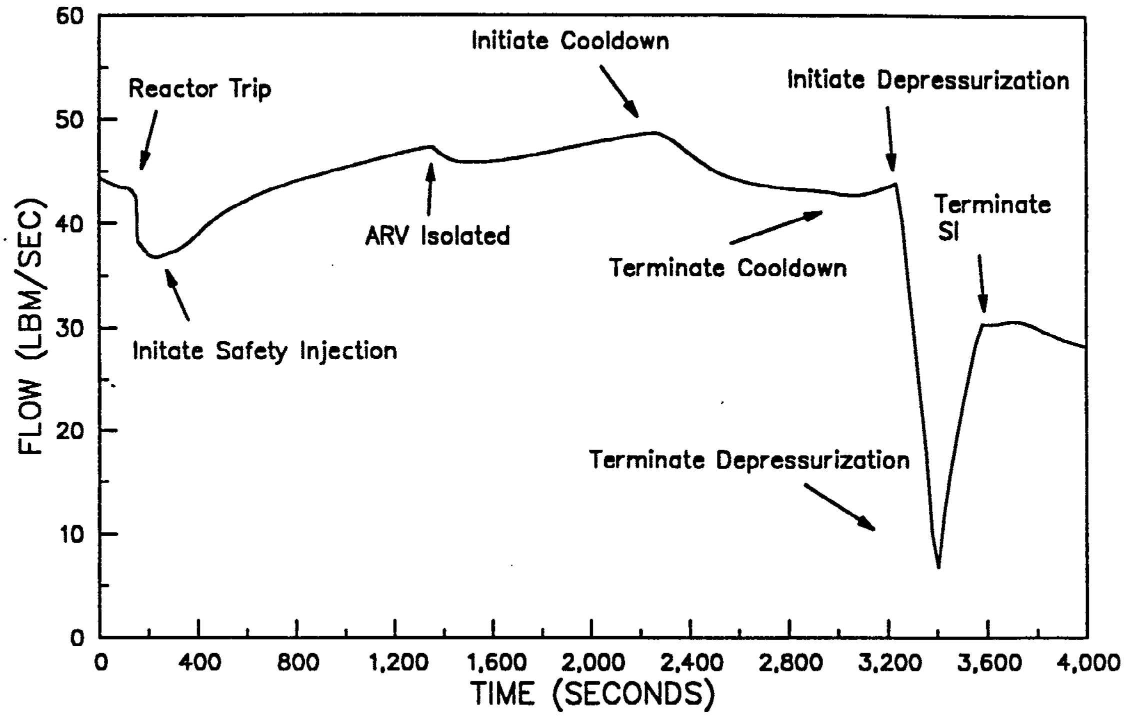
FIGURE 15.6-3q FAULTED STEAM GENERATOR STEAM FLOW TRANSIENT SGTR W/ STUCK-OPEN ATMOSPHERIC RELIEF VALVE



Rev. 7

WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.6-3r FAULTED STEAM GENERATOR AFW AND NR LEVEL TRANSIENT SGTR W/ STUCK-OPEN ATMOSPHERIC RELIEF VALVE



Rev. 7

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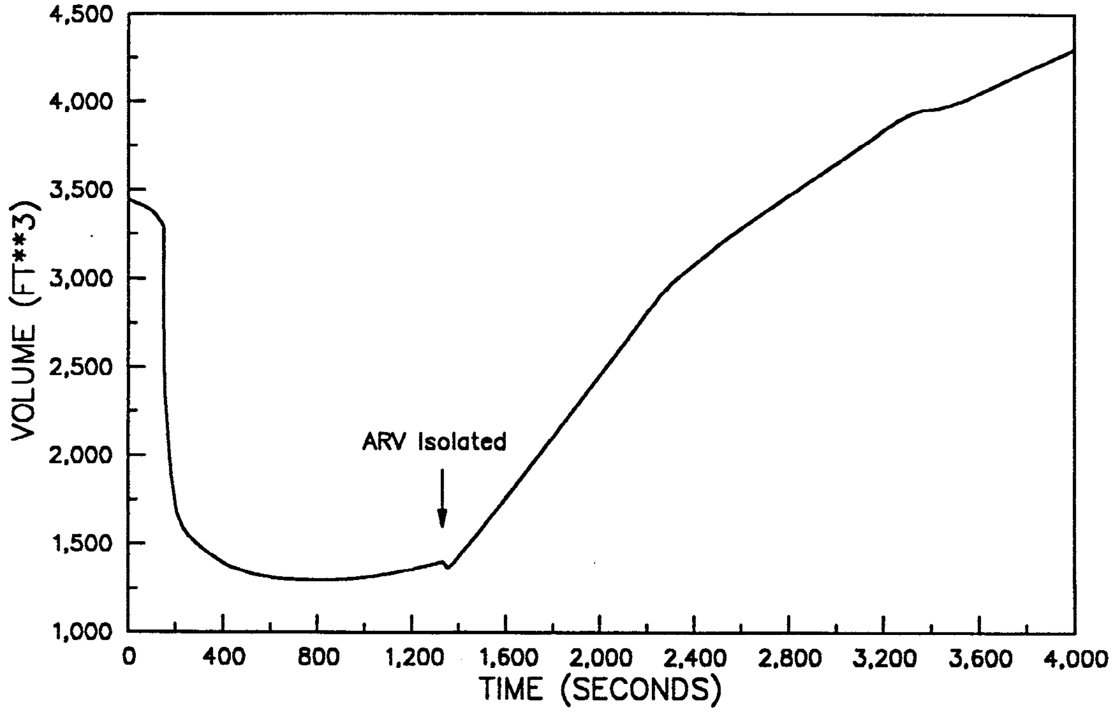
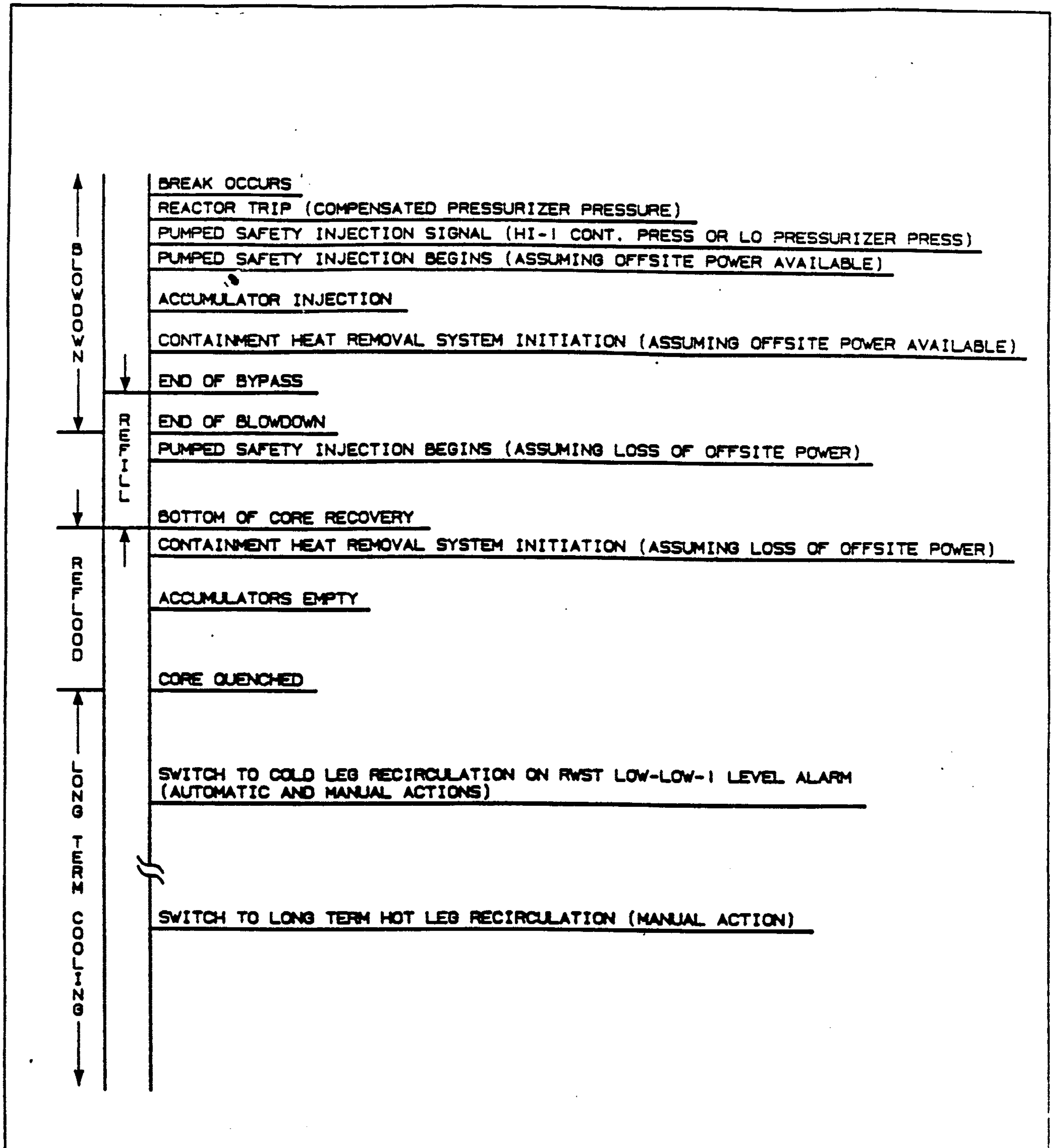
FIGURE 15.6-3s FAULTED STEAM GENERATOR TOTAL BREAK FLOW TRANSIENT SGTR W/ STUCK-OPEN ATMOSPHERIC RELIEF VALVE



Rev. 7

WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.6-3t FAULTED STEAM GENERATOR MIXTURE VOLUME TRANSIENT SGTR W/ STUCK-OPEN ATMOSPHERIC RELIEF VALVE



Rev. 7

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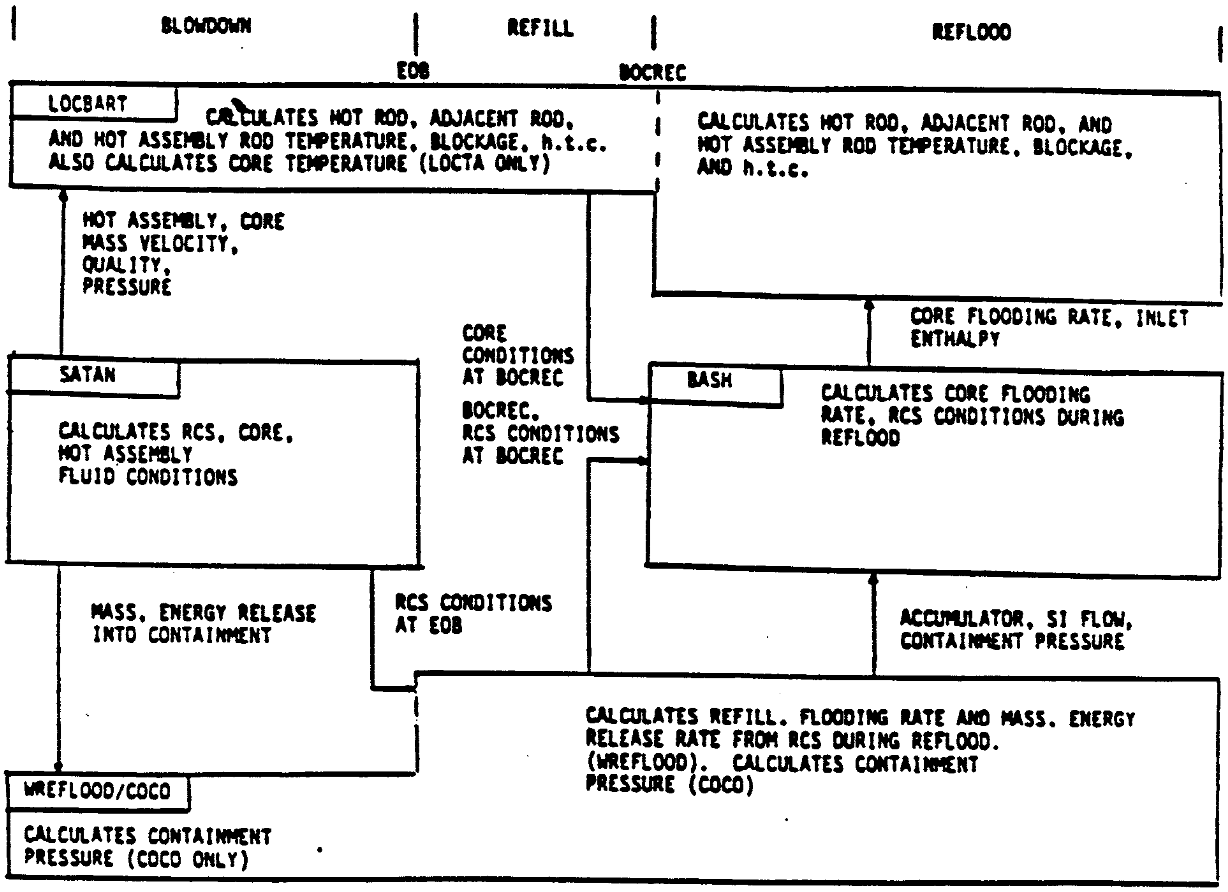
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-4
SEQUENCE OF EVENTS FOR LARGE
BREAK LOCA ANALYSIS



1981 Model with BASH

Rev. 7

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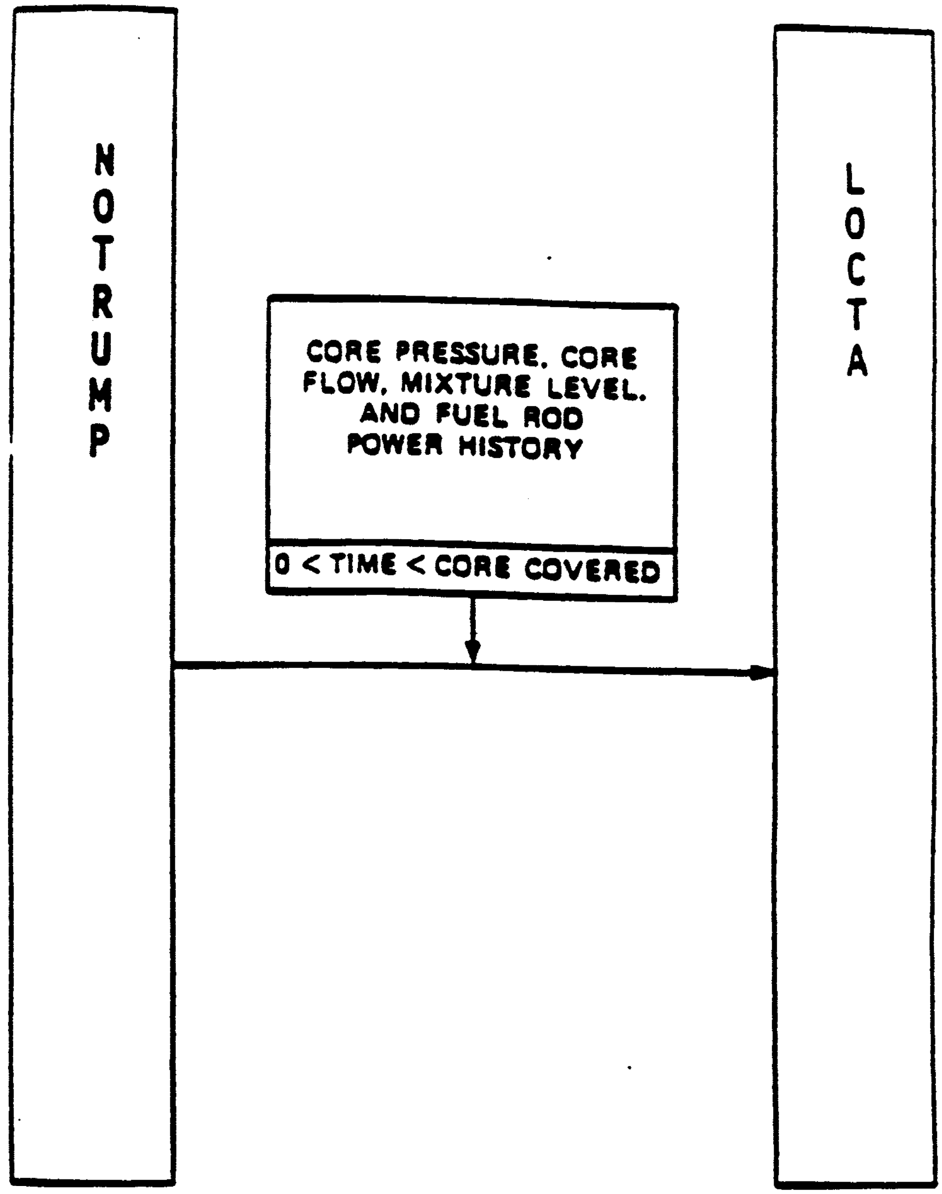
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-5

CODE INTERFACE DESCRIPTION FOR LARGE BREAK MODEL



Rev. 7

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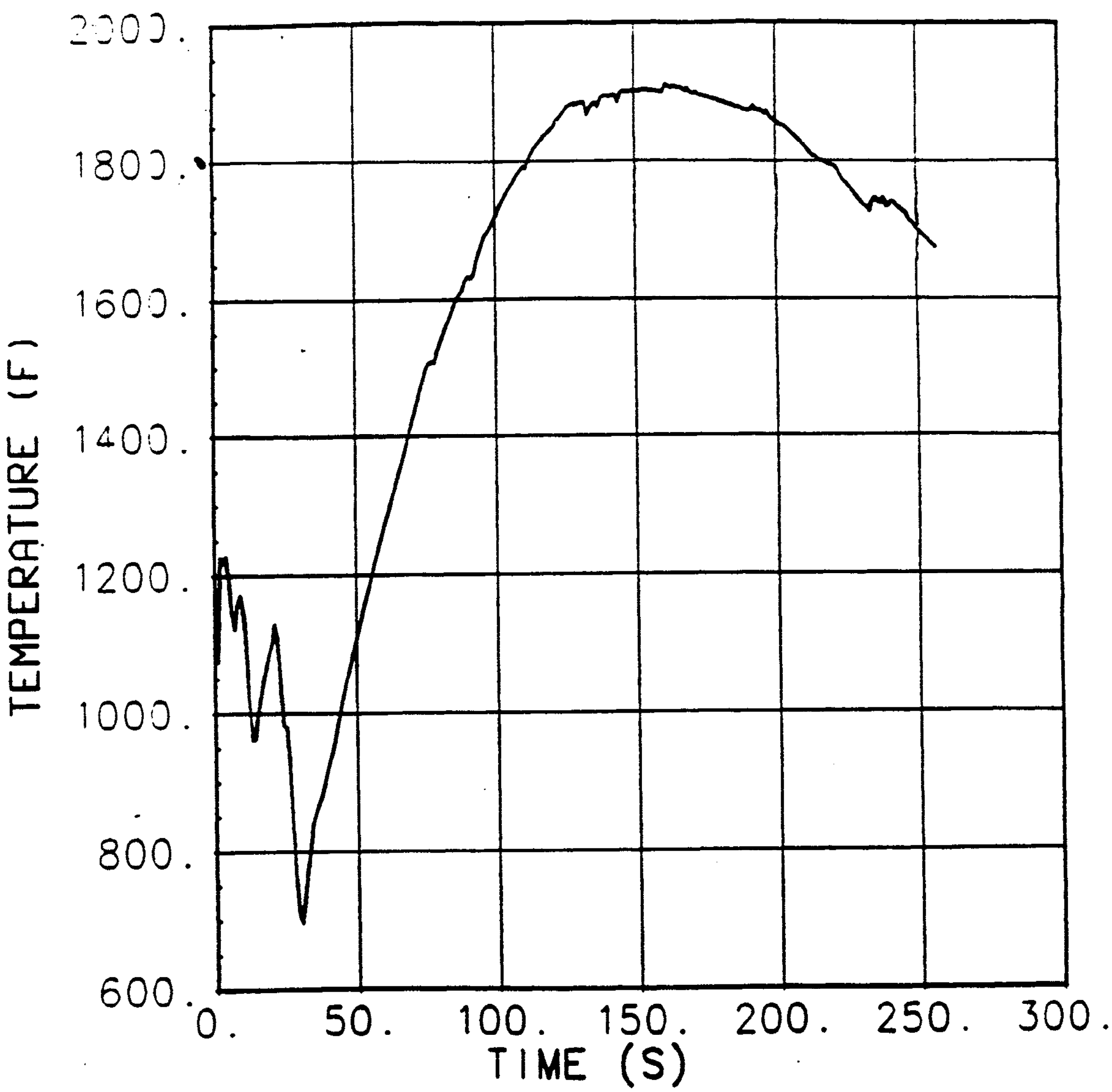
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-6
CODE INTERFACE DESCRIPTION FOR SMALL BREAK MODEL



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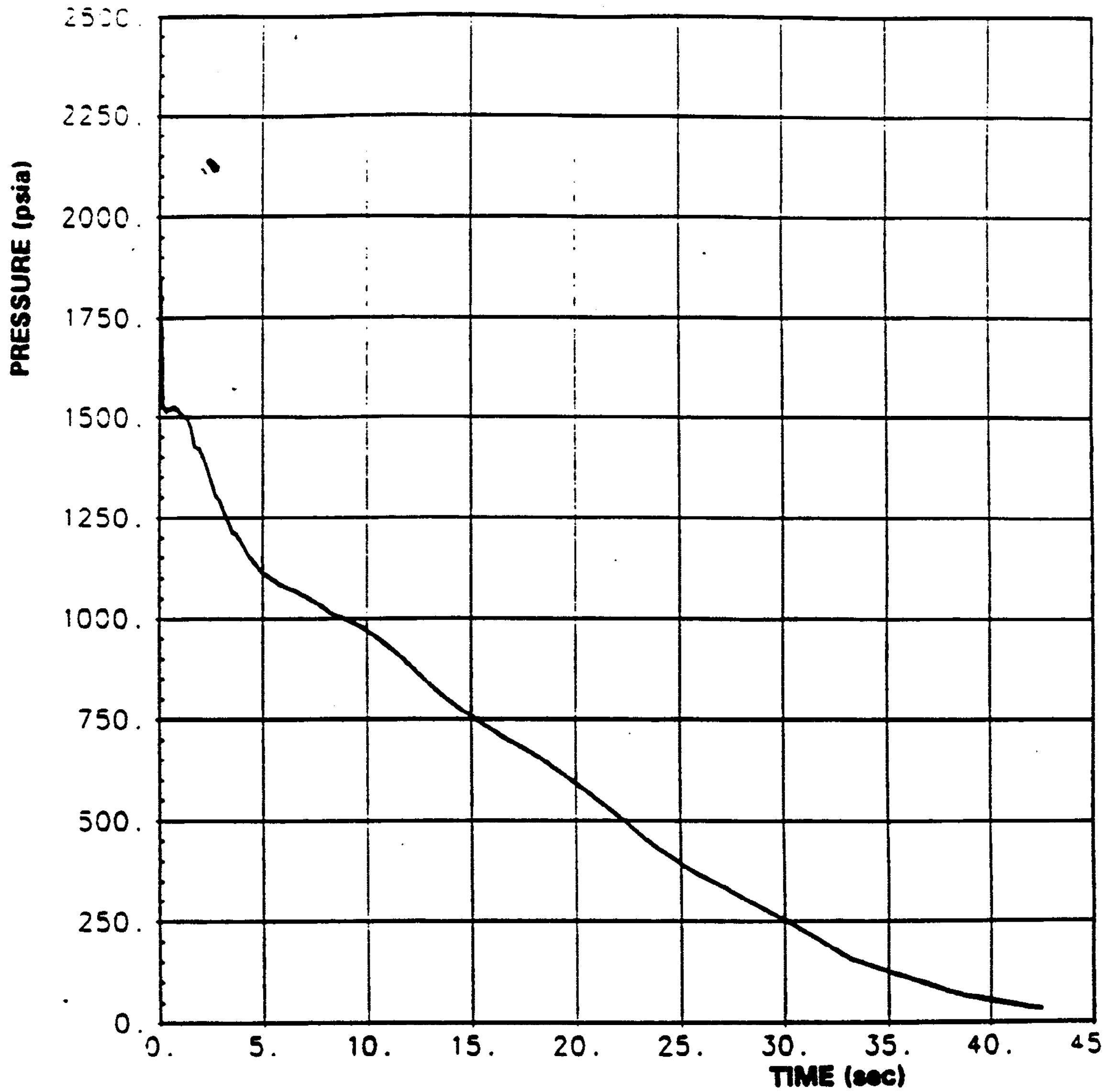
WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-7
CLAD TEMPERATURE TRANSIENT
HOT SPOT, REDUCED T_{AVG}
DECLG(C_D =0.4), MIN SAFEGUARDS



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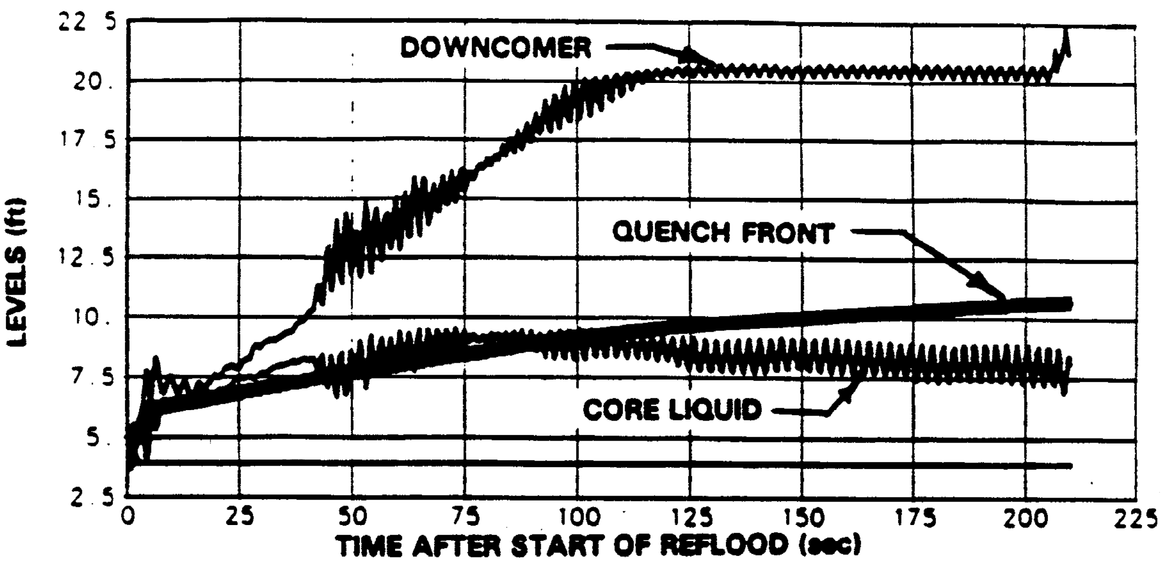
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-8
CORE PRESSURE
REDUCED T_{AVG}
DECLG(C_D = 0.4), MIN SAFEGUARDS



Rev. 7

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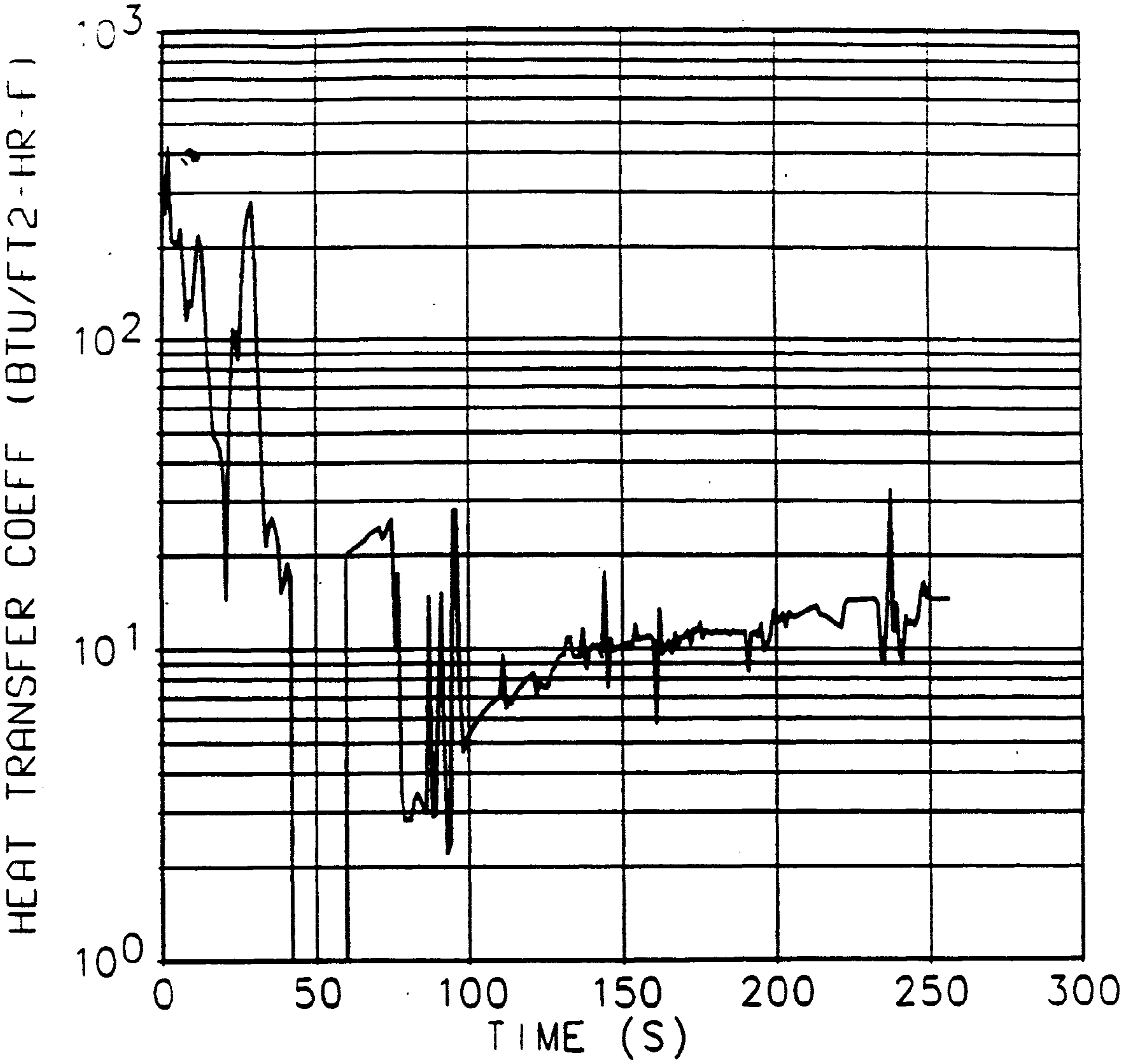
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-9

REFLOOD MIXTURE LEVELS
REDUCED T_{AVG}
DECLG(C_D = 0.4), MIN SAFEGUARDS



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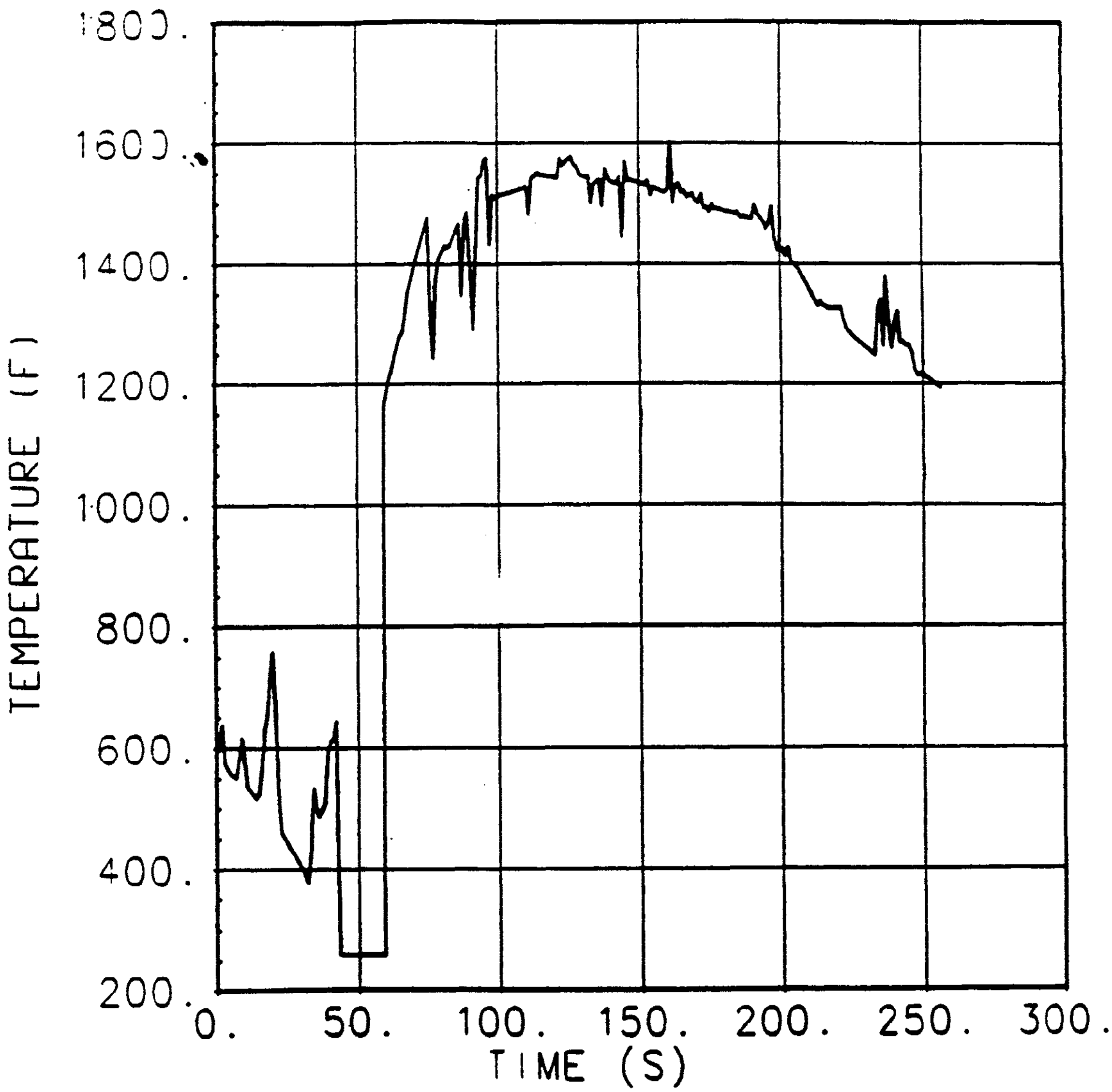
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-10

HEAT TRANSFER COEFFICENT
HOT SPOT, REDUCED T_{AVG}
DECLG(C_D =0.4), MIN SAFEGUARDS



Rev. 7

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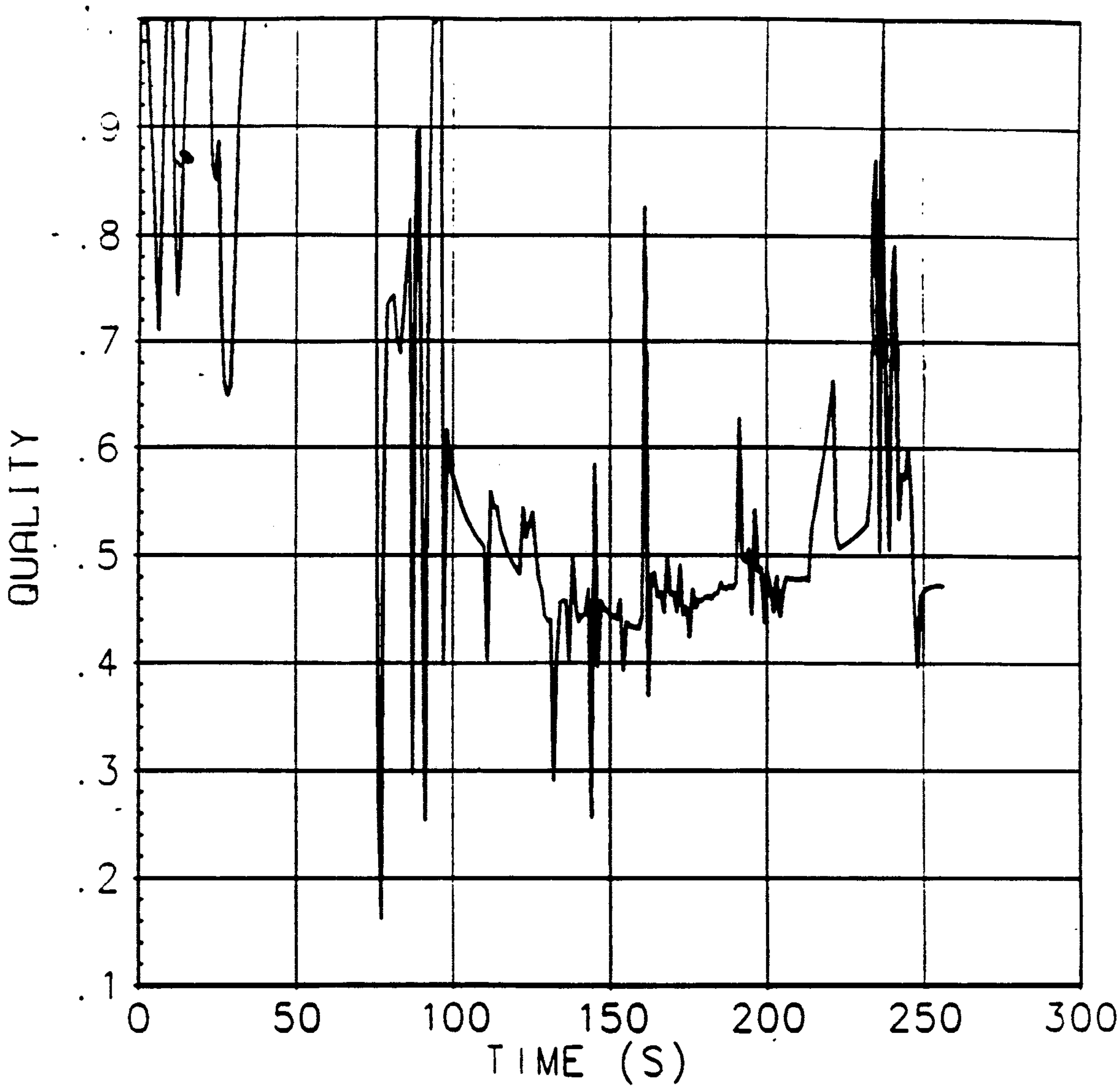
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-11

FLUID TEMPERATURE @ HOT SPOT

REDUCED T_{Avg}

DECLG(C_D =0.4), MIN SAFEGUARDS



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UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-12A

FLUID QUALITY @ HOT SPOT

REDUCED T_{AVG}

DECLG(C_D =0.4), MIN SAFEGUARDS

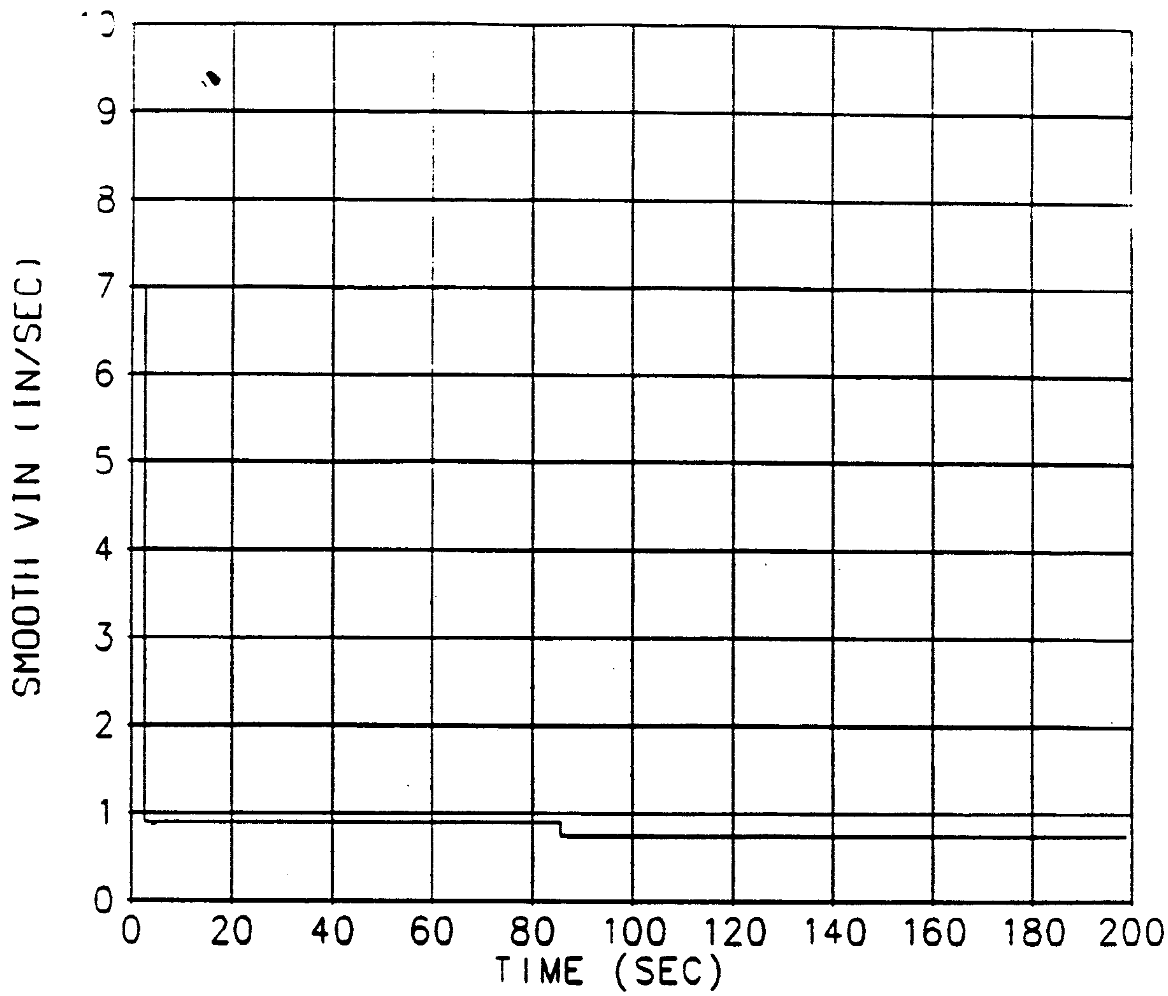

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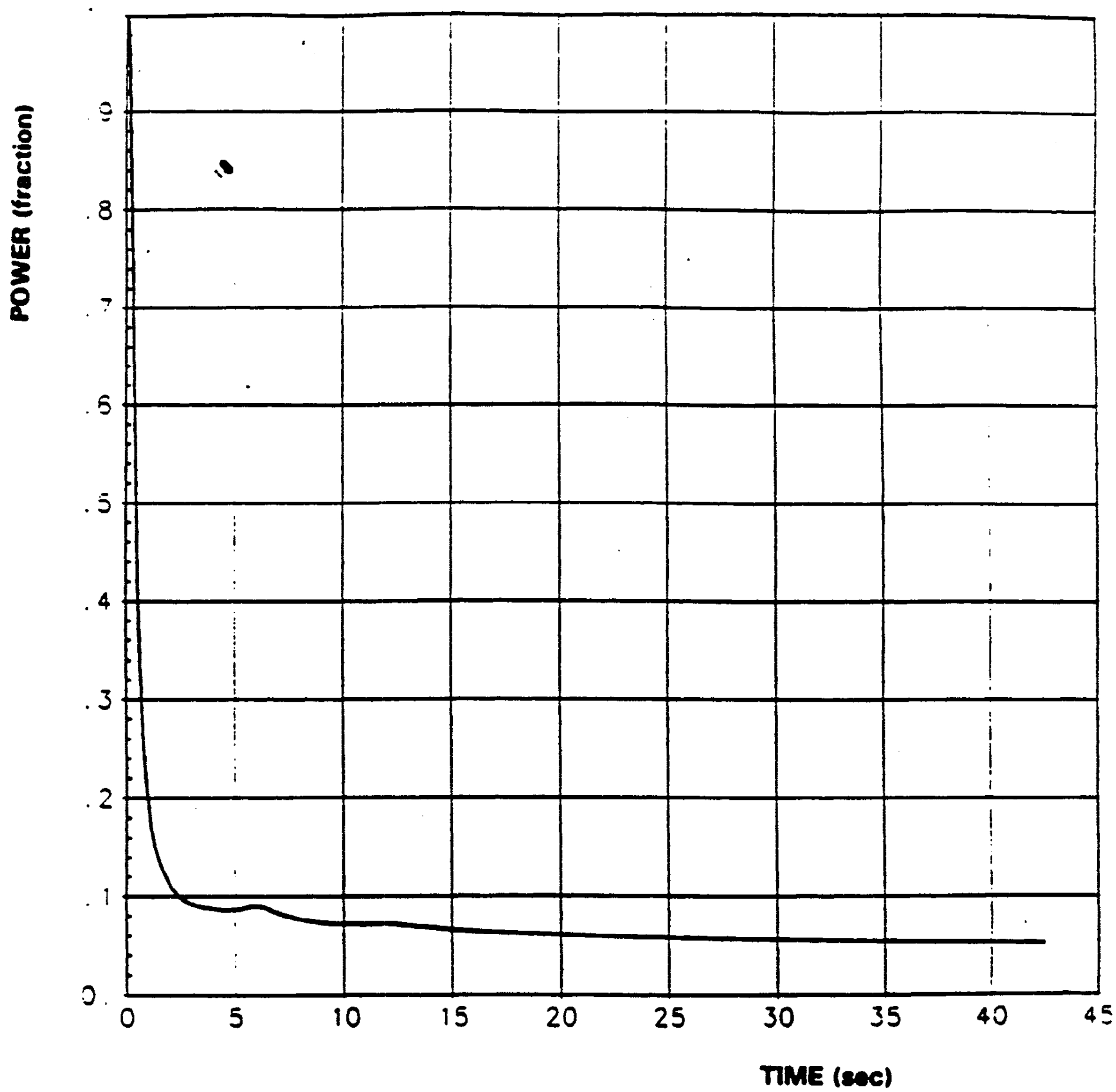
FIGURE 15.6-12B

CORE INLET FLOW VELOCITY

REDUCED T_{AVG}

DECLG(C_D = 0.4), MIN SAFEGUARDS



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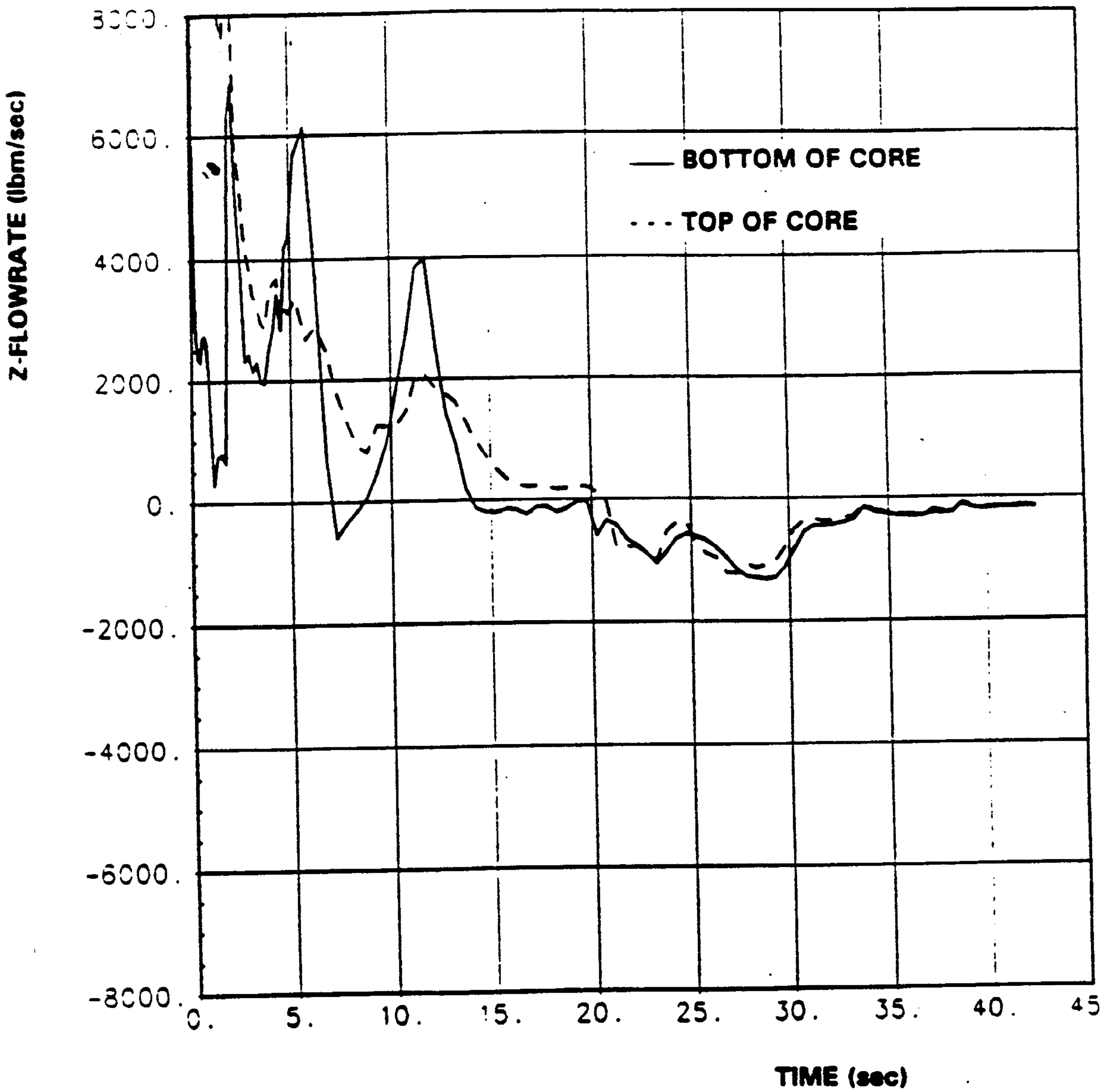
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-12C
CORE POWER TRANSIENT
REDUCED T_{AVG}
DECLG(C_D = 0.4), MIN SAFEGUARDS



Rev. 7

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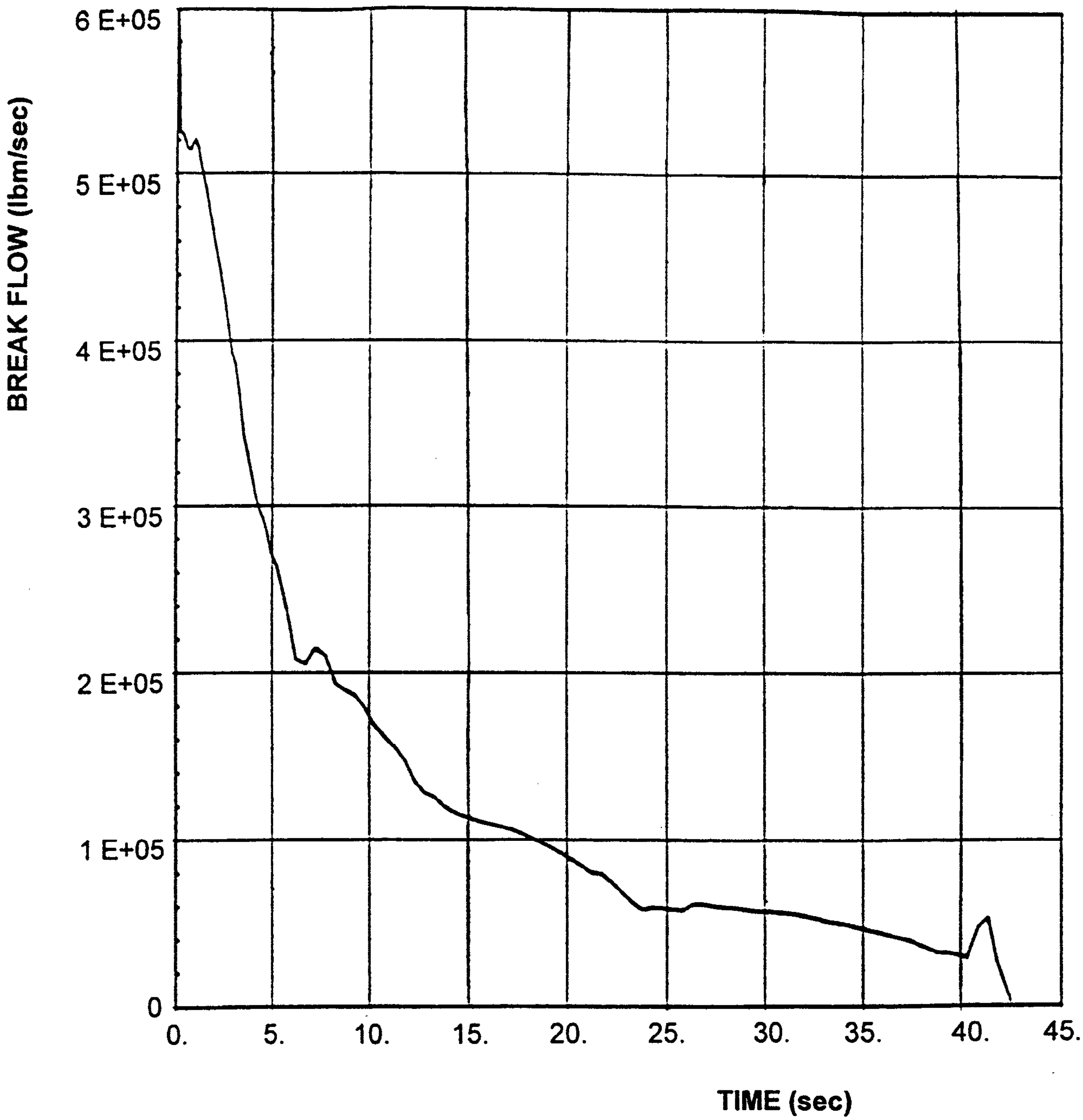
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-12D
CORE FLOWRATE
REDUCED T_{AVG}
DECLG(C_D = 0.4), MIN SAFEGUARDS



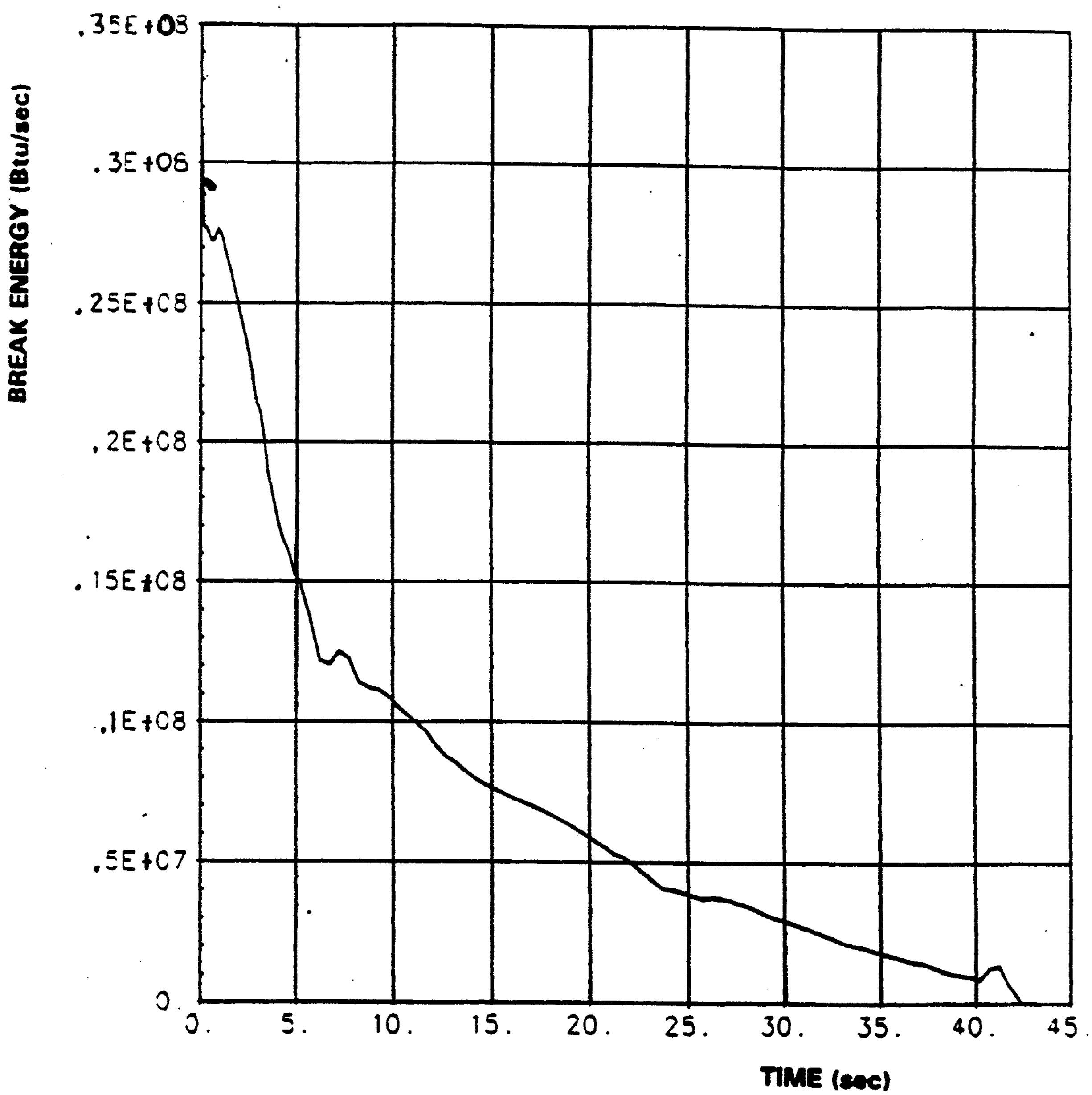
Rev. 12

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

Figure 15.6-12E

BREAK MASS FLOW RERATE
REDUCED T_{AVG}

DECLG(C_D = 0.4), MIN SAFEGUARDS



Rev. 7

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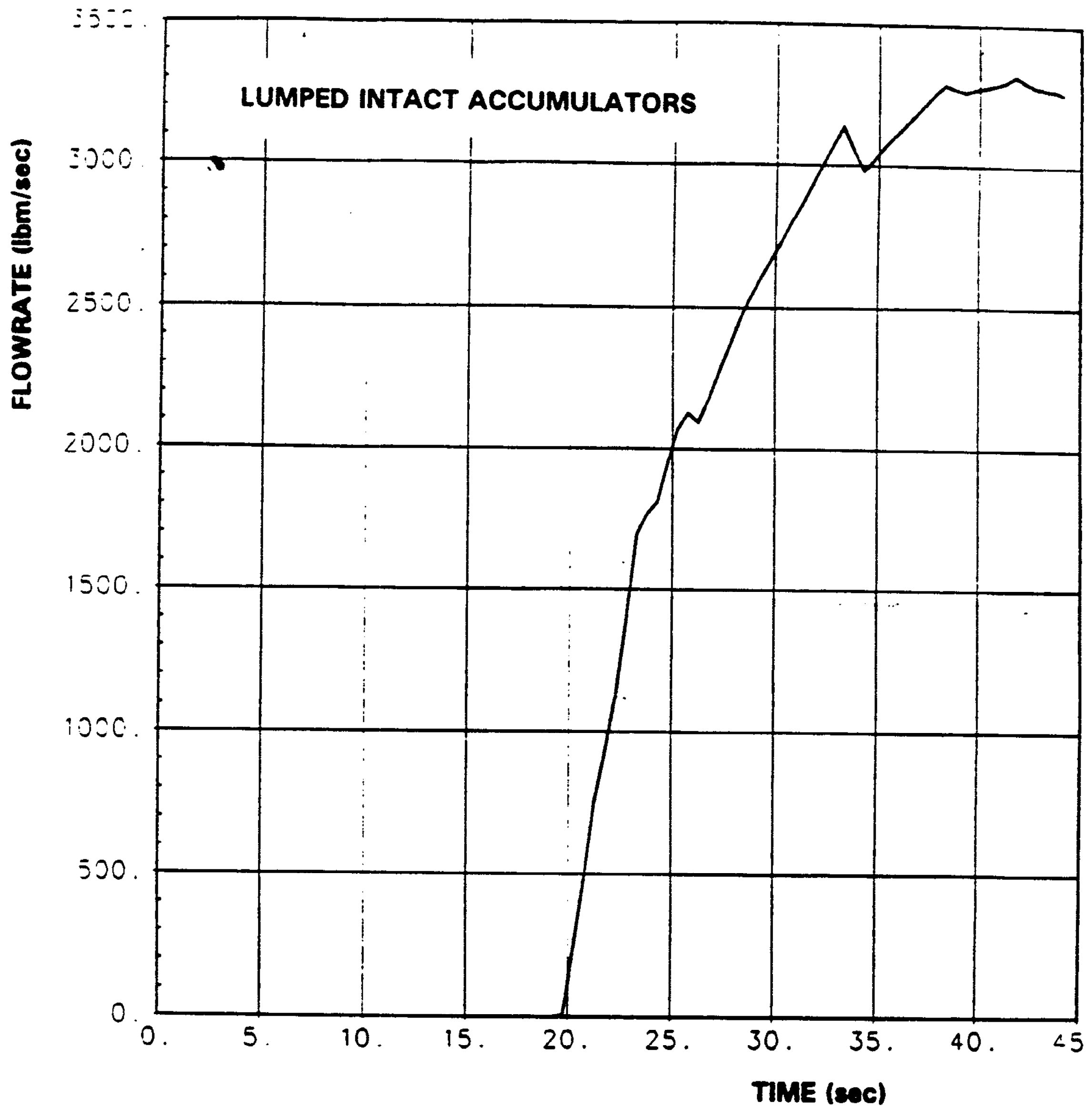
FIGURE 15.6-12F

BREAK ENERGY FLOW RATE

REDUCED T_{AVG}

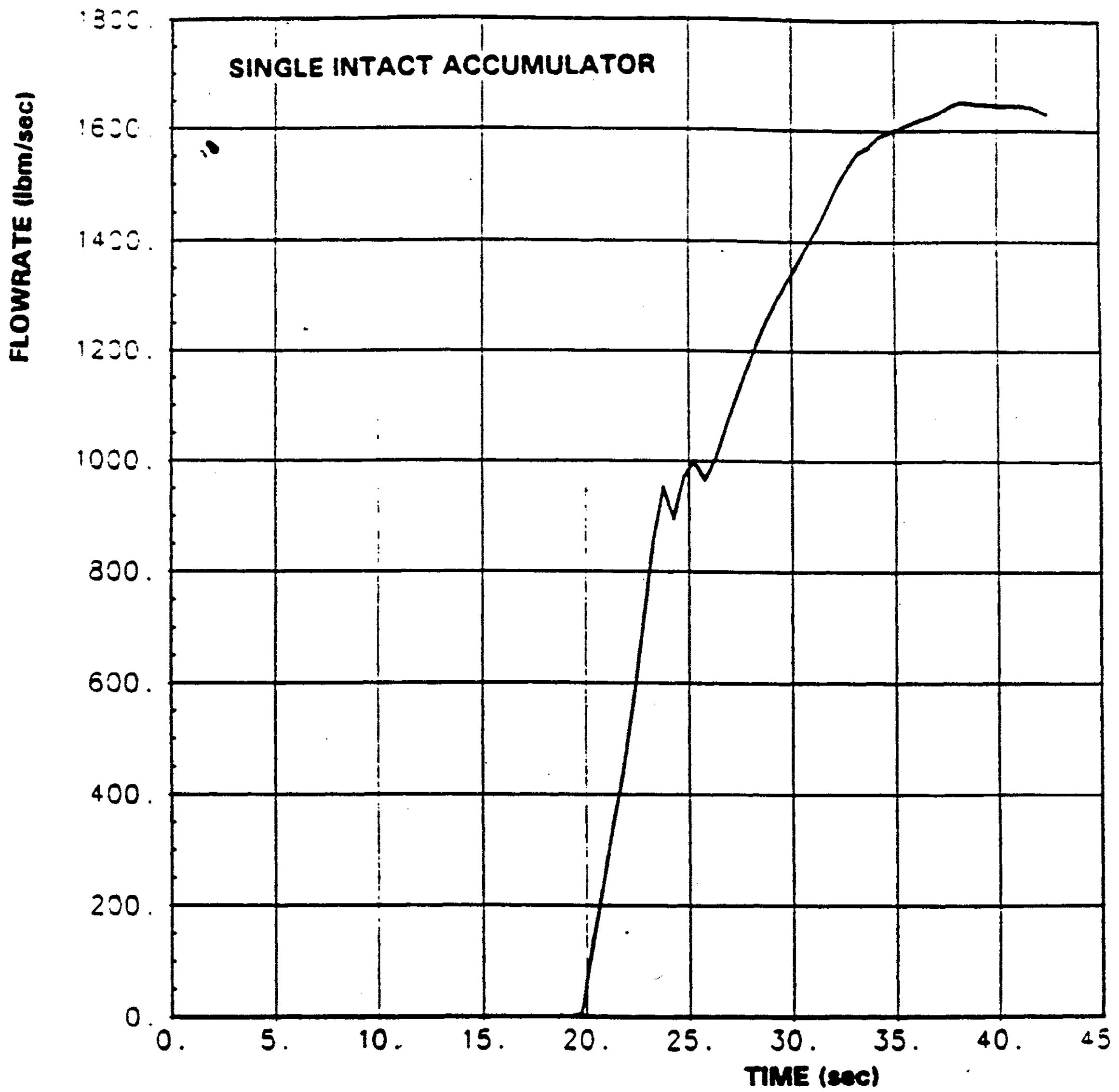
DECLG(C_D = 0.4), MIN SAFEGUARDS



Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-12G
ACCUMULATOR INJECTION
REDUCED T_{AVG}
DECLG(C_D =0.4), MIN SAFEGUARDS



Rev. 7

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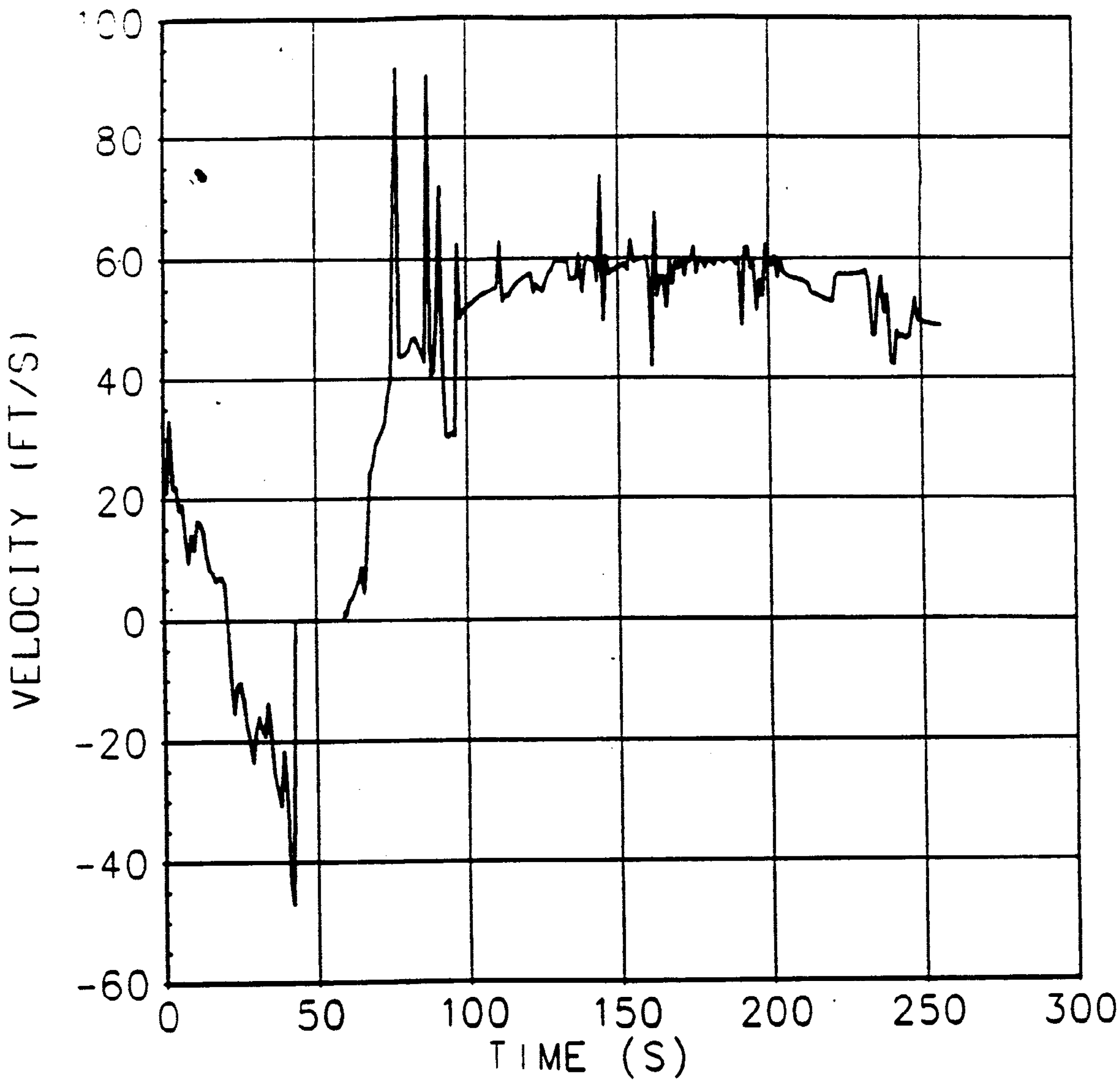
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-12H
ACCUMULATOR INJECTION
REDUCED T_{AVG}
DECLG(C_D = 0.4), MIN SAFEGUARDS



Rev. 7

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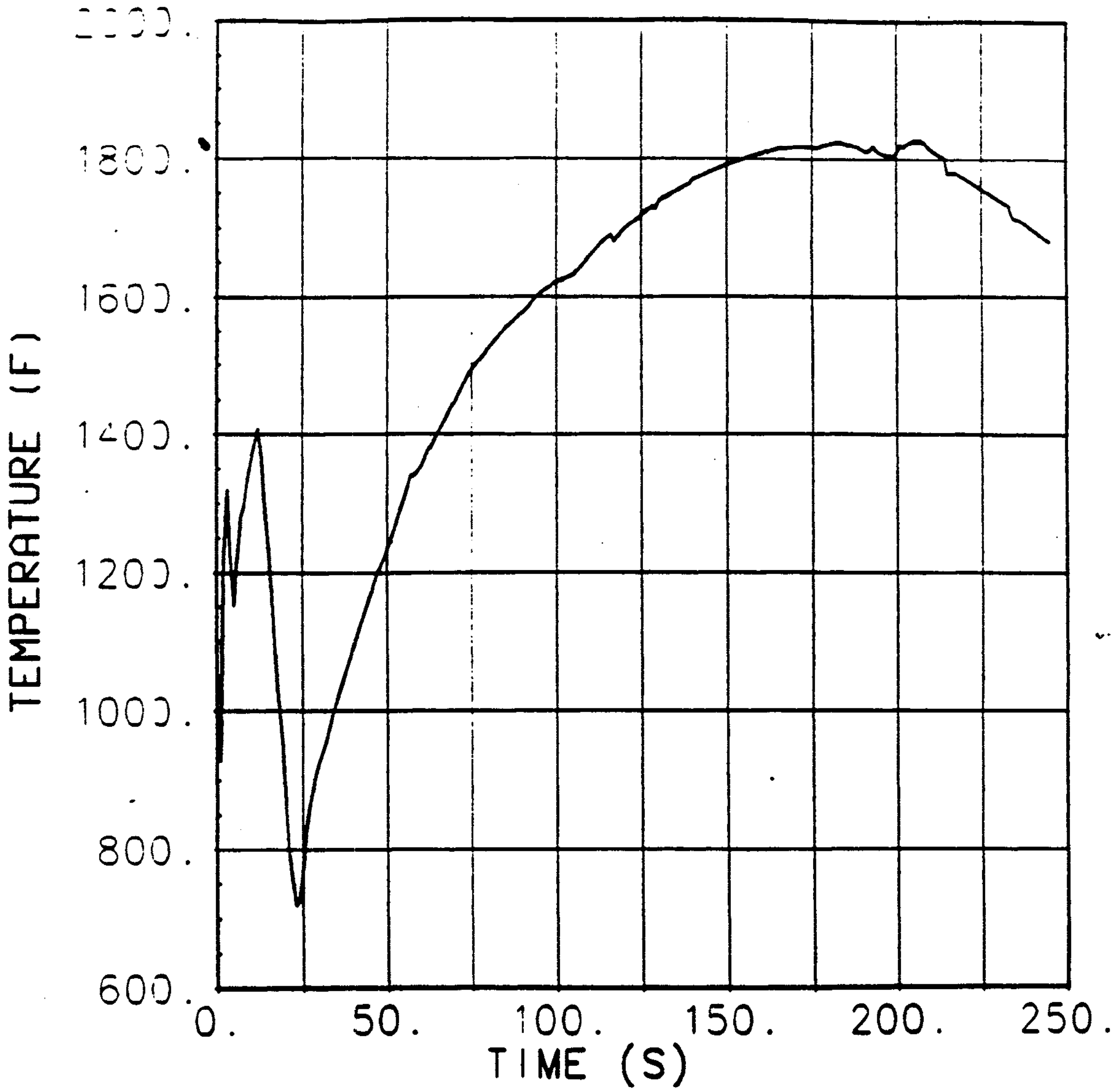
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-12I

FLUID VELOCITY @ HOT SPOT
REDUCED T_{AVG}
DECLG(C_D=0.4), MIN SAFEGUARDS



Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-13
CLAD TEMPERATURE TRANSIENT HOT SPOT

REDUCED T_{AVG}

DECLG(C_D = 0.4), MIN SAFEGUARDS

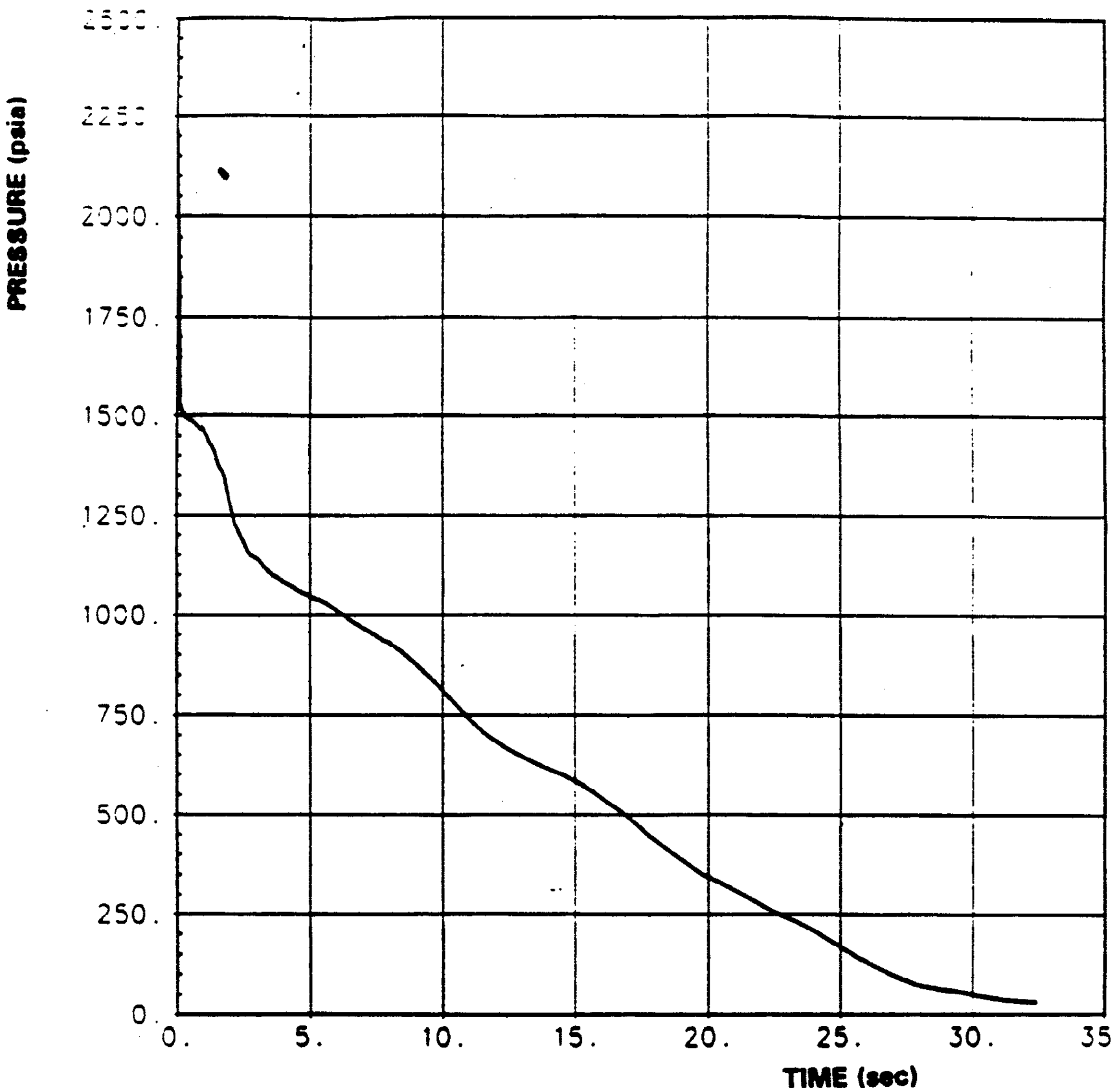


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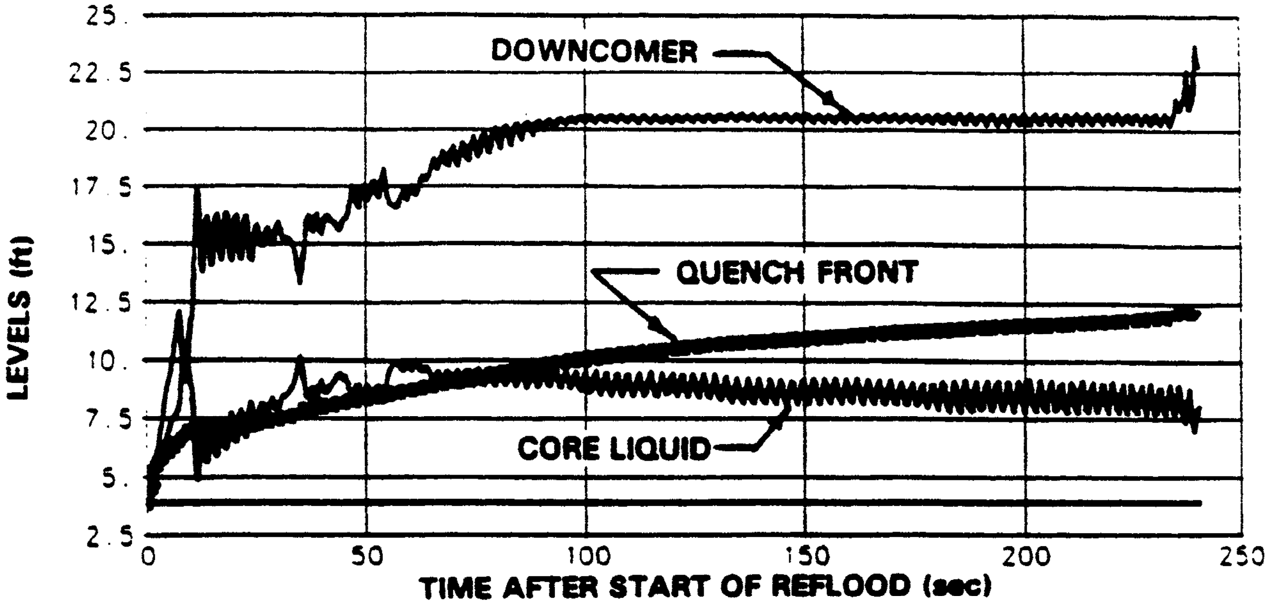
FIGURE 15.6-14

CORE PRESSURE

REDUCED T_{AVG}, DECLG(C_D=0.6)



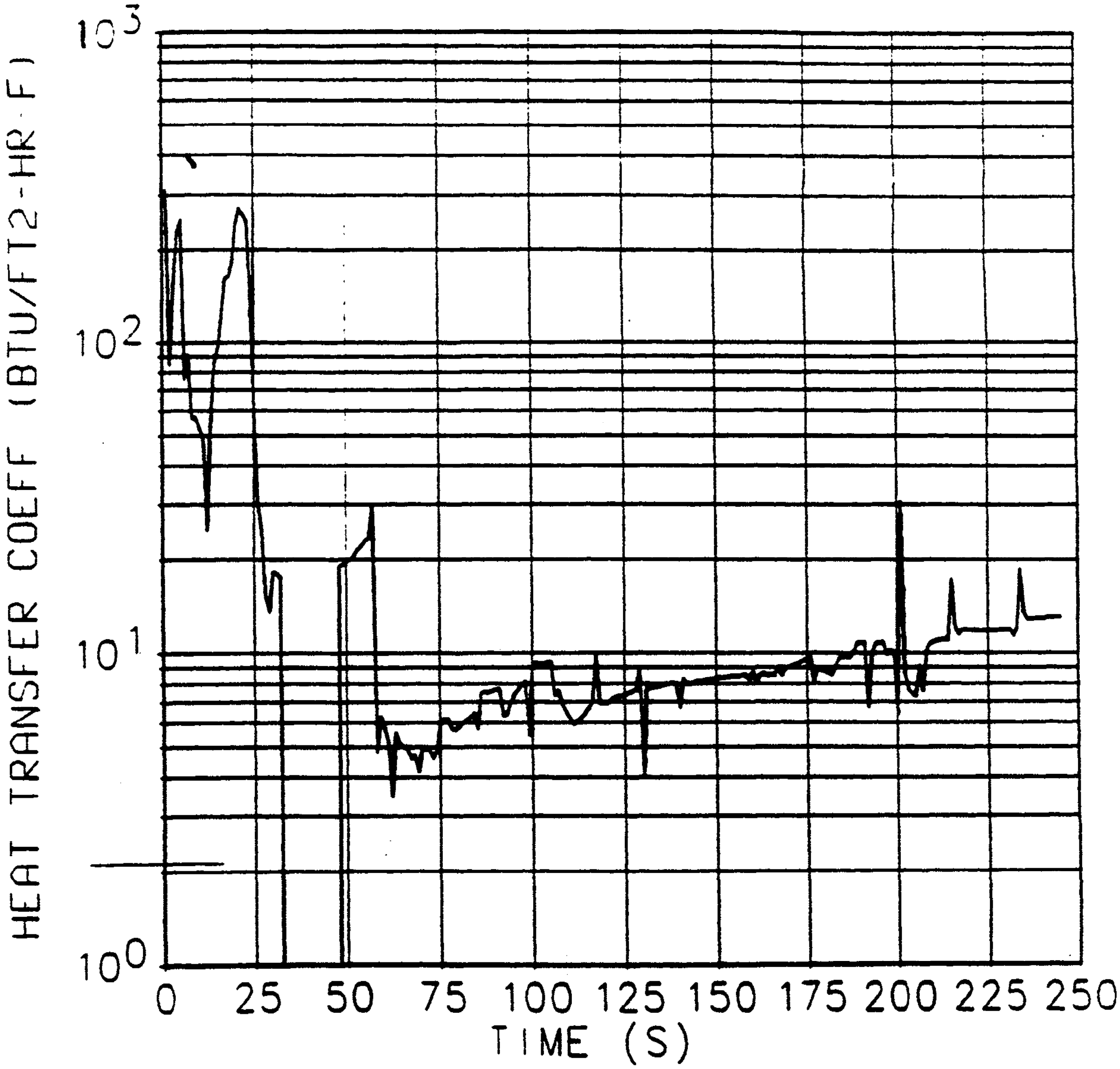
Rev. 12

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

Figure 15.6-15

REFLOOD MIXTURE LEVELS
REDUCED T_{AVG}

DECLG(C_D = 0.6), MIN SAFEGUARDS



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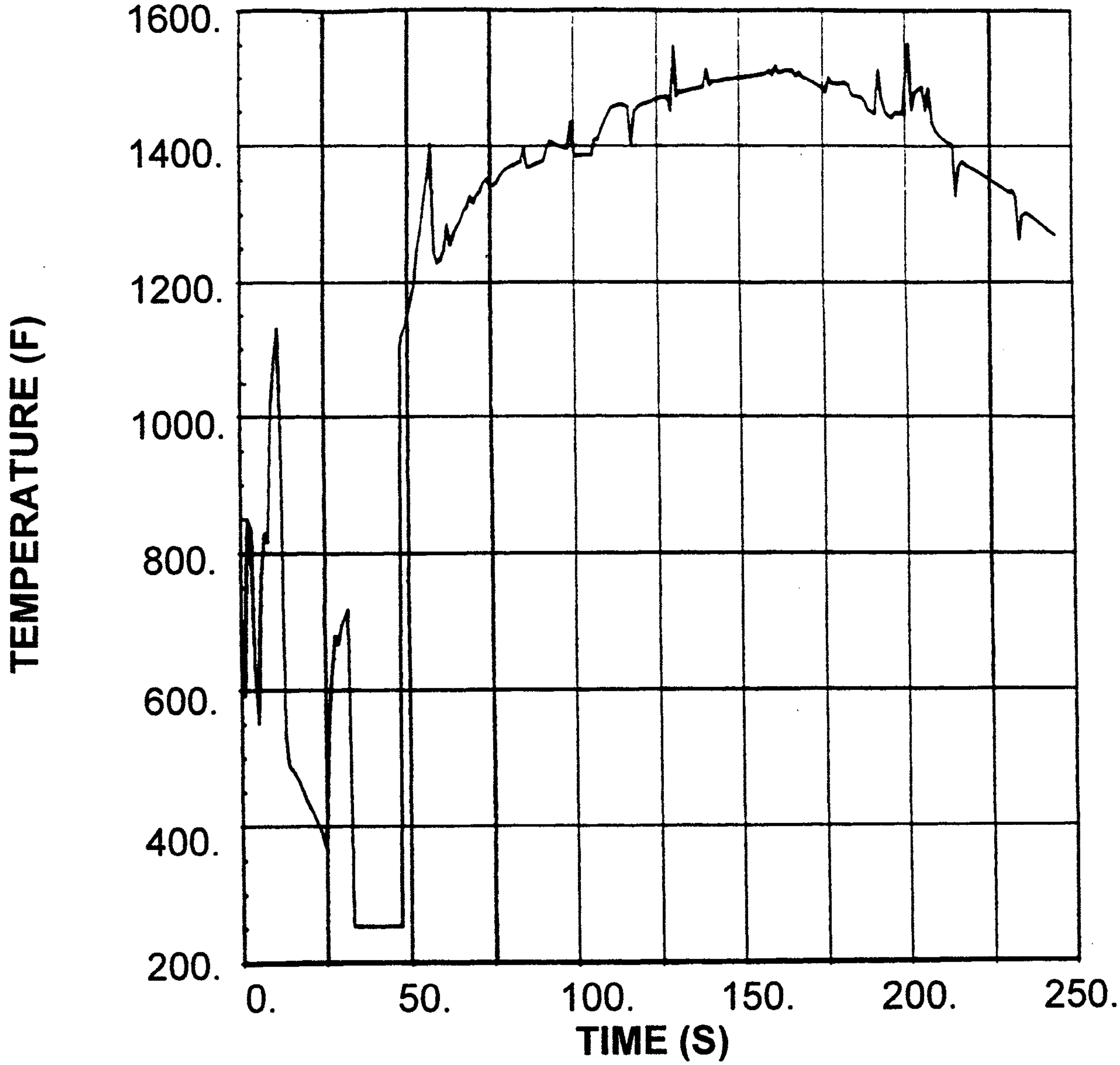
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-16

HEAT TRANSFER COEFFICIENT
HOT SPOT, REDUCED T_{AVG}
DECLG(C_D-0.6), MIN SAFEGUARDS



Rev. 12

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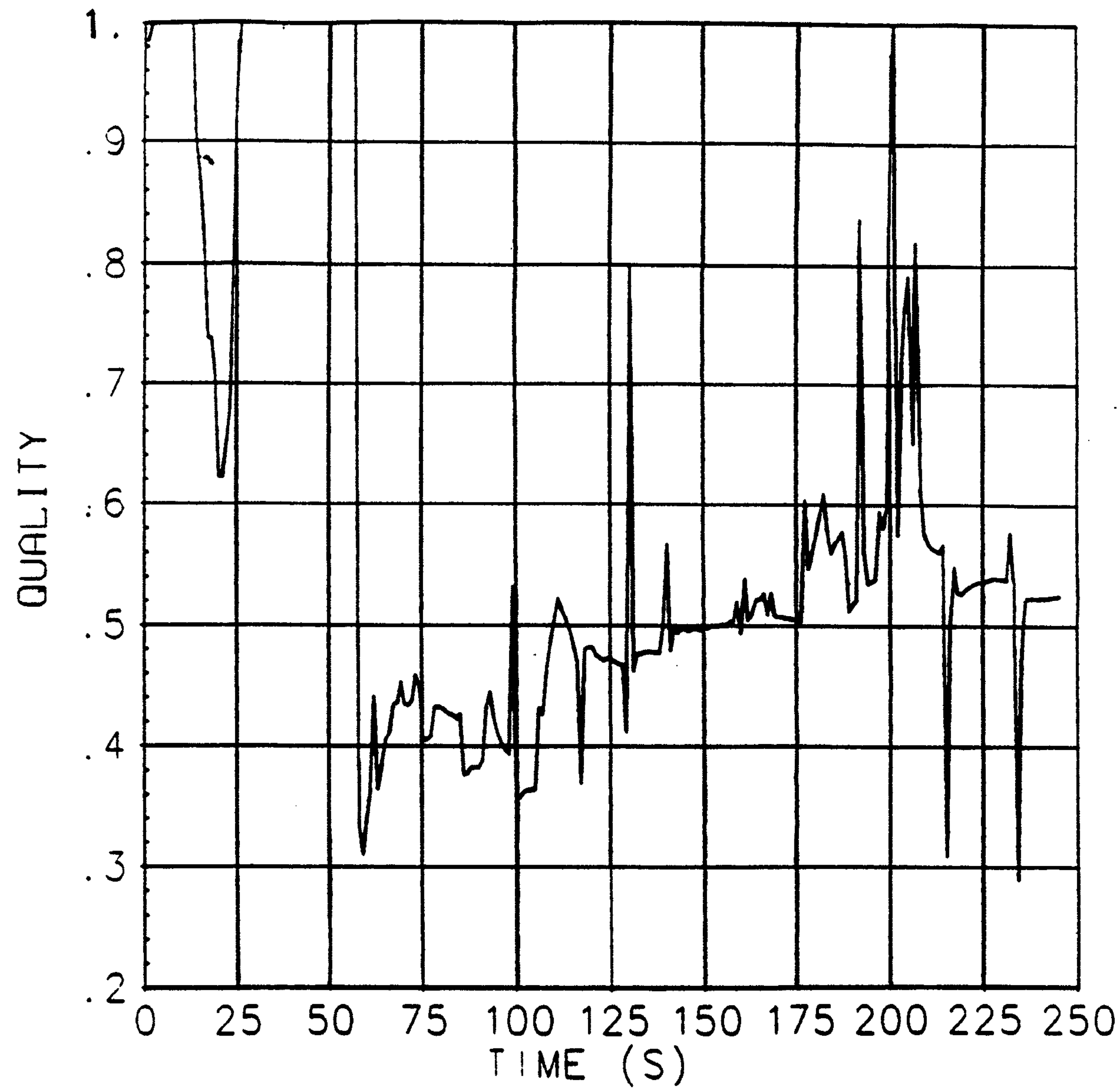
UPDATED SAFETY ANALYSIS REPORT

Figure 15.6-17

FLUID TEMPERATURE @ HOT SPOT
REDUCED T_{AVG}

DECLG(C_D = 0.6), MIN SAFEGUARDS



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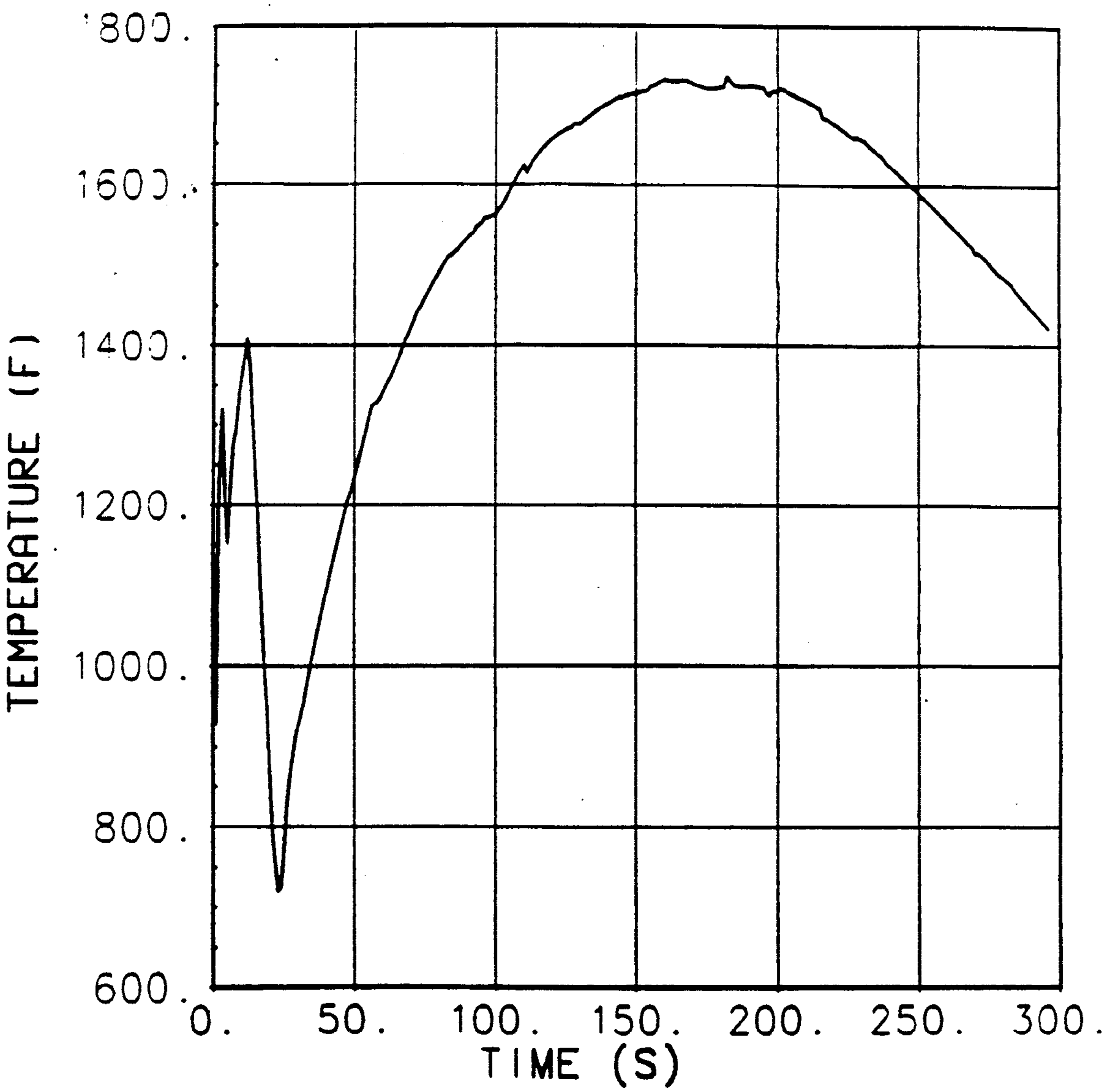
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-18

FLUID QUALITY @ HOT SPOT
REDUCED T_{AVG}
DECLG(C_D=0.6), MIN SAFEGUARDS

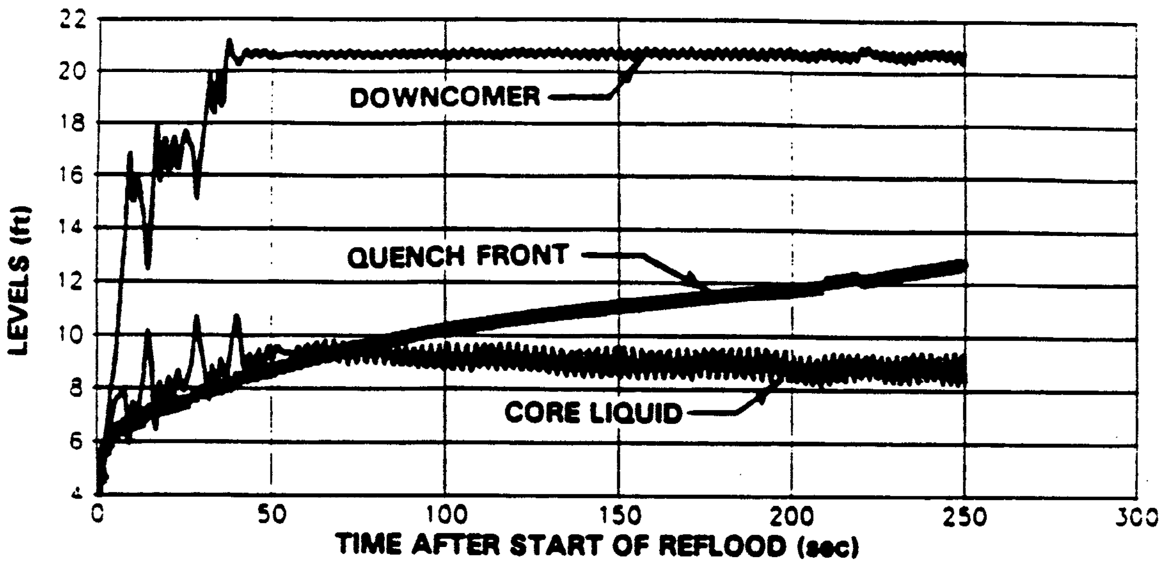


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-19

CLAD TEMPERATURE TRANSIENT
HOT SPOT, REDUCED T_{AVG}
DECLG(C_D=0.6), MAX SAFEGUARDS

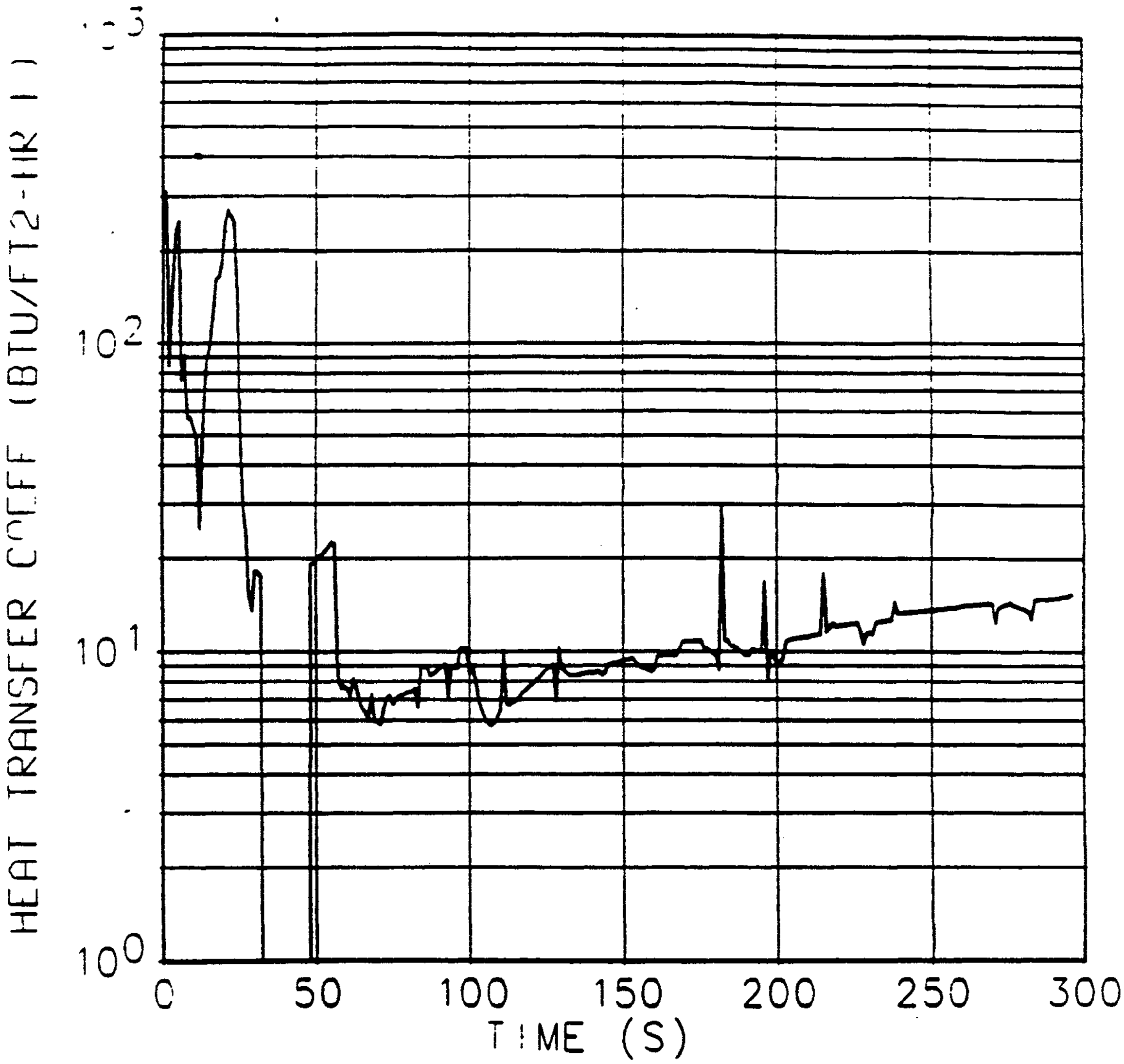


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-20

REFLOOD MIXTURE LEVELS
REDUCED T_{AVG}
DECLG(C_D=0.6), MAX SAFEGUARDS

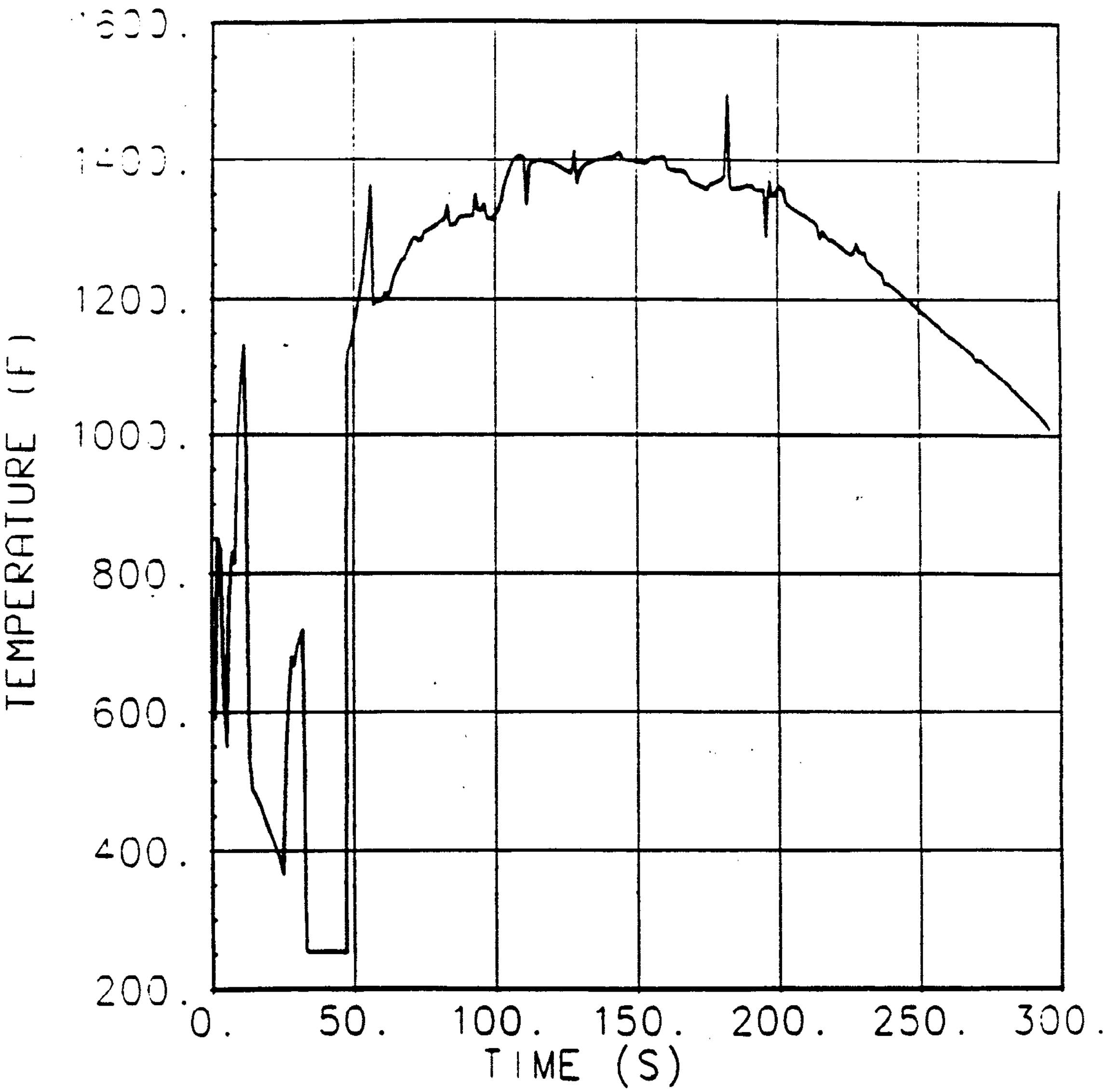


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-21

HEAT TRANSFER COEFFICIENT
HOT SPOT, REDUCED T_{AVG}
DECLG(CD=0.6), MAX SAFEGUARDS

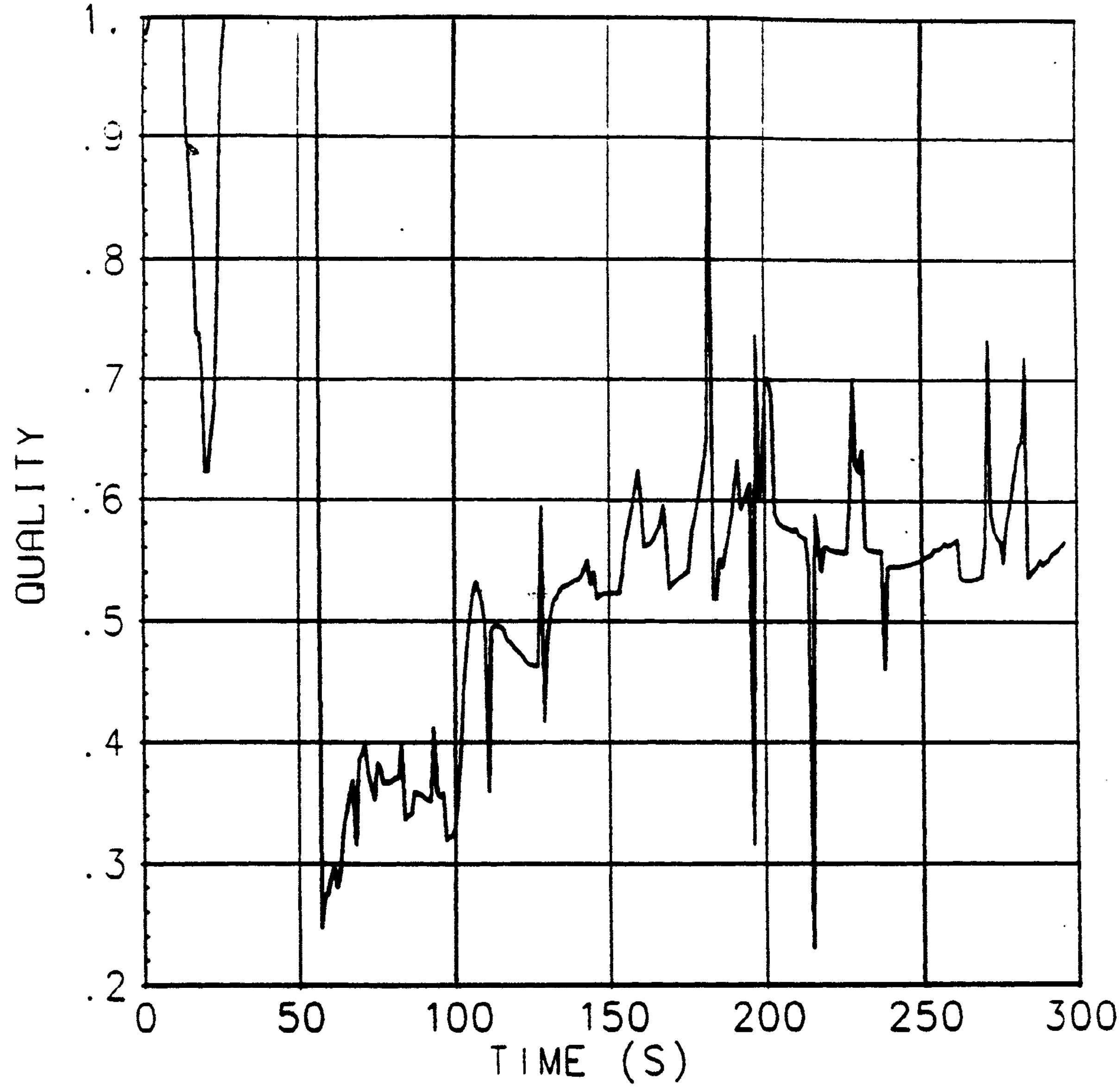


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-22

FLUID TEMPERATURE @ HOT SPOT
REDUCED T_{AVG}
DECLG(C_D=0.6), MAX SAFEGUARDS

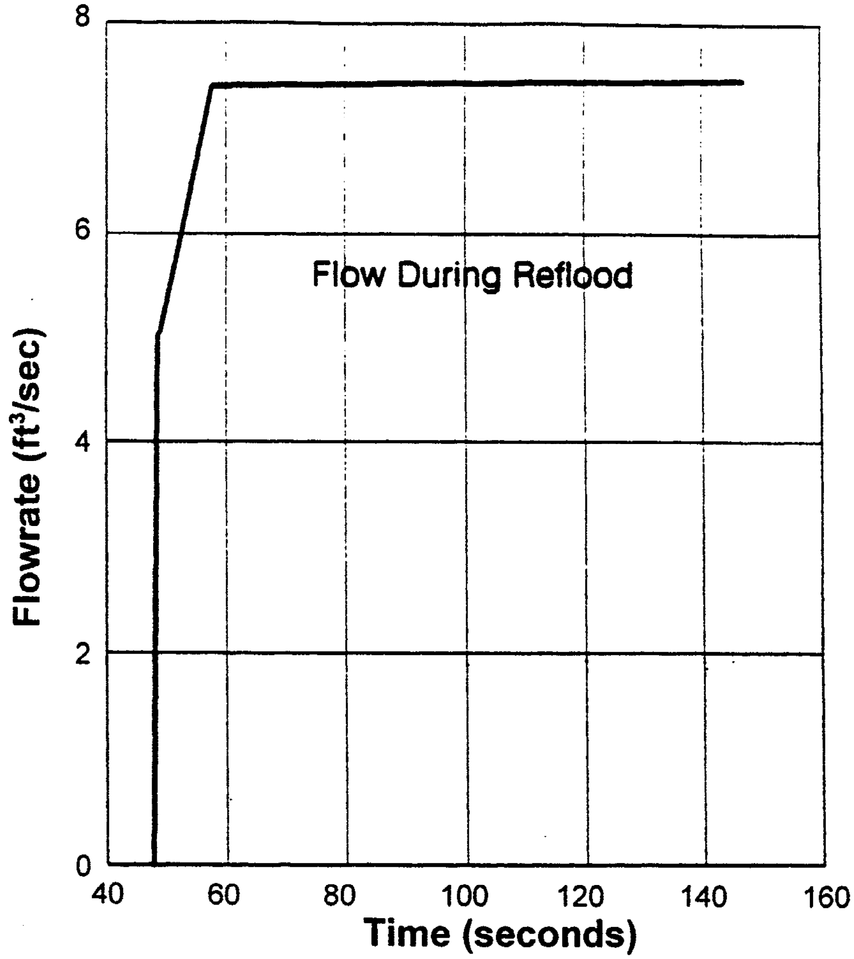


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-23

FLUID QUALITY @ HOT SPOT
REDUCED T_{AVG}
DECLG(C_D=0.6), MAX SAFEGUARDS



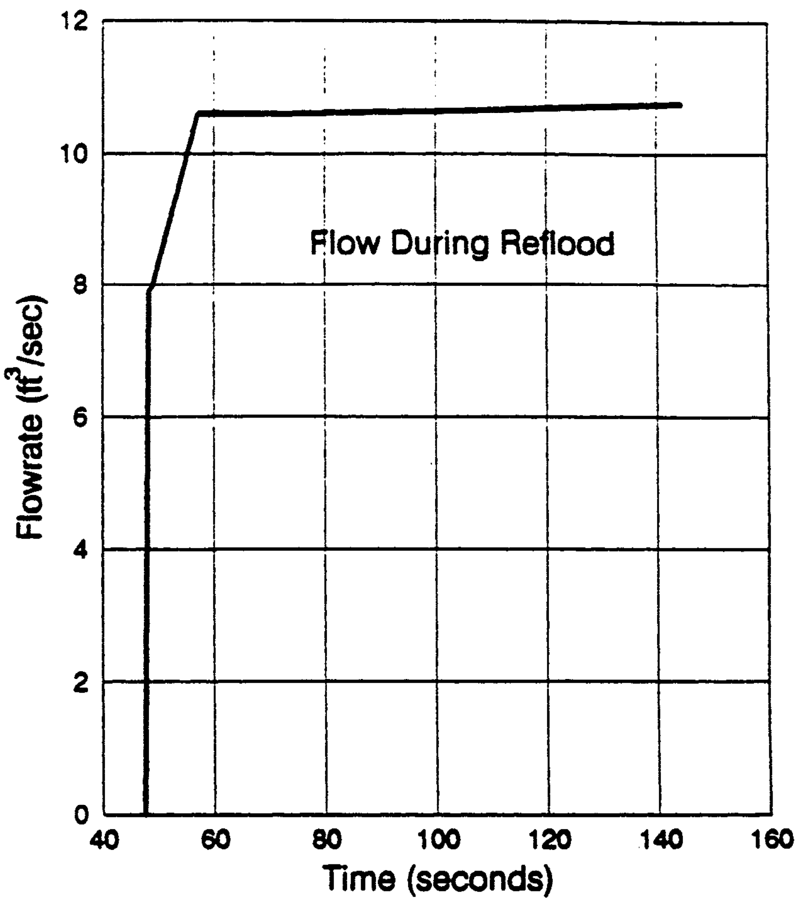
Rev. 12

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

Figure 15.6-24a

PUMPED SAFETY INJECTION
REDUCED T_{AVG}

DECLG(C_D = 0.6), MIN SAFEGUARDS

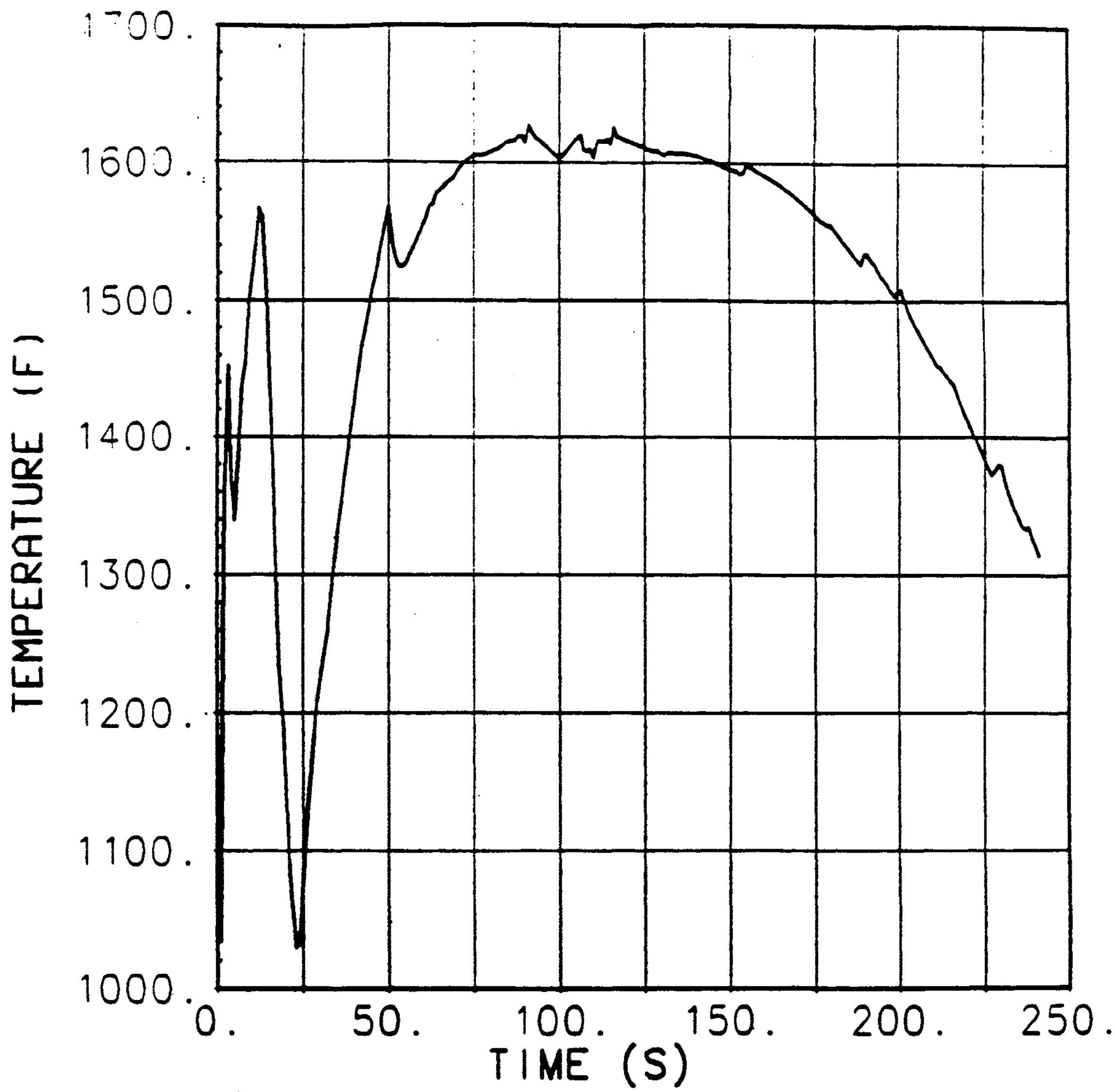


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-24B

PUMPED SAFETY INJECTION
REDUCED T_{AVG}
DECLG(C_D=0.6), MAX SAFEGUARDS

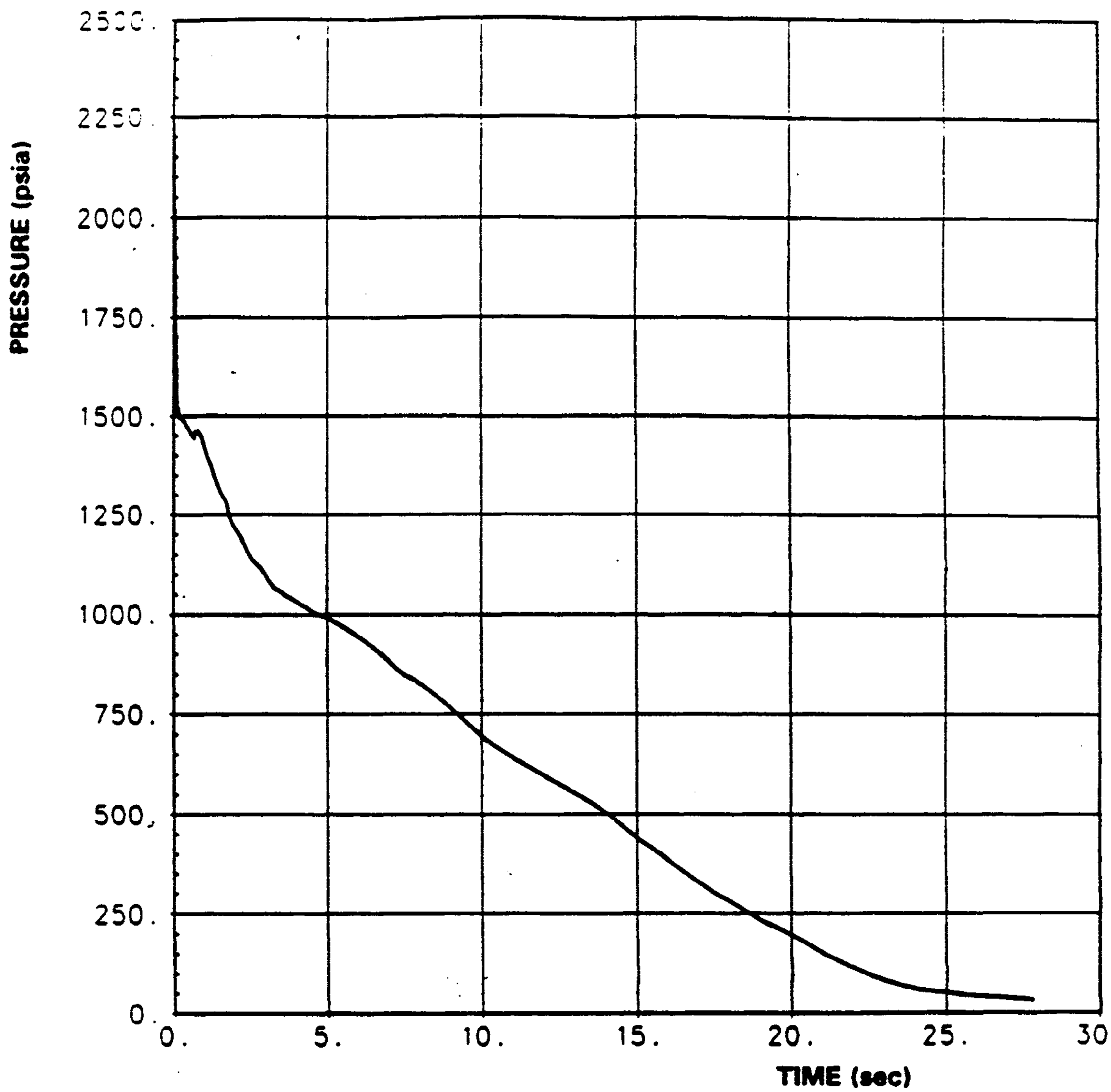


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-25

CLAD TEMPERATURE TRANSIENT
HOT SPOT, REDUCED T_{AVG}
DECLG(C_D=0.8), MIN SAFEGUARDS

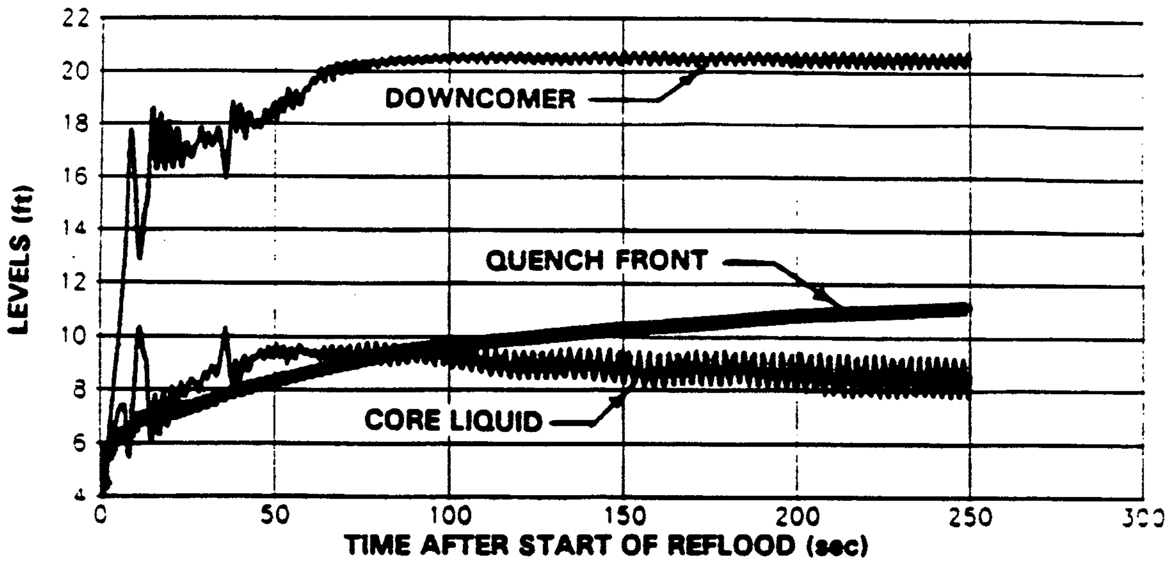


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-26

CORE PRESSURE REDUCED T_{AVG}
DECLG(C_D=0.8), MIN SAFEGUARDS

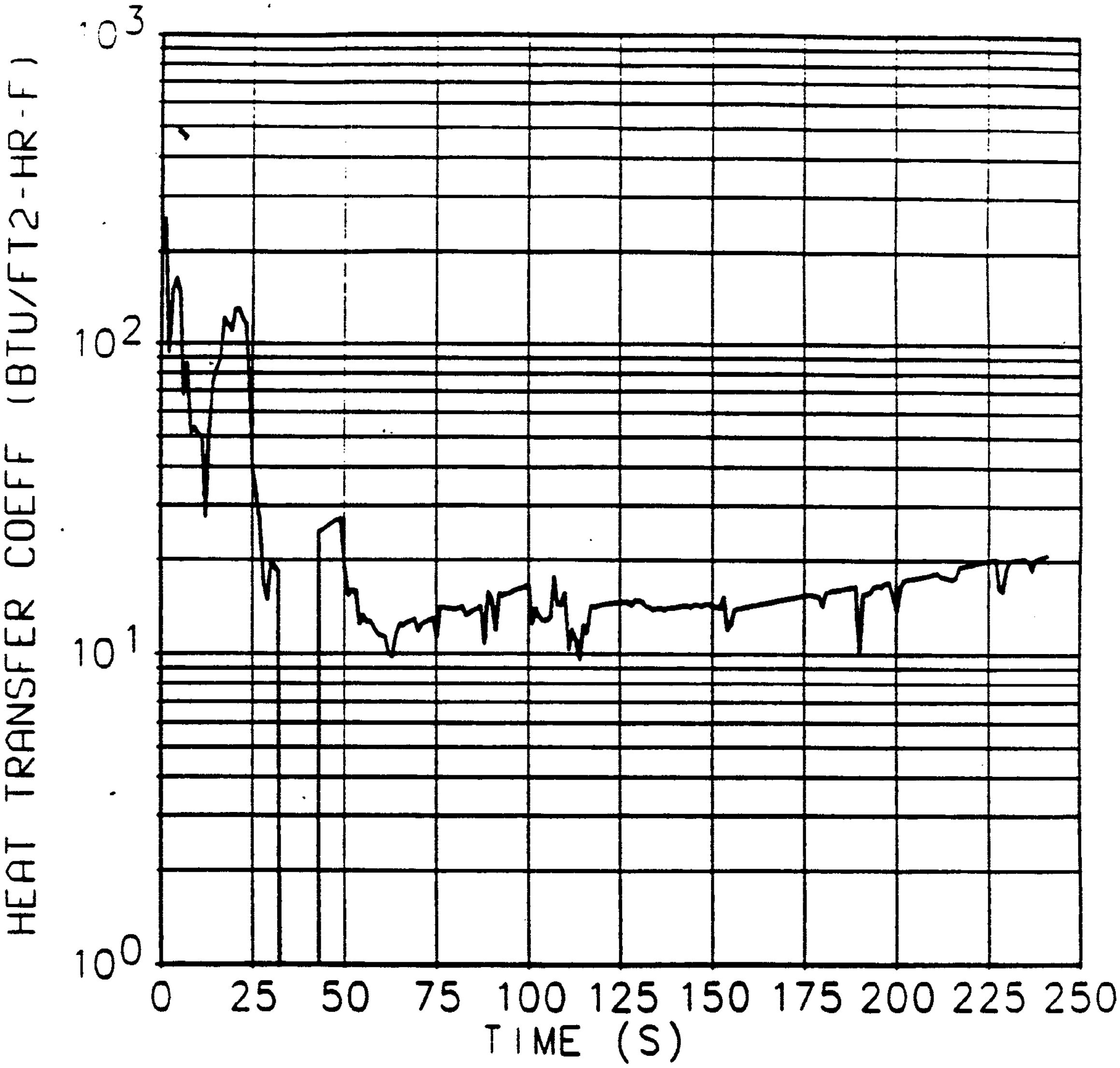


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-27

REFLOOD MIXTURE LEVELS
REDUCED T_{AVG}
DECLG(C_D=0.8), MIN SAFEGUARDS

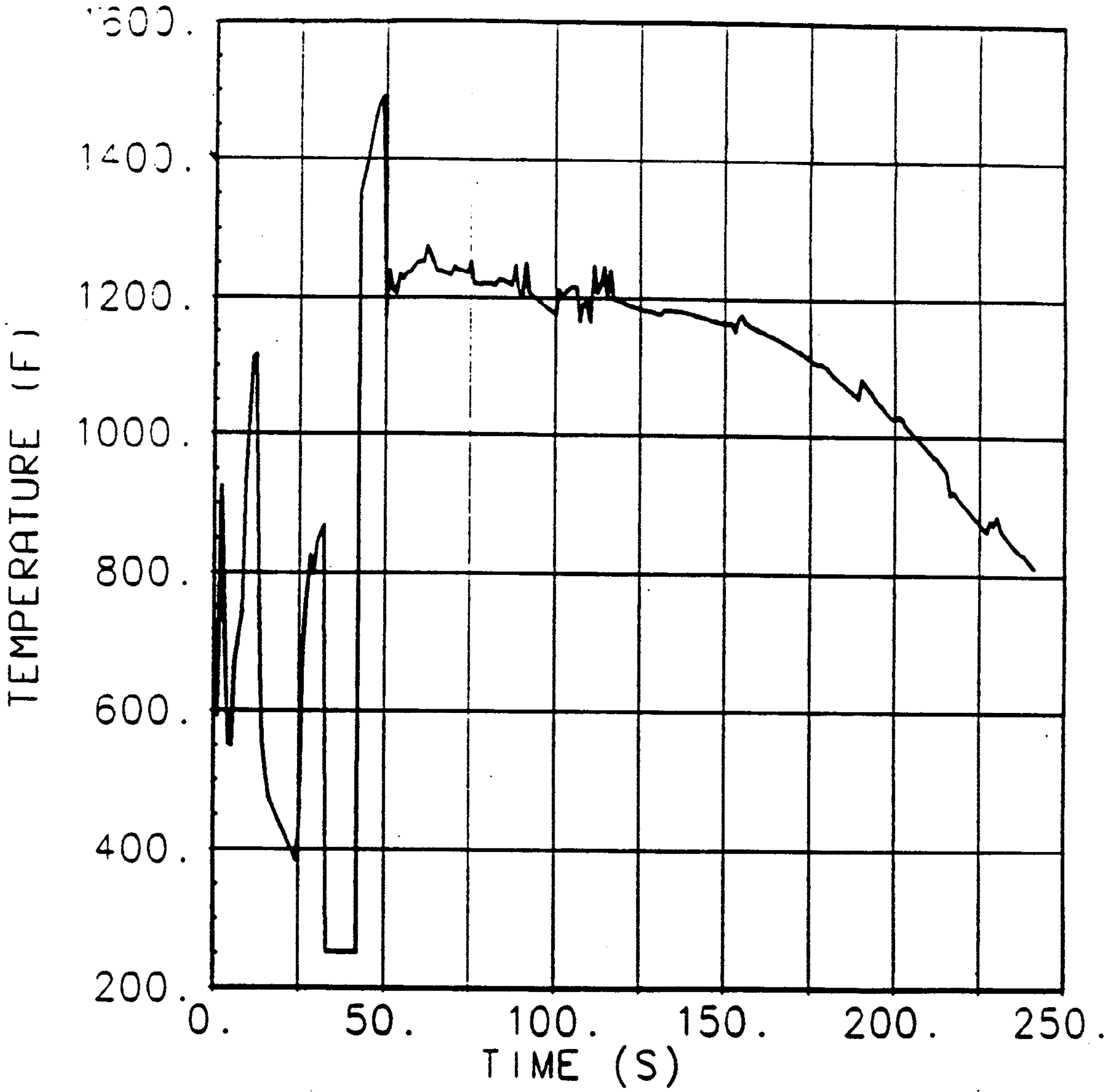


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-28

HEAT TRANSFER COEFFICIENT
HOT SPOT, REDUCED T_{AVG}
DECLG(C_D=0.8), MIN SAFEGUARDS

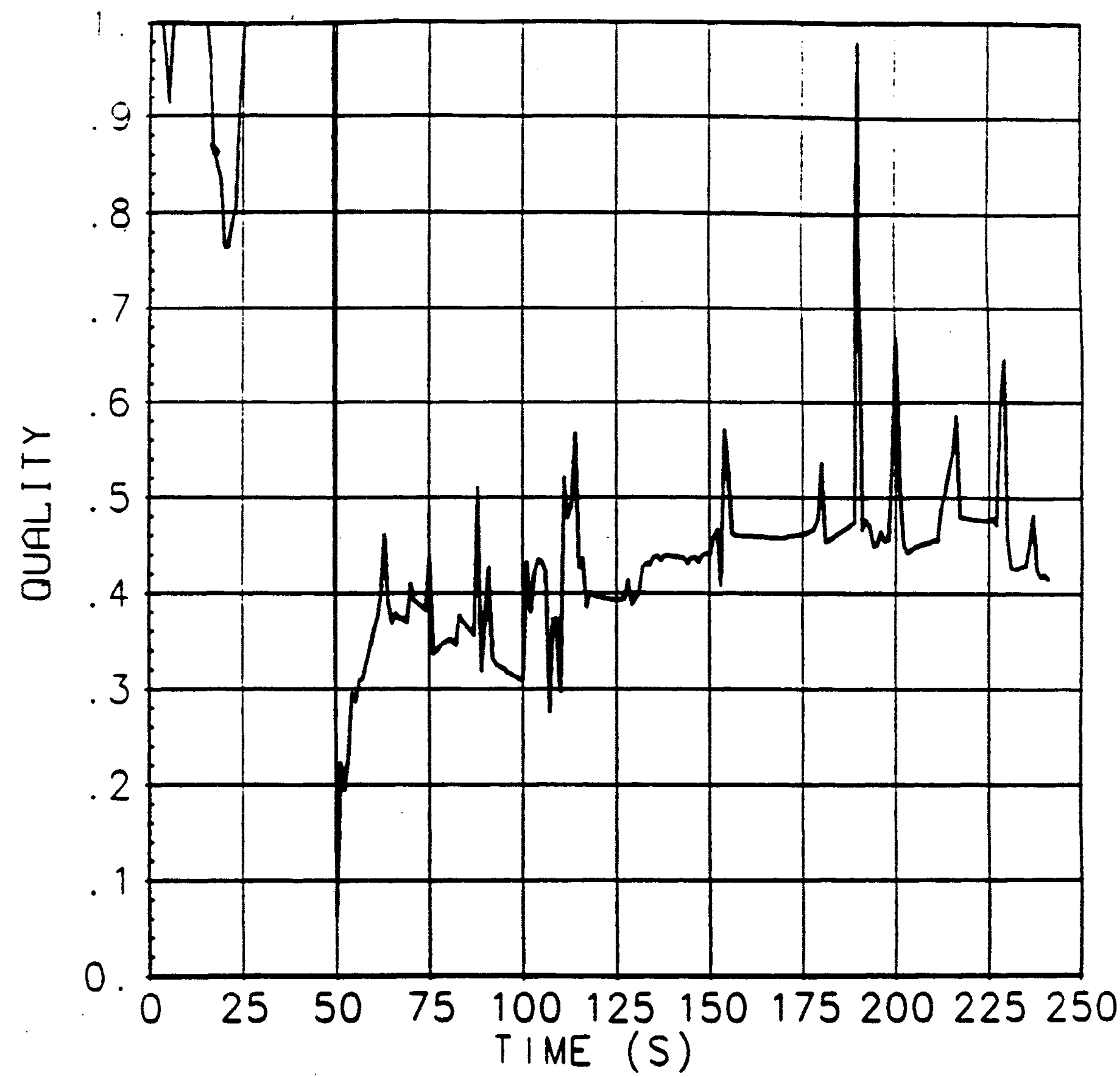


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-29

FLUID TEMPERATURE @ HOT SPOT
REDUCED T_{AVG}
DECLG(C_D=0.8), MIN SAFEGUARDS

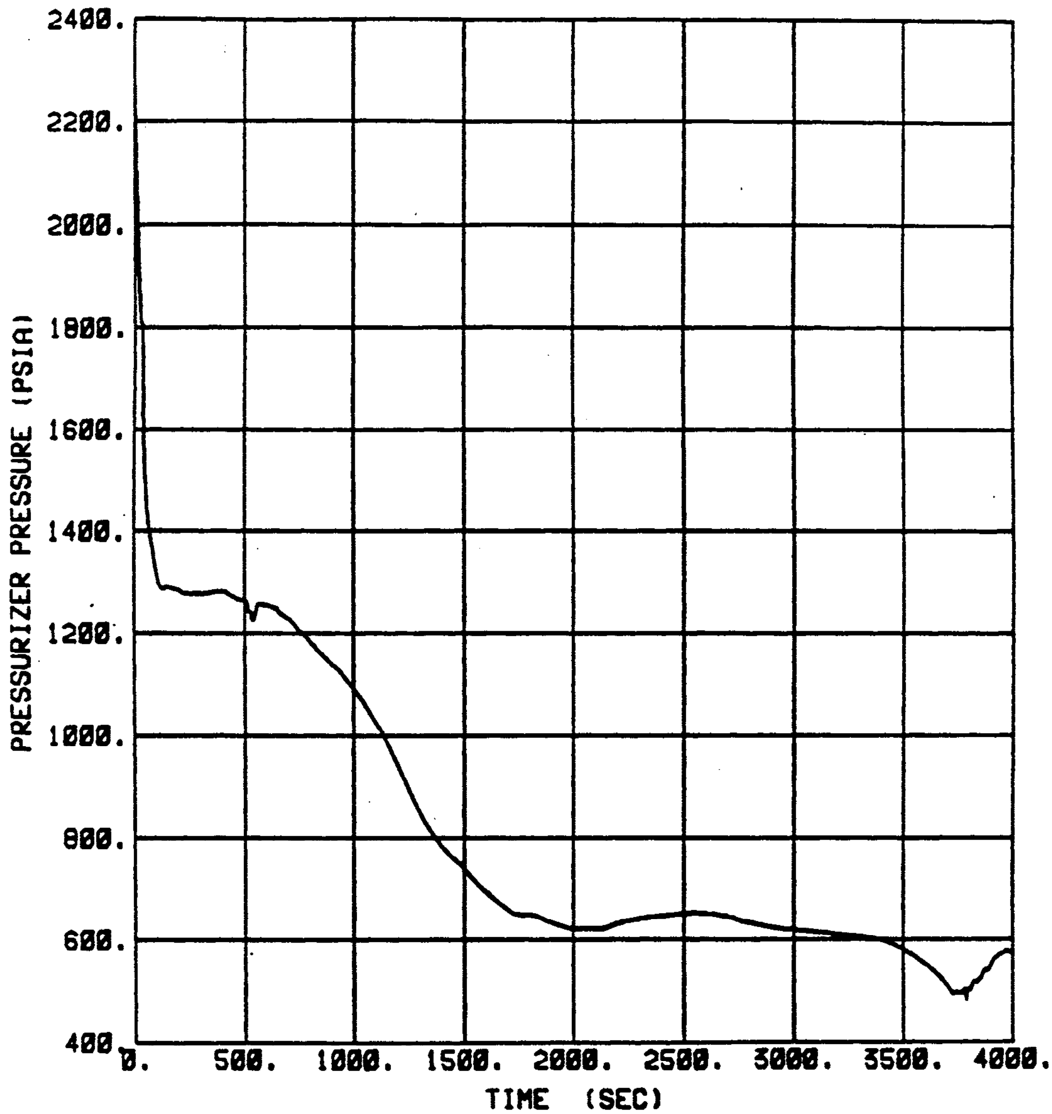


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-30

FLUID QUALITY @ HOT SPOT
REDUCED T_{AVG}
DECLG(C_D=0.8), MIN SAFEGUARDS

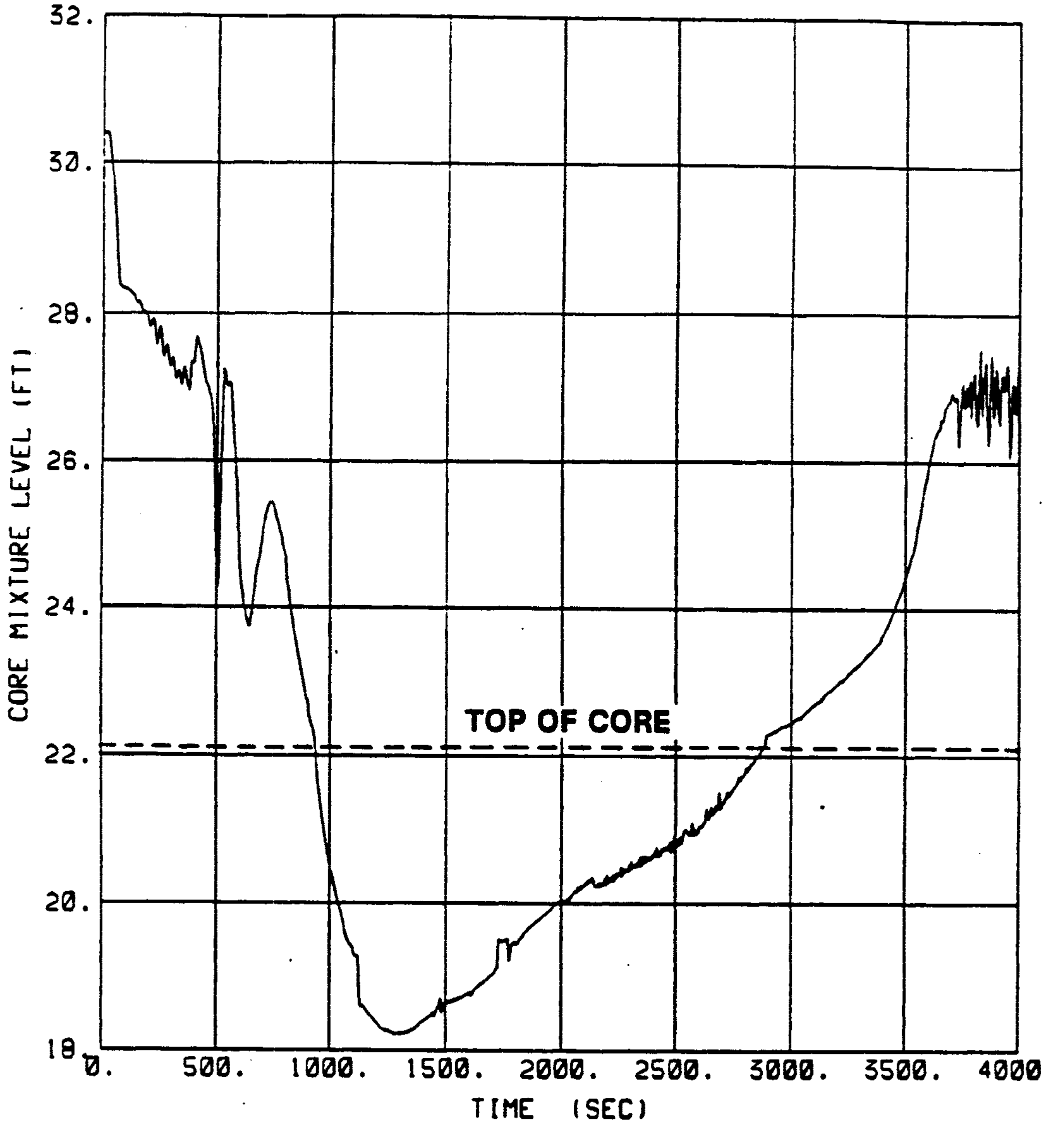


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-31

REACTOR COOLANT SYSTEM
DEPRESSURIZATION TRANSIENT
(3-INCH BREAK)

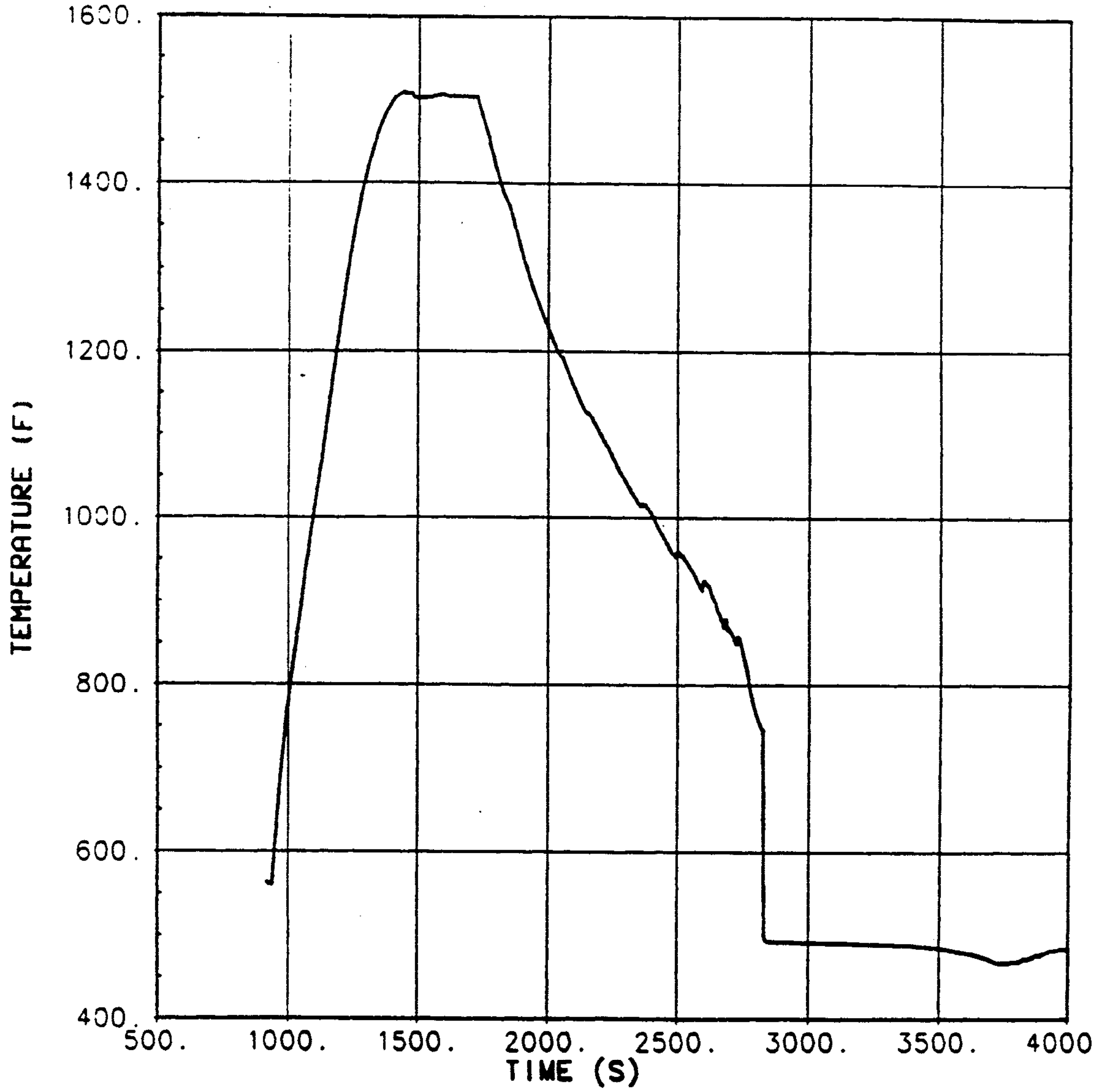


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-32

CORE MIXTURE LEVEL (3-INCH BREAK)

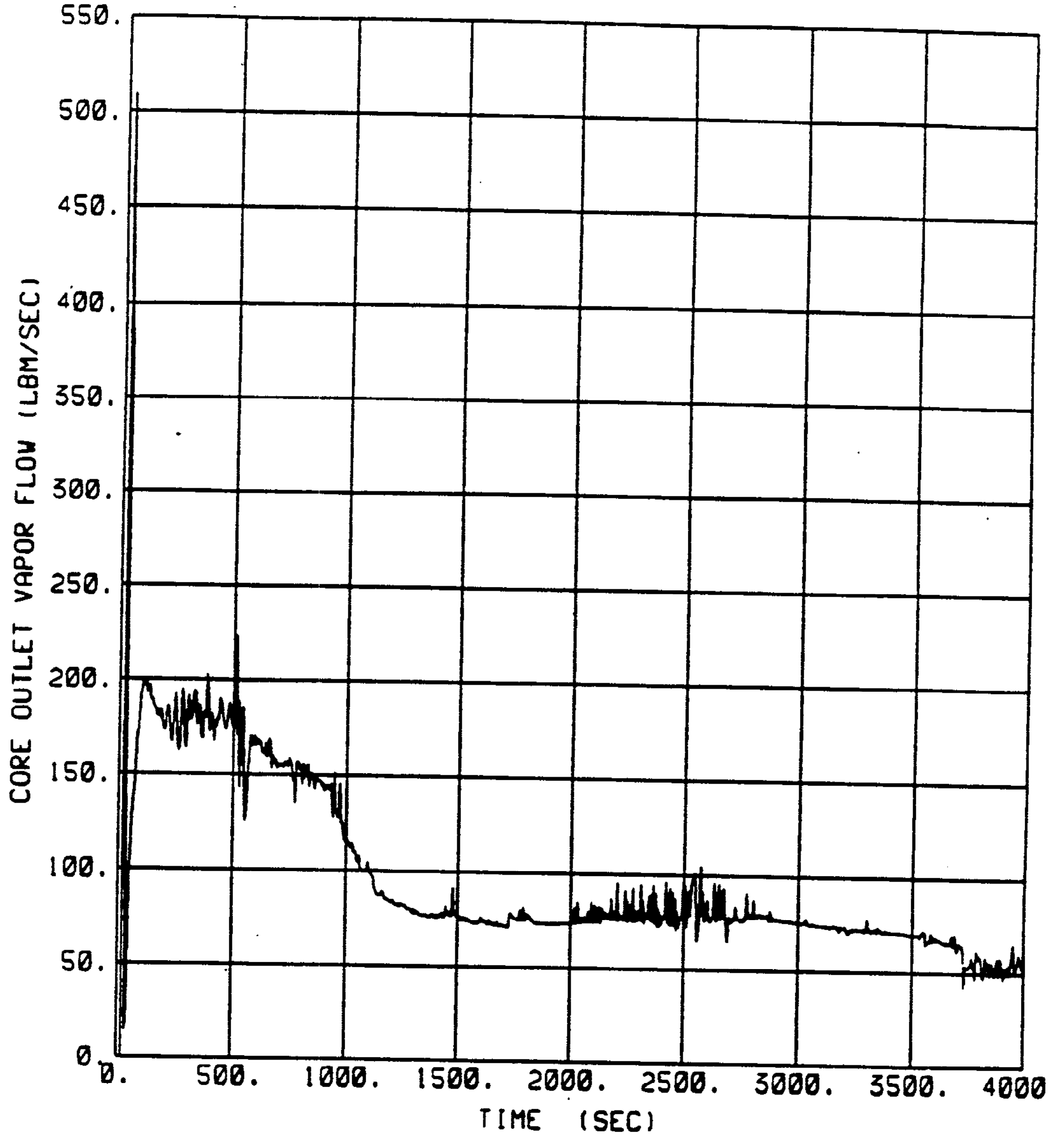


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-33

**CLAD TEMPERATURE TRANSIENT
HOT SPOT
(3-INCH BREAK)**



Rev. 7

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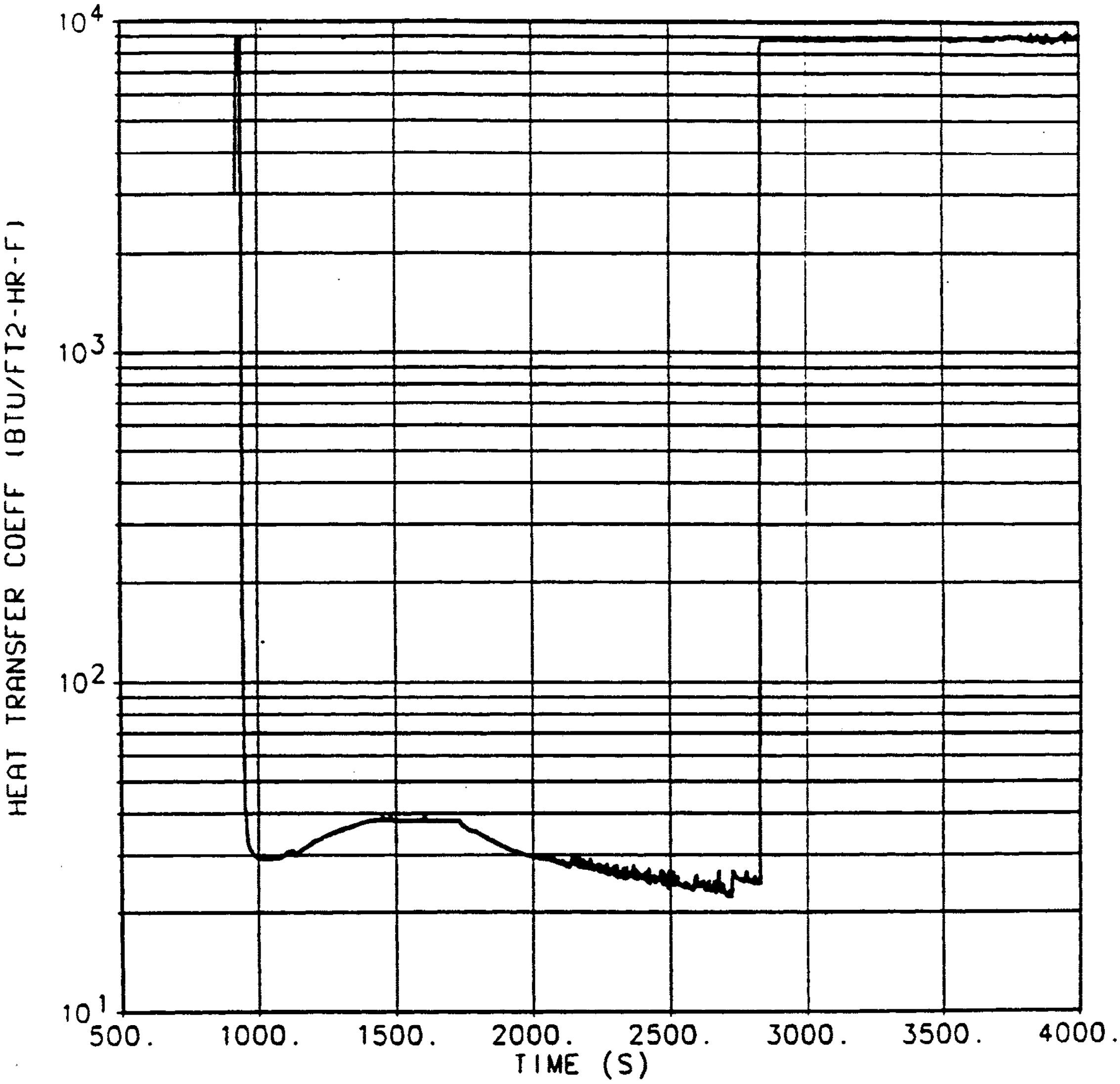
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-34

STEAM FLOW

(3-INCH BREAK)



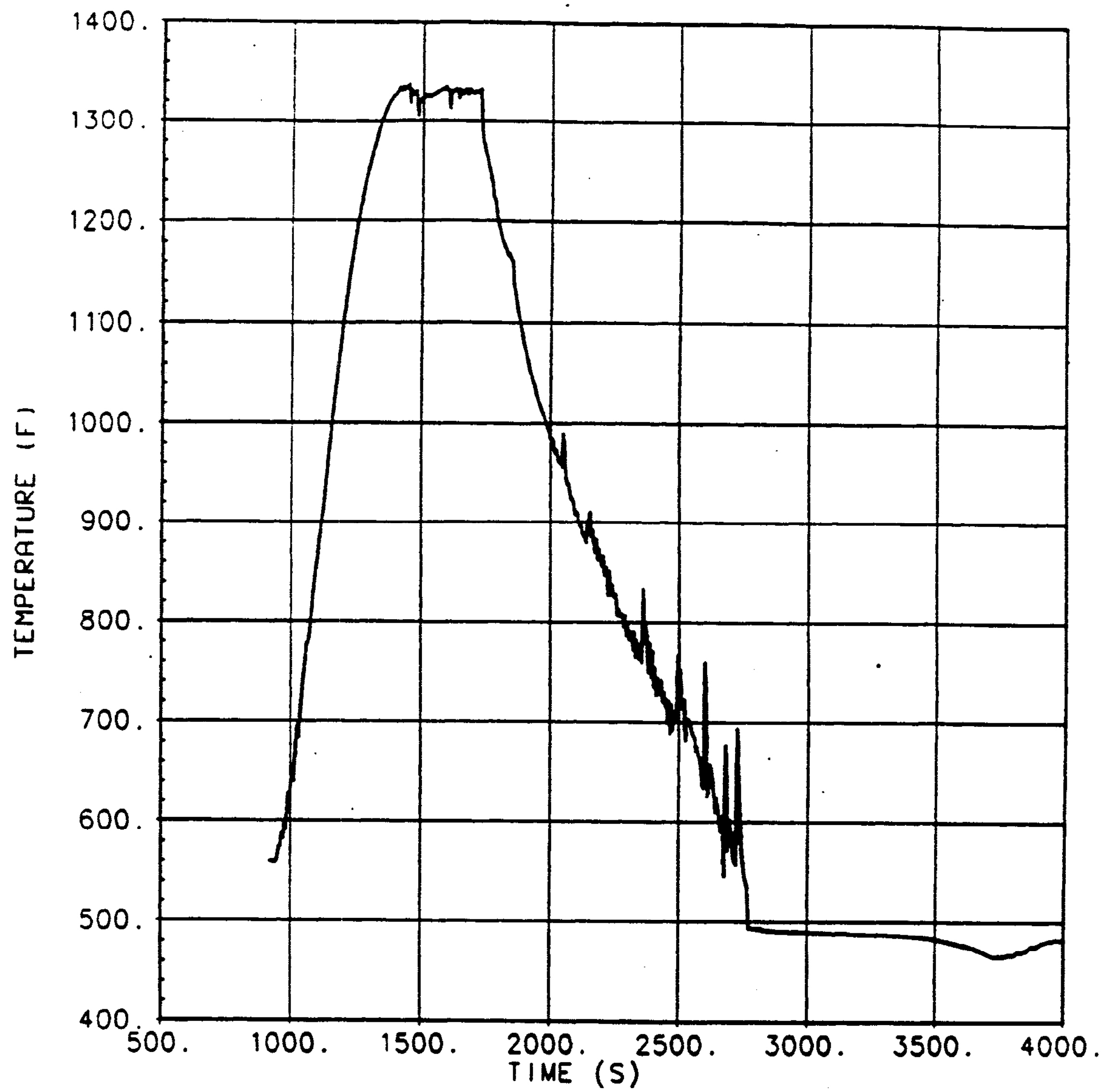
Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-35

ROD FILM HEAT TRANSFER COEFFICIENT

(3-INCH BREAK)



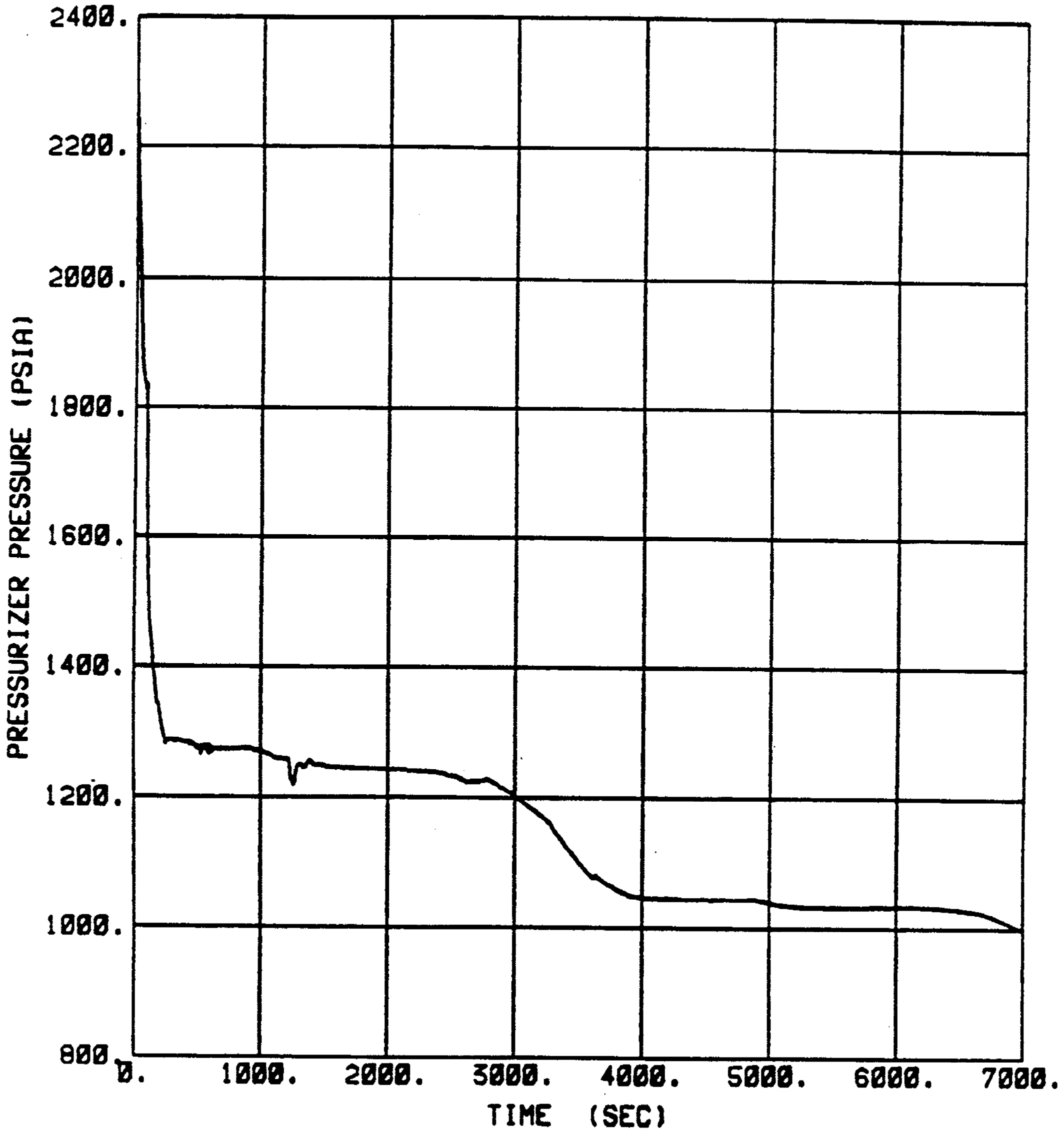
Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-36

HOT SPOT FLUID TEMPERATURE

(3-INCH BREAK)

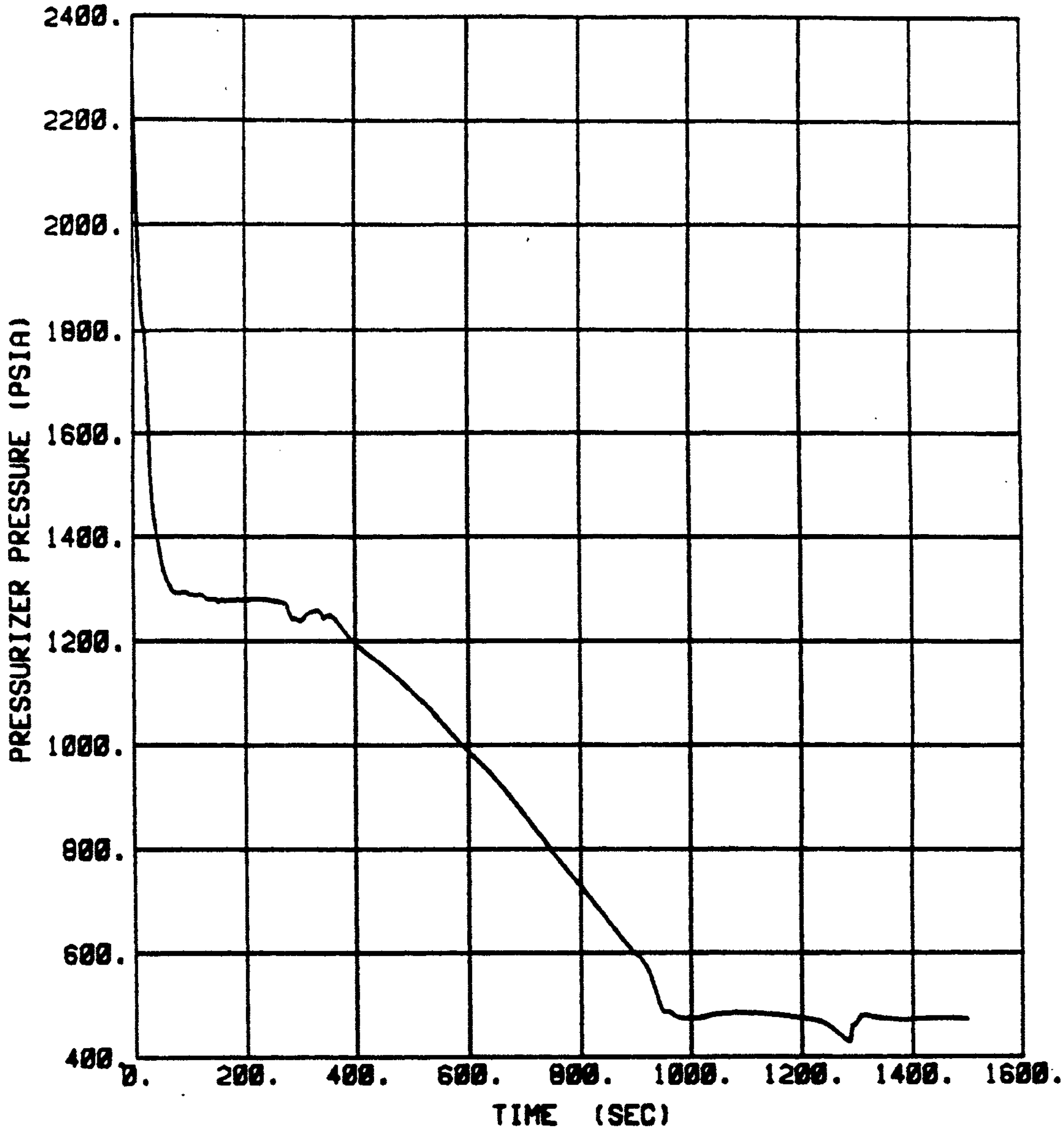


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-37

REACTOR COOLANT SYSTEM
DEPRESSURIZATION TRANSIENT
(2-INCH BREAK)

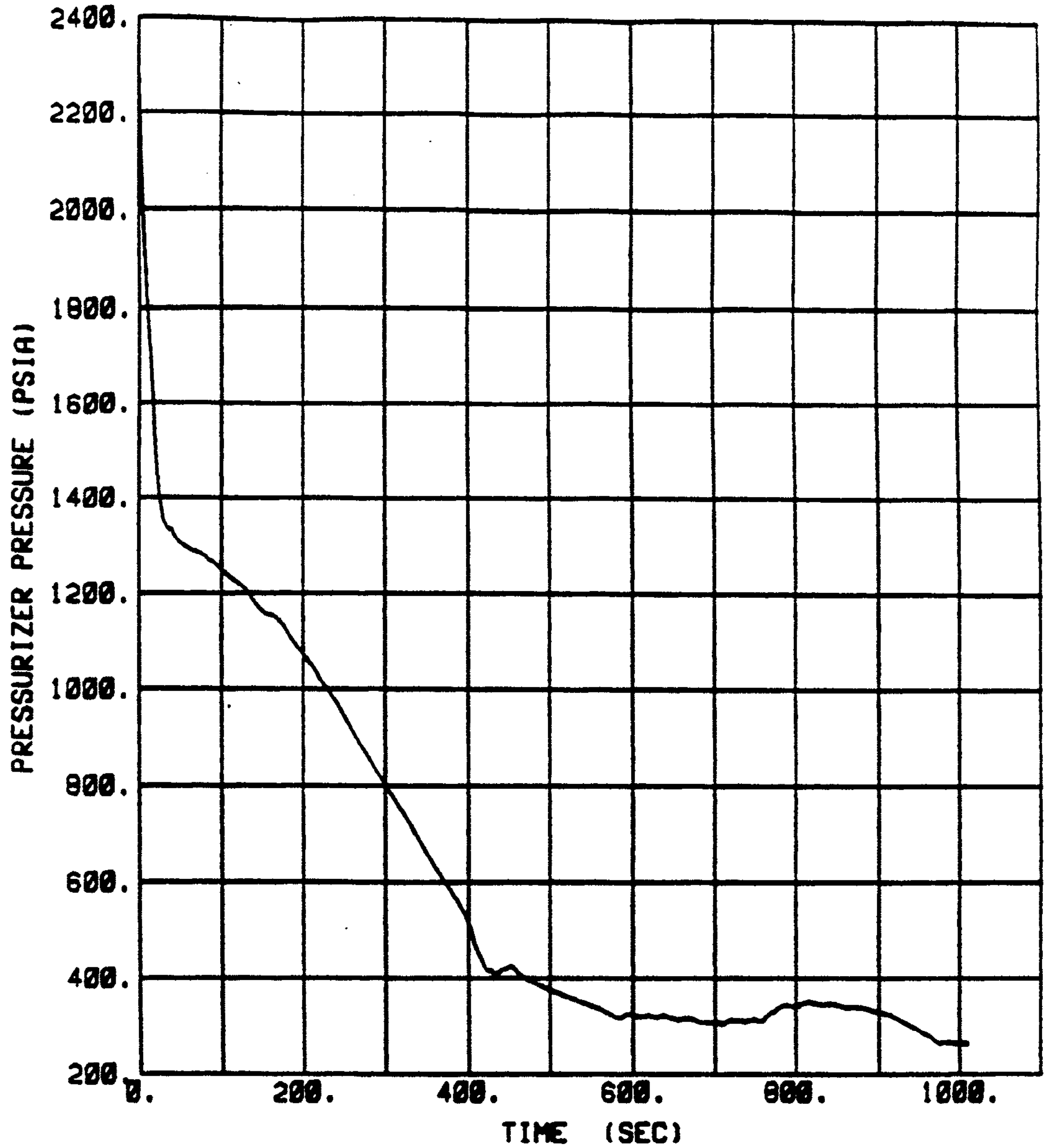


Rev. 12

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6.38

REACTOR COOLANT SYSTEM
DEPRESSURIZATION TRANSIENT
(4-INCH BREAK)

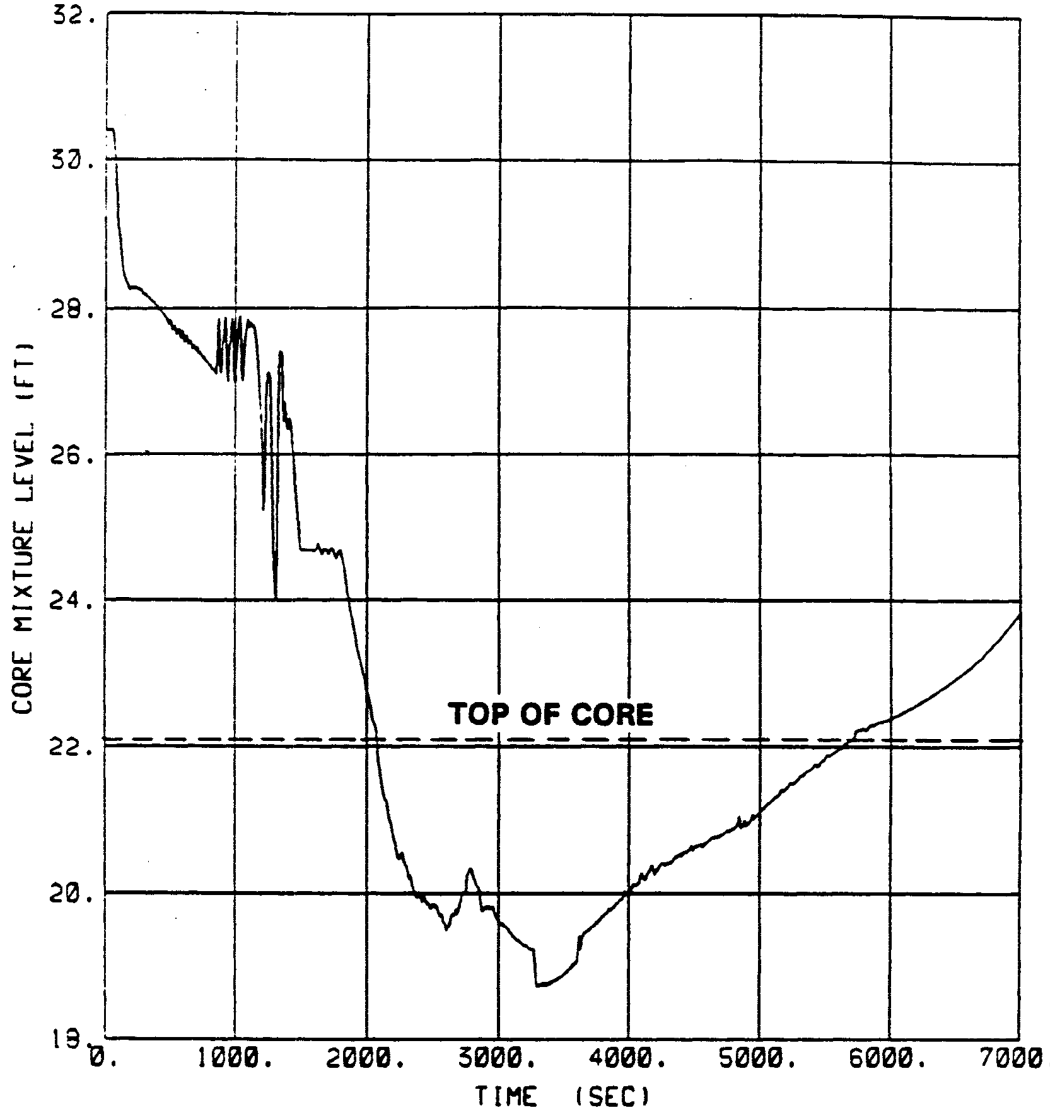


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-39

REACTOR COOLANT SYSTEM
DEPRESSURIZATION TRANSIENT
(6-INCH BREAK)



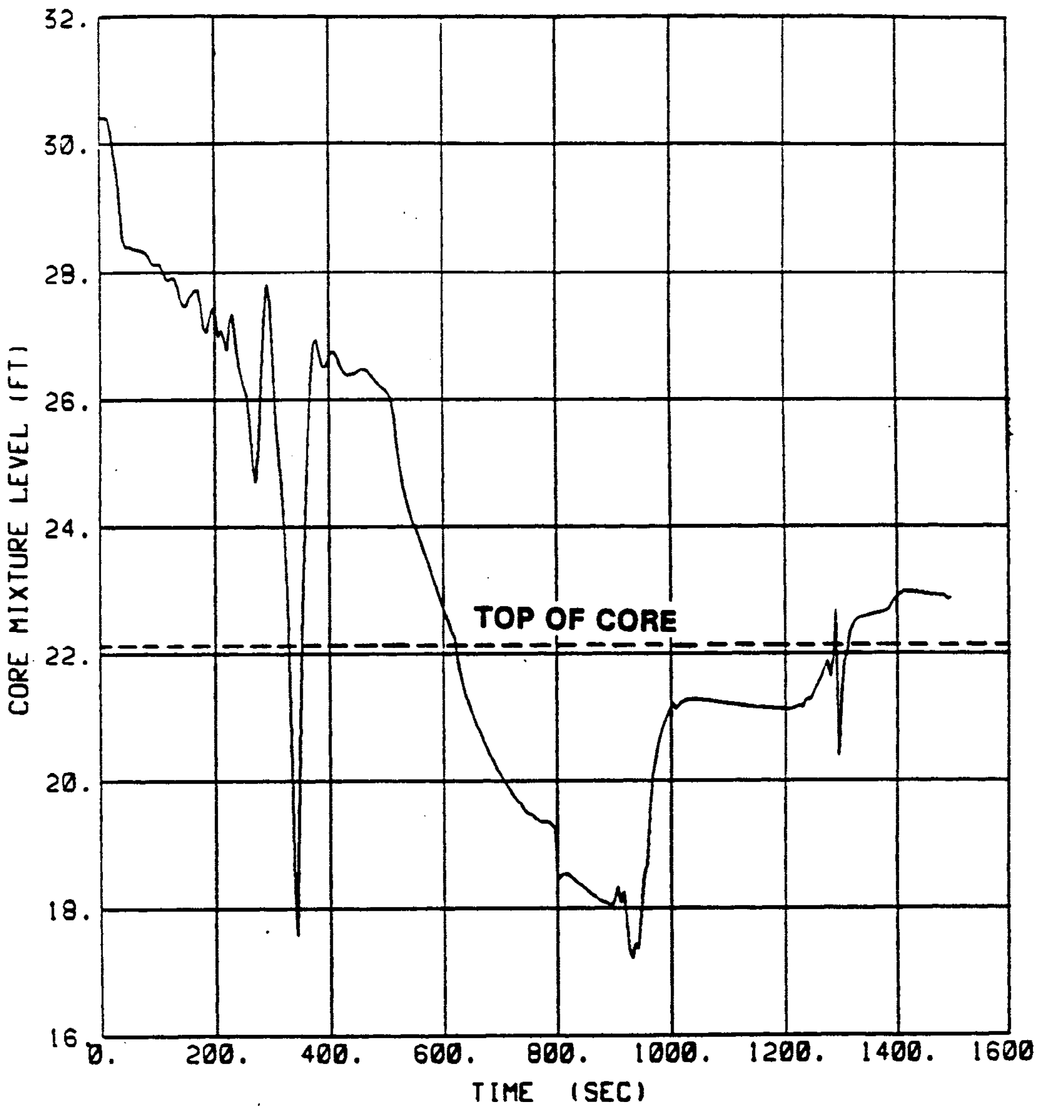
Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-40

CORE MIXTURE LEVEL

(2-INCH BREAK)



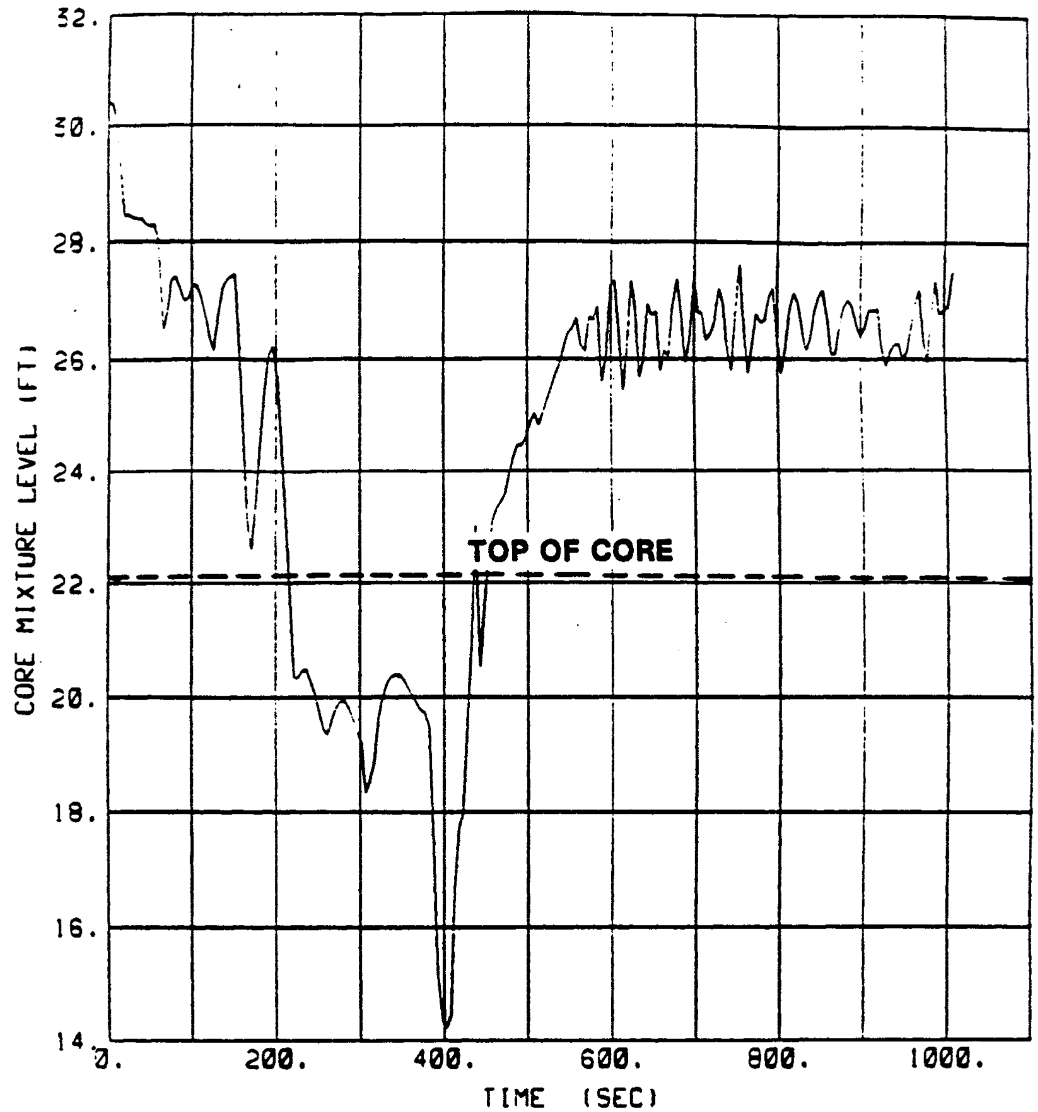
Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-41

CORE MIXTURE LEVEL

(4-INCH BREAK)



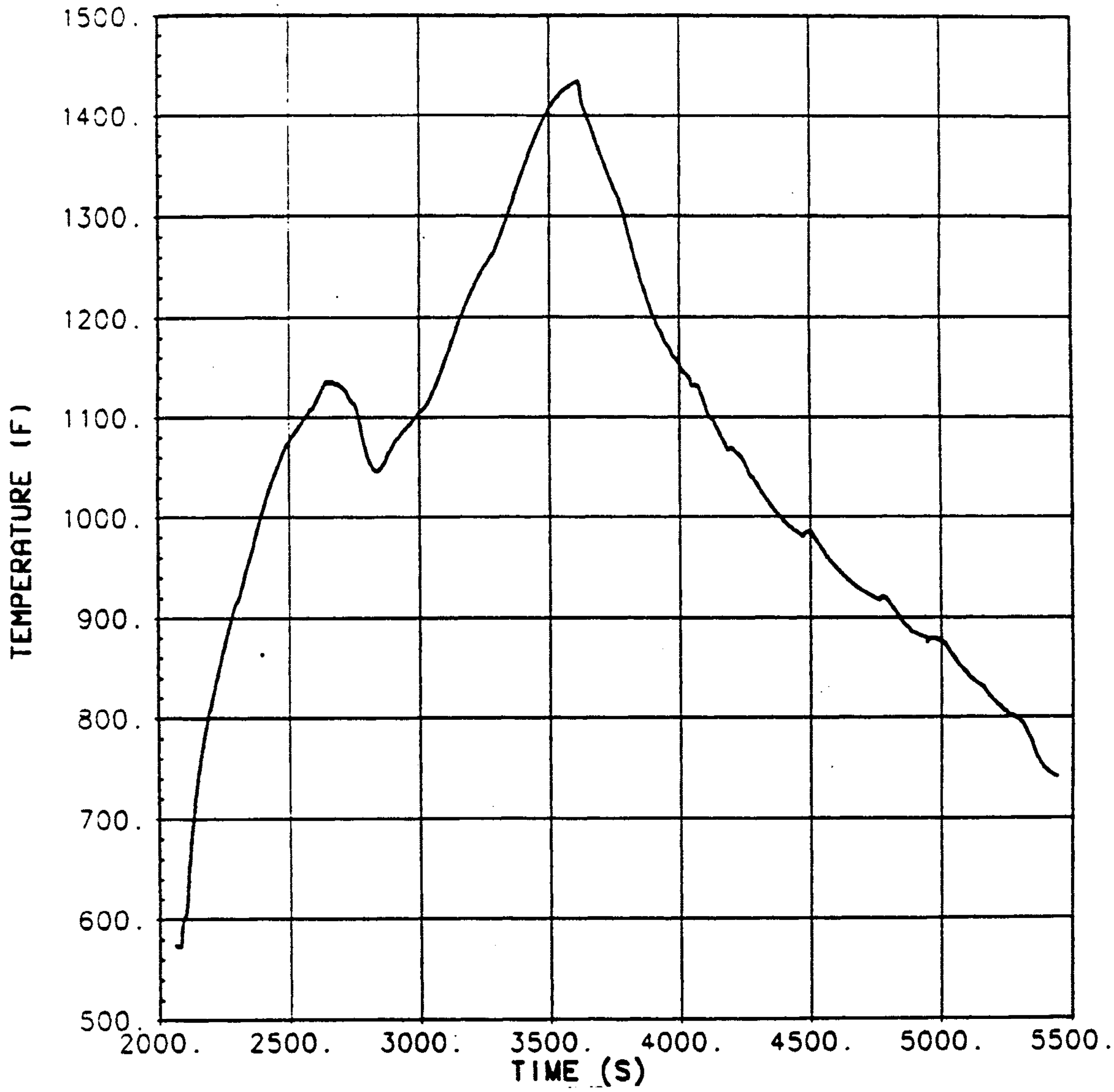
Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-42

CORE MIXTURE LEVEL

(6-INCH BREAK)

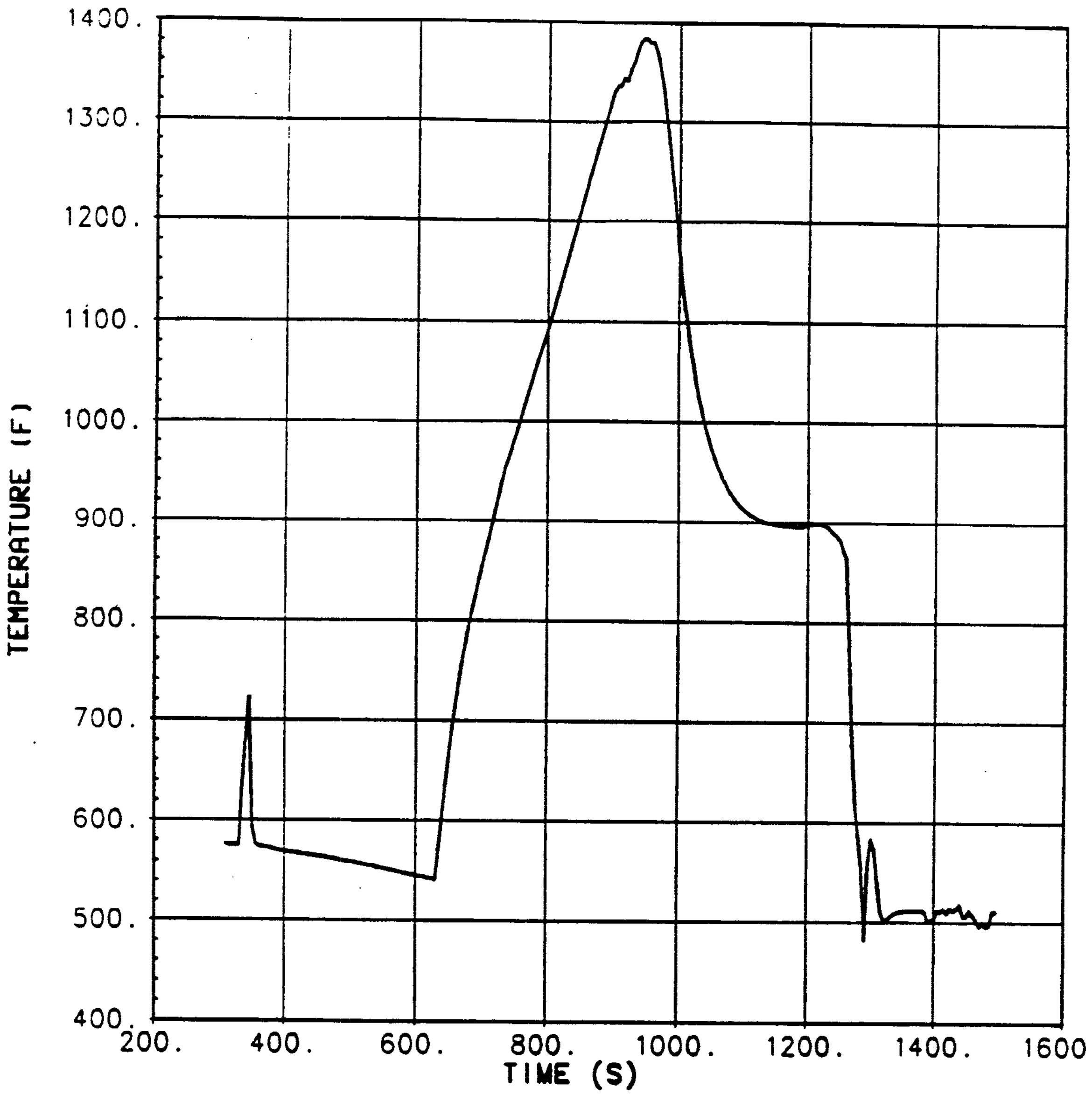


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-43

CLAD TEMPERATURE TRANSIENT
HOT ROD
(2-INCH BREAK)

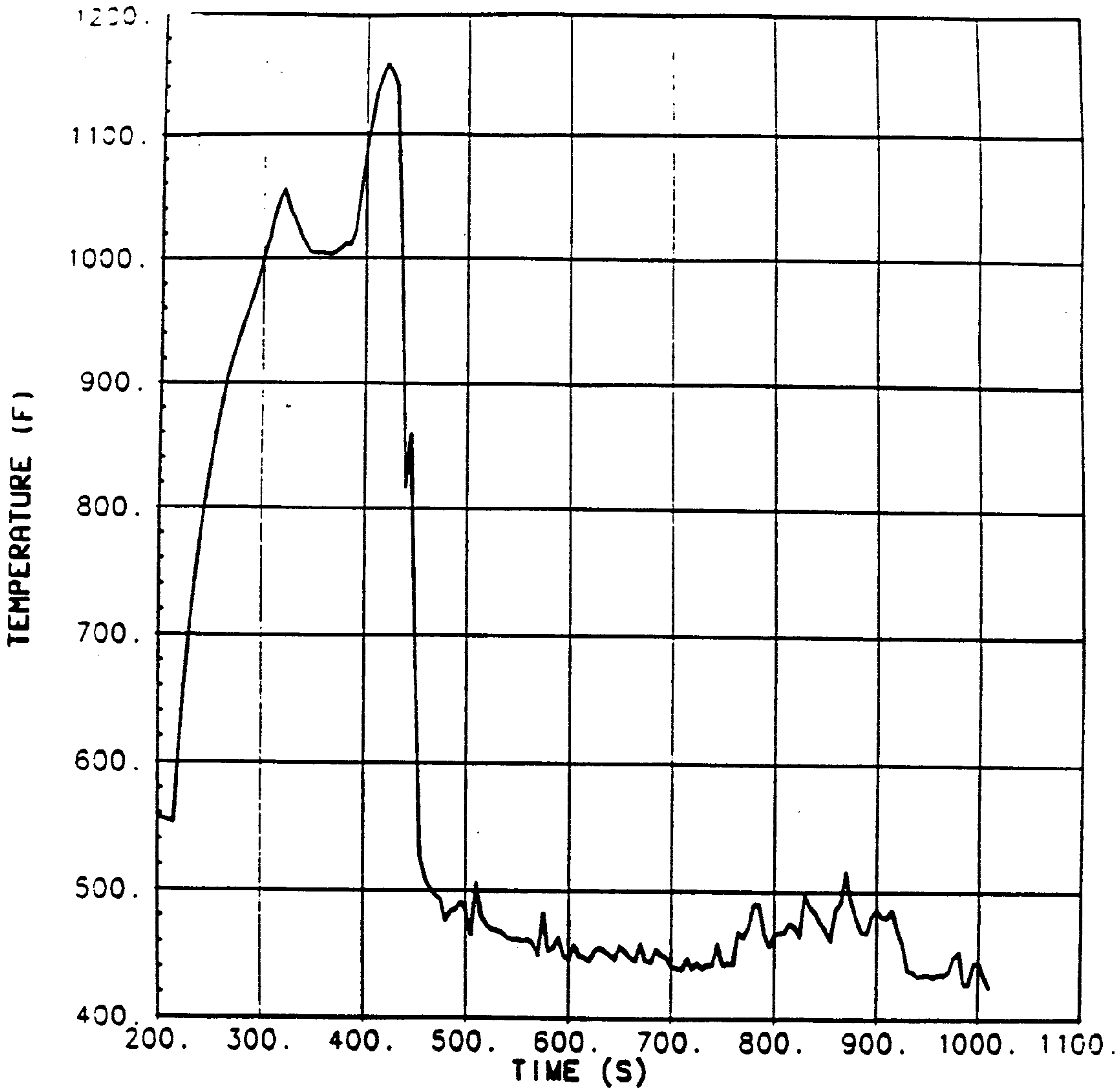


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-44

CLAD TEMPERATURE TRANSIENT
HOT ROD
(4-INCH BREAK)

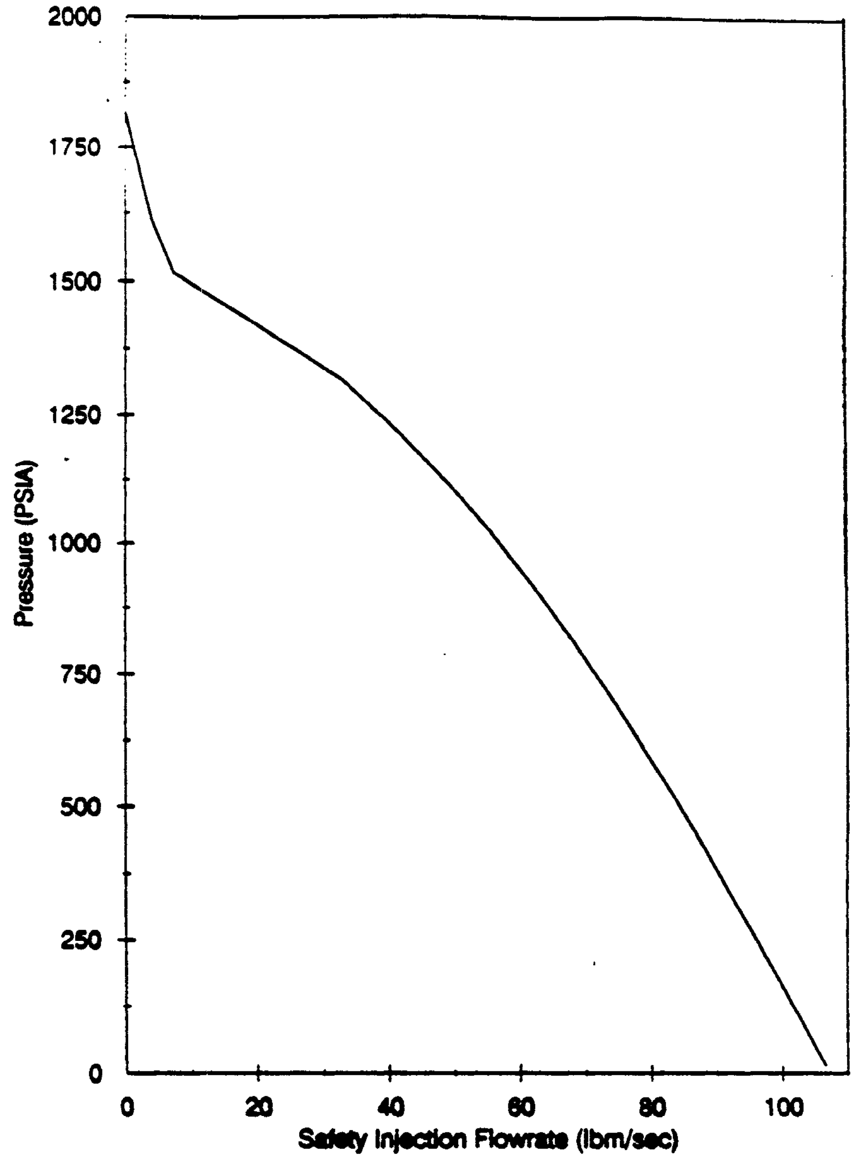


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-45

CLAD TEMPERATURE TRANSIENT
HOT ROD
(6-INCH BREAK)



Rev. 7

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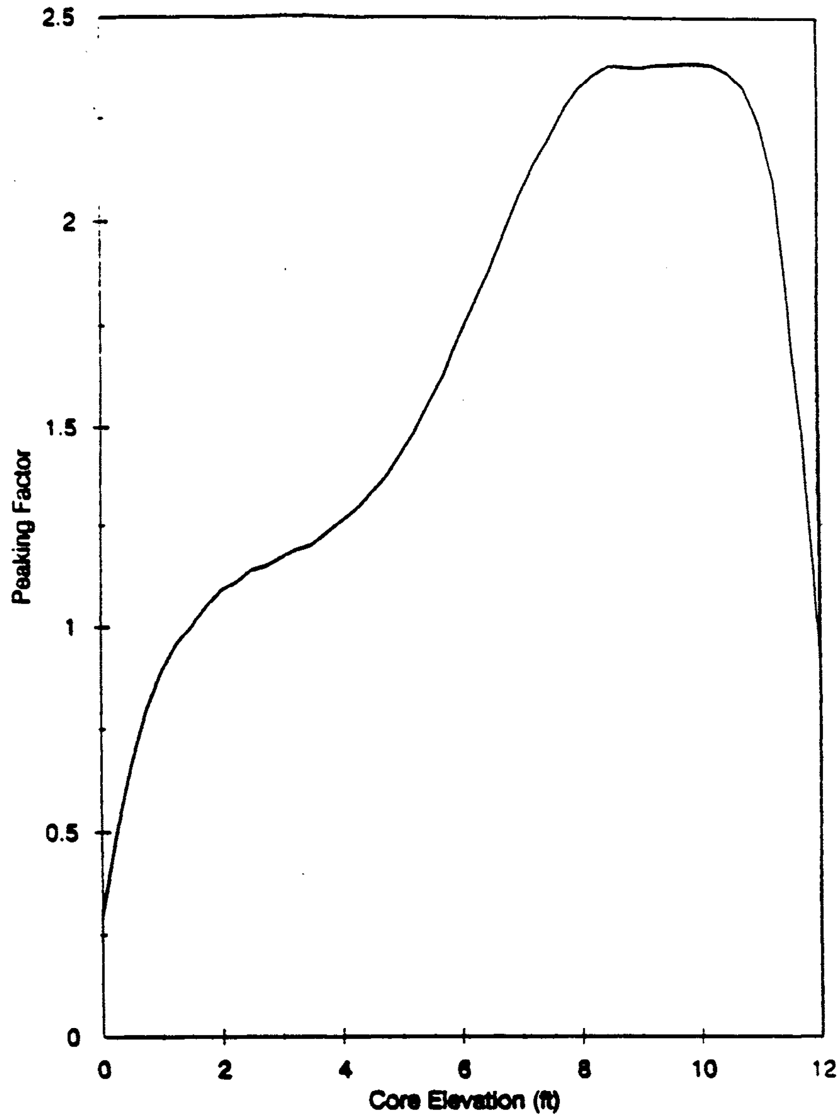
FIGURE 15.6-46

SAFETY INJECTION FLOWRATE

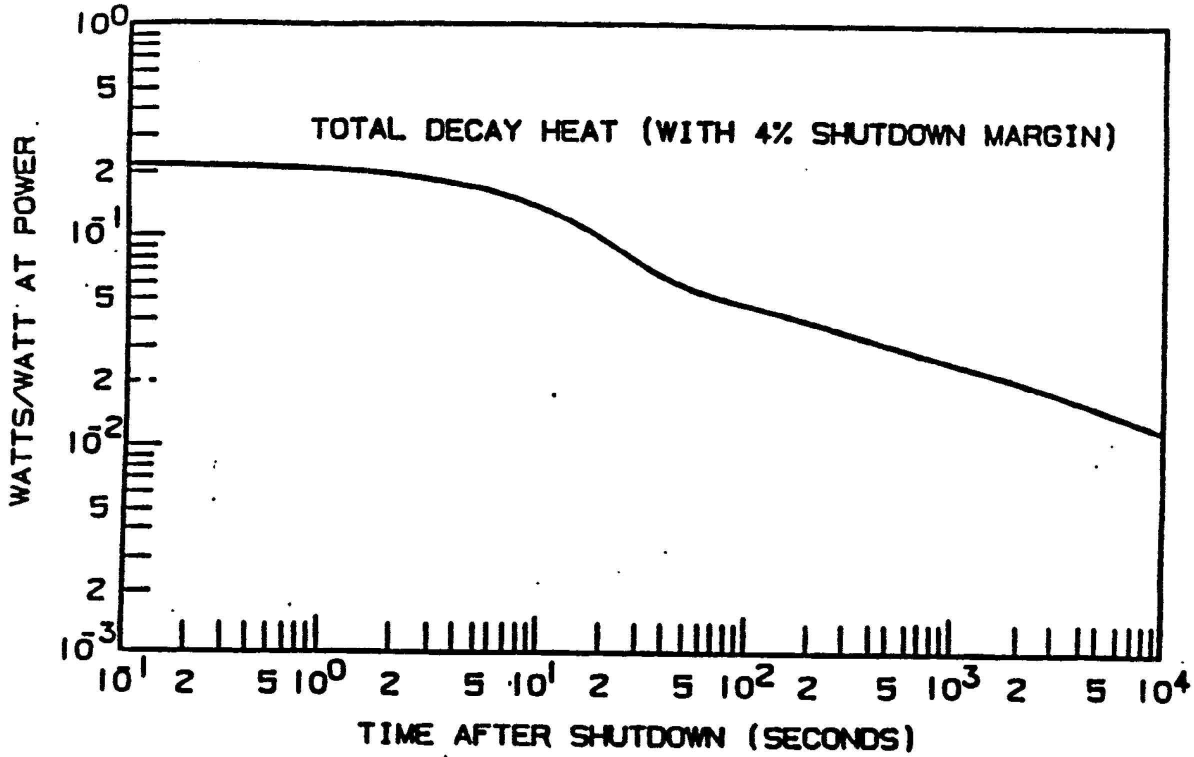


Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-47

SMALL BREAK POWER DISTRIBUTION



Rev. 7

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-48

CORE POWER (INCLUDING RESIDUAL FISSION) AFTER REACTOR TRIP (APPLIES TO ALL SMALL BREAKS)

15.7 RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT

This class of accident can be caused by any of the following events:

a. Radioactive gas waste system leak or failure - this is an ANS Condition III event.

b. Radioactive liquid waste system leak or failure - this is an ANS Condition III event.

c. Postulated radioactive release due to liquid tank failures - this is an ANS Condition IV event.

d. Fuel handling accident - this is an ANS Condition IV event.

e. Spent fuel cask drop accidents - this is an ANS Condition III event.

All of the accidents in this section have been analyzed. It has been determined that the most severe radiological consequences will result from the fuel handling accident analyzed in Section 15.7.4.

15.7.1 RADIOACTIVE WASTE GAS DECAY TANK FAILURE

15.7.1.1 Identification of Causes

This accident is an infrequent fault. Its consequences are considered in this section. The accident is defined as an unexpected and uncontrolled release of radioactive xenon and krypton fission product gases stored in a waste gas decay tank as a consequence of a failure of a single gas tank or associated piping.

15.7.1.2 Sequence of Events and System Operations

During a refueling shutdown, the radioactive gases are stripped from the primary coolant and are stored in the gas decay tanks. After the transfer has been completed, the tank is assumed to fail. This releases all of the contents of the tank to the radwaste building. Also, since the tanks are isolated from each other, the only radioactivity released is from the failed tank. For conservatism, the tank is assumed to fail after 40 years, releasing the peak inventory expected in the tank.

15.7.1.3 Core and System Performance

This accident occurs when the reactor is in the shutdown condition. There is no impact on the core or its system performance.

15.7.1.4 Barrier Performance

The only barrier between the released activity and the environment is the radwaste building. During the course of this accident, the radwaste building is assumed to remain intact. This means that the only method of release is through the radwaste building ventilation system.

15.7.1.5 Radiological Consequences

15.7.1.5.1 Method of Analysis

15.7.1.5.1.1 Physical Model

Radioactive waste gas decay tanks are used in the design to permit the decay of radioactive gases as a means of reducing or preventing the release of radioactive materials to the atmosphere. To evaluate the radiological consequences of the gaseous waste processing system, it is postulated that there is an accidental release of the contents of one of the waste gas decay tanks resulting from a rupture of the tank or from another cause, such as operator error or valve malfunction. The gaseous waste processing system is so designed that the tanks are isolated from each other during use, limiting the quantity of gas released in the event of an accident by preventing the flow of radioactive gas between the tanks.

The principal radioactive components of the waste gas decay tanks are the noble gases krypton and xenon, the particulate daughters of some of the krypton and xenon isotopes, and trace quantities of halogens. The maximum amount of waste gases stored in any one tank occurs after a refueling shutdown, at which time the waste gas decay tanks store the radioactive gases stripped from the reactor coolant.

The maximum content of a gas decay tank which is conservatively assumed for the purpose of computing the noble gas inventory available for release given in Table 15.7-3. Rupture of the waste gas decay tank is assumed to occur immediately upon completion of the waste gas transfer, releasing the entire contents of the tank to the radwaste building. For the purposes of evaluating the accident, it is assumed that all the activity is released directly to the environment during the 2-hour period immediately following the accident.

15.7.1.5.1.2 Assumptions and Conditions

The major assumptions and parameters assumed in the analysis are itemized in Table 15.7-3.

In the evaluation of the waste gas decay tank rupture, the fission product accumulation and release assumptions of Regulatory Guide 1.24 have been used. Table 15.7-1 provides a comparison of the assumptions used in the analysis to those of Regulatory Guide 1.24. The assumptions related to the release of radioactive gases from the postulated rupture of a waste gas decay tank are:

a. The reactor has been operating at full core power with 1 percent defective fuel, and a shutdown to cold condition has been conducted prior to the accident.

b. All noble gas activity has been removed from the reactor coolant system and transferred to the gas decay tank that is assumed to fail.

c. The maximum content of the waste gas decay tank was conservatively assumed to calculate the isotopic activities given in Table 15.7-3 for the accumulated radioactivity in the gaseous waste processing system after 40 years' operation and immediately following plant shutdown. The source term determination does take into account degassing of the reactor coolant system at shutdown.

d. The failure is assumed to occur immediately upon completion of the waste gas transfer, releasing the entire contents of the tank to the radwaste building.

e. The dose is calculated as if the release were from the radwaste building at ground level during the 2-hour period immediately following the accident. No credit for radioactive decay is taken.

15.7.1.5.1.3 Mathematical Models Used in the Analysis

The 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 the onsite meteorological measurement programs described in Section 2.3.

c. The thyroid inhalation and total-body immersion doses to a receptor at the exclusion area boundary or outer boundary of the low-population zone were analyzed, using the models described in Appendix 15A, Sections 15A.2.4 and 15A.2.5, respectively.

15.7.1.5.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

For the purposes of evaluating the radiological consequences due to the postulated waste gas decay tank rupture, the resultant activity is conservatively assumed to be released directly to the environment during the 2-hour period immediately following the occurrence of the accident. This is a considerably higher release rate than that based on the actual building exhaust ventilation rate. Therefore, the results of the analysis are based on the most conservative pathway available.

15.7.1.5.2 Identification of Uncertainties and Conservatisms in the Analysis

The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of a waste gas decay tank rupture result from assumptions made involving the release of the waste gas from the decay tank and the meteorology present at the site during the course of the accident.

a. The accumulated activity in the gaseous waste processing system after 40 years' operation and immediately following plant shutdown with zero decay assumed to be in the waste gas decay tank is based on 1 percent failed fuel, which is eight times greater than that assumed under normal operating conditions.

b. It is assumed that the waste gas decay tank fails immediately after the transfer of the noble gases from the reactor coolant to the waste gas decay tank is complete. These assumptions result in the greatest amount of noble gas activity available for release to the environment.

c. The noble gas activity contained in the ruptured waste gas decay tank was assumed to be released over a 2-hour period immediately following the accident. This is a conservative assumption. If the contents of the tank were assumed to mix uniformly with the volume of air within the radwaste building where the decay tanks are located, then, using the actual building exhaust ventilation rate, a considerable amount of holdup time would be gained. However, no credit for radioactive decay is taken. This reduces the amount of noble gas activity released to the environment due to natural decay. Also no credit for iodine removal by the non-safety grade radwaste building HVAC charcoal adsorbers has been taken.

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 meteorological conditions assumed will 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, will be conservative.

15.7.1.5.3 Conclusions

15.7.1.5.3.1 Filter Loading

Since the accumulated iodine activity in the waste gas decay tanks is negligible, filter loading due to a waste gas decay tank rupture does not establish the necessary design margin for the radwaste building exhaust or the control room intake filters. Hence, the respective filter loadings were not evaluated.

15.7.1.5.3.2 Dose to Receptor at the Exclusion Area Boundary and the Low-Population Zone Outer Boundary

The radiological consequences resulting from the occurrence of a postulated waste gas decay tank rupture have been conservatively analyzed, using assumptions and models described in previous sections.

The total-body dose due to immersion and the thyroid dose due to inhalation have been analyzed for the 0-2-hour dose at the exclusion area boundary and for the duration of the accident at the low-population zone outer boundary. The results are listed in Table 15.7-4. The resultant doses are a small fraction (≤ 10 percent) of the exposure limits set forth in 10 CFR 100.

15.7.2 RADIOACTIVE LIQUID WASTE SYSTEM LEAK OR FAILURE

15.7.2.1 <u>Identification of Causes</u>

This is an infrequent fault because, although it is unlikely to happen, the potential for release of significant amounts of radioactivity is present. The accident may be caused by an equipment malfunction or tank failure or rupture.

15.7.2.2 Sequence of Events and System Operation

The radioactive liquid tank is assumed to fail. This releases a maximum of 80 percent of the tank capacity to the equipment compartment.

15.7.2.3 Core and System Performance

This accident does not affect the core or the core system performance.

15.7.2.4 Barrier Performance

There are no barriers to the release of radioactivity from the radwaste building.

15.7.2.5 Radiological Consequences

15.7.2.5.1 Method of Analysis

15.7.2.5.1.1 Physical Model

The liquid radwaste tanks are used as a means of collecting waste to be: 1) processed through the liquid radwaste system, 2) pumped to the Solid Radwaste System, or 3) discharged from the plant. To evaluate the radiological consequences of the liquid waste processing system, it is postulated that there is an accidental release of the contents of one of the tanks.

Table 11.1-6 provides an inventory and the concentrations of stored radioactivity in all the liquid tanks. In the analyses, it is assumed that the liquid contents of the tank are released to the radwaste building and, subsequently, the airborne activity is released to the environment during the 2-hour period immediately following the tank failure.

Two tanks have been analyzed for this accident, and the radiological consequences for both tanks are provided. The boron recycle holdup tank was selected because it contained the maximum total inventory, and therefore, the highest whole body exposures. In addition a hypothetical tank containing the maximum possible amount of iodine was analyzed in order to determine the thyroid exposures. Although the primary spent resin tank contains the highest inventory of airborne and soluble iodine, it is considered extremely unlikely that all the iodine activity will become airborne in case the tank fails. The assumptions, conditions, and mathematical models described in this section are identical for both tanks, except as stated.

15.7.2.5.1.2 Assumptions and Conditions

The major assumptions and parameters assumed in this analysis are listed below and in Tables 15.7-5 and 15A-1:

a. The isotopic inventory of the ruptured tank is taken from Table 11.1-6, and is based on 1-percent failed fuel. The isotopic inventory of the ruptured hypothetical tank is also based on 1-percent failed fuel, but was calculated assuming that all of the iodine in the streams entering the liquid radwaste system are concentrated in a hypothetical tank where the only means of depletion is radioactive decay. These input streams are shown on USAR Figure 11.1A-2 sheets 2, 3, and 4.

b. The tank failure is assumed to occur when the contents of the tank are at a maximum.

c. The doses are calculated as if the release were from the radwaste building at ground level during the 2-hour period immediately following the accident. No credit is taken for radioactive decay during holdup in the tank or in transit to the site boundary.

d. For the boron recycle holdup tank 100 percent of all noble gas activity in the tank is released while 10 percent of the iodine activity is released as airborne activity. 100 percent of all iodine activity is released from the hypothetical liquid waste tank.

e. Credit for iodine removal by non-safety grade radwaste building HVAC charcoal adsorber is not taken.

15.7.2.5.1.3 Mathematical Models Used in the Analysis

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 the onsite meteorological measurement programs described in Section 2.3, and are provided in Table 15A-2.

c. The thyroid inhalation dose and total-body immersion dose to a receptor at the exclusion area boundary or outer boundary of the low-population zone were analyzed, using the models described in Appendix 15A, Sections 15A.2.4 and 15A.2.5, respectively.

15.7.2.5.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

For the purposes of evaluating the radiological consequences due to the postulated liquid radwaste tank rupture, the resultant activity is conservatively assumed to be released directly to the environment during the 2-hour period immediately following the occurrence of the accident. This is a considerably higher release rate than that based on the actual building exhaust ventilation rate. Therefore, the results of the analysis are based on the most conservative pathway available.

15.7.2.5.2 Identification of Uncertainties and Conservatisms in the Analysis

The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of the liquid radwaste tank rupture result from assumptions made involving the release of the radioactivity from the tanks and the meteorology assumed for the site.

a. It was assumed that the liquid radwaste tank fails when the inventory in the tank is a maximum. This assumption results in the greatest amount of activity available for release to the environment.

b. The contents of the ruptured tank are assumed to be released over a 2-hour period immediately following the accident. If the contents of the tank were assumed to mix uniformly with the volume of air within the radwaste building where the tanks are located, then, using the actual building exhaust ventilation rate, a considerable amount of holdup time would be gained. This reduces the amount of activity released to the environment due to the natural decay. Also, no credit for iodine removal by the radwaste building HVAC charcoal adsorbers is taken.

c. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that meteorological conditions assumed will be present during the course of the accident for any extended period of time.

d. A tank is assumed to have collected liquid waste based on operation at 100-percent power with 1 percent failed fuel for an extended period of time, which is eight times higher than under normal operating conditions.

15.7.2.5.3 Conclusions

15.7.2.5.3.1 Filter Loadings

The filter loading due to a liquid radwaste tank rupture does not establish the necessary design margin for the control room intake filters. Thus, the filter loading was not evaluated.

15.7.2.5.3.2 Doses to Receptor at the Exclusion Area Boundary and the Low-Population Zone Outer Boundary

The radiological consequences resulting from the occurrence of a postulated liquid radwaste tank rupture have been conservatively analyzed, using assumptions and models described in previous sections.

The total-body dose due to immersion and the thyroid dose due to inhalation have been analyzed for the 0-2-hour dose at the exclusion area boundary and for the duration of the accident at the low-population zone outer boundary. The results are listed in Table 15.7-6. The resultant dose is a small fraction ($\leq$ 10 percent) of the exposure limits set forth in 10 CFR 100.

15.7.3 POSTULATED RADIOACTIVE RELEASE DUE TO LIQUID TANK FAILURES

This analysis is presented in Section 2.4.13.3.

15.7.4 FUEL HANDLING ACCIDENTS

The postulated fuel handling accident has been analyzed for two cases: Case 1, a fuel handling accident outside the containment, and Case 2, a fuel handling accident inside the reactor building.

15.7.4.1 Identification of Causes and Accident Description

The accident is defined as the dropping of a spent fuel assembly onto the fuel storage area floor or refueling pool floor, resulting in the rupture of the cladding of all the fuel rods in the assembly despite many administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures.

15.7.4.2 Sequence of Events and Systems Operations

The first step in fuel handling is the achievement of plant cold safe shutdown of the reactor. After a radiation survey of the containment, the disassembly of the reactor vessel is started. After disassembly is complete, the first fuel handling is started. It is estimated that the earliest time to first fuel transfer after shutdown is 76 hours.

The postulated fuel handling accident is assumed to occur during a core offload at least 76 hours after shutdown in either the reactor containment building, or in the fuel building subsequent to the transfer of a fuel assembly through the fuel storage pool transfer gate and prior to placement in a fuel storage pool storage rack designated location.

15.7.4.3 Core and System Performance

As fuel damage occurs outside the reactor vessel in either the reactor containment building or fuel building, a postulated fuel handling accident does not impair the safe operation of the reactor or its associated systems.

15.7.4.4 Barrier Performance

A barrier between the released activity and the environment is the reactor building and the fuel building. Since these buildings are designed seismic Category I, it is safe to assume that during the course of a fuel handling accident their integrity is maintained. This means that the pathway for release of radioactivity for a postulated accident in the fuel building is initially via auxiliary/fuel building normal exhaust system. After it is isolated on a high radiation signal, the release pathway is via the ESF emergency filtration system. For a postulated accident in the reactor building, since the containment personnel airlock (PAL) doors are allowed to be open during core alterations or movement of irradiated fuels, portion of the gaseous effluent escaping from the refueling water pool in the Reactor Containment Building could be released to the environment via the open personnel hatch until one of the containment PAL doors is closed. The fuel storage pool and the refueling pool provide minimum decontamination factors of 100 for iodine.

15.7.4.5 Radiological Consequences

15.7.4.5.1 Method of Analysis

15.7.4.5.1.1 Physical Model

The possibility of a fuel-handling accident is remote because of the many administrative controls and physical limitations imposed on the fuel-handling operations (refer to Section 9.1.4). All refueling operations are conducted in accordance with prescribed procedures.

When transferring irradiated fuel from the core to the fuel storage pool for storage, the reactor cavity and refueling pool are filled with borated water at a boron concentration equal to that in the fuel storage pool, which ensures subcritical conditions in the core even if all rod cluster control (RCC) assemblies were withdrawn. After the reactor head and rod cluster control drive shafts are removed, fuel assemblies are lifted from the core, transferred vertically to the refueling pool, placed horizontally in a conveyor car and pulled through the transfer tube and canal, upended and transferred through the fuel storage pool transfer gate, then lowered into steel racks for storage in the fuel storage pool in a pattern which precludes any possibility of a criticality accident.

Fuel-handling manipulators and hoists are designed so that the fuel cannot be raised above a position that provides an adequate water shield depth for radiation protection of operating personnel.

The containment, fuel building, refueling cavity, refueling pool, and fuel storage pool are designed to seismic Category I requirements, which prevent the structures themselves from failing in the event of a safe shutdown earthquake. The spent fuel storage racks are also designed to prevent any credible external missile from reaching the stored irradiated fuel. The fuel-handling manipulators, cranes, trollies, bridges, and associated equipment above the water cavities through which the fuel assemblies move are designed to prevent this equipment from generating missiles and damaging the fuel. The construction of the fuel assemblies precludes damage to the fuel should portable or hand tools drop on an assembly.

The only time the postulated fuel-handling accident could occur is during the transfer of a fuel assembly from the core to its storage position in the fuel storage pool. The facility is designed so that heavy objects, such as the spent fuel shipping cask, cannot be carried over or tipped over onto the irradiated fuel stored in the fuel storage pool. Only one fuel assembly is handled at a time by the refueling machine, transfer system, or spent fuel pool bridge crane. Movement of equipment handling the fuel is kept at low speeds while exercising caution that the fuel assembly does not strike another object or structure during transfer from the core to its storage position. In the unlikely event that an assembly becomes stuck in the transfer tube, natural convection will maintain adequate cooling.

a. Reactor Building Accident

During fuel-handling operations, the containment is kept in an isolatable condition, with all penetrations to the outside atmosphere either closed or capable of being closed on an alarm signal from one of the redundant radiation monitors, indicating that radioactivity is above the prescribed limits.

In addition to the area radiation monitors in the containment, portable monitors capable of sounding audible alarms are to be located in the fuel-handling area. Should a fuel assembly be dropped and release activity above a prescribed level, the radiation monitors would sound an audible alarm, the containment would be isolated, and personnel would be evacuated.

The purge and vent lines are automatically closed on a containment isolation signal, thus minimizing the escape of any radioactivity. During movement of irradiated fuel and core alterations, both containment personnel airlock doors and the service air and breathing air containment penetrations are allowed to remain open. Stringent administrative controls are imposed to close the service air and breathing air valves and ensure at least one door will be available to perform its safety function, following evacuation in the event of an accident.

b. Fuel Building Accident

In the fuel building, a fuel assembly could be dropped in the transfer canal or in the fuel storage pool.

In addition to the area radiation monitor located on the bridge over the fuel storage pool, portable radiation monitors capable of emitting audible alarms are located in this area during fuel-handling operations. The doors in the fuel building are closed to maintain controlled leakage characteristics in the fuel storage pool region during operations involving irradiated fuel. Should a fuel assembly be dropped in the canal or in the pool and release radioactivity above a prescribed level, the radiation monitors sound an alarm,

If one of the redundant discharge vent radiation monitors indicates that the radioactivity in the vent discharge is greater than the prescribed levels, an alarm sounds and the auxiliary/fuel building normal exhaust is switched to the ESF Emergency Exhaust system to allow the spent fuel pool ventilation to exhaust through the ESF charcoal filters to remove most of the halogens prior to discharging to the atmosphere via the unit vent. The supply ventilation system servicing the fuel storage pool area is automatically shut down, thus ensuring controlled leakage to the atmosphere through charcoal adsorbers (refer to Section 9.4).

The probability of a fuel-handling accident is very low because of the safety features, administrative controls, and design characteristics of the facility, as previously mentioned.

15.7.4.5.1.2 Assumptions and Conditions

The major assumptions and parameters assumed in the analysis are itemized in Tables 15.7-7 and 15A-1.

In the evaluation of the fuel-handling accident, all the fission product release assumptions of Regulatory Guide 1.25 have been followed. Table 15.7-2 provides a comparison of the design to the requirements of Regulatory Guide 1.25. The following assumptions, related to the release of fission product gases from the damaged fuel assembly, were used in the analyses:

a. The dropped fuel assembly is assumed to be the assembly containing the peak fission product inventory. All the fuel rods contained in the dropped assembly are assumed to be damaged. In addition, for the analyses for the accident in the reactor building the dropped assembly is assumed to damage 20 percent of the rods of an additional assembly.

b. The assembly fission product inventories are based on a radial peaking factor of 1.65.

c. The accident occurs 76 hours after shutdown, which is the earliest time fuel-handling operations can begin. Radioactive decay of the fission product inventories was taken into account during this time period.

d. Only that fraction of the fission products which migrates from the fuel matrix to the gap and plenum regions during normal operation was assumed to be available for immediate release to the water following clad damage.

e. The gap activity released to the fuel pool from the damaged fuel rods consists of 10 percent of the total noble gases other than Kr-85, 30 percent of the Kr-85, and 10 percent of the total radioactive iodine other than I-131, 12 percent of the I-131, contained in the fuel rods at the time of the accident.

f. The pool decontamination factor is 1.0 for noble gases.

g. The effective pool decontamination factor is 100 for iodine assuming at least 23 feet of water above the top of the damaged fuel assemblies is maintained in the pool. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop.

h. The iodine above the fuel pool is assumed to be composed of 75 percent inorganic and 25 percent organic species.

i. The activity which escapes from the pool is assumed to be available for release to the environment in a time period of 2 hours.

j. No credit for decay or depletion during transit to the site boundary and outer boundary of the low-population zone is assumed.

k. No credit is taken for mixing or holdup in the fuel building atmosphere. The filter efficiency for the ESF emergency filtration system is assumed to be 82.5 percent which is based on the assumption of the failure of the humidity control system.

l. The fuel building is switched from the auxiliary/fuel building normal exhaust system to the ESF emergency exhaust system within one minute from the time the activity reaches the exhaust duct. The activity released before completion of the switchover is assumed to be discharged directly to the environment with no credit for filtration or dilution.

m. For the inside the reactor building case, the containment personnel airlock doors are assumed to be open at the time of the accident. For added conservatism, the gaseous effluent escaping from the refueling water pool in the Reactor Containment Building is assumed to be released immediately to the environment through the open personnel hatch and the adjacent Auxiliary Building without mixing in the surrounding atmosphere. The activity releases continue until the containment personnel airlock doors are closed (assumed to be accomplished within two hours). The Auxiliary Building atmosphere is normally exhausted through filter absorbers designed to remove iodine. However, no credit is taken for iodine removal by the atmosphere filtration system filters. It is also assumed that no containment coolers or hydrogen mixing fans are operating and 99.99% of the activity escaping from the pool to the containment building is released to the environment over a two-hour period following the accident.

15.7.4.5.1.3 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, Section 15A.2.

b. The atmospheric dispersion factors are calculated, based on the onsite meteorological measurements programs described in Section 2.3 and are provided in Table 15A-2.

c. The thyroid inhalation and total-body immersion doses to a receptor located at the exclusion area boundary and outer boundary of the low population zone are described in Appendix 15A, Sections 15A.2.4 and 15A.2.5, respectively.

15.7.4.5.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

For evaluating the radiological consequences due to the postulated fuel-handling accident in the fuel building and reactor building, the resultant activity is conservatively assumed to be released to the environment during the 0-2-hour period immediately following the occurrence of the accident. This is a considerably higher release rate than that based on the actual ventilation rate. Therefore, the results of the analysis are based on the most conservative pathway available.

15.7.4.5.2 Identification of Uncertainties and Conservatisms in Analysis

The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of a fuel-handling accident result from assumptions made involving the amount of fission product gases available for release to the environment and the meteorology present at the site during the course of the accident. The most significant of these assumptions are:

a. It is assumed in the analysis that all the fuel rods in the dropped assembly are damaged. This is a highly conservative assumption since, transferring fuel under strict fuel handling procedures, only under the worst possible circumstances could the dropping of a spent fuel assembly result in damage to all the fuel rods contained in the assembly.

b. The fission product gap inventory in a fuel assembly is dependent on the power rating of the assembly and the temperature of the fuel. It has been conservatively assumed that the core has been operating at 100 percent for the entire burnup period. The gap activities are listed in Table 15A-3.

c. Iodine removal from the released fission product gas takes place as the gas rises to the pool surface through the body of liquid in the fuel storage and refueling water pools. The extent of iodine removal is determined by mass transfer from the gas phase to the surrounding liquid and is controlled by the bubble diameter and contact time of the bubble in the solution. The values used in the analysis result in a release of activity approximately a factor of 5 greater than anticipated.

   Radioactive material from the refueling water pool in the reactor containment building is assumed to be released directly to the environment through the open personnel hatch and the adjacent auxiliary building, without mixing in the surrounding atmosphere. Radioactive material is assumed to be released from the auxiliary building or from the fuel building over a two-hour time period.

d. The ESF emergency filtration system charcoal filters are known to operate with at least a 99-percent efficiency. This means a further reduction in the iodine concentrations and thus a reduction in the thyroid doses at the exclusion area boundary and the outer boundary of the low-population zone for the fuel handling accident in the fuel building.

e. The containment purge exhaust system has charcoal adsorber units which filter any containment purge release. However, no credit has been taken for its capability (90-percent efficiency, minimum) since these units are not specifically designed to seismic Category I criteria. It is expected that for any event which would produce a catastrophic failure of the charcoal adsorber unit to the extent that its filtering capability would be negated would also result in the purge exhaust fan becoming inoperable. Therefore, failure within the purge exhaust system would terminate any high volume release from the containment. In fact, the purge exhaust fan is considerably more likely to be inoperable following any postulated event than the failure of a passive charcoal adsorber unit. Thus, although no credit in the analysis has been given for the normal purge exhaust filters, any release prior to containment isolation would be filtered.

f. There is also conservatism in the time to first fuel transfer. Despite the fact that fuel could be transferred at 76 hours, it is probable that fuel handling will begin sometime later.

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 meteorological conditions assumed will 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.

15.7.4.5.2.1 Filter Loadings

The ESF filtration systems which function to limit the consequences of a fuel-handling accident in the fuel building are the ESF emergency filtration system and the control room filtration system.

The activity loadings on the control room charcoal adsorbers as a function of time have been evaluated for the loss-of-coolant accident, Section 15.6.5. Since these filters are capable of accommodating the design basis LOCA fission product iodine loadings, more than adequate design margin is available with respect to postulated fuel-handling accident releases.

The activity loadings on the ESF filtration system charcoal adsorbers have been evaluated in accordance with Regulatory Guide 1.52, which limits the maximum loading to 2.5 mg of iodine per gram of activated charcoal.

15.7.4.5.2.2 Doses to Receptor at the Exclusion Area Boundary and Low-Population Zone Outer Boundary

The potential radiological consequences resulting from the occurrence of a postulated fuel-handling accident occurring in the fuel building and in the reactor building have been conservatively analyzed, using assumptions and models described in previous sections. The total-body dose due to immersion from direct radiation and the thyroid dose due to inhalation have been analyzed for the 0-2-hour dose at the exclusion area boundary and for the duration of the accident (0 to 2 hours) at the low-population zone outer boundary. The results are listed in Table 15.7-8. The resultant doses are well within the guideline values of 10 CFR 100.

15.7.5 SPENT FUEL CASK DROP ACCIDENTS

The design of the spent fuel cask handling equipment is such that no cask could be dropped more than the equivalent of 30 feet in the air. Therefore, no cask rupture will occur and thus no radioactivity will be released. Refer to Section 9.1.4 for a description of the spent fuel shipping procedures.

TABLE 15.7-1

DESIGN COMPARISON TO THE REGULATORY POSITIONS OF REGULATORY GUIDE 1.24 "ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL CONSEQUENCES OF A PRESSURIZED WATER REACTOR RADIOACTIVE GAS STORAGE TANK FAILURE" REVISION 0, DATED MARCH 23, 1972

Regulatory Guide 1.24 Position	Design
1. The assumptions related to the release of radioactive gases from the postulated failure of a gaseous waste storage tank are:	
a. The reactor has been operating at full power with one percent defective fuel and a shutdown to cold condition has been conducted near the end of an equilibrium core cycle. As soon as possible after shutdown, all noble gases have been removed from the primary cooling system and transferred to the gas decay tank that is assumed to fail.	1.a Complies.
b. The maximum content of the decay tank assumed to fail should be used for the purpose of computing the noble gas inventory in the tank. Radiological decay may be taken into account in the computation only for the minimum time period required to transfer the gases from the primary system to the decay tank.	1.b Complies.
c. The failure is assumed to occur immediately upon completion of the waste gas transfer, releasing the entire contents of the tank to the building. The assumption of the release of the noble gas inventory from only a single tank is based on the premise that all gas decay tanks will be isolated from each other whenever they are in use.	1.c Complies.

TABLE 15.7-1 (Sheet 2)

Regulatory Guide 1.24 Position	Design
d. All of the noble gases are assumed to leak out of the building at ground level over a 2-hour time period.	1.d Complies.
2. The atmospheric diffusion assumptions for ground level releases are: a. The basic equation for atmospheric diffusion from a ground level point source is: $\chi/Q = \frac{1}{\pi u \sigma_y \sigma^z}$ Where: χ = the short term average centerline value of the ground level concentration (curies/m^3) Q = amount of material released (curies/sec) u = windspeed (meters/sec) σ_y = the horizontal standard deviation of the plume (meters) [See Figure V-1, page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.]	2. Short-term atmospheric dispersion factors corresponding to a ground level release and accident conditions were calculated based on on-site meteorological measurement programs described in Section 2.3. The dispersion factors are in compliance with the methodology described in Regulatory Guide 1.145 and represent the worst of the 5 percent overall site meteorology and the 0.5 percent worst sector meteorology.

TABLE 15.7-1 (Sheet 3)

Regulatory Guide 1.24 Position	Design
σ_z = the vertical standard deviation of the plume (meters) [See Figure V-2, page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.]	
b. For ground level releases, atmospheric diffusion factors[1] used in evaluating the radiological consequences of the accident addressed in this guide are based on the following assumptions: (1) windspeed of 1 meter/sec; (2) uniform wind direction (3) Pasquill diffusion category F.	
c. Figure 1 is a plot of atmospheric diffusion factors (χ/Q) versus distance derived by use of the equation for a ground level release given in regulatory position 2.a. above under the meteorological conditions given in regulatory position 2.b. above.	
3. The following assumptions and equations may be used to obtain conservative approximations of external whole body dose from radioactive clouds: a. External whole body doses are calculated using "Infinite Cloud" assumptions, i.e., the dimensions of the cloud are assumed to be large compared to the distances that the gamma rays and beta particles travel.	3. Dose factors given in Regulatory Guide 1.109 for noble gases and iodine thyroid dose factors; iodine whole body dose factors were calculated with 5 cm body tissued attenuation; see Table 15A-4.

TABLE 15.7-1 (Sheet 4)

Regulatory Guide 1.24 Position	Design
The dose at any distance from the reactor is calculated based on the maximum ground level concentration at that distance.	
For an infinite uniform cloud containing curies of beta radioactivity per cubic meter, the beta dose rate in air at the cloud center is:[2]	
$_{\beta}D'_{\infty} = 0.457\ \overline{E}_{\beta}\chi$	
Where:	
$_{\beta}D'_{\infty}$ = beta dose rate from an infinite cloud (rad/sec)	
$\overline{E}_{\beta}$ = average beta energy per disintegration (Mev/dis)	
χ = concentration of beta or gamma emitting isotope in the cloud (curie/m^3)	
Because of the limited range of beta particles in tissue, the surface body dose rate from beta emitters in the infinite cloud can be approximated as being one-half this amount or:	
$_{\beta}D'_{\infty} = 0.23\ \overline{E}_{\beta}\chi$	
For gamma emitting material the dose rate in air at the cloud center is:	
$_{\gamma}D'_{\infty} = 0.507\overline{E}_{\gamma}\chi$	

TABLE 15.7-1 (Sheet 5)

Regulatory Guide 1.24 Position	Design
Where: $_{\gamma}D'_{\infty}$ = gamma dose rate from an infinite cloud (rad/sec) $\overline{E}_{\gamma}$ = average gamma energy per disintegration (Mev/dis) However, because of the presence of the ground, the receptor is assumed to be exposed to only one-half of the cloud (semi-infinite) and the equation becomes: $_{\gamma}D' = 0.25\ \overline{E}_{\gamma}\chi$ Thus, the total beta or gamma dose to an individual located at the center of the cloud path may be approximated as: $_{\beta}D_{\infty} = 0.23\ \overline{E}_{\beta}\Psi$ or $_{\gamma}D_{\infty} = 0.25\ \overline{E}_{\gamma}\Psi$ Where is the concentration time integral for the cloud (curie sec/m3).	
b. The beta and gamma energies emitted per disintegration, as given in Table of Isotopes,3 are averaged and used according to the methods described in ICRP Publication 2.	

TABLE 15.7-1 (Sheet 6)

Regulatory Guide 1.24 Position	Design
[1]These diffusion factors should be used until adequate site meteorological data are obtained. In some cases, available information on such site conditions as meteorology, topography and geographical location may dictate the use of more restrictive parameters to insure a conservative estimate of potential offsite exposures.	
[2]Meteorology and Atomic Energy - 1968, Chapter 7.	
[3]C. M. Lederer, J. M. Hollander, and I. Perlman, Table of Isotopes, Sixth Edition (New York: John Wiley and Sons, Inc., 1967).	

TABLE 15.7-2

DESIGN COMPARISON TO THE REGULATORY POSITIONS OF REGULATORY GUIDE 1.25 "ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT IN THE FUEL HANDLING AND STORAGE FACILITY FOR BOILING AND PRESSURIZED WATER REACTORS" REVISION 0, DATED MARCH 23, 1972

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
1. The assumptions[1] related to the release of radioactive material from the fuel and fuel storage facility as a result of a fuel handling accident are:		
a. The accident occurs at a time after shutdown identified in the technical specifications as the earliest time fuel handling operations may begin. Radioactive decay of the fission product inventory during the interval between shutdown and commencement of fuel handling operations is taken into consideration.	Complies, except the time after shutdown is identified in Section 9.1.4.2.3. Accident occurs 76 hours after shutdown.	Complies, except the time after shutdown is identified in Section 9.1.4.2.3. Accident occurs 76 hours after shutdown.
b. The maximum fuel rod pressurization[2] is 1200 psig.	Complies.	Complies.
c. The minimum water depth[2] between the top of the damaged fuel rods and the fuel pool surface is 23 feet.	Complies. Water depth is greater than 23 feet.	Complies. Water depth is greater than 23 feet.

TABLE 15.7-2 (Sheet 2)

	Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
d.	All of the gap activity in the damaged rods is released and consists of 10% of the total noble gases other than Kr-85, 30% of the Kr-85, and 10% of the total radioactive iodine in the rods at the time of the accident. For the purpose of sizing filters for the fuel handling accident addressed in this guide, 30% of the I-127 and I-129 inventory is assumed to be released from the damaged rods.	Complies.	Complies.
e.	The values assumed for individual fission product inventories are calculated assuming full power operation at the end of core life immediately preceding shutdown and such calculation should include an appropriate radial peaking factor. The minimum acceptable radial peaking factors are 1.5 for BWR's and 1.65 for PWR's.	Complies. A peaking factor of 1.65 is used.	Complies. A peaking factor of 1.65 is used.
f.	The iodine gap inventory is composed of inorganic species (99.75%) and organic species (.25%).	Complies.	Complies.
g.	The pool decontamination factors for the inorganic and organic species are 133 and 1, respectively, giving an overall effective decontamination factor of 100 (i.e., 99% of the total iodine	Complies.	Complies.

TABLE 15.7-2 (Sheet 3)

	Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
	released from the damaged rods is retained by the pool water). This difference in decontamination factors for inorganic and organic iodine species results in the iodine above the fuel pool being composed of 75% inorganic and 25% organic species.		
h.	The retention of noble gases in the pool is negligible (i.e., decontamination factor of 1).	Complies. A decontamination factor of 1 is used.	Complies. A decontamination factor of 1 is used.
i.	The radioactive material that escapes from the pool to the building is released from the building[3] over a 2-hour time period.	Complies. A 0-2 hour release from the pool to the building to the environment is assumed.	The containment shutdown purge lines are automatically isolated upon detection of high radioactivity in the containment. It is conservatively assumed that isolation does not occur until 25 seconds after the release. The containment minipurge lines are assumed to automatically isolate in less than 25 seconds after the release. Therefore, the greatest portion of the activity is contained in the reactor building following the event.
j.	If it can be shown that the building atmosphere is exhausted through adsorbers designed to remove iodine, the removal efficiency is 90% for inorganic species and 70% for organic species.[4]	Not applicable; complies with Regulatory Guide 1.52 as described in Table 9.4-2.	No credit is taken for the normal purge filters.

TABLE 15.7-2 (Sheet 4)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
k. The effluent from the filter system passes directly to the emergency exhaust system without mixing5 in the surrounding building atmosphere and is then released (as an elevated plume for those facilities with stacks[6]).	Complies.	Complies.
2. The assumptions for atmospheric diffusion are: a. Ground Level Releases (1) The basic equation for atmospheric diffusion from a ground level point source is: $\chi/Q = \dfrac{1}{\pi u \sigma_y \sigma_z}$ Where: χ = the short term average centerline value of the ground level concentration (curies/m^3) Q = amount of material released (curies/sec) u = windspeed (meters/sec) σ_y = the horizontal standard deviation of the plume (meters) [See Figure V-1,	Short-term atmospheric dispersion factors corresponding to ground level release and accident conditions were based on meteorological measurement programs described in Section 2.3. The dispersion factors are in compliance with the methodology described in Regulatory Guide 1.145 and represent the worst of the 5 percent overall site meteorology and the 0.5 percent worst sector meteorology.	

TABLE 15.7-2 (Sheet 5)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.]		
σ_z = the vertical standard deviation of the plume (meters) [See Figure V-2, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.]		
(2) For ground level releases, atmospheric diffusion factors[7] used in evaluating the radiological consequences of the accident addressed in this guide are based on the following assumptions: (a) windspeed of 1 meter/sec; (b) uniform wind direction; (c) Pasquill diffusion category F.		
(3) Figure 1 is a plot of atmospheric diffusion factors (χ/Q) versus distance derived by use		

TABLE 15.7-2 (Sheet 6)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
of the equation for a ground level release given in regulatory position 2.a.(1) and under the meteorological conditions given in regulatory position 2.a.(2).		
(4) Atmospheric diffusion factors for ground level releases may be reduced by a factor ranging from one to a maximum of three (see Figure 2) for additional dispersion produced by the turbulent wake of the reactor building. The volumetric building wake correction as defined in Subdivision 3-3.5.2 of Meteorology and Atomic Energy-1968, is used with a shape factor of 1/2 and the minimum cross-sectional area of the reactor building only.		
b. Elevated Releases		
(1) The basic equation for atmospheric diffusion from an elevated release is: $$\chi/Q = \frac{e^{-h^2/2\sigma_z^2}}{\pi u \sigma_y \sigma_z}$$	Not applicable. Ground level releases were assumed.	Not applicable. Ground level releases were assumed.

TABLE 15.7-2 (Sheet 7)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
Where: χ = the short term average centerline value of the ground level concentration (curies/m^3) Q = amount of material released (curies/sec) u = windspeed (meters/sec) σ_y = the horizontal standard deviation of the plume (meters) [See Figure V-1, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.] σ_z = the vertical standard deviation of the plume (meters) [See Figure V-2, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.] h = effective height of release (meters)		

TABLE 15.7-2 (Sheet 8)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
(2) For elevated releases, atmospheric diffusion factors[7] used in evaluating the radiological consequences of the accident addressed in this guide are based on the following assumptions: (a) windspeed of 1 meter/sec; (b) uniform wind direction; (c) envelope of Pasquill diffusion categories for various release heights; (d) a fumigation condition exists at the time of the accident.[8]		
(3) Figure 3 is a plot of atmospheric diffusion factors versus distance for an elevated release assuming no fumigation, and Figure 4 is for an elevated release with fumigation.		
(4) Elevated releases are considered to be at a height equal to no more than the actual stack height. Certain site conditions may exist, such as surrounding elevated topography or nearby structures, which will have the effect of reducing the effective stack height. The degree of stack height reduction will be evaluated on an individual case		

TABLE 15.7-2 (Sheet 9)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
3. The following assumptions and equations may be used to obtain conservative approximations of thyroid dose from the inhalation of radioiodine and external whole body dose from radioactive clouds:		
a. The assumptions relative to inhalation thyroid dose approximations are:	Complies. See Appendix 15A, Section 15A.2.4.	Complies. See Appendix 15A, Section 15A.2.4.
(1) The receptor is located at a point on or beyond the site boundary where the maximum ground level concentration is expected to occur.		
(2) No correction is made for depletion of the effluent plume of radioiodine due to deposition on the ground, or for the radiological decay or radioiodine in transit.		
(3) Inhalation thyroid doses may be approximated by use of the following equation: $D = \frac{F_q IFPBR(\chi/Q)}{(DF_p)(DF_f)}$ Where: D = thyroid dose (rads)		

TABLE 15.7-2 (Sheet 10)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
F_g = fraction of fuel rod iodine inventory in fuel rod void space (0.1)		
I = core iodine inventory at time of accident (curies)		
F = fraction of core damaged so as to release void space iodine		
P = fuel peaking factor		
B = Breathing rate = 3.47 x 10^{-4} cubic meters per second (i.e., 10 cubic meters per 8 hour work day as recommended by the ICRP)		
DF_p = effective iodine decontamination factor for pool water		
DF_f = effective iodine decontamination factor for filters (if present)		
χ/Q = atmospheric diffusion factor at receptor location (sec/m^3)		

TABLE 15.7-2 (Sheet 11)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
R = adult thyroid dose conversion factor for the iodine isotope of interest (rads per curie). Dose conversion factors for Iodine 131-135 are listed in Table 1.[9] These values were derived from "standard man" parameters recommended in ICRP Publication 2.[10]		
TABLE 1 Adult Inhalation Thyroid Dose Conversion Factors	Table 1; the thyroid dose conversion factors given in Regulatory Guide 1.109 are used.	Table 1; the thyroid dose conversion factors given in Regulatory Guide 1.109 are used.

Iodine Isotope	Conversion Factor (R) (Rads/curie inhaled)
131	1.48×10^6
132	5.35×10^4
133	4.0×10^5
134	2.5×10^4
135	1.24×10^5

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
b. The assumptions relative to external whole body dose approximations are:	Complies. See Appendix 15A, Section 15A.2.5.	Complies. See Appendix 15A, Section 15A.2.5.
(1) The receptor is located at a point on or beyond the site boundary where the maximum ground level concentration is expected to occur.		

TABLE 15.7-2 (Sheet 12)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
(2) External whole body doses are calculated using "Infinite Cloud" assumptions, i.e., the dimensions of the cloud are assumed to be large compared to the distances that the gamma rays and beta particles travel. The dose at any distance from the reactor is calculated based on the maximum ground level concentration at that distance. For an infinite uniform cloud containing curies of beta radioactivity per cubic meter, the beta dose rate in air at the cloud center is:[11] $_{\beta}D'_{\infty} = 0.457\ \bar{E}_{\beta}\chi$ Where: $_{\beta}D'_{\infty}$ = beta dose rate from an infinite cloud (rad/sec) $\bar{E}_{\beta}$ = average beta energy per disintegration (Mev/dis) χ = concentration of beta or gamma emitting isotope in the cloud (curie/m^3)	(2) The whole-body dose factors for gammas given in Regulatory Guide 1.109 are used; for iodines, the whole-body dose factors for gammas with credit for 5 cm body tissue attenuation are used. See Table 15A-4 for dose conversion factors.	

TABLE 15.7-2 (Sheet 13)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
Because of the limited range of beta particles in tissue, the surface body dose rate from beta emitters in the infinite cloud can be approximated as being one-half this amount or: $_{\beta}D'_{\infty} = 0.23\ \bar{E}_{\beta}\chi$ For gamma emitting material the dose rate in tissue at the cloud center is: $_{\gamma}D'_{\infty} = 0.507\ \bar{E}_{\gamma}\chi$ Where: $_{\gamma}D'_{\infty}$ = gamma dose rate from an infinite cloud (rad/sec) $\bar{E}_{\gamma}$ = average gamma energy per disintegration (Mev/dis) However, because of the presence of the ground, the receptor is assumed to be exposed to only one-half of the cloud (semi-infinite) and the equation becomes: $_{\gamma}D' = 0.25\ \bar{E}_{\gamma}\chi$		

TABLE 15.7-2 (Sheet 14)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
Thus, the total beta or gamma dose to an individual located at the center of the cloud path may be approximated as: $_{\beta}D_{\infty} = 0.23\ \bar{E}_{\beta}\Psi$ or $_{\gamma}D_{\infty} = 0.23\ \bar{E}_{\gamma}\Psi$ Where Ψ is the concentration time integral for the cloud (curie sec/m^3).		
(3) The beta and gamma energies emitted per disintegration, as given in Table of Isotopes,[12] are averaged and used according to the methods described in ICRP Publication 2.		

Notes:

[1]The assumptions given are valid only for oxide fuels of the types currently in use and in cases where the following conditions are not exceeded:

a. Peak linear power density of 20.5 kW/ft for the highest power assembly discharged.

b. Maximum center-line operating fuel temperature less than 4500 F for this assembly.

TABLE 15.7-2 (Sheet 15)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
c. Average burnup for the peak assembly of 25,000 MWD/ton or less (this corresponds to a peak local burnup of about 45,000 MWD/ton).		

[2]For release pressures greater than 1200 psig and water depths less than 23 feet, the iodine decontamination factors will be less than those assumed in this guide and must be calculated on an individual case basis using assumptions comparable in conservatism to those of this guide.

[3]The effectiveness of features provided to reduce the amount of radioactive material available for release to the environment will be evaluated on an individual case basis.

[4]These efficiencies are based upon a 2-inch charcoal bed depth with 1/4 second residence time. Efficiencies may be different for other systems and must be calculated on an individual case basis.

[5]Credit for mixing will be allowed in some cases; the amount of credit will be evaluated on an individual case basis.

[6]Credit for an elevated release will be given only if the point of release is (a) more than two and one-half times the height of any structure close enough to affect the

TABLE 15.7-2 (Sheet 16)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
dispersion of the plume or (b) located far enough from any structure which could affect the dispersion of the plume. For those plants without stacks the atmospheric diffusion factors assuming ground level release given in regulatory position 2.b should be used.		
[7]These diffusion factors should be used until adequate site meteorological data are obtained. In some cases, available information on such site conditions as meteorology, topography and geographical location may dictate the use of more restrictive parameters to ensure a conservative estimate of potential offsite exposures.		
[8]For sites located more than 2 miles from large bodies of water such as oceans or one of the Greak Lakes, a fumigation condition is assumed to exist at the time of the accident and continue for 1/2 hour. For sites located less than 2 miles from large bodies of water a fumigation condition is assumed to exist at the time of the accident and continue for the duration of the release (2 hours).		

TABLE 15.7-2 (Sheet 17)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)

[9]Dose conversion factors taken from "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, J. J. DiNunno, R. E. Baker, F. D. Anderson, and R. L. Waterfield (1962).

[10]Recommendations of the International Commission on Radiological Protection, "Report of Committee II on Permissible Dose for Internal Radiation (1959)," ICRP Publication 2, (New York: Permagon Press, 1960).

[11]Meteorology and Atomic Energy-1968, Chapter 7.

[12]C. M. Lederer, J. M. Hollander, and I. Perlman, Table of Isotopes, Sixth Edition (New York: John Wiley and Sons, Inc. 1967).

TABLE 15.7-3

PARAMETERS USED IN EVALUATING
THE RADIOLOGICAL CONSEQUENCES OF A
WASTE GAS DECAY TANK RUPTURE

I.	Source Data	
	a. Core power level, MWt	3,565
	b. Failed fuel, percent	1
II.	Atmospheric Dispersion Factors	See Table 15A-2
III.	Activity Release Data	

Isotope	0-2 hr (Ci)
I-131	4.02E-2
I-132	0.00
I-133	3.50E-2
I-134	0.00
I-135	1.39E-2
Xe-131m	9.05E+2
Xe-133m	1.14E+3
Xe-133	7.09E+4
Xe-135m	5.99E+1
Xe-135	1.15E+3
Xe-137	3.42E-1
Xe-138	4.40
Kr-83m	2.26E+1
Kr-85m	1.85E+2
Kr-85	4.75E+3
Kr-87	3.64E+1
Kr-88	2.25E+2
Kr-89	1.76E-1

TABLE 15.7-4

RADIOLOGICAL CONSEQUENCES OF A
WASTE GAS DECAY TANK RUPTURE

	Doses (rem)
Exclusion Area Boundary (0-2 hr)	
Thyroid	3.65E-3
Whole body	1.30E-1
Low Population Zone Outer Boundary (duration)	
Thyroid	4.86E-4
Whole body	1.74E-2

TABLE 15.7-5

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LIQUID RADWASTE TANK FAILURE

I.	Source Data	
	a. Core power level, MWt	3,565
	b. Failed fuel, percent	1
II.	Atmospheric Dispersion Factors	See Table 15A-2
III.	Activity Release Data	
	a. Noble gas activity, percent of tank contents	100
	b. Iodine gas activity, percent of tank contents	
	1. Boron recycle holdup tank	10
	2. Hypothetical Liquid Waste tank	100
	c. Tank contents subject to release	
	1. Boron recycle holdup tank	Table 11.1-6 (sheet 13)
	2. Primary evaporator bottoms tank	Table 11.1-6 (sheet 16)
	d. Activity released to the environment	
	1. Boron recycle holdup tank	

Isotope	0-2 hr (Ci)
I-131	4.96E-1
I-132	7.90E-3
I-133	1.13E-1
I-134	7.25E-4
I-135	2.09E-2
Xe-131m	3.35E+2
Xe-133m	1.40E+2
Xe-133	1.68E+4
Xe-135m	1.06E-1
Xe-135	4.40E+1
Xe-137	6.97E-3
Xe-138	9.38E-2
Kr-83m	5.02E-1
Kr-85m	4.93
Kr-85	1.59E+3
Kr-87	9.06E-1
Kr-88	5.80
Kr-89	3.10E-3

TABLE 15.7-5 (Sheet 2)

2. Primary evaporator bottoms tank

Isotope	0-2 hr (Ci)
I-131	1.92E+01
I-132	9.939E-24
I-133	7.464E-03
I-134	3.192E-61
I-135	7.190E-09
Xe-131m	0.000E+00
Xe-133m	0.000E+00
Xe-133	0.000E+00
Xe-135m	0.000E+00
Xe-135	0.000E+00
Xe-137	0.000E+00
Xe-138	0.000E+00
Kr-83m	0.000E+00
Kr-85m	0.000E+00
Kr-85	0.000E+00
Kr-87	0.000E+00
Kr-88	0.000E+00
Kr-89	0.000E+00

TABLE 15.7-6

RADIOLOGICAL CONSEQUENCES OF A LIQUID RADWASTE TANK FAILURE

	Dose (rem)
Boron Recycle Tank	
Exclusion Area Boundary (0-2 hr)	
Thyroid	4.01E-2
Whole-body	2.48E-2
Low Population Zone Outer Boundary (duration)	
Thyroid	5.35E-3
Whole-body	3.31E-3
Hypothetical Liquid Waste Tank	
Exclusion Area Boundary (0-2 hr)	
Thyroid	1.49E+00
Low Population Zone Outer Boundary (duration)	
Thyroid	2.00E-1

TABLE 15.7-7

PARAMETERS USED IN EVALUATING
THE RADIOLOGICAL CONSEQUENCES OF A
FUEL-HANDLING ACCIDENT

	In Fuel Building	In Reactor Building
I. Source Data		
a. Core power level, MWt	3,565	3,565
b. Radial peaking factor	1.65	1.65
c. Decay time, hours	76	76
d. Number of fuel assemblies affected	1.0	1.2
e. Fraction of fission product gases contained in the gap region of the fuel assembly	Per R.G. 1.25 and NUREG/CR-5009	Per R.G. 1.25 and NUREG/CR-5009
II. Atmospheric Dispersion Factors	See Table 15A-2	See Table 15A-2
III. Activity Release Data		
a. Percent of affected fuel assemblies gap activity released	100	100
b. Pool decontamination factors		
1. Iodine	100	100
2. Noble gas	1	1
c. Filter efficiency, percent	82.5*	0
d. Building mixing volumes assumed, percent of total volume	0	0
e. Activity release period, hrs	2	2

*NOTE: The postulated fuel handling accident in the Fueling Building was analyzed with a reduced filter efficiency, based upon the single failure assumption that one of the emergency Exhaust Filter-Adsorber units is operating with a failed heater or humidistat.

TABLE 15.7-7 (Sheet 2)

h. Activity released to the environment

Isotope	Fuel Building Building 0-2 hr (Ci)	Reactor 0-2 hr (Ci)
I-131	1.84E+2	8.86E+2
I-133	3.28E+1	1.58E+2
Xe-131m	7.18E+2	8.61E+2
Xe-133m	1.90E+3	2.28E+3
Xe-133	1.10E+5	1.32E+5
Xe-135	1.22E+2	1.45E+2
Kr-85	2.62E+3	3.14E+3

TABLE 15.7-8

RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT

	Dose (rem)
In Fuel Building	
Exclusion Area Boundary (0-2 hr)	
Thyroid	1.48E+1
Whole-body	1.72E-1
Low Population Zone Outer Boundary (duration)	
Thyroid	1.97E+0
Whole-body	2.30E-2
In Reactor Building	
Exclusion Area Boundary (0-2 hr)	
Thyroid	7.09E+1
Whole-body	2.04E-1
Low Population Zone Outer Boundary (duration)	
Thyroid	9.46E+0
Whole-body	2.72E-2

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

An ATWS is an anticipated operational occurrence (such as loss of normal feedwater, loss of load, turbine trip, inadvertent control rod withdrawal, loss of AC power, and loss of condenser vacuum) that is accompanied by a failure of the Reactor Protection System (RPS) to shut down the reactor. The most severe ATWS scenarios are those in which there was a complete loss of normal feedwater and the loss of load. The primary safety concern for these transients is the potential for high pressures within the RCS. Based on generic analyses, acceptable consequences would result, that is, RCS pressures less than the pressure (3200 psig) corresponding to the ASME Boiler and Pressure Vessel Code Level C service limit stress criteria, provided that the turbine trips and auxiliary feedwater flow is initiated in a timely manner. Normally, these features would be actuated by the RPS. However, if a common mode failure in the RPS incapacitates AFW flow initiation and/or turbine trip (in addition to prohibiting a scram), then an alternate method of providing AFW flow and turbine trip is required. These two functions are provided by AMSAC, which is required by the final ATWS Rule in 10CFR50.62 (c)(1) for Westinghouse PWRs, as discussed below.

The ATWS Rule (10CFR50.62(b)) requires specific improvements in the design and operation of commercial nuclear power facilities to reduce the probability of failure to shut down the reactor following anticipated transients and to mitigate the consequences of an ATWS event. 10CFR50.62 "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants," was effective July 26, 1984. Paragraph (c)(6) of the ATWS Rule requires that detailed information to demonstrate compliance with the requirements be submitted to the Director, Office of Nuclear Reactor Regulation (NRR). In accordance with paragraph (c)(6) of the ATWS Rule, the Operating Agent provided information to the NRC by letter, dated March 20, 1987. The letter forwarded the detailed description of the ATWS Mitigation System Actuation Circuitry (AMSAC) proposed for installation at the Wolf Creek Generating Station (WCGS). Supplemental information on the AMSAC design was also submitted to the NRC in letters dated April 16, 1987 and October 5, 1987, for review. WCGS installed equipment (AMSAC) from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary feedwater system and initiate a turbine trip under conditions indicative of an ATWS.

On December 16, 1987, the NRC notified the Operating Agent by letter that the staff had completed its review of the submittal and concluded that the AMSAC design for the WCGS was acceptable and in compliance with the ATWS Rule, 10CFR50.62.

APPENDIX 15A

ACCIDENT ANALYSIS RADIOLOGICAL CONSEQUENCES EVALUATION MODELS AND PARAMETERS

15A.1 GENERAL ACCIDENT PARAMETERS

This section 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 (Ref. 1) methodology and represent the worst of the 5 percent overall site meteorology and the 0.5 percent worst-sector meteorology and these are given in Table 15A-2 (see Section 2.3.4 for additional details on meteorology). The core and gap inventories are given in Table 15A-3. The thyroid (via inhalation pathway), beta skin, and total-body (via submersion pathway) dose factors based on References 2, 3a and 3b are given in Table 15A-4.

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 flow paths, and the equations for dose calculations. Two major release models are considered: (1) a single holdup system with no internal cleanup and (2) a holdup system wherein a two-region spray model is used for internal cleanup.

15A.2.1 ACCIDENT RELEASE PATHWAYS

The release pathways for the major accidents are given in Figure 15A-1. The accidents and their pathways are as follows:

LOCA: Immediately following a postulated loss-of-coolant accident (LOCA), the release of radioactivity from the containment is to the environment with the containment spray and ESF systems in full operation. The release in this case is calculated using Equation (8) which takes into account a two-region spray model within the containment. The release of radioactivity to the environment due to assumed ESF system leakages in the auxiliary building will be via ESF filters and is calculated using Equation (5). In addition, the release of radioactivity to the environment due to assumed ECCS boundary valves leakage through RWST is calculated using Equation (11).

WGDTR: The activity release to the environment due to waste gas decay tank rupture (WGDTR) will be direct and unfiltered, with no holdup. The release pathway is A'-D. The total activity release in this case is therefore assumed to be the initial source activity itself.

FHA: The release to the environment due to a fuel handling accident (FHA) in the fuel building is via filters following the actuation of the emergency exhaust system. The release pathway is B-C-D. Since the release is calculated without any credit for holdup in the fuel building, the total release will be the unfiltered release for the first minute plus the product of the initial activity and the filter nonremoval efficiency fraction (for noble gases, the nonremoval efficiency fraction is 1). The release of radioactivity to the environment due to FHA inside the containment is direct and unfiltered, via the A'-D pathway, and occurs over a two-hour period (actually, the release is via the non-safety graded filters). The release is calculated using Equation (8) based on a two-region spray model.

CAE: Radioactivity release to the environment due to the control assembly ejection (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-D and A'-D.

MSLB, SGTR: Radioactivity releases to the environment due to main steam line break (MSLB) or steam generator tube rupture (SGTR) accidents are direct and unfiltered with no holdup via the A'-D pathway. The activity release calculations for these accidents are complex, involving spiking effects, time-dependent flashing fractions, and scrubbing of flashed activities; the release calculations are described in those sections that address these accidents.

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.

$A_1(0)$ = initial source activity at time t_o, Ci

$A_1(t)$ = source activity at time t seconds, Ci

$$A_1(t) = A_1(0)e^{-\lambda_1 t} \qquad (1)$$

where λ_1 = total removal constant from primary holdup system, sec^{-1}

$$\lambda_1 = \lambda_d + \lambda_{1\ell} + \lambda_r \qquad (2)$$

where λ_d = decay removal constant, sec^{-1}

$\lambda_{1\ell}$ = primary holdup leak or release rate, sec^{-1}

λ_r = internal removal constant (i.e., sprays, plateout, etc.), sec^{-1}

Thus, the direct release rate to the atmosphere from the primary holdup system

$$R_u(t) = \lambda_{1\ell} A_1(t) \qquad (3)$$

$R_u(t)$ = unfiltered release rate (Ci/sec)

The integrated activity release is the integral of the above equation.

$$IAR(t) = \int_o^t R_u(t) = \int_o^t \lambda_1 A_1(o) e^{-\lambda_1 t} \qquad (4)$$

This yields:

$$IAR(t) = \left(\lambda_{1\ell} A_1(o) \; / \; \lambda_1\right)\left(1 - e^{-\lambda_1 t}\right) \qquad (5)$$

15A.2.3a 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 a sprayed and unsprayed region 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 of the form

$$\frac{dA_i}{dt} = -\sum_{j=1}^{K_i} -\lambda_{ij} A_i \; -\sum_{\ell=1}^{-n-1} Q_{i\ell} \frac{Ai}{Vi} \; +\sum_{\ell=1}^{n-1} Q_{i\ell} \frac{Ai}{Vi} \qquad (6)$$

where

A_i = fission product activity in volume i, Ci

n = number of volumes considered in the model

Q_{il} = transfer rate from volume i to volume l, cc/sec

V_i = volume of the ith compartment, cc

l_{ij} = removal rate of the jth removal process in volume i, sec^{-1}

K_i = total number of removal processes in volume i

This system of equations is readily solved if the coefficients are known.

For a two-region model, the above system reduces to

$$\frac{dA_1}{dt} = -\sum_{j=1}^{K_1} l_{1j}A_1 - Q_{12}\frac{A_1}{V_1} + Q_{21}\frac{A_2}{V_2} \qquad (6a)$$

$$\frac{dA_2}{dt} = -\sum_{j=1}^{K_1} l_{2j}A_2 - Q_{21}\frac{A_2}{V_2} + Q_{12}\frac{A_1}{V_1} \qquad (6b)$$

Upon solving this coupled set of differential equations numerically, the release rate of activity is found from

$$R(t) = \lambda_{1\ell}\ A(t) \qquad (7)$$

The integrated activity released from time t_0 - t_1 is shown in the following equation which is solved numerically.

$$IAR = \int_{t_0}^{t_1} R(t)dt \qquad (8)$$

15A.2.3b TWO-REGION RELEASE MODEL FOR LEAKAGE THROUGH RWST

It is assumed that the activity released to the holdup system (in this case, the containment recirculation sump) instantaneously diffuses to uniformly occupy the sump volume. Removal mechanisms from the sump include decay and release (i.e., leakage) to the RWST. The release rate from the RWST to the environment is given by

$$R_2(t) = f\, \lambda_{2\ell}\, A_2(t) \qquad (9)$$

where $R_2(t)$ = the unfiltered release rate from the RWST vent, Ci/sec

f = assumed percent of radioiodine released to the RWST that becomes airborne

$\lambda_{2\ell}$ = release rate constant for leakage from the RWST to the environment, based on an assumed leak rate from the sump that is uniformly mixed in the RWST volume, sec^{-1}

$A_2(t)$ = RWST activity, Ci

The RWST activity can be calculated as

$A_2(t)$ = Activity in RWST + Activity from sump - Activity released to environment

$$= A_2(t-\Delta t)e^{-\lambda_d \Delta t} + \lambda_{1\ell} A_1(t-\Delta t)\Delta t - R_2(t-\Delta t)\Delta t \qquad (10)$$

where $\lambda_{1\ell}$ = release rate constant for leakage from the uniformly mixed sump to the RWST, based on an assumed leak rate from the sump to the RWST, sec^{-1}

$A_1(t)$ = containment sump activity, Ci

λ_d = decay removal constant, sec^{-1}

The integrated release from the RWST is given by

$$IAR_2(t) = \int_0^t R_2(t)dt = f\, \lambda_{2\ell} \int_0^t A_2(t)dt \qquad (11)$$

and is calculated numerically by using Equation (10).

15A.2.4 OFFSITE THYROID DOSE CALCULATION MODEL

Offsite thyroid doses are calculated using the equation:

$$D_{TH} = \sum_i DCF_{Thi} \sum_j (IAR)_{ij} \ (BR)j \ (c/Q)j \qquad (12)$$

where

$(IAR)_{ij}$ = integrated activity of isotope i released* during the time interval j in Ci

and $(BR)_j$ = breathing rate during time interval j in meter^3/second

$(c/Q)j$ = offsite atmospheric dispersion factor during time interval j in second/meter^3

$(DCF)^{Thi}$ = thyroid dose conversion factor via inhalation for isotope i in rem/Ci

DTh = thyroid dose via inhalation in rems

*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.

15A.2.5 OFFSITE TOTAL-BODY DOSE CALCULATIONAL MODEL

Assuming a semi-infinite cloud of gamma emitters, offsite total-body doses are calculated using the equation:

$$D_{TB} = \sum_i DCF_{gi} \sum_j (IAR)_{ij} \quad (c/Q)_j$$

where

$(IAR)_{ij}$ = integrated activity of isotope i released* during the j^{th} time interval in Ci

and $(c/Q)_j$ = offsite atmospheric dispersion factor during time interval j in second/meter^3

$(DCF)_{gi}$ = total-body gamma dose conversion factor for the i^{th} isotope in rem-meter^3/Ci-sec

D_{TB} = total-body dose in rems

15A.3 CONTROL ROOM RADIOLOGICAL CONSEQUENCES CALCULATIONAL MODELS

Only radiation doses to a control room operator due to postulated LOCA are presented in this chapter since 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

Make-up air is brought into the control room via the control room filtration system which draws in air from the control building. Outside air is brought into the control building through safety grade filters via the control room pressurization fan. Some unfiltered air also may leak into the control building via an assumed inleakage rate. The activity concentrations at the control building intake for each time interval are found by multiplying the activity release to the environment by the appropriate c/Q for that time interval. The flow path model is shown below.

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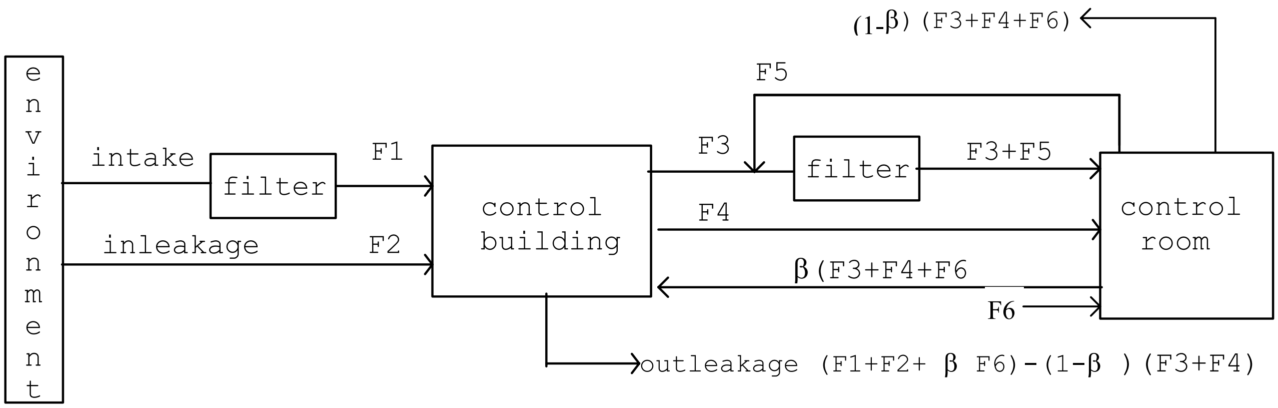
*No credit is taken for cloud depletion by ground deposition and radioactive decay during transport to the exclusion area boundary or the outer boundary of the low-population zone.



Once activity is brought into the control building, mixing within the control building is afforded by the control room pressurization fan. The control room filtration system fan takes air from the control building and the control room (recirculation) and discharges to the control room through the control room filtration safety grade filters. The radiological analysis input parameters are provided in Table 15A-1.

The control room ventilation isolation signal (CRVIS) starts both trains of the control room pressurization system and the control room filtration system. For the determination of dose to control room personnel, the worst single failure has been ascertained to be the failure of the filtration fan in one of the two filtration system trains.

Prior to operator action, a potential pathway would exist allowing air from the control building to enter the control room, bypassing the control room filtration filters. Operator action is required to ensure no bypass pathways then exist for unfiltered air to enter the control room.

Owning to this single failure of the control room filtration fan, the assumed failure of one of the two containment spray (CS) trains, and two of the four hydrogen mixing subsystem fans, inherent in the LOCA analysis parameters given in Table 15.6-6 should not be applied in this analysis. With both trains of CS and four hydrogen mixing fans operating, more volumetric coverage of the containment spray and more mixing between the new sprayed and unsprayed regions would be expected, thereby giving much greater iodine removal within the containment atmosphere. However, the doses to control room personnel have been based on the LOCA analysis parameters given in Table 15.6-6.

The activity in the control building and control room is calculated by solving the following coupled set of first order differential equations.

$$\frac{dA_{CB}(t)}{dt} = \left[(1-\eta)F_1 + F_2\right]\chi/Q\left[R(t)\right] + \beta\lambda_{4\ell}A_{CR}(t) - \lambda_3 A_{CB}(t)$$

$$\frac{dA_{CR}(t)}{dt} = \left[(1-\eta)\lambda_{3f} + \lambda_{3u}\right]A_{CB}(t) - \lambda_4 A_{CR}(t) + F_6(\chi/Q)\left[R(t)\right]$$

where

$A_{CB}(t)$ = activity in control building at time t, curies

$A_{CR}(t)$ = activity in control room at time t, curies

η = filter efficiency, fraction

F_1 = filtered intake rate, $meter^3/sec$

F_2 = unfiltered intake (inleakage), $meter^3/sec$

χ/Q = atmospheric dispersion factor, $sec/meter^3$

$R(t)$ = activity release rate in Ci/sec as given in Equation 3 of Section 15A.2.2 or Equations 7 and 9 of Sections 15A.2.3a and 15A2.3b

λ_3 = $\lambda_d + \lambda_{3\ell} + \lambda_{3f} + \lambda_{3u}$, total removal rate from the control building, sec^{-1}

λ_d = isotopic decay constant, sec^{-1}

$\lambda_{3\ell}$ = outleakage to atmosphere from the control building (= $(F1 + F2 + \beta F_6 - (1-\beta)(F3 + F4))/V_{CB}$ with V_{CB} being control building mixing volume in $meter^3$), sec^{-1}

λ_{3f} = filtered flow from control building into control room ($=F_3/V_{CB}$, F_3 in $meter^3/sec$), sec^{-1}

λ_{3u} = unfiltered flow from control building into control room ($=F_4/V_{CB}$, F_4 in $meter^3/sec$), sec^{-1}

λ_4 = $\lambda_d + \lambda_r + \lambda_{4\ell}$, total removal rate from the control room, sec^{-1}

λ_r = recirculation removal rate ($=\eta F_5/V_{CR}$ with F_5 being recirculation flow rate in meter^3/sec through filter with efficiency η and V_{CR} being control room volume in meter^3), sec^{-1}

$\lambda_{4\ell}$ = leakage to control building from the control room ($=[F_3 + F_4]/V_{CR}$), sec^{-1}

β = fraction of control room outleakage which returns to the control building mixing volume

F_6 = control room direct unfiltered intake, meter^3/sec

Upon solving this coupled set of differential equations, the integrated activity in the control room (IA_{CR}) is determined by the expression

$$IA_{CR(t)} = \int_0^t A_{CR}(t)dt$$

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.3.2 CONTROL ROOM THYROID DOSE CALCULATIONAL MODEL

Control room thyroid doses via inhalation pathway are calculated using the following equation:

$$D_{Th-CR} = \frac{BR}{V_{CR}} \sum_i DCF_{Thi} \sum_j (IA_{CRij}) \times O_j$$

where

D_{Th-CR} = control room thyroid dose in rem

and

BR = breathing rate assumed to be always 3.47×10^{-4} meter^3/second

V_{CR} = volume of the control room in cubic meters

DCF_{Thi} = thyroid dose conversion factor for adult via inhalation in rem/Ci for isotope i

IA_{CRij} = integrated activity in control room in Ci-sec for isotope i during time interval j

O_j = control room occupancy fraction during time interval j

15A.3.3 CONTROL ROOM BETA-SKIN DOSE CALCULATIONAL MODEL

The beta-skin doses to a control room operator are calculated using the following equation:

$$D_{b-CR} = \frac{1}{V_{CR}} \sum_i DCF_{bi} \sum_j (IA_{CRij}) \times O_j$$

where D_{b-CR} and DCF_{bi} are the beta-skin doses in the control room

in rem and the beta-skin dose conversion factor for isotope i in rem-meter^3/Ci-sec, respectively. The other symbols are explained in Section 15A.3.2.

15A.3.4 CONTROL ROOM TOTAL-BODY DOSE CALCULATION

Due to the finite structure of the control room, the total-body gamma doses to a control room operator will be substantially less than what they would be due to immersion in an infinite cloud of gamma emitters. The finite cloud gamma doses are calculated using Murphy's method (Ref. 4) which models the control room as a hemisphere. The following equation is used:

$$D_{TB-CR} = \frac{1}{V_{CR}(GF)} \sum_i DCF_{gi} \sum_j (IA_{CRij}) \times O_j$$

where

GF = dose reduction due to control room geometry factor

GF = $1173/(V_1)^{0.338}$

V_1 = volume of the control room in cubic feet

D_{TB-CR} = total-body dose in the control room in rem,

and other quantities have been defined in subsections 15A.2.5 and 15A.3.2.

15A.3.4.1 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 Section 7.5.4 of Reference 5 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.

3a. Kocher, D.C., "Nuclear Decay Data for Radionuclides Occurring in Routine Releases from Nuclear Fuel Cycle Facilities," ORNL/NUREG/TM-102, August 1977.

3b. Berger, M.J., "Beta-Ray Dose in Tissue-Equivalent Material Immersed in a Radioactive Cloud," Health Physics, Vol. 26, pp. 1-12, January 1974.

4. 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, August 1974.

5. "Meteorology and Atomic Energy 1968," D. H. Slade (ed.), USAEC Report, TID 24190, 1968.

TABLE 15A-1

PARAMETERS USED IN ACCIDENT ANALYSIS

I.	General	
	1. Core power level, Mwt	3565
	2. Full-power operation, days per cycle	510
	3. Number of fuel assemblies in the core	193
	4. Maximum radial peaking factor	1.65
	5. Percentage of failed fuel	1.0
	6. Steam generator tube leak, lb/hr	500
II.	Sources	
	1. Core inventories, Ci	Table 15A-3
	2. Gap inventories, Ci	Table 15A-3
	3. Primary coolant specific activities, Ci/gm	Table 11.1-5
	4. Primary coolant activity, technical specification limit for iodines - I-131 dose equivalent, μCi/gm	1.0
	5. Secondary coolant activity technical specification limit for iodines - I-131 dose equivalent, μCi/gm	0.1
III.	Activity Release Parameters	
	1. Free volume of containment, ft^3	2.5×10^6
	2. Containment leak rate	
	i. 0-24 hours, % per day	0.2
	ii. after 24 hrs, % per day	0.1
IV.	Control Room Dose Analysis (for LOCA)	
	1.Control building	
	i. Mixing volume, cf	239,000
	ii. Filtered intake, cfm	
	Prior to operator action (0-1.5 hours)	$\geq$1350
	After operator action (1.5 hours-720 hours)	$\geq$675
	iii. Unfiltered inleakage, cfm	$\leq$300
	iv. Filter efficiency (all forms of iodine), %	95

TABLE 15A-1 (Sheet 2)

2. Control room		
i.	Volume, cf	100,000
ii.	Filtered flow from control building, cfm	≤550(*)
iii.	Unfiltered flow from control building, cfm	
	Prior to operator action	≤550(*)
	(0-1.5 hours) After operator action (1.5 hours - 720 hours)	0
iv.	Filtered recirculation, cfm	≥1250
v.	Filter efficiency (all forms of iodine), %	95

(*)NOTE: Flows possible per train with two trains in operation. Each train is balanced for 400 cfm.

3. Total Flow Summary (Filtered plus unfiltered)		
i.	0-1.5 hours	
	a. Control Room Pressurization	≥1650(**)
	b. Control Room Filtration	≤1100
ii.	1.5 hours - 720 hours	
	a. Control Room Pressurization	≥975(**)
	b. Control Room Filtration	≤550

(**)NOTE: Includes 300 cfm of unfiltered Control Bldg inleakage.

4. Emergency Exhaust Filter Adsorber Unit Efficiency(all forms of iodine), %		90

V. Miscellaneous

1.Atmospheric dispersion factors, χ/Q sec/m^3		Table 15A-2
2.Dose conversion factors		
i.	total body and beta skin, rem-meter^3/ Ci-sec	Table 15A-4
ii.	thyroid, rem/Ci	Table 15A-4
3.Breathing rates, meter^3/sec		
i.	control room at all times	3.47×10^{-4}
ii.	offsite	
	0-8 hrs	3.47×10^{-4}
	8-24 hrs	1.75×10^{-4}
	24-720 hrs	2.32×10^{-4}
4.Control room occupancy fractions		
0-24 hrs		1.0
24-96 hrs		0.6
96-720 hrs		0.4

TABLE 15A-2

LIMITING SHORT-TERM ATMOSPHERIC DISPERSION FACTORS
(χ/Qs) FOR ACCIDENT ANALYSIS
(sec/meter^3)

Location Type/ Time Interval (hrs)	χ/Q (Sec/Meters^3)
Site boundary	
0-2	1.5E-4
Low-population zone	
0-8	2.0E-5
8-24	1.3E-5
24-96	5.4E-6
96-720	1.5E-6
Control room (via containment leakage)	
0-8	5.3E-4
8-24	3.6E-4
24-96	6.6E-5
96-720	0
Control room (via unit vent exhaust)	
0-8	1.lE-4
8-24	6.8E-5
24-96	1.7E-5
96-720	0

TABLE 15A-3

FUEL AND ROD GAP INVENTORIES - CORE (Ci)

Core

Isotope	Fuel	Gap
I-131	9.46E+7	9.46E+6
I-132	1.37E+8	1.37E+7
I-133	1.95E+8	1.95E+7
I-134	2.15E+8	2.15E+7
I-135	1.83E+8	1.83E+7
Kr-83m	1.24E+7	1.24E+6
Kr-85m	2.67E+7	2.67E+6
Kr-85	1.02E+6	3.05E+5
Kr-87	5.16E+7	5.16E+6
Kr-88	7.28E+7	7.28E+6
Kr-89	8.94E+7	8.94E+6
Xe-131m	1.01E+6	1.01E+5
Xe-133m	6.06E+6	6.06E+5
Xe-133	1.95E+8	1.95E+7
Xe-135m	3.77E+7	3.77E+6
Xe-135	4.70E+7	4.70E+6
Xe-137	1.71E+8	1.71E+7
Xe-138	1.64E+8	1.64E+7

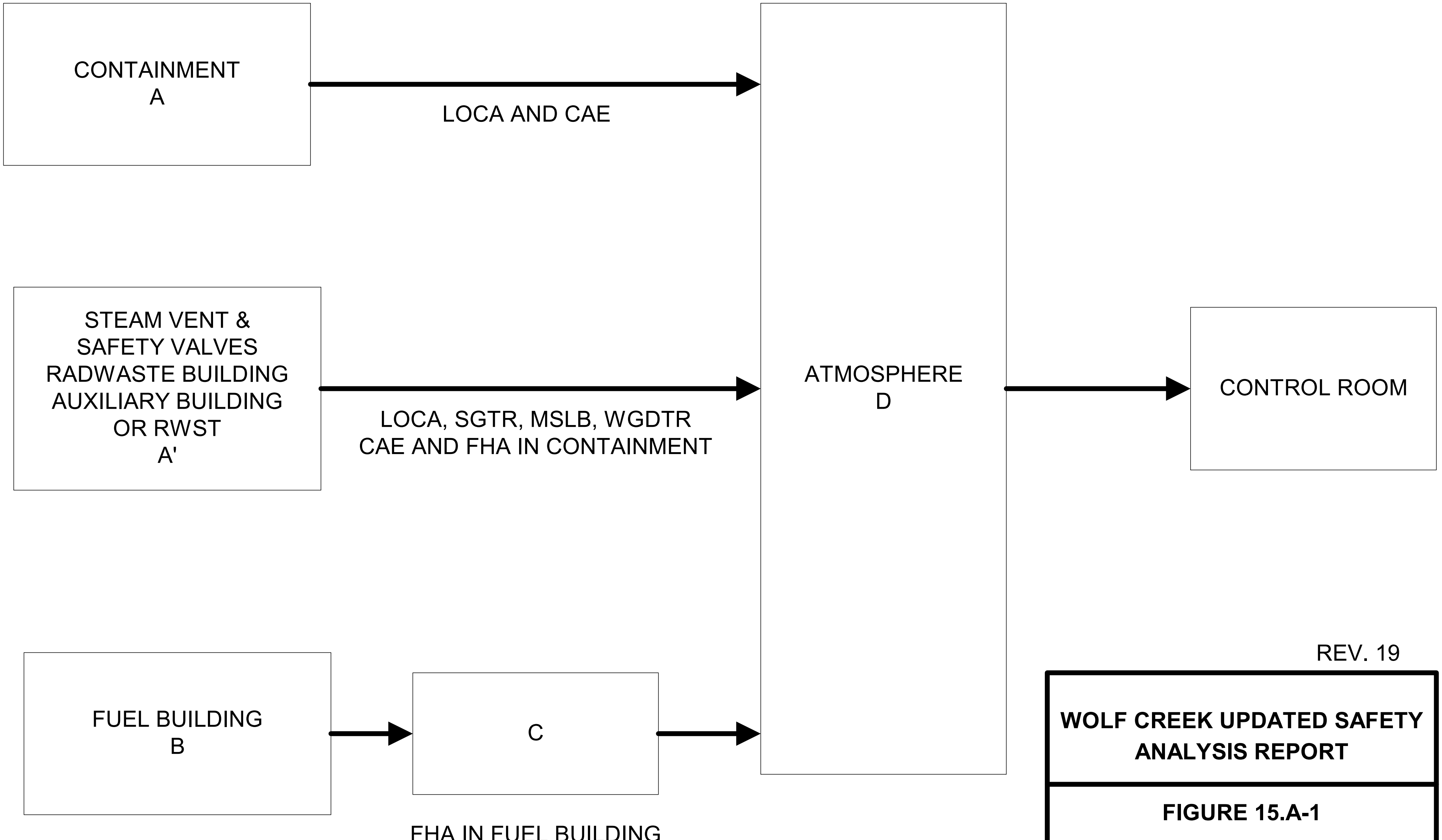
*Gap activity is assumed to be 10 percent of core activity for all isotopes except for Kr-85; for Kr-85 it is assumed to be 30 percent of the core activity. However, gap activity for I-131 is assumed to be 12% instead of 10% of the core activity for fuel handling accident, locked rotor accident and rod ejection accident analyses to account for extended burnup fuel.

WOLF CREEK

TABLE 15A-4

DOSE CONVERSION FACTORS USED IN ACCIDENT ANALYSIS

Nuclide	Total Body $\frac{\text{Rem-meter}^3}{\text{Ci-sec}}$	Beta Skin $\frac{\text{Rem-meter}^3}{\text{Ci-sec}}$	Thyroid Rem/Ci
I-131	8.72E-2	3.17E-2	1.49E+6
I-132	5.13E-1	1.32E-1	1.43E+4
I-133	1.55E-1	7.35E-2	2.69E+5
I-134	5.32E-1	9.23E-2	3.73E+3
I-135	4.21E-1	1.29E-1	5.60E+4
Kr-83m	2.40E-6	0	NA
Kr-85m	3.71E-2	4.63E-2	NA
Kr-85	5.llE-4	4.25E-2	NA
Kr-87	1.88E-1	3.09E-1	NA
Kr-88	4.67E-1	7.52E-2	NA
Kr-89	5.27E-1	3.20E-1	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-137	4.51E-2	3.87E-1	NA
Xe-138	2.80E-1	1.31E-1	NA



REV. 19

WOLF CREEK UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.A-1

RELEASE PATHWAYS