

# Davis-Besse Unit 1 Updated Final Safety Analysis Report

## SECTION 15

### TABLE OF CONTENTS

<u>Section</u>	<u>Title</u>	<u>Page</u>
15.0	ACCIDENT ANALYSIS	15.1-1
15.1	GENERAL	15.1-1
15.1.1	<u>Accident Classifications</u>	15.1-1
15.1.2	<u>Reactor Protection</u>	15.1-2
15.1.3	<u>Uncertainties</u>	15.1-4
15.1.4	<u>Radiological Consequences</u>	15.1-4
15.1.5	<u>Systems Interdependency</u>	15.1-4
15.2	CLASS 1 – EVENTS LEADING TO NO RADIOACTIVE RELEASE AT EXCLUSION AREA BOUNDARY	15.2-1
15.2.1	<u>Uncontrolled Control Rod Assembly Group Withdrawal from a Subcritical Condition (Startup Accident)</u>	15.2-4
15.2.1.1	Identification of Causes	15.2-4
15.2.1.2	Analyses of Effects and Consequences	15.2-4
15.2.1.2.1	Safety Evaluation Criteria	15.2-4
15.2.1.2.2	Methods of Analysis	15.2-5
15.2.1.2.3	Results of Analysis	15.2-5
15.2.1.3	Effects of Changes	15.2-5
15.2.2	<u>Uncontrolled Control Rod Assembly Group Withdrawal at Power</u>	15.2-8
15.2.2.1	Identification of Causes	15.2-8
15.2.2.2	Analysis of Effects and Consequences	15.2-8
15.2.2.2.1	Safety Evaluation Criteria	15.2-8
15.2.2.2.2	Methods of Analysis	15.2-8
15.2.2.2.3	Results of Analysis	15.2-9
15.2.2.2.4	Impact of Replacement Steam Generators	15.2-10
15.2.3	<u>Control Rod Assembly Misalignment (Stuck-Out, Stuck-In, or Dropped Control Rod Assembly)</u>	15.2-12
15.2.3.1	Identification of Causes	15.2-12
15.2.3.2	Analysis of Effects and Consequences	15.2-12
15.2.3.2.1	Safety Evaluation Criteria	15.2-12
15.2.3.2.2	Methods of Analysis	15.2-12
15.2.3.2.3	Results of Analysis	15.2-13
15.2.3.2.4	Additional Analyses	15.2-13
15.2.3.2.5	Impact of Replacement Steam Generators	15.2-14
15.2.4	<u>Makeup and Purification System Malfunction</u>	15.2-16
15.2.4.1	Identification of Causes	15.2-16
15.2.4.2	Analysis of Effects and Consequences	15.2-17
15.2.4.2.1	Safety Evaluation Criteria	15.2-17
15.2.4.2.2	Methods of Analysis	15.2-17
15.2.4.2.3	Results of Analysis	15.2-17
15.2.4.2.4	Impact of Replacement Steam Generators	15.2-18

# Davis-Besse Unit 1 Updated Final Safety Analysis Report

## TABLE OF CONTENTS (CONTINUED)

<u>Section</u>	<u>Title</u>	<u>Page</u>
15.2.5	<u>Loss of Forced Reactor Coolant Flow (Partial, Complete, and Single Reactor Coolant Pump Locked Rotor)</u>	15.2-21
15.2.5.1	Identification of Causes	15.2-21
15.2.5.2	Analysis of Effects and Consequences	15.2-21
15.2.5.2.1	Safety Evaluation Criteria	15.2-21
15.2.5.2.2	Methods of Analysis	15.2-22
15.2.5.2.3	Results of Analysis	15.2-23
15.2.5.2.4	Reanalysis of Locked Rotor Accident	15.2-24
15.2.5.2.5	Reanalysis of Total Loss of Flow Accident	15.2-25
15.2.5.2.6	Three Pump Locked Rotor Accident	15.2-26
15.2.5.2.7	Partial Loss of Flow Accident Reanalysis	15.2-26
15.2.5.2.8	Impact of Replacement Steam Generators	15.2-27
15.2.6	<u>Startup of an Inactive Reactor Coolant Loop (Pump Startup Accident)</u>	15.2-30
15.2.6.1	Identification of Causes	15.2-30
15.2.6.2	Analysis of Effects and Consequences	15.2-30
15.2.6.2.1	Safety Evaluation Criteria	15.2-30
15.2.6.2.2	Methods of Analysis	15.2-30
15.2.6.2.3	Results of Analysis	15.2-30
15.2.6.2.4	Additional Analyses	15.2-31
15.2.6.2.5	Impact of Replacement Steam Generators	15.2-31
15.2.7	<u>Loss of External Electrical Load and/or Turbine Trip</u>	15.2-33
15.2.7.1	Identification of Causes	15.2-33
15.2.7.2	Analysis of Effects and Consequences	15.2-33
15.2.7.2.1	Safety Evaluation Criteria	15.2-33
15.2.7.2.2	Methods of Analysis	15.2-33
15.2.7.2.3	Results of Analysis	15.2-33
15.2.7.3	Plant Changes and Effects	15.2-34
15.2.7.3.1	Identification of Changes	15.2-34
15.2.7.3.2	Effects of Changes	15.2-34
15.2.7.3.3	Impact of Replacement Steam Generators	15.2-35
15.2.7.4	Additional Analysis	15.2-35
15.2.8	<u>Loss of Normal Feedwater</u>	15.2-35
15.2.8.1	Identification of Causes	15.2-35
15.2.8.2	Analysis of Effects and Consequences	15.2-36
15.2.8.2.1	Safety Evaluation Criteria	15.2-36
15.2.8.2.2	Methods of Analysis	15.2-36
15.2.8.2.3	Results of Analysis	15.2-36
15.2.8.3	Plant Changes and Effects	15.2-41
15.2.8.3.1	Identification of Changes	15.2-41
15.2.8.3.2	Effects of Plant Changes	15.2-41
15.2.8.4	Reanalysis of Loss of Feedwater (LOFW) Event	15.2-42
15.2.8.4.1	Need for Reanalysis	15.2-42
15.2.8.4.2	Analysis Results	15.2-42
15.2.8.5	Impact of Replacement Steam Generators	15.2-44
15.2.9	<u>Loss of all AC Power to the Station Auxiliaries (Station Blackout)</u>	15.2-48
15.2.9.1	Identification of Causes	15.2-48

# Davis-Besse Unit 1 Updated Final Safety Analysis Report

## TABLE OF CONTENTS (CONTINUED)

<u>Section</u>	<u>Title</u>	<u>Page</u>
15.2.9.2	Analysis of Effects and Consequences	15.2-48
15.2.9.2.1	Safety Evaluation Criteria	15.2-48
15.2.9.2.2	Methods of Analysis	15.2-48
15.2.9.2.3	Results of Analysis	15.2-48
15.2.9.2.4	Impact of Replacement Steam Generators	15.2-49
15.2.10	<u>Excessive Heat Removal Due to Feedwater System Malfunction</u>	15.2-49
15.2.10.1	Identification of Causes	15.2-49
15.2.10.2	Analysis of Effects and Consequences	15.2-50
15.2.10.2.1	Safety Evaluation Criteria	15.2-50
15.2.10.2.2	Methods of Analysis	15.2-50
15.2.10.2.3	Results of Analysis	15.2-50
15.2.10.2.4	Effects of Plant Changes	15.2-52
15.2.10.2.5	Additional Analyses	15.2-53
15.2.10.2.6	Impact of Replacement Steam Generators	15.2-53
15.2.11	<u>Excessive Load Increase</u>	15.2-55
15.2.11.1	Identification of Causes	15.2-55
15.2.11.2	Analysis of Effects and Consequences	15.2-55
15.2.11.2.1	Safety Evaluation Criteria	15.2-55
15.2.11.2.2	Results of Analysis	15.2-55
15.2.11.2.3	Comparison of Excessive Load Increase Accident to Inadvertent Opening of Pressurizer Safety Valve Accident	15.2-56
15.2.11.2.4	Impact of Replacement Steam Generators	15.2-57
15.2.12	<u>Anticipated Variations in the Reactivity of the Reactor</u>	15.2-57
15.2.12.1	Identification of Causes	15.2-57
15.2.12.2	Analysis of Effects and Consequences	15.2-57
15.2.12.2.1	Safety Evaluation Criteria	15.2-57
15.2.12.2.2	Methods of Analysis	15.2-57
15.2.12.2.3	Results of Analysis	15.2-58
15.2.12.2.4	Impact of Replacement Steam Generators	15.2-58
15.2.13	<u>Failure of Regulating Instrumentation</u>	15.2-60
15.2.13.1	Accident Analysis	15.2-60
15.2.13.2	Impact of Replacement Steam Generators	15.2-60
15.2.14	<u>External Causes</u>	15.2-60
15.2.14.1	Accident Analysis	15.2-60
15.2.14.2	Impact of Replacement Steam Generators	15.2-60
15.3	CLASS 2 – EVENTS LEADING TO SMALL TO MODERATE RADIOACTIVE RELEASES AT EXCLUSION AREA BOUNDARY	15.3-1
15.3.1	<u>Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes which Actuates Emergency Core Cooling</u>	15.3-3
15.3.1.1	Accident Analysis	15.3-3
15.3.1.2	Effects of Plant Changes	15.3-3
15.3.1.3	Impact of Replacement Steam Generators	15.3-3

# Davis-Besse Unit 1 Updated Final Safety Analysis Report

## TABLE OF CONTENTS (CONTINUED)

<u>Section</u>	<u>Title</u>	<u>Page</u>
15.3.2	<u>Minor Secondary System Pipe Break</u>	15.3-4
15.3.2.1	Accident Analysis	15.3-4
15.3.2.2	Impact of Replacement Steam Generators	15.3-4
15.3.3	<u>Inadvertent Loading of a Fuel Assembly Into an Improper Position</u>	15.3-4
15.3.3.1	Identification of Causes	15.3-4
15.3.3.2	Analysis	15.3-5
15.3.3.3	Impact of Replacement Steam Generators	15.3-6
15.4	CLASS 3 – DESIGN BASIS ACCIDENTS	15.4-1
15.4.1	<u>Waste Gas Decay Tank Rupture</u>	15.4-4
15.4.1.1	Identification of Causes	15.4-4
15.4.1.2	Analysis of Effects and Consequences	15.4-4
15.4.1.2.1	Safety Evaluation Criteria	15.4-4
15.4.1.2.2	Methods of Analysis	15.4-4
15.4.1.2.3	Results of Analysis	15.4-5
15.4.1.2.4	Impact of Replacement Steam Generators	15.4-5
15.4.2	<u>Steam Generator Tube Rupture</u>	15.4-7
15.4.2.1	Identification of Causes	15.4-7
15.4.2.2	Accident Analysis	15.4-7
15.4.2.2.1	Safety Evaluation Criteria	15.4-7
15.4.2.2.2	Methods of Analysis	15.4-7
15.4.2.2.3	Results of Analysis	15.4-8
15.4.2.2.4	Environmental Consequences	15.4-9
15.4.2.2.5	Consequences of Less Severe Ruptures	15.4-9
15.4.2.2.6	Effects of Plant Changes	15.4-9
15.4.2.2.6.1	24 Month Fuel Cycle	15.4-9
15.4.2.2.6.2	Steam Generator Replacement	15.4-9
15.4.3	<u>CRA Ejection Accident</u>	15.4-13
15.4.3.1	Identification of Causes	15.4-13
15.4.3.2	Accident Analysis	15.4-15
15.4.3.2.1	Safety Evaluation Criteria	15.4-15
15.4.3.2.2	Methods of Analysis	15.4-15
15.4.3.2.3	Results of Analysis	15.4-16
15.4.3.2.4	Energy Required to Produce Further Reactor Coolant System Damage	15.4-17
15.4.3.2.5	Conclusions	15.4-18
15.4.3.2.5.1	Partial Coolant Flow Condition	15.4-18
15.4.3.2.6	Environmental Consequences	15.4-18
15.4.3.2.7	Additional Analyses	15.4-19
15.4.3.2.8	Impact of Replacement Steam Generators	15.4-21
15.4.4	<u>Steam Line Break</u>	15.4-26
15.4.4.1	Identification of Causes	15.4-26
15.4.4.2	Accident Analysis	15.4-26
15.4.4.2.1	Safety Evaluation Criteria	15.4-26
15.4.4.2.2	Methods of Analysis	15.4-26
15.4.4.2.3	Results of Analysis	15.4-29

# Davis-Besse Unit 1 Updated Final Safety Analysis Report

## TABLE OF CONTENTS (CONTINUED)

<u>Section</u>	<u>Title</u>	<u>Page</u>
15.4.4.2.3.1	Minor secondary pipe break	15.4-29
15.4.4.2.3.2	Double-ended main steam line break	15.4-29
15.4.4.2.3.3	Containment Vessel pressure	15.4-31
15.4.4.2.4	Environmental Consequences	15.4-32
15.4.4.2.5	Conclusions	15.4-32
15.4.4.2.6	Additional Analyses	15.4-32
15.4.4.2.6.1	Control Room Habitability	15.4-32
15.4.4.2.6.2	Partial Coolant Flow	15.4-33
15.4.4.2.6.3	Steam Line Break (inside containment) Dual S/G Blowdown	15.4-33
15.4.4.2.6.4	Minimum Reactivity Margin Evaluation	15.4-34
15.4.4.2.6.5	Comparison of Controlling Parameters With and Without Offsite Power	15.4-35
15.4.4.2.6.6	Steam Line Break Concurrent with Operator Error Allowing Continued Feedwater	15.4-35
15.4.4.2.6.7	Moderator Coefficient Evaluation	15.4-37
15.4.4.2.6.8	Reanalysis of Steam Line Break in Containment	15.4-38
15.4.4.3	Plant Changes and Effects	15.4-38
15.4.4.3.1	Post June 9, 1985 Loss of Feedwater Event	15.4-38
15.4.4.3.2	Lowering the RPS Low Pressure Trip Setpoint	15.4-39
15.4.4.3.3	Raising the Maximum Allowable Steam Generator Water Level	15.4-39
15.4.4.3.4	Lowering the SFAS RCS Low Pressure Trip Setpoint	15.4-41
15.4.4.3.5	Impact of Replacement Steam Generators	15.4-41
15.4.5	<u>Break in Instrument Lines or Lines from Primary System that Penetrate Containment</u>	15.4-46
15.4.5.1	Identification of Causes	15.4-46
15.4.5.2	Analysis of Effects and Consequences	15.4-46
15.4.5.2.1	Safety Evaluation Criterion	15.4-46
15.4.5.2.2	Methods of Analysis	15.4-46
15.4.5.2.3	Environmental Consequences	15.4-47
15.4.5.3	Effects of Plant Changes	15.4-47
15.4.5.3.1	Lowering the RPS Low Pressure Reactor Trip Setpoint	15.4-47
15.4.5.3.2	Lowering the SFAS Low Pressure Reactor Trip Setpoint	15.4-48
15.4.5.3.3	Impact of Replacement Steam Generators	15.3-49
15.4.6	<u>Major Rupture of Pipes Containing Reactor Coolant up to and Including Double-Ended Rupture of the Largest Pipe in the Reactor Coolant System (Loss-of-Coolant Accident)</u>	15.4-51
15.4.6.1	Accident Analysis	15.4-51
15.4.6.2	Safety Evaluation Criterion	15.4-51
15.4.6.3	Environmental Analysis of Loss-of-Coolant Accident	15.4-51
15.4.6.4	Maximum Hypothetical Accident	15.4-51
15.4.6.5	Effects of Engineered Safety Features Leakage During the Maximum Hypothetical Accident	15.4-59
15.4.6.6	Control Room Habitability	15.4-63
15.4.6.7	Partial Loop Flow LOCA	15.4-63
15.4.6.8	Additional Analyses and Related Plant Modifications	15.4-64

# Davis-Besse Unit 1 Updated Final Safety Analysis Report

## TABLE OF CONTENTS (CONTINUED)

<u>Section</u>	<u>Title</u>	<u>Page</u>
15.4.6.8.1	DELETED	
15.4.6.8.2	Positive Moderator Temperature Coefficient (MTC) Analysis	15.4-64
15.4.6.8.3	DELETED	
15.4.6.8.4	DELETED	
15.4.6.8.5	Impact of Replacement Steam Generators	15.4-64
15.4.7	<u>Fuel-Handling Accident</u>	15.4-64
15.4.7.1	Identification of Causes	15.4-64
15.4.7.2	Accident Analysis – Accident Outside Containment	15.4-65
15.4.7.2.1	Safety Evaluation Criterion	15.4-65
15.4.7.2.2	Methods of Analysis	15.4-65
15.4.7.2.3	Results of Analysis	15.4-65
15.4.7.2.4	Environmental Consequences	15.4-66
15.4.7.2.5	Additional Analyses	15.4-66
15.4.7.2.5.1	Effects of Extended Fuel Cycles, Fuel Burnup and Increased Fuel Enrichments	15.4-66
15.4.7.2.5.2	Results	15.4-66
15.4.7.2.5.3	DELETED	
15.4.7.2.6	Impact of Replacement Steam Generators	15.4-68
15.4.7.3	Accident Analysis – Accident Inside Containment	15.4-73
15.4.7.3.1	Safety Evaluation Criterion	15.4-73
15.4.7.3.2	Analysis	15.4-73
15.4.7.3.3	Environmental Consequences	15.4-74
15.4.7.3.4	Additional Analysis	15.4-74
15.4.7.3.4.1	Effects of Extended Fuel Cycle, Fuel Burnup and Increased Fuel Enrichments	15.4-74
15.4.7.3.4.2	Results	15.4-74
15.4.7.3.4.3	Control Room Dose Analysis	15.4-75
15.4.7.3.5	Impact of Replacement Steam Generators	15.4-75
15.4.8	<u>Effects of Toxic Material Release on the Control Room</u>	15.4-79
15.4.8.1	Identification of Causes	15.4-79
15.4.8.2	Accident Analysis	15.4-79
15.4.8.3	Results of the Analysis	15.4-83
15.4.8.3.1	Impact of Replacement Steam Generators	15.4-83
15.5	REFERENCES	15.5-1
APPENDIX 15-A – RADIATION SOURCES		15.A-1

# Davis-Besse Unit 1 Updated Final Safety Analysis Report

## LIST OF TABLES

<u>Table</u>	<u>Title</u>	<u>Page</u>
15.1-1	Digital Computer Programs/Analog Simulations	15.1-5
15.1-2	Parameters Applicable to All Accidents in the Accident Analysis	15.1-8
15.1-3	Worst Possible Control Parameters at Time of In Core Life	15.1-11
15.2-1	Class 1 Events	15.2-2
15.2.1-1	Startup Accident Parameters	15.2-7
15.2.1-2	Summary of Startup Accident Analysis	15.2-7
15.2.2-1	CRA Withdrawal Accident Parameters	15.2-11
15.2.2-2	Summary of CRA Withdrawal Accident Analysis	15.2-11
15.2.3-1	Dropped CRA Accident Parameters	15.2-15
15.2.3-2	Summary of Maximum Numerical Values for Dropped CRA Accident	15.2-15
15.2.4-1	Moderator Dilution Accident Parameters	15.2-19
15.2.4-2	Summary of Moderator Dilution Accident Analysis	15.2-20
15.2.4-3	Table of Moderator Dilution Accident for Partial Pump Operation	15.2-20
15.2.5-1	Loss-of-Coolant-Flow Accident Parameters	15.2-28
15.2.5-2	Locked Rotor Accident Parameters	15.2-28
15.2.5-3	Summary of Loss-of-Coolant-Flow Accident Analysis	15.2-29
15.2.5-4	Natural Circulation Capability	15.2-29
15.2.6-1	Pump Startup Accident Parameters	15.2-32
15.2.6-2	Summary of Pump Startup Accident Analysis	15.2-32
15.2.8-1	Loss of Normal Feedwater Accident Parameters	15.2-45

# Davis-Besse Unit 1 Updated Final Safety Analysis Report

## LIST OF TABLES (CONTINUED)

<u>Table</u>	<u>Title</u>	<u>Page</u>
15.2.8-2	Summary of Loss of Normal Feedwater Analysis	15.2-45
15.2.8-3	Loss of Normal Feedwater Accident Parameters for the Reanalysis	15.2-45
15.2.8-4	Sequence of Events for Peak RCS Pressure Case	15.2-46
15.2.8-5	Sequence of Events for Peak Pressurizer Level Case	15.2-47
15.2.10-1	Excessive Heat Removal Accident Parameters	15.2-54
15.2.12-1	Uncompensated Operating Reactivity Change Parameters	15.2-59
15.2.12-2	Uncompensated Reactivity Disturbances	15.2-59
15.3-1	Class 2 Events	15.3-2
15.4-1	Class 3 Events	15.4-2
15.4.1-1	Resultant Doses from Waste Gas Tank Rupture	15.4-6
15.4.1-2	Activity Released Due to Waste Gas Tank Rupture (Ci)	15.4-6
15.4.2-1	Steam Generator Tube Failure Parameters	15.4-11
15.4.2-2	Summary of Steam Generator Tube Failure Analysis	15.4-12
15.4.2-3	Resultant Doses from Steam Generator Tube Rupture	15.4-12
15.4.3-1	Control Rod Assembly Ejection Accident Parameters	15.4-22
15.4.3-2	Nominal Values of Input Parameters for CRA Ejection Accident Analysis	15.4-22
15.4.3-3	Comparison of Space-Dependent and Point Kinetics Results of Fuel Enthalpy	15.4-23



Davis-Besse Unit 1 Updated Final Safety Analysis Report

LIST OF TABLES (CONTINUED)

<u>Table</u>	<u>Title</u>	<u>Page</u>
15.4.3-4	Summary of Control Rod Assembly Ejection Accident Analysis	15.4-23
15.4.3-5	Reactor Vessel Parameters	15.4-24
15.4.3-6	Resultant Doses from a CRA Ejection Accident	15.4-25
15.4.4-1	Steam Line Failure Parameters	15.4-42
15.4.4-2	Mass and Energy Releases for Building Pressure Analysis	15.4-42
15.4.4-3	Summary of Steam Line Failure Analysis	15.4-43
15.4.4-4	Resultant Doses from a Steam Line Failure in Mode 1	15.4-43
15.4.4-4a	Resultant Doses from a Steam Line Failure in Mode 3 With SG Level at 96% Operate Range	15.4-43
15.4.4-5	Minimum Reactivity Margins for Various Main Steam Line Break Situations	15.4-44
15.4.4-6	Steam Line Failure with concurrent MSSV Failure Parameters	15.4-45
15.4.5-1	Activity Released to Auxiliary Building from Letdown Line Rupture	15.4-50
15.4.5-2	Resultant Doses from Letdown Line Rupture	15.4-50
15.4.6-1	Resultant Doses from Maximum Break Size LOCA	15.4-61
15.4.6-2	Resultant Doses from MHA	15.4-62
15.4.6-3	DELETED	
15.4.6-4	DELETED	
15.4.7-1	Fuel-Handling Accident Parameters Outside Containment	15.4-69
15.4.7-1a	Fuel Handling Accident Assumptions – Outside Containment	15.4-70
15.4.7-2	Resultant Doses from Fuel-Handling Accident Outside Containment	15.4-71

# Davis-Besse Unit 1 Updated Final Safety Analysis Report

## LIST OF TABLES (CONTINUED)

<u>Table</u>	<u>Title</u>	<u>Page</u>
15.4.7-2a	Resultant Doses from Fuel Handling Accident Outside Containment – Extended Fuel Burnup (60,000 MWD/MTU)	15.4-71
15.4.7-3	Activity Released to the Atmosphere Due to the Postulated Fuel-Handling Accident Outside Containment (Ci)	15.4-72
15.4.7-4	Resultant Doses from Fuel-Handling Accident Inside Containment	15.4-76
15.4.7-4a	Resultant Doses from Fuel Handling Accident Inside Containment – Extended Fuel Burnup (60,000 MWD/MTU)	15.4-76
15.4.7-5	Activity Released to the Atmosphere Due to the Postulated Fuel-Handling Accident Inside Containment (Ci)	15.4-77
15.4.7-6	Fuel Assembly-Average Fission Product Activities (Curies) for Extended Fuel Burnup (60,000 MWD/MTU)	15.4-78

# Davis-Besse Unit 1 Updated Final Safety Analysis Report

## LIST OF FIGURES

<u>Figure</u>	<u>Title</u>
15.1-1	Normalized Rod Worth Versus Time
15.2.1-1	Startup Accident from 10-9 Rated Power for a Reactivity Addition Rate of $1.65 \times 10^{-4}$ (Dk/k)/s High Pressure Reactor Trip is Actuated
15.2.1-2	Startup Accident From 10-9 Rated Power for a Reactivity Addition Rate of $7.19 \times 10^{-4}$ (Dk/k)/s (Simultaneous Withdrawal of All CRA's): High Flux Reactor Trip is Actuated
15.2.1-3	Peak Thermal Power Versus Reactivity Addition Rate for a Startup Accident From 10-9 Rated Power
15.2.1-4	Peak Neutron Power Versus Reactivity Addition Rate for a Startup Accident from 10-9 Rated Power
15.2.1-5	Peak Thermal Power Versus Doppler Coefficient for a Startup Accident with a Constant Reactivity Addition Rate of $1.65 \times 10^{-4}$ (Dk/k)/s From 10-9 Rated Power
15.2.1-6	Peak Thermal Power Versus Moderator Coefficient for a Startup Accident with a Constant Reactivity Addition Rate of $1.65 \times 10^{-4}$ (Dk/k)/s From 10-9 Rated Power
15.2.1-7	Peak Thermal Power Versus Doppler Coefficient for a Startup Accident with a Reactivity Addition Rate of $7.19 \times 10^{-4}$ (Dk/k)/s (Simultaneous Withdrawal of all CRA's) From 10-9 Rated Power
15.2.1-8	Peak Thermal Power Versus Moderator Coefficient for a Startup Accident with a Reactivity Addition Rate of $7.19 \times 10^{-4}$ (Dk/k)/s (Simultaneous Withdrawal of all CRA's) From 10-9 Rated Power
15.2.2-1	CRA Withdrawal Accident From Rated Power for a Reactivity Addition Rate of $2.3 \times 10^{-4}$ (Dk/k)/s; High Flux Reactor Trip is Actuated
15.2.2-2	Peak Pressure Versus Reactivity Addition Rate for a CRA Withdrawal Accident From Rated Power
15.2.2-3	Peak Pressure Versus Trip Delay Time for a CRA Withdrawal Accident from Rated Power with a Constant Reactivity Addition Rate of $2.3 \times 10^{-4}$ (Dk/k)/s

## Davis-Besse Unit 1 Updated Final Safety Analysis Report

### LIST OF FIGURES (CONTINUED)

<u>Figure</u>	<u>Title</u>
15.2.2-4	Peak Pressure Versus Doppler Coefficient for a CRA Withdrawal Accident From Rated Power with a Constant Reactivity Addition Rate of $2.3 \times 10^{-4} \Delta k/k/s$
15.2.2-5	Peak Pressure Versus Moderator Coefficient for a CRA Withdrawal Accident From Rated Power with a Constant Reactivity Addition Rate of $2.3 \times 10^{-4} \Delta k/k/s$
15.2.2-6	Maximum Neutron and Thermal Power for an All-CRA Withdrawal Accident From Various Initial Power Levels
15.2.2-7	Peak Fuel Temperature in Average Rod and Hot Spot for an All-CRA Withdrawal Accident From Various Initial Power Levels
15.2.2-8	Peak Reactor Coolant Pressure vs. Power for All CRA Croup Withdrawal
15.2.3-1	0.65% $\Delta k/k$ CRA Drop From Rated Power at EOL Condition
15.2.5-1	Percent Reactor Coolant Flow as a Function of Time After Loss of Pump Power
15.2.5-2	Percent Neutron Power Versus Time Following Reactor Trip
15.2.5-3	Minimum DNBR Which Occurs During a Four Pump Coastdown From Various Initial Power Levels
15.2.5-4	Neutron Power, Flow, and Reactor System Pressure for a Locked Rotor Accident, BOL Parameters
15.2.5-5	DNB Ratio Versus Time for Locked Rotor Accident From 102 Percent of Rated Power
15.2.6-1	Two Pump Startup From 60% Power and 49% Flow
15.2.7-1	Loss of External Load at Rated Power With Automatic Power Run Back
15.2.8-1	Loss of All Feedwater From Rated Power

# Davis-Besse Unit 1 Updated Final Safety Analysis Report

## LIST OF FIGURES (CONTINUED)

<u>Figure</u>	<u>Title</u>
15.2.8-1a	Steam Generator Collapsed Level (Loop 1)
15.2.8-1b	Hot Leg Temperature (Loop 1)
15.2.8-1c	Hot Leg Pressure (Loop 1)
15.2.8-1d	Pressurizer Collapsed Liquid Level
15.2.8-2	Feedwater Line Break With Offsite Power Available
15.2.8-3	Feedwater Line Break With Offsite Power Available
15.2.8-4	Feedwater Line Break With Offsite Power Available
15.2.8-5	Feedwater Line Rupture With Offsite Power Available - Case I
15.2.8-6	Feedwater Line Break With Offsite Power Available
15.2.8-7	Feedwater Line Break With Loss of Offsite Power at Time of Rupture
15.2.8-8	Feedwater Line Rupture With Loss of Offsite Power at Time of Rupture - Case II
15.2.8-9	Feedwater Line Break With Loss of Offsite Power at Time of Rupture
15.2.8-10	Feedwater Line Break With Loss of Offsite Power at Time of Rupture
15.2.8-11	Feedwater Line Rupture With Loss of Offsite Power at Time of Trip - Case III
15.2.8-12	Feedwater Line Break With Loss of Offsite Power at Time of Trip
15.2.8-13	Feedwater Line Break With Loss of Offsite Power at Time of Trip
15.2.8-14	Feedwater Line Break With Loss of Offsite Power at Time of Trip
15.2.9-1	Loss of All A.C. Power While Operating at Rated Power

# Davis-Besse Unit 1 Updated Final Safety Analysis Report

## LIST OF FIGURES (CONTINUED)

<u>Figure</u>	<u>Title</u>
15.2.10-1	Response of Reactor Coolant System to Feedwater Temperature Decrease at Rated Power
15.2.10-2	Response of Reactor Coolant System to Feedwater Flow Increase to No Load Condition
15.2.10-3	Excessive Heat Removal Due to 115 Percent FW Flow
15.3.3-1	Radial x Local Assembly Power Distribution - Case 3A
15.3.3-2	Radial x Local Assembly Power Distribution - Case 3B
15.4.3-1	Peak Neutron Power as a Function of Ejected Control Rod Assembly Worth
15.4.3-2	Peak Thermal Power as a Function of Ejected Control Rod Assembly Worth
15.4.3-3	Peak Enthalpy of Hottest Fuel Rod as a Function of Ejected Control Rod Assembly Worth
15.4.3-4	Peak Neutron Power as a Function of Doppler Coefficient for an Ejected CRA Worth of 0.65 Percent $\Delta k/k$ at Both $10^{-3}$ Rated Power and Rated Power
15.4.3-5	Peak Thermal Power as a Function of Doppler Coefficient for an Ejected CRA Worth of 0.65 Percent $\Delta k/k$ at Both $10^{-3}$ Rated Power and Rated Power
15.4.3-6	Peak Neutron Power as a Function of Moderator Coefficient for an Ejected CRA Worth of 0.65 Percent $\Delta k/k$ at Both $10^{-3}$ Rated Power and Rated Power
15.4.3-7	Peak Thermal Power as a Function of Moderator Coefficient for an Ejected CRA Worth of 0.65 Percent $\Delta k/k$ at Both $10^{-3}$ Rated Power and Rated Power
15.4.3-8	Peak Thermal Power as a Function of Trip Delay Time for an Ejected CRA Worth of 0.65 Percent $\Delta k/k$ at Both $10^{-3}$ Rated Power and Rated Power

## Davis-Besse Unit 1 Updated Final Safety Analysis Report

### LIST OF FIGURES (CONTINUED)

<u>Figure</u>	<u>Title</u>
15.4.3-9	Percent Pins Experiencing DNB as a Function of Ejected Control Rod Assembly Worth at Rated Power, BOL
15.4.4-1	Double-Ended Rupture of 36-Inch Steam Line Between Steam Generator and Main Steam Isolation Valve
15.4.4-2	Double-Ended Rupture of 36-Inch Steam Line Between Steam Generator and Main Steam Isolation Valve
15.4.4-3	Double-Ended Rupture of 36-Inch Steam Line Between Steam Generator and Main Steam Isolation Valve
15.4.4-4	Main Steam Line Break Temperature vs. Time
15.4.4-5	Time From Rupture to Trip Versus Steam Line Break Size
15.4.4-6	Double-Ended Rupture of 36-Inch Steam Line Between S/G and MSIV With Stuck Open Safety Valve
15.4.4-7	Double-Ended Rupture of 36-Inch Steam Line Between S/G and MSIV With Stuck Open Safety Valve
15.4.4-8	Double-Ended Rupture of 36-Inch Steam Line Between S/G and MSIV With Stuck Open Safety Valve
15.4.4-9	Pressure Transient at SFRCS Pressure Tap Location
15.4.4-10	Pressure Transient at SFRCS Pressure Tap Location
15.4.5-1	Reactor Coolant Pressure as a Function of Time for the Complete Severance of a Letdown Line
15.4.6-1	Cladding Temp for 8.55 ft <sup>2</sup> Break at RCP Discharge During Partial Flow
15.4.6-2	Core Flow for 8.55 ft <sup>2</sup> Break at RCP Discharge During Partial Flow
15.4.8-1	Chlorine Concentration After a Tank Car Rupture
15.4.8-2	Chlorine Concentration After a Pipe Line Break

SECTION 15

15.0 ACCIDENT ANALYSIS

15.1 GENERAL

NOTE: Chapter 15 is updated by adding new discussions to reflect plant changes and new analyses at the end of an accident section. The discussions of the original licensing basis remain to provide a historical perspective of the accident analysis.

License Amendment No. 278 increased core rated thermal power by 1.63% from 2772 MWt to 2817 MWt, based on the use of more accurate Caldon LEFM CheckPlus™ instrumentation for heat balance measurement. Each accident was evaluated with respect to the power uprate and was found to meet the required event-specific acceptance criteria, as described in Reference 62.

Both normal and abnormal operations of the various systems and components and the susceptibility of individual components to malfunction or failure are discussed in prior sections of the USAR. This chapter summarizes and further explores the consequences of these abnormal situations or failures. All accidents and environmental consequences have been evaluated for a design core power of 2817 MWt, which corresponds to the nuclear heat load portion of the ultimate steam capacity of the main turbine. The transient conditions resulting from all accidents are analyzed to such an extent that they are shown to be:

- a. Inherently terminated;
- b. Terminated by the operation of the Reactor Protection System which is designed to maintain the integrity of the fuel and/or the Reactor Coolant System; and/or
- c. Terminated by the operation of engineered safety features, which are designed to maintain the integrity of the core and/or the Containment Vessel and to reduce the potential offsite doses to the public when one or more of the protective barriers are not effective.

15.1.1 Accident Classifications

The full spectrum of abnormal situations and accidents is divided into three classes in accordance with their anticipated frequency and their radiological consequences as follows:

- a. Class 1 - Events Leading to No Radioactivity Release at Exclusion Area Boundary.
- b. Class 2 - Events Leading to Small to Moderate Radioactivity Release at Exclusion Area Boundary.
- c. Class 3 - Design Basis Accidents.

The events examined have been taken from the listing of accidents required by the Atomic Energy Commission (AEC) Safety Analysis Report (SAR) guide, issued February, 1972. This listing agrees closely with the accidents required by the American Nuclear Society (ANS) Pressurized Water Reactor (PWR) criteria (ANSI N18.2). Although all the accidents recommended by the ANS PWR criteria are not required by the AEC SAR guide, they are incorporated as a part of other accidents which are required by the SAR Guide. An attempt has



been made to classify all accidents in accordance with both the requirements of the AEC SAR guide and the ANS PWR criteria. Since the ANS PWR criteria classifies accidents according to frequency or probability of occurrence, and the AEC SAR guide classifies accidents according to severity of radioactive release, the accidents in this chapter are grouped into classes which best exemplify their nature according to the philosophy of the ANS PWR criteria and the AEC SAR guide.

The basic principle which is demonstrated in relating design requirements to each category of accidents is that the more probable abnormal situations or accidents result in the least radiological risk to the public health and safety, and those extreme situations having the potential for the greatest risk to the public are very improbable.

#### 15.1.2 Reactor Protection

In each accident analysis a description of cause and effect, and order of occurrence for the postulated event is provided. The amount of dependence on the Reactor Protection System (RPS) operation and the conservative values of important design parameters such as reactivity feedback coefficients are indicated in each analysis. Each accident analysis states the RPS or SFAS function used to terminate the transient. Reactor system variables which are monitored in order to provide core protection are summarized in Chapter 7.

The computer codes used in each analysis are listed in Table 15.1-1 or in the list of references for each accident. All codes listed that have been used in the analysis have been submitted in topical report form for code approval or been approved by the NRC for single application use.

The effects of failure of RPS functions during anticipated transients and accidents are discussed in B&W Topical Report BAW-10016 (September 1972), Analysis of Anticipated Transients Without Trip, and BAW-10099 (December 1974), Babcock and Wilcox Anticipated Transients Without Scram Analysis. The effects of failure to obtain the primary reactor trip signal are discussed in B&W Topical Report BAW-10019 (September 1970), Systematic Failure Study of Reactor Protection System. These reports were performed for typical B&W nuclear steam systems to identify and evaluate the potential events which have the greatest possibility to damage the Reactor Coolant System, given that no control rod motion occurs when a reactor trip is expected. These reports are now grouped under the overall anticipated transients without scram (ATWS) subject, and are representative of early ATWS event risk evaluation studies. Overall compliance with 10 CFR 50.62, Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants, is summarized in Reference 39.

All systems or functions utilized in these accident analyses have been designed in a manner such that a single failure of an active component will not prevent them from meeting the performance requirements used in this analysis. No Integrated Control System (ICS) is required for reactor protection, because all accidents were analyzed without ICS (maloperation or failures in the ICS or ICS-controlled equipment are assumed in the analysis if they produce more serious consequences). Operator action for maintaining hot shutdown conditions or cooldown to cold shutdown conditions is required only where adequate time and instruments indications are available to the operator.

Initial conditions for power, flow, pressure, and inlet temperature used in the transient analysis are provided in Table 15.1-2. DNBR for steady-state operation as a function of power level is given in Table 4.4-4. These nominal values are considered appropriate for determination of the most representative average system responses.

To ensure that the transient results are conservative, RPS trip values used in the accident analyses, as given in Table 15.1-2, are based on setpoint values inclusive of maximum measurement errors given in the Technical Specifications. For hot channel calculations of DNBR, transient pressures and temperatures are additionally corrected by lowering the system pressure by -65 psia and raising the coolant inlet temperature by +2°F to account for control band and instrumentation errors.

For accidents which result in reactor trip, a minimum tripped CRA worth for either beginning or end of core life is assumed. The minimum tripped CRA worth accounts for the moderator deficit, Doppler deficit, and the reduction in CRA worth to produce a 1%  $\Delta k/k$  subcritical margin at hot shutdown with the maximum worth CRA stuck out of the core. The values of these individual deficits are given in Chapter 4 for both BOL and EOL conditions. The normalized control rod worth curve used to analyze each event in Chapter 15 is shown in Figure 15.1-1. The assumed axial power profile used to generate the core reactivity versus control rod position was a balanced cosine-shaped curve, generated by allowing the flux to redistribute following each reactivity (rod insertion) step. Xenon redistribution was not accounted for; however, since rods for trip are inserted in a few seconds, this assumption is reasonable. The primary conservatism employed in the rod worths used for the safety analysis is that only the rod worth necessary to overcome temperature effects from HFP to HZP and to provide a 1%  $\Delta k/k$  shutdown margin is assumed to be available. This is very conservative compared to the actual available rod worth. Furthermore, analysis has shown that the rate of neutron power decrease is faster with larger rod worths, which, combined with the parameterization of trip delay times, would compensate for differences in axial power shapes affecting the rate at which shutdown reactivity is added.

The operation imbalance limits and control rod insertion limits relating to axial power profile are specified in the Core Operating Limits Report. Operator requirements for maintaining flux shape are included in the Technical Specifications.

For each transient and accident, a summary (Table 15.1-3) of the time in core life (BOL or EOL) during which each controlling parameter would be at its worst is provided.

The criterion, adopted in these accident analyses to ensure that the Reactor Coolant System boundary integrity is maintained, is that the system pressure shall remain below code pressure limits. The ASME Code Section III pressure limit is 110 percent of the Reactor Coolant System design pressure, 2500 psig (see Chapter 5). The safety limit thus established is 2750 psig.

The criterion, adopted in these accident analyses to ensure that no fuel damage occurs, is that a DNBR greater than 1.3 must be maintained throughout the transient. As demonstrated in Chapter 4, a DNBR of 1.3 corresponds to the 112% of 2772 MWt design overpower condition. Thus if the thermal power during the transient does not exceed 112% of 2772 MWt, there will be no fuel damage. If the DNBR goes below 1.3 during a transient, the gap activity for all of the fuel rods with a DNBR of less than 1.3 is assumed to be released. If the DNBR goes below 1.0 during a transient, the cladding temperature is evaluated to confirm the structural integrity of the fuel rod. These DNBR values represent safety limits and are discussed further in the Technical Specifications. An additional fuel damage criterion used in the CRA Ejection Accident (Subsection 15.4.3) is the fuel enthalpy. Fuel enthalpy is used because of the fast rate of energy addition. For the CRA ejection accident, the fuel integrity is maintained if the peak enthalpy of the hottest rod is less than 210 cal/gm, the threshold energy for the zirconium-water reaction. Above 210 cal/gm the next threshold is 280 cal/gm, above which the fuel rod will probably not be intact. The "safety margin" for any transient is the difference between the peak

value of the controlling parameter and the 112% of 2772 MWt thermal power, 1.3 DNBR, 210 cal/gm or 2750 psig safety limit.

#### 15.1.3 Uncertainties

The evaluation of each accident is based upon conservative engineering assumptions to provide margin for uncertainties in calculational methods. RPS trip values used in the accident analyses are based on the maximum setpoint value plus the maximum uncertainties in measurement. The uncertainties associated with equipment and instrumentation performance are discussed in Chapter 7. Uncertainties in calculated values of parameters are considered by a sensitivity analysis for those parameters.

#### 15.1.4 Radiological Consequences

An evaluation of environmental effects is presented for each accident which results in offsite radiation exposures in excess of normal operating releases. In general, two hour thyroid and whole body doses at the exclusion area boundary and thirty day doses at the outer boundary of the low population zone are given for these accidents. Additional dose values (as described in the AEC SAR Guide) are presented in the analysis when they are required in the overall evaluation of the consequences of a particular accident. Dose values are also presented for analyses made in accordance with AEC Safety Guides governing certain accidents. The atmospheric dispersion factors used in calculating these doses are given in Section 2.3.

A description of the physical and mathematical models employed in calculating radiation source terms is given in Chapter 11. Radiation source terms used in the dose calculations and their calculational basis are summarized in Appendix 15A. The radiation source terms include individual isotopic activities of fission products in fuel, fuel rod gap, reactor coolant and secondary system.

#### 15.1.5 Systems Interdependency

The design and interdependency of the various safety feature actuated systems are discussed in Chapter 6. Each of these systems has been designed with sufficient capacity, structural integrity, and redundancy that a single malfunction or failure of an active component within any one or each system will not compromise the intended operation of the other systems which are directly or indirectly involved in controlling the fission product release or limiting the leakage from the containment vessel. Whenever engineered safety features are used in the accident analysis, systematic malfunctions or failures are taken into account by assuming the minimum performance level.

TABLE 15.1-1

Digital Computer Programs/Analog Simulations

15.2.1 Startup Accident

- Ref. 1 Bingham, B. E. and Rhyne, W. R., "KAPP4-Digital Computer Program for Solution of Reactor Kinetics and Primary System Pressure Response," BAW-10068, June 1973.
- Ref. 2 Hsii, Y. H., Watson, C. E., Vasudevan, N., Busby, S. E., and Trent, R. L., "-CADDs- Computer Application to Direct Digital Simulation of Transients in PWRs With or Without Scram - Revision 1," BAW-10098, Rev. 1, January 1978.
- Ref. 3 Framatome Technologies Group, "RELAP5/MOD2 - B&W for Safety Analysis of B&W-Designed Pressurized Water Reactors," BAW-10193P-A, January 2000.

15.2.2 CRA Withdrawal Accident

Reference 1 - KAPP4

15.2.3 Dropped CRA Accident

Reference 1 - KAPP4

15.2.4 Moderator Dilution Accident

Reference 1 - KAPP4

15.2.5 Loss of Coolant Flow

Reference 1 - KAPP4

- Ref. 3 Morgan, C. D., Cheatwood, H. C., Gloudemans, J. R., "RADAR - Reactor Thermal and Hydraulic Analysis During Reactor Flow Coastdown," BAW-10069A, Rev. 1 (October 1974).
- Ref. 4 Galan, Y. J., Miller, C. K., "SPLIT - Digital Steady-State Flow Distribution Code for Various Primary System Combinations," BAW-10071A, September 1974.
- Ref. 5 M. R. Grandia, "PUMP - Analog-Hybrid Reactor Coolant Hydraulic Transient Model," BAW-10073A, Rev. 1, March 1976.
- Ref. 6 LYNXT- Core Transient Thermal-Hydraulic Program," BAW-10156-A, February 1986.
- Ref. 11 VIPRE-01 "A Thermal-Hydraulic Analysis Code for Reactor Cores," EPRI NP-2511-CCM-A, July 1985.

TABLE 15.1-1 (Continued)

Digital Computer Programs/Analog Simulations

15.2.6	<u>Pump Startup Accident</u> Reference 1 - KAPP4
15.2.7	<u>Loss of Load Accident</u> Ref. 7 Christy, P. T., Galan, V. J., "POWER TRAIN - General Hybrid Simulation for Reactor Coolant and Secondary System Transient Response," BAW-10070, July 1973.
15.2.8	<u>Loss of Feedwater</u> Reference 1 - KAPP 4
Ref. 12	RELAP5/MOD 2 approved for specific use in "Safety Evaluation by the Office of Nuclear Reactor Regulation Relating to the Change of Technical Specification 3/4.7.1.2," Log 2440, November 18, 1987.
Ref. 13	RELAP5/MOD2-B&W (Approved for use per Reference 54 (Section 15.5))
15.2.9	<u>Station Blackout</u> Reference 7 - POWER TRAIN
15.2.10	<u>Excessive Heat Removal</u> Reference 7 - POWER TRAIN
15.2.11	<u>Excessive Load Increase</u> Reference 7 - POWER TRAIN
15.2.12	<u>Normal Reactivity Changes</u> Reference 1 - KAPP4
15.3.1	<u>Loss of Coolant Accident - Small Break</u> Ref. 8 "CRAFT - Description of Model for Equilibrium LOCA Analysis Program," BAW-10030, Babcock and Wilcox, Lynchburg, Va., October 1971. (Historical)
Ref. 13	RELAP5-MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis, BAW-10164P-A, Rev. 4, November 2002.

TABLE 15.1-1 (Continued)

Digital Computer Programs/Analog Simulations

15.3.2	<u>Minor Secondary System Pipe Break</u>
Ref. 9	J. A. Redfield, <u>et al</u> , "FLASH -2: A Fortran IV Program for the Digital Simulation of a Multinode Reactor Plant During Loss-of-Coolant," WAPD-TM-666, April 1967.
15.4.1	<u>Waste Gas Decay Tank Rupture</u>
Ref. 10	The method used to calculate all coolant activities is described in detail in Chapter 11.
15.4.2	<u>Steam Generator Tube Rupture</u>
	Reference 10 - Coolant Activities
15.4.3	<u>CRA Ejection Accident</u>
	Reference 1 - KAPP4 Reference 10 - Coolant Activities
15.4.4	<u>Steam Line Break</u>
	Reference 9 - FLASH 2 Reference 10 - Coolant Activities
15.4.6	<u>Loss of Coolant Accident</u>
	Reference 8 - CRAFT (Historical)
	All other methods and assumptions are described in BAW-10034, Rev. 3 (May 1972) "Multinode Analysis of B&W's 2568-MWt Nuclear Plants During a Loss-of-Coolant Accident," and BAW-10105, Rev. 1 (July 1975) "ECCS Evaluation of B&W's 177-FA Raised-Loop NSS - Revision 1." (Historical)
	Beginning with Cycle 13, the LOCA methods and assumptions are described in BAW-10192PA, Rev. 0 (July 1998), "BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants."

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 15.1-2

Parameters Applicable to All Accidents in the Accident Analysis<sup>(1)</sup>

Initial power for accident analysis, MWt	2772 (see notes 4 and 5)
Initial RC system pressure, psia	2200
Initial RC system flow, lb/hr	Ref. Table 4.4-4
Initial RC inlet temperature, °F	
Rated power	Ref. Table 4.4-4
Hot shutdown	Ref. Subsection 4.3
Initial steam generator operation conditions	Ref. Subsection 5.5.2, Table 15.4.4-1
High flux trip value, % rated power	112 (of 2772 MWt)
High flux trip delay, sec	0.4
High pressure trip value, psia	2430
High pressure trip delay, sec	0.6
Low pressure trip value, psia	1900
Low pressure trip delay, sec	0.6
Power/RC pumps trip delay, sec	0.62*
Power/flow trip value	1.08
Power/flow trip delay, sec	1.95
Pressure relief capability, lb/hr of steam	Ref. Subsections 10.3, 10.4
Pressurizer level	(see note 3)
Nominal moderator coeff. at BOL, $\Delta k/k/^\circ F$	$0.13 \times 10^{-4}$ (see note 4)
Nominal moderator coeff. at EOL, $\Delta k/k/^\circ F$	$-3 \times 10^{-4}$ (see note 2)

\*Trip delay time has been changed to 800 msec. for analysis of Section 15.2.5.2.5.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 15.1-2 (Continued)

Parameters Applicable to All Accidents in the Accident Analysis

Nominal Doppler coeff. at BOL, $\Delta k/k/^\circ F$	-1.28x10 <sup>-5</sup> (see note 4)
Nominal Doppler coeff. at EOL, $\Delta k/k/^\circ F$	-1.45x10 <sup>-5</sup> *
CR travel time to 2/3 insertion, sec	1.4
Minimum tripped rod worth at BOL, % $\Delta k/k$	2.26
Minimum tripped rod worth at EOL, % $\Delta k/k$	3.36
Core thermal power at which cladding damage is assumed to occur, % rated power	112 (of 2772 MWt)
Applicable Critical Heat Flux correlation	W-3 (see note 4)
DNBR at steady-state design overpower (112% of 2772 MWt) at full coolant flow	1.41 (see note 4)

\* Doppler Coefficient for Steam Line Break was  $-1.77 \times 10^{-5} \Delta k/k/^\circ F$  for greater conservatism.

- (1) Parameters given in this table are used in all the accidents unless specified under each accident.
- (2) For the Dropped Control Rod Assembly, Inactive RCS Pump Startup, and Control Rod Assembly (CRA) Ejection, a Hot Full Power (HFP) moderator coefficient of  $-4.0 \times 10^{-4} \Delta k/k/^\circ F$  has been used in additional applicable analyses (Reference 31, FTI Document 51-1201479-00). For the Excessive Heat Removal Due To Feedwater System Malfunction event, the limit shown in the table above is applicable at both HFP and Hot Zero Power (HZP) conditions. For the Steam Line Break, a limiting temperature coefficient (combination of moderator and Doppler coefficients) of  $-3.1 \times 10^{-4} \Delta k/k/^\circ F$  has been used at HZP conditions and colder.
- (3) The assumed initial pressurizer level is significant for many transients. Since normal operating pressurizer level has changed several times since beginning of plant life, a consistent value is not used by all analysis. During power operation 220 inches was used in the loss of normal feedwater event analysis (Reference 46, FTI Document 32-1171148-00). Lower levels are assumed during startup, based on startup accident analysis.



TABLE 15.1-2 (Continued)

Parameters Applicable to All Accidents in the Accident Analysis

- (4) Large and small break LOCA analyses were performed using the RELAP5/MOD2-B&W-based evaluation model. The analysis modeled full and mixed cores of Mark-B-HTP fuel. The Critical Heat Flux correlation and DNBR limit are fuel type dependent. In these analyses, the core power was increased to 102% of 2966 MWt and 20% overall SG tube plugging. These calculations also included a 0.0 pcm/°F MTC and a Doppler coefficient (at 1420°F) of  $-2.0 \times 10^{-5} \Delta k/k/^\circ F$ .
- (5) License Amendment No. 278 increased core rated thermal power by 1.63% from 2772 MWt to 2817 MWt, based on the use of more accurate instrumentation for heat balance measurement.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 15.1-3

Worst Possible Control Parameters at Time of In Core Life

<u>Accident</u>	<u>Controlling parameter</u>	<u>Time in life when effect is Maximized</u>
Startup accident	Mod. coeff.	BOL
	Doppler coeff.	BOL
CRA withdrawal accident	Mod. coeff.	BOL
	Doppler coeff.	BOL
Mod dilution	Boron conc.	BOL
Loss of flow	Fuel Temp.	BOL
Pump startup accident	Mod. coeff.	EOL
	Doppler coeff.	EOL
Loss of Load	Mod. coeff.	BOL
Loss of Feedwater	Mod. coeff.	BOL
Station blackout	Decay heat rate	EOL
Excessive heat removal	Mod. coeff.	EOL
Excessive load increase	Mod. coeff.	EOL
Waste gas tank rupture	Coolant activity	EOL
S. G. Tube Rupture	Mod. coeff.	EOL
	Coolant activity	EOL
CRA ejection accident	Mod. coeff.	BOL
	Doppler coeff.	BOL
	Gap activity	EOL
Steam line break	Mod. coeff.	EOL
Loss of coolant	Mod. coeff.	BOL
	Decay heat rate	EOL
	Fuel temp.	BOL*
	Coolant activity	EOL
Fuel handling accident	Gap activity	EOL

\* The small break LOCA analyses model BOL fuel temperatures.  
The large break LOCA is analyzed at different times in core life (see Reference 51 for details).

TABLE 15.1-3 (Continued)

Worst Possible Control Parameters at Time of In Core Life

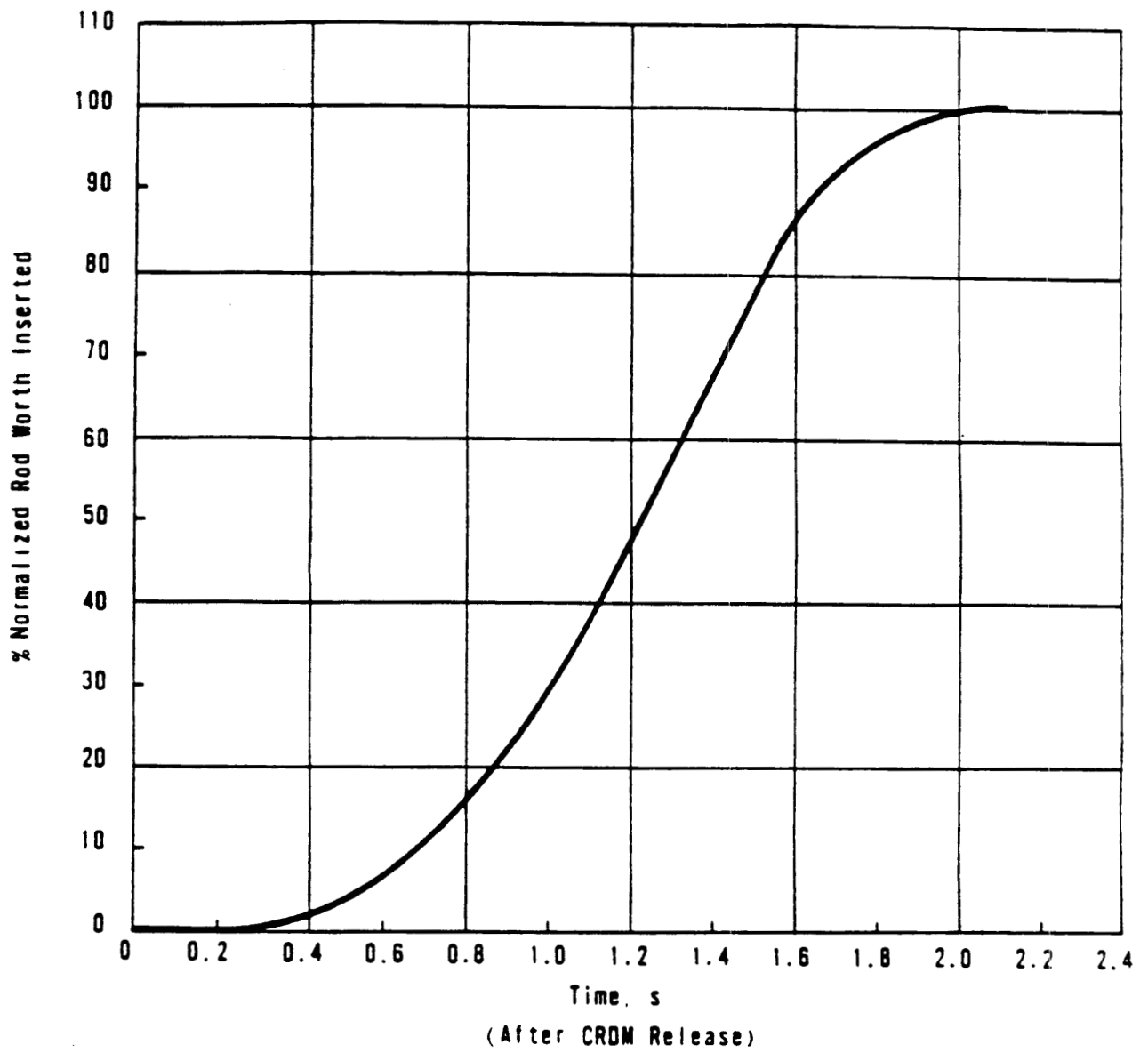
The following reactivity coefficients were assumed for each of the Chapter 15.0 events unless otherwise noted in an individual accident analysis:

	<u>BOL</u>	<u>EOL</u>
Moderator coefficient, $\Delta k/k/^\circ\text{F}$	$0.13 \times 10^{-4}$	$-3.0 \times 10^{-4}$ (see notes 1 and 2)
Doppler coefficient, $\Delta k/k/^\circ\text{F}$	$-1.28 \times 10^{-5}$	$-1.45 \times 10^{-5}$

For each accident where the results are very sensitive to moderator and Doppler coefficient variations, the analysis includes a parametric study on the reactivity coefficients.

The expected ranges of the reactivity coefficients are given in Appendix 4B for each cycle.

- (1) For the Dropped Control Rod Assembly, Inactive RCS Pump Startup, and Control Rod Assembly (CRA) Ejection, a Hot Full Power (HFP) moderator coefficient of  $-4.0 \times 10^{-4} \Delta k/k/^\circ\text{F}$  has been used in additional applicable analyses (Reference 31 and Reference 66). For the Excessive Heat Removal Due To Feed water System Malfunction event, the limit shown in the table above is applicable at both HFP and Hot Zero Power (HZP) conditions. For the Steam Line Break, a limiting temperature coefficient (combination of moderator and Doppler coefficients) of  $-3.1 \times 10^{-4} \Delta k/k/^\circ\text{F}$  has been used at HZP conditions and colder.
- (2) The current LOCA analyses were performed at a power level corresponding to 102% of 2966 MWt. Included in these analyses was a sensitivity study on moderator temperature coefficient as a function of core power level.



DAVIS-BESSE NUCLEAR POWER STATION  
NORMALIZED ROD WORTH VERSUS TIME

FIGURE 15.1-1

REVISION 0  
JULY 1982

15.2 CLASS 1 - EVENTS LEADING TO NO RADIOACTIVE RELEASE AT EXCLUSION AREA BOUNDARY

Class 1 events are abnormal operational transients which do not result in the following:

- a. Fuel failures in excess of those expected during normal operation.
- b. A breach of the fuel cladding (which leads to fission product release) or a breach of the Reactor Coolant System boundary.
- c. Operation of any engineered safety feature.
- d. Significant off-site radiation exposures.
- e. Propagation into the more serious Class 2 or 3 events.

Table 15.2-1 summarizes the potential abnormalities categorized as Class 1 events and shows a list of parameters used throughout the accident analysis.

TABLE 15.2-1

Class 1 Events

<u>Event</u>	<u>Analysis Assumption</u>	<u>Effect</u>
Uncontrolled Control Rod Assembly Group Withdrawal from a Subcritical Condition.	Uncontrolled single-group and all-group CRA withdrawal from sub-criticality with the reactor at zero power; only high flux and high RC pressure trips were used to terminate the accident.	Power rise terminated by negative Doppler effect, high Reactor Coolant System pressure trip or over power trip.
Uncontrolled Control Rod Assembly Group Withdrawal at Power.	Uncontrolled single-group and all-group CRA withdrawal with the reactor at rated power; only high flux and high RC pressure trips were used to terminate the accident.	Power rise terminated by over-power trip or high Reactor Coolant System pressure trip.
Control Rod Assembly Misalignment (Stuck-out, Stuck-in, or Dropped Control Rod Assembly).	Maximum worth control rod assembly dropped into core with the reactor at rated power, near middle-of-life condition. Stuck-out CRA worth considered in calculating the shutdown margin.	Subcriticality can be achieved if one CRA is stuck out. Dropped CRA does not result in reactor trip towards end of life condition.
Makeup and Purification System Malfunction.	Uncontrolled addition of unborated water to the Reactor Coolant System due to failure of equipment designed to limit flow rate and total water addition.	Slow change of power terminated by reactor trip on high coolant temperature or pressure. During shutdown a decrease in shutdown margin occurs, but criticality does not occur.
Loss of Forced Reactor Coolant Flow.	Reactor Coolant System flow decreases because of mechanical or electrical failure in one or more reactor coolant pumps.	Reactor is protected by the power/imbalance/flow and power/RC pumps trip.

TABLE 15.2-1 (Continued)

Class 1 Events

<u>Event</u>	<u>Analysis Assumption</u>	<u>Effect</u>
Start-up of an Inactive Reactor Coolant Loop.	Two reactor coolant pumps started with reactor at 60% of rated power and end-of-life conditions.	Reactor power and coolant pressure transient produced by increase in flow does not result in a reactor trip.
Loss of External Electrical Load and/or Main Turbine Trip.	A load rejection condition is considered.	Power reduction without reactor or turbine trip.
Loss of Normal Feedwater.	Main feedwater flow to steam generators is lost.	Reactor trips on high reactor coolant temperature or pressure.
Loss of all AC Power to the Station Auxiliaries.	A complete loss of all AC power is considered.	Immediate reactor trip on loss of power to control rod assemblies.
Excessive Heat Removal Due to Feedwater System Malfunctions.	A reduction in feedwater temperature and increase in feedwater flow are considered.	Reactor trips on high flux or low reactor coolant pressure.
Excessive Load Increase.	A small steam line leak to the atmosphere and to the condenser is considered.	Reactor trips on high flux or low reactor coolant pressure.
Anticipated variations in the Reactivity of the Reactor.	Automatic control system is inoperative or unused.	Change in Reactor Coolant System average temperature. Reactor trips if change is uncompensated.
Failure of Regulating Instrumentation.	Adverse functioning of regulating instrumentation in primary and secondary system is considered.	All induced transients are covered within the accident analysis of Chapter 15.
External Causes.	Storms or earthquakes are taken into consideration in design of station.	Evaluated as part of station design.

15.2.1 Uncontrolled Control Rod Assembly Group Withdrawal from a Subcritical Condition (Startup Accident)

15.2.1.1 Identification of Causes

The objective of a normal startup is to bring a subcritical reactor to the critical or slightly supercritical condition, and then to increase power in a controlled manner until the desired power level and system operating temperature are obtained. During a startup, an uncontrolled reactivity addition could cause a power excursion. This excursion is terminated by negative Doppler effect if no other protective action operates. No power excursion will result if a similar uncontrolled reactivity addition occurs during refueling. The boron concentration in the core is maintained during reactor vessel head removal and while loading and unloading fuel from the reactor at not less than that boron concentration required to shut down the reactor to a 1%  $\Delta k/k$  subcritical condition if all Control Rod Assemblies were removed. Transients occurring prior to reactor vessel head removal are adequately considered under the following analysis of the startup accident.

The following design provisions minimize the possibility of inadvertent continuous rod withdrawal and limit the potential power excursions:

- a. The control system is designed so that only one Control Rod Assembly (CRA) group can be withdrawn at a time, except that there is a 25 percent overlap in travel between two regulating CRA groups successively withdrawn. This overlap occurs near the minimum worth positions for each group since one group is near the beginning of travel.
- b. Control Rod Assembly withdrawal rate is limited.
- c. A high startup rate withdrawal stop and alarm are provided in the source range.
- d. A high startup rate withdrawal stop and alarm are provided in the intermediate range.

Even though design provisions (a), (c), and (d) above are available they were not used in the analysis. The withdrawal rate of 30 inches/minute was used to calculate the reactivity insertion rates.

15.2.1.2 Analyses of Effects and Consequences

15.2.1.2.1 Safety Evaluation Criteria

The safety evaluation criteria for this accident are:

1. The reactor thermal power shall not exceed 3104 MWt (112 percent of 2772 MWt).
2. The Reactor Coolant System pressure shall not exceed code pressure limits (2750 psig, which is 110% of design pressure).



#### 15.2.1.2.2 Methods of Analysis

The Startup Accident was originally analyzed using the KAPP4 computer code. The accident was re-analyzed after the TMI-2 accident using the CADDS computer program. This re-analysis was due to changing the PORV and Primary Safety Valve (PSV) setpoints. Due to several noted concerns with the inputs used in that analysis, the accident was again re-analyzed using the RELAP5/Mod2 B&W computer code (Reference 62). The RELAP computer code was used in compliance with all limitations and restrictions included in the Topical Report BAW-10193P-A (Reference 56).

The inputs to the analysis included a maximum allowable positive moderator coefficient of  $0.9\text{E-}04$  delta  $\text{k/k}^\circ\text{F}$ , the RCS design flow rate of 380,000, a reactor trip setpoint of 2400 psia, and a PSV setpoint of 2500 psig. These inputs were used to resolve identified concerns with the previous analysis. No changes were made to the remaining inputs.

#### 15.2.1.2.3 Results of Analysis

A spectrum of reactivity insertion rates (RIR) were used to determine the point at which the RPS high flux trip and the RPS high pressure trip occur concurrently. This yields the largest RCS pressure transient. At a RIR of  $1.925\text{ E-}04$  delta  $\text{k/k/second}$ , a peak pressure of 2750.2 psia was calculated. This is less than the maximum allowed pressure of 2764.4 psia (2750 psig). The peak core neutron power<sup>1</sup> was 67.74% of 2772 MWt. The analysis demonstrates that the safety evaluation criteria of Section 15.2.1.2.1 are met under all bounding conditions.

The previous analyses of the Startup Accident included parametric evaluations to determine sensitivities. The results of those analyses are included for information as Figures 15.2.1-1 through 15.2.1-8.

The limiting case results using RELAP5 for power and pressure are included as Figures 15.2.1-9 and 15.2.1-10, respectively.

Table 15.2.1-1 presents the Startup Accident Input Parameters used in Reference 62. Table 15.2.1-2 provides a summary of the results of the analysis.

The RELAP5 program was not utilized to evaluate two and three pump operation. The previous analyses demonstrated that the full flow case was bounding. It was determined that the plant limit on DNB will be met during two and three pump operation.

It is concluded that the reactor is completely protected against any startup accident involving the withdrawal of any or all Control Rod Assemblies, since in no case does the thermal power approach the design overpower condition, and the peak pressure never exceeds the allowable limits.

#### 15.2.1.3 Effects of Changes

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. That evaluation concluded that the replacement Steam Generator design provides small beneficial effects for the startup event. These include the increased primary volume, which benefits the RCS pressure predictions, and improved heat transfer capacity, which

---

<sup>1</sup> Neutron power is defined as the total power from fission.

## Davis-Besse Unit 1 Updated Final Safety Analysis Report

lessens the overheating consequences. If credited in the analysis, these beneficial effects would result in the event being slightly less severe. Therefore, the existing analysis remains applicable with the replacement Steam Generators installed.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 15.2.1-1

Startup Accident Parameters

Maximum CRA Speed, in/min	30*
Maximum number of full length CRAs	53**
Maximum CRA worth of all CRAs % $\Delta k/k$	**
Maximum reactivity addition rate, (all 53 CRAs at max speed), $\Delta k/k/sec$	$7.19 \times 10^{-4}$ *
Nominal CRA worth of single group when Reactor is critical, % $\Delta k/k$	2.30*
Reactivity addition rate for single CRA Group ( $\Delta k/k/sec$ )	$1.65 \times 10^{-4}$ *
Doppler Coefficient at rated power ( $\Delta k/k/^\circ F$ )	$-1.34 \times 10^{-5}$
Moderator coefficient at rated power ( $\Delta k/k/^\circ F$ )	$+0.9 \times 10^{-4}$
Trip parameters	
Delay for high pressure trip, sec	0.6
Delay for high flux trip, sec	0.5

\* from original plant analysis

\*\* confirmed to be bounded each reload

TABLE 15.2.1-2

Summary of Startup Accident Analysis

Reactivity Addition Rate	Peak Thermal Power	Peak Pressure
$\Delta k/k/sec$	% of Rated	psia
$1.925 \times 10^{-4}$	64.74	2750.2

<sup>1</sup> Neutron power is defined as the total power from fission.

15.2.2 Uncontrolled Control Rod Assembly Group Withdrawal at Power

15.2.2.1 Identification of Causes

A rod withdrawal accident presupposes an operator error or equipment failure resulting in accidental withdrawal of a Control Rod Assembly group while the reactor is at rated power. As a result, the power level increases, the reactor coolant and fuel rod temperature increases, and, if the withdrawal is not terminated by the operator or the protection system, core damage would eventually occur.

The following provisions are made in the design to indicate this accident:

- a. High reactor coolant outlet temperature alarms.
- b. High Reactor Coolant System pressure alarms.
- c. High pressure level alarms.

Even though these design provisions are available, they were not used in the analysis. Only the high flux level and high pressure trip of the Reactor Protection System were used in the analysis.

15.2.2.2 Analysis of Effects and Consequences

15.2.2.2.1 Safety Evaluation Criteria

The safety evaluation criteria for this accident are:

1. The reactor thermal power shall not exceed 112% of a nominal power level of 2772 MWt.
2. The Reactor Coolant System shall not exceed code pressure limits.

15.2.2.2.2 Methods of Analysis

A B&W digital computer code ("KAPP4 - Digital Computer Program for Solution of Reactor Kinetics and Primary System Pressure Response," BAW-10068, June 1973) was used to determine the characteristics of this accident. Included were a complete kinetics model, pressure model, average fuel rod model, steam demand model with secondary coastdown to decay heat level, coolant transport model, and a simulation of the instrumentation for pressure and flux trip. The initial conditions were normal rated power operation without automatic control. CRA withdrawal was modelled as a constant reactivity addition rate. The CRA's were assumed to be moving outward along the steepest part of the CRA worth versus CRA-travel curve at 30 inches/minute in order to calculate reactivity addition rates corresponding to the 3.2% ( $\Delta k/k$ ) single group worth and to the total worth of all CRA's. Only the Doppler and moderator coefficients of reactivity were used as feedback. The nominal values used for the main parameters in evaluating this accident are given in Table 15.2.2-1. For trip, the minimum control rod assembly worth that satisfies the criterion for a shutdown margin of 1%  $\Delta k/k$  at the hot standby condition is used throughout the analysis.

### 15.2.2.2.3 Results of Analysis

Figure 15.2.2-1 shows the results of the nominal CRA group withdrawal (constant reactivity addition rate of  $2.3 \times 10^{-4}$  ( $\Delta k/k$ ) / sec) from rated power. The nominal CRA group withdrawal from rated power is assumed to be the composite groups 4 and 5. The transient is terminated by a high neutron flux level trip, and the reactor thermal power is limited to well below the design overpower. The changes in the parameters are quite small, as shown in Table 15.2.2-2.

A sensitivity analysis of important parameters was performed around this nominal case, and the resultant Reactor Coolant System pressure responses are shown in Figures 15.2.2-2 through 15.2.2-5.

Figure 15.2.2-2 shows the pressure variation for a very wide range of CRA withdrawal rates, i.e., from approximately  $6 \times 10^{-6}$  ( $\Delta k/k$ ) / sec to approximately  $1 \times 10^{-3}$  ( $\Delta k/k$ ) / sec). For the very rapid rates, the neutron flux trip is the primary protective device for the reactor core. It also protects the system against high pressure during fast CRA withdrawal accidents. The high-pressure trip is relied upon for the slower transients. In no case does the thermal power exceed the design overpower.

Figures 15.2.2-3 through 15.2.2-5 show the pressure response to variations in the trip delay time, Doppler coefficient, and moderator coefficient, respectively. In all cases the high neutron flux level trip is actuated.

An analysis has been performed extending the evaluation of the CRA withdrawal accident for various fractional initial power levels up to rated power. In this evaluation, simultaneous withdrawal of all CRA's was assumed, giving the maximum possible reactivity addition rate, (a constant reactivity addition rate of  $7.19 \times 10^{-4}$  ( $\Delta k/k$ ) / sec). This rate is significantly higher than that used for the cases evaluated for withdrawal of a single group (Table 15.2.2-1). The results of this analysis are shown in Figures 15.2.2-6 and 15.2.2-7.

As seen in Figure 15.2.2-6, the peak thermal power occurs for the rated power case and is below the design overpower. The peak neutron power for all cases slightly overshoots the assumed trip level. Figure 15.2.2-7 shows that the maximum fuel temperatures reached in the average rod and the hot spot are well below melting. Even in the most severe case, which occurs at rated power, the average fuel temperature increases only a few degrees. Therefore, it is readily concluded that no fuel damage will result from the simultaneous withdrawal of all CRA's from any initial power level. Figure 15.2.2-8 shows reactor coolant pressure as a function of power for all CRA group withdrawal accidents.

For two and three reactor coolant pump operation, several reactivity insertion rates were analyzed. These included the reactivity addition rates due to withdrawing the maximum worth CRA group and withdrawing all Control Rod Assemblies.

The worst CRA withdrawal accident is the one with the slowest reactivity addition rate, because it permits the thermal power to keep up with the neutron power.

The maximum thermal power for three pump operation is approximately 81%. The maximum thermal power for two pump operation (1 Pump/Loop) is approximately 54%. The DNB ratio will not drop below the 1.30 limit for either of these two cases as specified by the limits shown in the plant Technical Specifications.

## Davis-Besse Unit 1 Updated Final Safety Analysis Report

This analysis demonstrates that the high-pressure trip and the high flux level trip adequately protect the reactor against any CRA withdrawal accident from any power level. In no case does the thermal power approach the design overpower condition, and the peak pressure never exceeds code allowable limits.

### 15.2.2.2.4 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. That evaluation concluded that the increased heat transfer capacity and the increased Reactor Coolant System flow associated with the replacement Steam Generators would improve the DNB margin for this event. Therefore, the existing analyses remain applicable with the replacement Steam Generators installed.

TABLE 15.2.2-1

CRA Withdrawal Accident Parameters

High pressure trip level (core outlet), psia	2,430
High flux trip level, %	112 (See Note)
Trip delay time (high pressure trip), sec	0.6
Trip delay time (high flux trip), sec	0.4
CRA insertion time (2/3 insertion), sec	1.4
Doppler coefficient, ( $\Delta k/k$ ) /°F	-1.28x10 <sup>-5</sup>
Moderator coefficient, ( $\Delta k/k$ ) /°F	+0.13x10 <sup>-4</sup>
CRA speed, in./min	30
CRA group worth, % $\Delta k/k$	3.2
One CRA group reactivity addition rate, ( $\Delta k/k$ ) /sec	2.3x10 <sup>-4</sup>
Maximum reactivity addition rate of all 53 Control Rod Assemblies, ( $\Delta k/k$ ) /sec	7.19x10 <sup>-4</sup>

NOTE: The analysis as presented was performed with an initial core power of 2772 MWt and a high flux setpoint of 112% of this value. License Amendment No. 278 increased core rated thermal power by 1.63%, from 2772 MWt to 2817 MWt. The event was not reanalyzed in support of the power uprate. However, the high flux setpoint was reduced to 110.2% of 2817 MWt to preserve the overpower limit.

TABLE 15.2.2-2

Summary of CRA Withdrawal Accident Analysis

Reactivity Addition Rate, ( $\Delta k/k$ ) /sec	Peak Thermal Power, % rated power	Peak system Pressure increase, psi
2.3 x10 <sup>-4</sup>	106.1	33
7.19 x10 <sup>-4</sup>	105.0	16

NOTE: As shown on Figure 15.2.2-2, lower Reactivity Addition Rates result in significantly greater peak system pressure increases than reported here.

15.2.3 Control Rod Assembly Misalignment (Stuck-Out, Stuck-In, or Dropped Control Rod Assembly)

15.2.3.1 Identification of Causes

In the event that a Control Rod Assembly cannot be moved, localized power peaking and shutdown margin must be considered. If a Control Rod Assembly is dropped into the core while operating the resulting transient must be examined.

Adequate hot shutdown margin is assured by requiring a subcriticality of 1%  $\Delta k/k$  with the Control Rod Assembly of greatest worth fully withdrawn from the core. The nuclear analysis reported in Chapter 4 of the SAR demonstrates that this has been satisfied. This has been analyzed in terms of the minimum tripped CRA worth available in the loss-of-coolant flow, Control Rod Assembly ejection, startup, CRA withdrawal, and steamline-failure accidents. In all cases the available CRA worth is sufficient to provide margins below any damage threshold.

A misaligned Control Rod Assembly is defined as a deviation of a CRA from its group reference position by more than an indicated 9 inches. This definition then covers both the action of dropping a CRA and of having a CRA stick while moving a group. The ICS action available is to inhibit all CRA-out motion. The details of this action are described in Chapter 7. Even though ICS action is available to prevent or mitigate this accident, the accident analysis was done without ICS action. No operator action is required.

15.2.3.2 Analysis of Effects and Consequences

15.2.3.2.1 Safety Evaluation Criteria

The safety evaluation criteria for this accident are:

1. The reactor thermal power shall not exceed 112 percent of a nominal power level of 2772 MWt.
2. The Reactor Coolant System pressure shall not exceed code pressure limits (2750 psig).

15.2.3.2.2 Methods of Analysis

A detailed B&W digital model ("KAPP4 - Digital Computer Program for Solution of Reactor Kinetics and Primary System Pressure Response," BAW-10068, June 1973) was used in the original analysis of the transient response to a dropped CRA. This program includes fuel pin, point kinetics, pressurizer, and loop models, including the steam generators.

The reactor was assumed to be operating at rated power when the CRA is dropped. To achieve the most adverse response in the original analysis, the most negative value of the moderator coefficient was used along with the maximum allowable CRA worth for rated power operation. Since the reactor does not trip, a bounding (less negative) power Doppler coefficient was assumed. The parameters used in the original analyses with End-of-Cycle conditions are shown in Table 15.2.3-1.



#### 15.2.3.2.3 Results of Analysis

The results of the original analysis are presented in Figure 15.2.3-1. The neutron power decreases causing a rapid decrease in both the core moderator temperature and the fuel temperature. These temperature decreases overcompensate for the worth of the CRA, and the neutron power rises slightly above the initial neutron power level. The neutron power then decreases to below the initial power level and eventually levels out at the initial power level. The thermal power response is similar to the neutron power; however, the thermal power level never exceeds the initial rated power value. Both the core moderator temperature and pressurizer pressure decrease during the transient and level out at a value lower than the initial value. Since the thermal power never exceeds the initial value and the pressure decreases during the transient, both safety evaluation criteria are met.

For three and two reactor coolant pump operation the maximum allowable power levels are lower than for four reactor coolant pump operation and the stuck-out, stuck-in, and dropped control rod assembly events are less severe. Also, because the initial power levels are lower, there is more margin for control available.

Cases have been run for CRA drops at beginning-of-life conditions and lower CRA worths. These transients are not included in this discussion because they produced less severe results than when end-of-life conditions and the maximum calculated CRA worth and later-in-life reactivity parameters were modeled.

#### 15.2.3.2.4 Additional Analyses

Additional evaluations (in Reference 31) were performed for the Dropped Control Rod Assembly event and these analyses were shown to be applicable to Davis-Besse Unit 1. An End of Cycle (EOC) Hot Full Power (HFP) Moderator Temperature Coefficient (MTC) of  $-4.0 \times 10^{-4} \Delta k/k/^\circ F$  was assumed, and the results of these analyses met the Safety Evaluation Criteria of Section 15.2.3.2.1.

In other analyses, several cases with different dropped rod worths were considered and for each worth, the MTC was varied over a range from BOC to EOC. At BOC conditions, the MTCs are not sufficiently negative to provide enough feedback, and the reactor will trip on low reactor coolant system (RCS) pressure. The reactor trip is followed by a recovery of Departure from Nucleate Boiling (DNB) margin. That is, a reactor trip will limit the severity of the dropped rod event. Later in the cycle, the increasingly negative MTCs cause the RCS pressure to decrease more slowly, delaying the time to trip. At some point near mid-cycle, there exists an MTC value that will stop the RCS pressure decrease just above the low-RCS-pressure trip setpoint, i.e., such that the low-RCS-pressure trip will not occur. The RCS temperature also decreases and, due to the positive reactivity insertion (decreasing coolant temperature coupled with a negative MTC), core neutron power will increase to a new steady state value above the initial value. As the burn up increases beyond MOC, the increasingly positive reactivity feedback, associated with a more negative MTC, can almost compensate for the worth of the dropped control rod and the RCS pressure and temperature decrease will be less severe. The resulting power increase will not be as high and the calculated minimum DNB ratio will increase. Therefore, reactivity coefficients at both BOC and EOC are less limiting for this event than those near MOC in terms of DNB margin.

The results of parametric analyses are summarized in Reference 53. The results of the analyses confirmed that, for a dropped rod worth of  $\leq 0.33 \Delta k/k$ , with MTC values corresponding to middle of cycle conditions, the low-RCS-pressure reactor trip will not occur. These analyses

also confirmed that the safety evaluation criteria will be met and the fuel rods will not experience DNB (i.e., the minimum DNB Ratio will be above the correlation limit) throughout the transient if the dropped rod worth is  $\leq 0.33\Delta$  k/k with an MTC and Doppler coefficient at MOC reactor conditions.

An additional dropped CRA accident analysis has been performed at an initial core power level of 3025 MWt using RELAP/MOD2-B&W (Reference 66). The analysis examined a bounding range of dropped CRA worths between -0.01 and -0.28 % $\Delta$ k/k. Conditions representing various times during a 24 month fuel cycle were also examined. A moderator temperature reactivity coefficient equal to  $-4.00 \text{ E-}04 \Delta\text{k/k/}^\circ\text{F}$  and a doppler reactivity coefficient of  $1.85 \text{ E-}05 \Delta\text{k/k/}^\circ\text{F}$  were utilized at the end-of-cycle.

For all dropped CRA cases analyzed, core neutron power, core thermal power, fuel temperatures, moderator temperatures, and RCS pressures all remained below their initial steady-state values. Also, no single dropped CRA worth or core burnup produced a set of limiting DNB conditions.

Results show that the acceptance criteria for the event (see Section 15.2.3.2.1) were satisfied for dropped rod worths between -0.01 and -0.28 % $\Delta$ k/k.

#### 15.2.3.2.5 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. That evaluation concluded that the increased heat transfer capacity and the increased Reactor Coolant System flow associated with the replacement Steam Generators would improve the DNB margin for this event. Therefore, the existing analyses remain applicable with the replacement Steam Generators installed.

TABLE 15.2.3-1

Dropped CRA Accident Parameters

Moderator coefficient, $(\Delta k/k)/^{\circ}F$	$-3.00 \times 10^{-4}$ (see note 1)
Doppler coefficient, $(\Delta k/k)/^{\circ}F$	$-1.45 \times 10^{-5}$
CRA worth at rated power, % $\Delta k/k$	0.65 (see note 1)

- (1) An EOC moderator coefficient of  $-4.0 \times 10^{-4} \Delta k/k/^{\circ}F$  was evaluated as acceptable per Reference 31. See Section 15.2.3.2.4 for discussions of analyses with MTC values typical of BOC and Middle-of-Cycle with different dropped rod worth assumptions.

TABLE 15.2.3-2

Summary of Maximum Numerical Values for Dropped CRA Accident \*

<u>CRA worth</u> <u>%<math>\Delta k/k</math></u>	<u>Maximum Neutron Power</u> <u>% of Rated</u>	<u>Minimum Pressure</u> <u>psia</u>	<u>Time-in-Life</u>
0.65	101	1940	EOC

\*See Section 15.2.3.2.4 for discussion of analyses with MTC values typical of BOC and Middle-of-Cycle with different dropped rod worth assumptions.

15.2.4 Makeup and Purification System Malfunction

15.2.4.1 Identification of Causes

Boron, in the form of boric acid, in the reactor coolant controls excess reactivity. The boron content of the reactor coolant is periodically reduced to compensate for fuel burnup. Dilution water is supplied to the Reactor Coolant System by the Makeup and Purification System. This system has several interlocks and alarms to prevent improper operation, as described in Chapter 7. Alarms are provided to annunciate that the interlock setpoints have been reached. These interlocks and alarms are as follows:

- a. Flow of dilution water to the makeup tank must be initiated by the operator with the Control Rod Assemblies withdrawn to a preset bank. The operator must position valves to connect the demineralized water supply to the makeup tank. The operator must preset a batch size and start the batch controller before the reactor coolant makeup batch stop valve will open to admit water to the Makeup and Purification System. Dilution water is added at flow rates up to the maximum makeup rate at operating pressure.

The original analysis used the primary water storage tank as a source of dilution water. The current source is the demineralized water. There are three demineralized water pumps but they are interlocked to allow running only two pumps at once. Even if the interlock failed, the combined capacity of the three pumps is less than 500 gpm at the pressure of concern.

- b. Flow of dilution water is automatically stopped when the total amount of dilution water has reached a preset value or when the CRA's have been inserted to a preset position.

The ability of ICS to terminate dilution based on programmed rod band and power has been removed. This circuit will terminate feed-and-bleed if any safety group is not 100% withdrawn and/or control rod group 5 is not greater than 25 % withdrawn.

- c. A continuous dilute permit light and diverting position lights on the console are on whenever continuous dilution is in progress. Thus, information concerning dilution is available to the operator, and the dilution can be terminated by the operator at any time.

Even though Integrated Control System (ICS) action is available to terminate this accident, the successive failures of this ICS action have been assumed in the accident analysis.

The values of system parameters used in evaluating this accident are listed in Table 15.2.4-1. The Makeup and Purification System normally has one pump in operation, which supplies makeup to the Reactor Coolant System and the required seal flow to the reactor coolant pumps. Thus, the total makeup flow available is normally limited by pump capacity. When the makeup rate is greater than the letdown rate, the net water increase will cause the pressurizer level control to close the makeup valves. The nominal moderator dilution event considered is the pumping of water with zero boron concentration from the makeup tank to the Reactor Coolant System.

It is possible, however, to have a slightly higher flow rate during transients when the system pressure is lower than the nominal value and the pressurizer level is below normal.

Furthermore, with a combination of multiple valve failures or maloperations, plus more than one makeup pump operating with reduced Reactor Coolant System pressure, the resultant inflow rate could be much higher than the normal rate.

(Table 15.2.4-1). This constitutes the maximum dilution accident.

#### 15.2.4.2 Analysis of Effects and Consequences

##### 15.2.4.2.1 Safety Evaluation Criteria

The Safety Evaluation criteria for this accident are:

- a. The reactor thermal power shall not exceed 112% of a nominal power level of 2772 MWt.
- b. The Reactor Coolant System pressure shall not exceed code pressure limits.
- c. A minimum shutdown margin of 1%  $\Delta k/k$  shall be maintained during refueling conditions

##### 15.2.4.2.2 Methods of Analysis

A B&W digital computer code ("KAPP4 - Digital Computer Program for Solution of Reactor Kinetics and Primary System Pressure Response," BAW-10068, June 1973) was used to determine the characteristics of this accident for the maximum rate of reactivity increases from the moderator dilution. The code included a complete point kinetics model and an average fuel rod model. The reactor is assumed to be operating at rated power with a maximum initial boron concentration in the Reactor Coolant System, as given in Table 15.2.4-1. The dilution water is uniformly distributed throughout the reactor coolant volume because of the very small discharge rate of makeup flow into the very large reactor coolant flow. The analysis is based on the maximum moderator coefficient, beginning-of-core life Doppler coefficient, and maximum initial Reactor Coolant System boron concentration. Both moderator coefficient and boron concentration values used are conservative. The effects of the three dilution rates discussed above on the reactor are tabulated in Table 15.2.4-2.

##### 15.2.4.2.3 Results of Analysis

The highest rate of dilution can be handled by the ICS which inserts CRAs to maintain the power level and the Reactor Coolant System temperature. Dilution will terminate when any safety group is not 100% withdrawn and/or control rod group 5 is not greater than 25% withdrawn.

If the reactor is under manual control with no CRA insertion or the ICS fails to insert the CRAs, these reactivity additions will cause a high-pressure or high-temperature trip, which will cause the makeup valve to close during feed and bleed operations, terminating the addition of deborated water to the makeup system. Peak pressures and thermal powers for this case are shown in Table 15.2.4-2 for the normal and the maximum dilution flow rates. The thermal power does not exceed the design overpower condition and the system pressure does not exceed code allowable limits. Therefore, moderator dilution accidents will not cause any damage to the Reactor Coolant System.

After a reactor trip, emptying a full makeup tank of deborated water into the Reactor Coolant System will result in a reactivity addition of 0.97%  $\Delta k/k$ . This effect will only be seen when the initial boron concentration is high, which is at the beginning-of-core life when sufficient CRA worth is available to shut down the reactor by several percent even with the highest worth CRA stuck out of the core. Thus, the minimum shutdown margin of 1%  $\Delta k/k$  subcritical would be maintained. Emptying a full makeup tank of deborated water into the Reactor Coolant System would require 9 minutes even at the maximum dilution rate considered.

During refueling or maintenance operations when the reactor closure head has been removed, the sources of dilute water to the makeup tank and therefore to the Reactor Coolant System are closed, and the makeup pumps are not operating. At the beginning of core life when the boron concentration is highest, the reactor is several percent subcritical with the maximum worth CRA stuck out. To demonstrate the ability of the reactor to accept moderator dilution during shutdown, the consequences of accidentally filling the makeup tank with dilution water and starting the makeup pumps have been evaluated. The entire water volume from the makeup tank could be pumped into the Reactor Coolant System (assuming that only the coolant in the reactor vessel is diluted); the reactor would still be several percent subcritical (Table 15.2.4-2).

The Moderator Dilution Accident Analysis information presented in Table 15.2.4-2, for both at-power and shutdown conditions, was for the initial core design. Cycle-specific analyses are performed to confirm the initial analyses remain bounding for each new core design. This conclusion is documented in the Reload Report (USAR Appendix 4B).

For two and three reactor coolant pump operation, the maximum moderator dilution accident was analyzed. As seen in Table 15.2.4-3, the rate of average Reactor Coolant System temperature change increases when fewer pumps are running.

The maximum rate of moderator dilution can be handled by the Automatic Control System, which would insert CRA's to maintain the rated power level and thus limit the Reactor Coolant System temperature rise.

#### 15.2.4.2.4 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. The increased heat transfer capacity of the replacement Steam Generators lessens the severity of the peak pressure and power response for the moderator dilution event at full power. In addition, the larger primary inventory for the ROTSG produces a slight benefit by slowing the dilution effects for a given dilution flow at full power and shutdown conditions. Therefore, the existing analyses remain applicable with the replacement Steam Generators installed.

TABLE 15.2.4-1

Moderator Dilution Accident Parameters

Dilution flow rate condition

- |    |  |     |
|----|--|-----|
| a. | Nominal dilution, gpm  | 70  |
| b. | Nominal dilution with low Reactor Coolant System pressure, gpm | 100 |
| c. | Maximum considered, gpm  | 500 |

Initial boron concentration in Reactor Coolant System, ppm (hot, clean, BOL) 1407

Boron reactivity worth, hot rated power, ppm/1% ( $\Delta k/k$ ) 100

Moderator coefficient, ( $\Delta k/k$ )/°F  $+0.13 \times 10^{-4}$

Doppler coefficient, ( $\Delta k/k$ )/°F  $-1.28 \times 10^{-5}$

Dilution valve interlock setpoints:

- a. Dilution valve may be opened when:
  - 1. CRA is withdrawn to a predetermined position, and
  - 2. Integrated flow timing device is set.
- b. Dilution valves automatically close when:
  - 1. CRA are less than predetermined position, or
  - 2. Integrated flow timing device has reached a preset value corresponding to a given integrated flow, or
  - 3. Reactor trips.

TABLE 15.2.4-2

Summary of Moderator Dilution Accident Analysis

1. Dilution at power:

<u>Condition</u>	<u>Dilution Water flow, gpm</u>	<u>Reactivity Rate (<math>\Delta k/k</math>)/sec</u>	<u>Average reactor coolant system temp change, °F/sec</u>
Normal	70	+1.80 x 10 <sup>-6</sup>	0.004
Low RCS pressure	100	+2.57 x 10 <sup>-6</sup>	0.006
Maximum considered	500	+1.29 x 10 <sup>-5</sup>	0.027

2. Dilution to trip:

<u>Dilution water flow, gpm</u>	<u>Peak thermal power, percent of rated</u>	<u>Peak pressure, psia</u>	<u>Time to trip, sec</u>
70	102.2	2435	126
500	105.6	2447	46

3. Dilution to shutdown: (1)

Initial shutdown margin	6.27% $\Delta k/k$
Final shutdown margin	3.17% $\Delta k/k$

(1) Historical value, refer to subsection 15.2.4.2.3.

TABLE 15.2.4-3

Table of Moderator Dilution Accident for Partial Pump Operation

<u>Condition</u>	<u>Dilution Water flow, gpm</u>	<u>Reactivity Rate (<math>\Delta k/k</math>)/sec</u>	<u>Average reactor coolant system temp change, °F/sec</u>
Maximum Considered			
For 2 pumps	500	+1.227 x 10 <sup>-5</sup>	+0.0768
For 3 pumps	500	+1.227 x 10 <sup>-5</sup>	+0.0472



15.2.5 Loss of Forced Reactor Coolant Flow (Partial, Complete, and Single Reactor Coolant Pump Locked Rotor)

15.2.5.1 Identification of Causes

The reactor coolant flow rate is reduced if one or more of the reactor coolant pumps fail. A loss of coolant flow can occur from mechanical failures or from a loss of electric power. With four independent pumps available, a mechanical failure in one pump will not affect operation of the others.

Each reactor coolant pump receives electric power from one of the two electrically separated buses, as discussed in Chapter 8. Loss of the unit auxiliary transformer to which the 13.8kV buses are normally connected will initiate a rapid transfer to the two startup transformers without loss of coolant flow. Faults in an individual pump motor or its power supply could cause a reduction in flow, but a complete loss of forced flow, the more conservative case, is extremely unlikely and would occur only if all offsite power and both startup transformers were lost. Power loss would cause immediate reactor trip independent of protection system actuation. Even though this event has a low probability of occurrence, the nuclear unit can sustain such a failure without core damage.

The loss of flow due to mechanical malfunction from any cause has been considered and analyzed as the locked rotor accident. The frequency of occurrence of this accident is expected to be the same as any gross mechanical failure of the primary system. For this reason only one pump is assumed to be affected.

The reactor is protected from the consequences of a reactor coolant pump failure(s) by the Reactor Protection System. The reactor is tripped if insufficient reactor coolant flow exists for the power level. The ICS initiates a power reduction upon pump failure to prevent reactor power from exceeding that permissible for the available flow. Even though ICS action is available to prevent or mitigate this accident, the accident analysis was done without ICS action.

15.2.5.2 Analysis of Effects and Consequences

15.2.5.2.1 Safety Evaluation Criteria

The safety evaluation criteria for this accident are:

1. For the failure of one or more reactor pumps, (i.e., coastdown) the minimum DNB ratio will not be less than the applicable DNBR design limit for the critical heat flux (CHF) correlation used:

$$W-3 = 1.30$$

$$B\&W-2 = 1.30$$

$$BWC = 1.18$$

$$\text{BHTP Thermal Design Limit (Statistical Core Design)} = 1.65$$

2. For the locked rotor accident, no fuel cladding failure shall occur.

#### 15.2.5.2.2 Methods of Analysis

The loss-of-coolant flow accident was analyzed using the following analog and digital computer programs: (a) "KAPP4 - Digital Computer Program for Solution of Reactor Kinetics and Primary System Pressure Response," BAW-10068, June 1973, (b) "RADAR - Reactor Thermal and Hydraulic Analysis During Reactor Flow Coastdown," BAW-10069A, Rev. 1 October 1974, (c) "SPLIT- Digital Steady-State Flow Distribution Code for Various Primary System Combinations," BAW-10071A, September 1974, and (d) "PUMP - Analog - Hybrid Reactor Coolant Hydraulic Transient Model," BAW-10073A, Rev. 1, March 1976. An analog simulation was used to determine the reactor flow rate following a loss of pumping power. Reactor power, coolant flow, and inlet temperature were input data to a digital program, which determined the core thermal characteristics during the flow coastdown.

The B&W digital computer model used to determine the neutron power following reactor trip includes six delayed neutron groups, control assembly worth and CRA insertion characteristics, and trip delay time. The analog model used to determine flow coastdown characteristics includes descriptions of flow-pressure drop relations in the reactor coolant loops. Pump flow characteristics were determined from manufacturer's zone maps. Flow-speed, flow-torque, and flow-head relationships were solved by affinity laws.

B&W has developed a computer code to calculate fuel temperature, cladding temperature, and DNB ratio as a function of time for reactor system transients. Input to the code consists of flow, power, inlet temperature, and system pressure as a function of time. The flow resulting from a coastdown is calculated by a standard B&W analog model that includes a simulation of the pump and its associated inertia along with all pressure drops around the loop. The power, inlet temperature, and system pressure are calculated by a standard B&W digital code that includes a point-kinetics model with a closed-loop simulation. The loop simulation includes a pressurizer model and a steam generator model, all connected through time delays to the kinetics calculations.

The core transient analysis code simulates the reactor core through the use of a two-channel model. Each channel consists of one fuel rod with its associated flow area and spacer grid geometry. Given the necessary input as stated above, the code will calculate a pressure drop across a typical reactor channel (average channel) as a function of time. This pressure drop is then imposed on the second channel (usually a hot channel) to determine hot channel flow and DNB ratio in addition to fuel and cladding temperatures. Compared to the average channel, the hot channel has greater heat generation and reduced flow areas, as well as statistical hot channel factors. The analytical fuel pin model contains a transient response calculation, while the hydraulic model considers the steady-state solutions of energy, mass and momentum balances at each time step.

The transient response is obtained by applying the changing flow, power, inlet temperature, and system pressure to the initial conditions of the average channel. This calculation yields the average channel pressure drop as a function of time, along with the hot channel power, inlet temperature, and system pressure. This yields the response of the hot channel in terms of cladding temperature, fuel temperature, and DNB ratio as a function of time.

The loss-of-coolant flow analysis has been carried out in the power range for coastdown from power levels between rated and the design overpower condition. The conditions used in the analysis are given in Table 15.2.5-1.

The conditions listed in Table 15.2.5-2 were used in thermal calculations to determine the minimum value that the DNB ratio would reach during the locked rotor accident.

#### 15.2.5.2.3 Results of Analysis

The key parameter for the loss of flow occurrence is the power level. For fewer than four pumps operating the initial power level will be lower and the initial DNBR will be higher, so for the loss of all flow the minimum DNBR will be higher for the partial pump mode than for the four pump mode.

The results of the four pump coastdown analysis show that the reactor can sustain a loss-of-coolant flow accident without damage to the fuel. The results of the evaluation are presented in Figures 15.2.5-1, 15.2.5-2, and 15.2.5-3. Figure 15.2.5-1 shows the percent reactor coolant flow as a function of time after loss of all pump power. Figure 15.2.5-2 shows the percent neutron power versus time following a reactor trip. Figure 15.2.5-3 shows the minimum DNBR which occurs during coastdown from various initial power levels using the minimum tripped CRA worth (assuming 1%  $\Delta k/k$  hot, shutdown margin). The degree of core protection during coastdown is indicated by comparing the minimum DNBR for the coastdown with the criterion value of 1.3.

Under normal conditions, the maximum reactor power level from which a loss-of-coolant flow accident could occur is 102% of 2772 MWt power. This power level provides an allowance of +2% 2772 MWt power for heat balance error. Even with this error, Figure 15.2.5-3 and Table 15.2.5-3 show that an acceptable minimum DNBR exists in the hot channel.

The Reactor Coolant System is capable of providing natural circulation flow after the pumps have stopped. The natural circulation characteristics of the Reactor Coolant System have been calculated with conservative values for all resistance and form loss factors. No voids are assumed to exist in the core or the reactor outlet piping. Table 15.2.5-4 shows the natural circulation flow capability as a function of the decay heat generation. These flows provide more than adequate heat transfer capability for core cooling and decay heat removal by the Reactor Coolant System.

The locked rotor accident is a rapid decrease in flow resulting from the instantaneous seizure of a reactor coolant pump rotor. The initial operating conditions are as given in Table 15.2.5-2. Figure 15.2.5-4 shows the flow coastdown, neutron power transient and the reactor system pressure as a function of time. The neutron power rises slightly but quickly drops after CRA insertion due to a power/imbalance/flow reactor trip. The reactor system pressure peaks after the neutron power peak and remains well below the code limits.

DNBR for the locked rotor accident is calculated by the RADAR (BAW-10069A, Rev 1) code, which analyzes an isolated hot channel. The initial DNBR is representative of the 102% of 2772 MWt power maximum design case, as discussed in Subsection 4.4.3.4.1 with the following exception: For thermal design of the core and for anticipated transients, such as the four-pump coastdown, maximum control and instrumentation errors on pressure and temperature are applied. For the locked rotor accident, which is considered to be a low probability event, the pressure and temperature errors are not considered. The maximum design condition assumes that the worst thermal, nuclear, and mechanical conditions exist simultaneously in a particular subchannel (the hot channel).

Thermal calculations performed to determine the minimum value of DNBR that would be reached during the locked rotor accident are shown in Figure 15.2.5-5. As the figure shows, the

DNBR initially decreases very rapidly from its initial value. After the flow transient ends, however, the DNBR decreases much less rapidly and reaches a new steady-state value. The transient is terminated by the power/imbalance/flow reactor trip, and as the figure shows, the DNBR increases in response to the decreasing power, going above 1.3 and continues to rise thereafter. The locked rotor is an accident of very low probability of occurrence for which a certain degree of fuel failure can be accepted. Criterion 2 of Subsection 15.2.5.2.1 states that no fuel cladding failure shall occur, meaning that the structural integrity of the fuel rod shall not be impaired. Since the DNBR in the analysis never goes below 1.0, both Criteria are met.

No fission product release is postulated for the locked rotor accident meaning that no activity is released to the atmosphere as a result of the accident. However, the gap activity for all of the fuel rods with a DNBR of less than 1.3 was assumed to be released to the coolant. If a fission product release did occur, it would be necessary to assume the breach of the Reactor Coolant System boundary before the environmental effects of the accident must be considered. A potential path of fission product escape would be primarily through secondary system generator tube leakage and then release to the atmosphere through the condenser air ejector.

The locked-rotor accident is not analyzed for two-pump operation because the results of the transient during this mode of operation would be less severe than if starting from the four-pump operation and rated power. During two-pump, steady-state operation, the reactor coolant system contains much more boron than during operation at full power since the regulating control rod groups are kept at approximately the same insertion limit. This provides increased shutdown margin during reactor trip. Core protection is ensured by the flux/flow reactor trip function.

Also, initial DNBR is much larger during two-pump operation than during four-pump operation. In addition, the heat flux distribution would be the same for operation with two and four pumps.

The initially higher DNBR, combined with identical hot channel factors and increased shutdown worth, indicates that the locked-rotor accident would yield less severe results for two-pump than for four-pump operation.

While analysis results for two reactor coolant pump operation are reported above, power operation with only two reactor coolant pumps running is not allowed by Davis-Besse License Condition C.3.a.

#### 15.2.5.2.4 Reanalysis of Locked Rotor Accident

Crossflow thermal-hydraulic analysis methodology was implemented for the Davis-Besse Station beginning with the cycle 6 reload licensing analysis. The implementation of crossflow modeling for thermal-hydraulic evaluations was performed by using the LYNX1 and LYNX2 computer codes for multi-pass modeling and LYNXT for single-pass modeling. These codes account for the mass and energy exchange between adjacent channels to more accurately predict coolant axial flow behavior. This compares to the closed-channel analysis performed by the RADAR code which does not account for the mass exchange between channels. Detailed descriptions of the crossflow codes and modeling techniques are provided in reference 22.

The assessment of the DNBR behavior during the Locked Rotor Event with four Reactor Coolant Pumps operating has been performed in accordance with the safety criteria and analysis methodology defined and/or referenced by Reference 23. The transient local coolant conditions within the core have been predicted using the LYNXT thermal-hydraulic code which considers the lateral crossflow occurring between and within fuel assemblies. This capability

permits the assessment of DNBR performance for cores composed of one fuel design and mixed cores composed of two or more fuel designs. The evaluation of the DNBR performance for the fuel in the core has been made using Statistical Core Design methodology which treats the occurrence of variable uncertainties statistically for a more realistic simulation than the traditional deterministic treatment. The Statistical Core Design methodology was first introduced for DNB assessment in fuel cycle 10.

Implementation of the Mark-B-HTP fuel assembly design commenced with fuel cycle 15. The BHTP critical heat flux correlation is utilized to predict DNBR for the Mark-B-HTP fuel assemblies. This correlation is applicable to fuel assemblies containing the M5™ HTP spacer grids. The Locked Rotor Accident was evaluated for the core being composed of Mark-B- HTP fuel (using HTP spacer grids) and Mark-B fuel (using conventional Mark-B type spacer grids). The BHTP CHF correlation was used in determining the DNBR behavior in the Mark-B-HTP fuel and the BWC CHF correlation was used in determining the DNBR behavior in the Mark-B fuel. The CHF design limits for the correlations are 1.132 and 1.18, respectively. The Statistical Design Limits for the respective correlations are 1.316 (BHTP) and 1.313 (BWC). These elevated design limits account for the incorporation of the DNB impact associated with variable uncertainties selected for statistical treatment.

Since the Mark-B-HTP fuel design has a higher pressure drop than the Mark-B fuel types, coolant tends to move from the Mark-B-HTP fuel towards the Mark-B fuel types within the core. Consequently, the DNBR performance of the Mark-B fuel is enhanced as a result of residing with the Mark-B-HTP fuel. The DNBR performance of the Mark-B-HTP fuel is reduced as a result of residing with the Mark-B fuel types. The methodology for assessing the DNBR performance in the mixed core condition is to quantify the DNBR behavior assuming a full core composed of Mark-B-HTP fuel and to incorporate the transition core penalty, if needed, by elevating its respective design criterion. The transition core penalty is equal to the maximum DNBR difference between a limiting fuel rod operating in a full-core of Mark-B-HTP fuel and in a Mark-B-HTP fuel assembly within a mixed core condition. A Thermal Design Limit that is greater than the Statistical Design Limit is selected for each reload core design to provide additional DNB margin. The Thermal Design Limit contains sufficient DNB margin to accommodate cycle-specific needs including any transition core DNB penalty.

Results show that the minimum DNBR is greater than the Thermal Design Limit. This result indicates that the structural integrity of the fuel rod is not paired. Thus, the safety evaluation criterion of Section 15.2.5.2.1 for the locked rotor accident is satisfied.

As discussed in Section 15.2.5.2.3, the Reactor Coolant System is capable of providing natural circulation flow after the pumps have stopped. Table 15.2.5-4 lists the natural circulation flow capability as a function of the decay heat generation. These flows were evaluated with respect to the implementation of the Mark-B-HTP fuel assembly design. The dominate pressure loss that must be overcome during natural circulation flow is due to the elevation head. Since the Mark-B-HTP fuel does not change the elevation head, the impact due to the unrecoverable pressure loss is negligible. Consequently, there is an insignificant change to the natural circulation flow due to the introduction of Mark-B-HTP fuel assemblies. Therefore, the natural circulation flows presented in Table 15.2.5-4 remain applicable for full and mixed cores of Mark-B-HTP fuel assemblies.

#### 15.2.5.2.5 Reanalysis of the Total Loss of Flow Accident

The Total Loss of Flow Accident (coastdown of four Reactor Coolant Pumps) is protected against by the High Flux/Number of Reactor Coolant Pumps On (power/pumps) trip function of

the Reactor Protection System. The most limiting total loss of flow transient occurs when the core is operating at rated thermal power. At this condition, the power/pump trip function trips the reactor upon sensing a loss of power to all four reactor coolant pumps. Because of the delay times associated with the various hardware components, the design analysis considers a total trip delay time which is defined as the time between the initial loss of flow and commencement of rod insertion.

Implementation of the Mark-B-HTP fuel assembly design commenced with fuel cycle 15. The BHTP critical heat flux correlation is utilized to predict DNBR for the Mark-B-HTP fuel assemblies. The BHTP correlation is applicable to fuel assemblies containing the M5™ HTP spacer grids. Because of the fuel assembly design change, the Total Loss of Flow Accident was reanalyzed using crossflow and statistical core design methodologies (see Section 15.2.5.2.4 for a discussion of mixed cores). The assessment of the DNBR behavior during the event has been performed in accordance with the safety criteria and analysis methodology defined and/or referenced by Reference 23. The Total Loss of Flow Accident was evaluated with LYNXT in a transient mode using a transient system response determined with RELAP5/MOD2. The minimum BHTP DNBR determined for the Total Loss of Flow Accident is greater than the Thermal Design Limit (see Section 15.2.5.2.4 for a discussion of the Thermal Design Limit) and demonstrates that the safety evaluation criterion of Section 15.2.5.2.1 is satisfied.

#### 15.2.5.2.6 Three Pump Locked Rotor Accident

Technical Specifications allow a positive moderator coefficient below 95 percent of rated thermal power (RTP). Although a positive moderator coefficient is unlikely for reload cores, the Locked Rotor Accident was analyzed with 1) an initial power of 75 percent of RTP, 2) three reactor coolant pumps operating, and 3) the most positive moderator coefficient allowed by Technical Specifications.

Implementation of the Mark-B-HTP fuel assembly design commenced with fuel cycle 15. The BHTP critical heat flux correlation is utilized to predict DNBR for the Mark-B-HTP fuel assemblies. This correlation is applicable to fuel assemblies containing the M5™ HTP spacer grids. Because of the fuel assembly design change, the Locked Rotor Accident with three Reactor Coolant Pumps operating was reanalyzed using crossflow and statistical core design methodologies (see Section 15.2.5.2.4 for a discussion of mixed cores). The assessment of the DNBR behavior during the event has been performed in accordance with the safety criteria and analysis methodology defined and/or referenced by Reference 23. Reactor coolant flow and core power behavior during the transient were computer using RELAP5/MOD2. These system parameters as a function of time were input to the LYNXT core thermal-hydraulic analysis code.

Results show that the minimum DNBR is greater than the Thermal Design Limit (see Section 15.2.5.2.4 for a discussion of the Thermal Design Limit). These results indicate that the structural integrity of the fuel rod is not impaired. Thus, the safety evaluation criterion of Section 15.2.5.2.1 for the locked rotor accident is satisfied.

#### 15.2.5.2.7 Partial Loss of Flow Accident Reanalysis

The One Pump Loss of Flow Accident is a decrease in flow resulting from the loss of one Reactor Coolant Pump, but without seizure of the pump (i.e., coastdown). This event is protected by the power/imbalance/flow trip function of the Reactor Protection System.

The most limiting DNB condition exists when the core is operating at rated thermal power. At this condition, the power/imbalance/flow trip function trips the reactor when the flux-to-flow setpoint is reached. Because of the deadband times, electronic errors, and delay times associated with the various hardware components, the design analysis considers a total trip delay time which is defined as the time between the initial flux-to-flow signal and commencement of rod insertion.

Implementation of the Mark-B-HTP fuel assembly design commenced with fuel cycle 15. The BHTP critical heat flux correlation is utilized to predict DNBR for the Mark-B-HTP fuel assemblies. This correlation is applicable to fuel assemblies containing the M5™ HTP spacer grids. Because of the fuel assembly design change, the One Pump Loss of Flow Accident was reanalyzed using crossflow and statistical core design methodologies (see Section 15.2.5.2.4 for a discussion of mixed cores). The assessment of the DNBR behavior during the event has been performed in accordance with the safety criteria and analysis methodology defined and/or referenced by Reference 23. Reactor coolant flow and core power behavior during the transient were computed using RELAP5/MOD2. These system parameters as a function of time were input to the LYNXT core thermal-hydraulic analysis code.

Results show that the minimum DNBR is greater than the Thermal Design Limit (see Section 15.2.5.2.4 for a discussion of the Thermal Design Limit). Thus, the safety evaluation criterion of Section 15.2.5.2.1 for the One Pump Loss of Flow accident is satisfied.

#### 15.2.5.2.8 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. The replacement Steam Generators have less Auxiliary Feedwater bypass flow and a small increase in heat transfer capability. Both of these effects would make this event less severe. Therefore, the existing analyses remain applicable with the replacement Steam Generators installed.

TABLE 15.2.5-1

Loss-of-Coolant-Flow Accident Parameters

Initial power

Maximum indicated power, %	100 (of 2772 MWt)
Maximum actual power, %	102 (of 2772 MWt)

System characteristics:

The initial pressure and inlet temperature are nominal values minus 65 psi and plus 2°F, respectively.

The trip delay time is 620 msec. \*

The percent of rated neutron power (at beginning of life) as a function of time after trip is shown in Figure 15.2.5-2.

The pump motor inertia is 70,000 lb-ft<sup>2</sup>.

\*Trip delay time has been changed to 800 msec. for analysis of Section 15.2.5.2.5.

TABLE 15.2.5-2

Locked Rotor Accident Parameters

Initial power, % rated power	102 (of 2772 MWt)
Initial flow, % design flow	100
Power / imbalance / flow trip delay time, sec.	1.95

System characteristics:

The initial pressure and inlet temperatures are nominal values.

Maximum design conditions were assumed for the thermal conditions.

The reactor is tripped by power / imbalance / flow.



TABLE 15.2.5-3

Summary of Loss-of-Coolant-Flow Accident Analysis

Minimum DNBR during coastdown for loss of all four reactor coolant pumps and locked rotor:

<u>Situation</u>	<u>Criterion</u>	<u>Result</u>
Four pump coastdown for 100% (of 2772 MWt) power and 100% flow	1.3	1.56
Four pump coastdown from 102% (of 2772 MWt) power and 100% flow	1.3	1.49
One pump locked rotor from 102% (of 2772 MWt) power and 100% flow	1.0	1.05

TABLE 15.2.5-4

Natural Circulation Capability

<u>Time after loss of power, sec</u>	<u>Decay heat core power, %</u>	<u>Natural circulation core flow available, % full flow</u>	<u>Flow required for decay heat removal, % full flow</u>
3.6x10 <sup>1</sup>	5	4.6	2.3
2.2x10 <sup>2</sup>	3	3.8	1.2
1.2x10 <sup>4</sup>	1	2.3	0.4
1.3x10 <sup>5</sup>	0.5	1.6	0.2

## 15.2.6 Startup of an Inactive Reactor Coolant Loop (Pump Startup Accident)

### 15.2.6.1 Identification of Causes

The classical cold water accident is not possible in this reactor because the reactor coolant flow path contains no check or isolation valves.

However, when the reactor is operated with one or more pumps not running, and these pumps are then started, the increased flow rate will cause the average core temperature to decrease. If the moderator coefficient is negative, then positive reactivity will be introduced into the core and a power rise will occur.

For the case of one or more idle pumps, the pump control circuitry has an interlock to prevent starting an idle pump if the reactor power is above 60%. This interlock ensures that a pressure trip setpoint will not be reached before reaching equilibrium conditions upon restarting a pump. This interlock was not included in the analysis.

### 15.2.6.2 Analysis of Effects and Consequences

#### 15.2.6.2.1 Safety Evaluation Criteria

The safety evaluation criteria for this accident are:

1. The core thermal power shall not exceed 112% of a nominal power level of 2772 MWt.
2. The Reactor Coolant System pressure shall not exceed code pressure limits (2750 psig). (2750 psig).

#### 15.2.6.2.2 Methods of Analysis

A detailed digital simulation ("KAPP4-Digital Computer Program for Solution of Reactor Kinetics and Primary System Pressure Response," BAW-10068, June 1973) of the plant was used to evaluate the transient response to this accident. The model includes point kinetics and a multiregion fuel pin model connected through time delays to a pressurizer model. A steam generator model was included.

The pump control circuitry interlock was not included in the analysis because it was assumed that the plant was operating with two pumps at 60% (of 2772 MWt) of rated power when the remaining two pumps were started. This is the maximum power level that can exist for two pumps in operation without a reactor trip. The startup of the idle pumps caused the system flow to increase from 49% to 100% of design flow in 9 seconds. It was found that the maximum temperature decrease for this case occurs in less than one coolant loop transit time. Conservative values of the moderator coefficient and the Doppler coefficient were assumed to exist at the time of the accident. The conditions used in this analysis are shown in Table 15.2.6-1.

#### 15.2.6.2.3 Results of Analysis

The results are shown in Figure 15.2.6-1. It is seen that the maximum neutron power is reached several seconds after the pumps are started, and the pressure peaks well below its trip point several seconds later. The mismatch between the heat removal in the steam generator

and the power generation causes this pressure rise. The thermal power lags the neutron power and reaches its maximum value several seconds after initiation of the accident. The results of this analysis are shown in Table 15.2.6-2. Since thermal power does not exceed rated power at full flow, and the pressure does not exceed the trip setpoint, the protection criteria are met.

With three pumps in operation the temperature decrease upon starting the idle pump is only about one-half the temperature change when two pumps are started. Therefore, the total reactivity insertion due to moderator temperature decrease will be much less, leading to a milder transient.

#### 15.2.6.2.4 Additional Analyses

Additional analyses (Reference 31) have been performed for the Inactive RCS Pump Startup event and these analyses have been shown to be applicable to Davis-Besse Unit 1. An End of Life (EOL) Hot Full Power (HFP) moderator coefficient of  $-4.0 \times 10^{-4} \Delta k/k/^\circ F$  was assumed in these analyses and, although the transient response is slightly more severe than that discussed above in Section 15.2.6.2.3, the results of these analyses continue to meet the Safety Evaluation Criteria of Section 15.2.6.2.1.

#### 15.2.6.2.5 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. That evaluation concluded that the only impact would be beneficial, due to the small increase in Reactor Coolant System flow associated with the replacement Steam Generators. Therefore, the existing analyses remain applicable with the replacement Steam Generators installed.

TABLE 15.2.6-1

Pump Startup Accident Parameters

Initial power level, %	60 (of 2772 MWt)
Initial flow rate, %	49
Moderator coefficient, ( $\Delta k/k$ ) /°F	$-3.00 \times 10^{-4}$ (see note 1)
Doppler coefficient, ( $\Delta k/k$ ) /°F	$-1.45 \times 10^{-5}$

(1) A moderator coefficient of  $-4.0 \times 10^{-4} \Delta k/k/^\circ F$  has also been used in more recent additional analyses (see Section 15.2.6.2.4) with acceptable results.

TABLE 15.2.6-2

Summary of Pump Startup Accident Analysis

Maximum neutron power, %	83
Maximum thermal power, %	73
Maximum pressure rise, psi	115

## 15.2.7 Loss of External Electrical Load and/or Turbine Trip

### 15.2.7.1 Identification of Causes

The station is designed to withstand the effects of a load rejection condition caused by separation from the transmission system without reactor or turbine trip. Load rejection may result from an abnormal variation in network frequency or other adverse disturbances in the power distribution network. Upon load rejection, the reactor power is automatically runback to a power corresponding to the steam generator low level limit. Steam relief permits sufficient energy removal from the reactor coolant system to prevent reactor trip and allow the reactor and turbine generator operation to stabilize. The emergency power systems available during this transient are described in Chapter 8.

### 15.2.7.2 Analysis of Effects and Consequences

#### 15.2.7.2.1 Safety Evaluation Criteria

The safety evaluation criteria for this accident are:

1. Fuel damage shall not occur.
2. The Reactor Coolant System pressure shall not exceed code pressure limits.

#### 15.2.7.2.2 Methods of Analysis

A B&W analog hybrid computer code ("POWERTRAIN - General Hybrid Simulation for Reactor Coolant and Secondary System Transient Response," BAW-10070, July 1973) was used to determine the characteristics of this accident. The program simulates reactor coolant and steam system operation. The reactor core model is based on a point kinetics solution with reactivity feedback for control rod assembly insertion, and Doppler and moderator temperature coefficients. A Reactor Coolant System model provides for heat transfer from the core to the steam generators with appropriate coolant loop transient times. The model includes a detailed description of the secondary steam side, including the flow and valve operation of the main steam and feedwater systems. Steam relief to the atmosphere through the atmospheric vent and safety valves is distinguished from steam relief through the turbine bypass valves. The model also permits simulation of power runback, turbine stop and governor valve operation, and feedwater pump and valve control.

#### 15.2.7.2.3 Results of Analysis

The unit has been designed to accommodate a loss of load condition without a reactor or turbine trip. Under circumstances where the external system deteriorates, as indicated by the system frequency deviation, the unit will automatically disconnect from the transmission system. When this occurs, a runback signal causes an automatic power reduction to a power corresponding to the steam generator low level limit. For the analysis, the power was runback to a power level corresponding to the steam generator low level limit as shown in Figure 15.2.7-1. Other actions that occur include:

1. All electrical loads, including the reactor coolant pumps, condenser circulating water pumps, condensate pumps, and other auxiliary equipment, will continue to obtain power from the unit generator. Feedwater is supplied to the steam generators by the steam-driven feed pumps.

2. As the electric load is dropped, the turbine generator accelerates and closes the governor valves and intercept valves. The unit frequency will peak at less than the overspeed trip point and decay back to set frequency in 40 to 50 seconds.
3. Following closure of the turbine governor valves and intercept valves, steam pressure increases to the turbine bypass valve setpoint and may increase to the steam system safety valve setpoint. Steam is relieved to the condenser and to the atmosphere. Steam venting to the atmosphere occurs for about 3 minutes following loss of load from 100% initial power until the turbine bypass can handle all excess steam generated. A steam relief permits energy removal from the Reactor Coolant System to prevent a high pressure reactor trip. The initial power runback is to a power corresponding to the steam generator low level limit, which is a higher power level than needed for the unit auxiliary load. This allows sufficient steam flow for regulating turbine speed control. Excess steam above unit auxiliary load requirements is rejected to the condenser by the turbine bypass valves.
4. During the short interval while the turbine speed is high, the vital electrical loads connected to the unit generator will undergo speed increases in proportion to the generator's frequency increase. All motors and electrical gear so connected will withstand the increases in frequency.
5. After the turbine generator has been stabilized at auxiliary load and set frequency, the station operator may reduce reactor power to the auxiliary load as desired.

The loss-of-load accident does not result in fuel damage or excessive pressures on the Reactor Coolant System.

### 15.2.7.3 Plant Changes and Effects

#### 15.2.7.3.1 Identification of Changes

The following plant changes have been made which affect the loss of external electrical load and/or turbine trip as discussed above.

1. The addition of the Anticipatory Reactor Trip System (ARTS) as described in Subsection 7.4.1.4.
2. Raising the Pilot Operated Relief Valve (PORV) lift setpoint from 2255 psig to 2450 psig.
3. As the result of License Amendment No. 278, the rated thermal power was increased to 2,817 MWt from 2,772 MWt.
4. The Once-Through Steam Generators were replaced in 18RFO.

#### 15.2.7.3.2 Effects of Changes

The addition of ARTS results in a reactor trip whenever a turbine trip occurs with the reactor at or greater than 45% FP. There is no ICS initiated reactor runback and the plant follows a normal post trip response.

The change was made to the PORV setpoint to avoid challenging the PORV. Upon a loss of external electrical load, the ICS will initiate a reactor runback. If accomplished successfully, the system response is bounded by the transient described in Subsection 15.2.7.2.3. For occurrences when the ICS fails to respond or for loss of load at high power levels with unsuccessful main feedwater reduction, the RCS pressure rise exceeds the RPS high pressure trip setpoint and the reactor trips. This transient is bounded by the loss of normal feedwater accident as described in Subsection 15.2.8.

The rated thermal power was increased to 2,817 MWt from 2,772 MWt. The RCS peak pressure for a Loss of External Load and/or Turbine Trip remained bounded by the Loss of Normal Feedwater accident. The Turbine Trip accident remained bounding with respect to Steam Generator secondary side pressure. Overpressure protection is detailed in Section 5.2.2.3.

The replacement of the steam generators during 18RFO modified the low level limit. An ICS runback of reactor power targets the power that corresponds to this level.

#### 15.2.7.3.3 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. That evaluation concluded that the original USAR conclusion, that this event is bounded by the Loss of Main Feedwater event, remained valid. Therefore, the replacement Steam Generators do not impact the existing analyses.

#### 15.2.7.4 Additional Analysis

An additional Turbine Trip analysis has been performed at an initial core power level of 3,025 MWt using RELAP5/MOD2-B&W. The analysis shows that the capability of the MSSVs is sufficient to prevent the Steam Generator Pressure from exceeding the ASME Code allowable (i.e., 1169.7 psia). Additional detail for this analysis is provided in Reference 67. Information on Overpressure Protection is detailed in Section 5.2.2.3.

### 15.2.8 Loss of Normal Feedwater

#### 15.2.8.1 Identification of Causes

A loss of feedwater accident results from either a reduction in or the complete loss of secondary feedwater to the steam generators. With loss or reduction of feedwater to the steam generators, the capability of the secondary system to remove the heat generated in the Reactor Coolant System is impaired. Reactor trip, however, occurs before the steam generator heat transfer capability is significantly reduced. Since the Auxiliary Feedwater System is also available to remove the residual heat generated following reactor trip, fuel and Reactor Coolant System boundary damage will not occur.

Loss of Feedwater events can be categorized as either loss of inventory events, which result from pipe breaks and cracks, or termination of flow events, without a concurrent loss of inventory. Termination of flow can be caused by events such as abnormal closure of a feedwater valve or the failure of a feedwater pump.

Sections 15.2.8.2 and 15.2.8.3 present analyses of loss of feedwater events due to both pipe breaks and due to termination of flow. Section 15.2.8.4 presents the results of a reanalysis of

the loss of feedwater due to termination of flow. The results presented in Section 15.2.8.4 supercede the portions of Sections 15.2.8.2 and 15.2.8.3 pertaining to termination of feedwater flow.

#### 15.2.8.2 Analysis of Effects and Consequences

##### 15.2.8.2.1 Safety Evaluation Criteria

The safety evaluation criteria for this accident are:

1. Fuel damage shall not occur.
2. Reactor Coolant System pressure shall not exceed code pressure limits.

##### 15.2.8.2.2 Methods of Analysis

(See Section 15.2.8.4 for reanalysis of loss of feedwater event)

A B&W digital computer code ("KAPP4 - Digital Computer Program for Solution of Reactor Kinetics and Primary System Pressure Response," BAW-10068, June 1973) was used to determine the characteristics of this accident. Included were a complete kinetics model, pressure model, average fuel rod model, steam demand model with secondary coastdown to decay heat level, coolant transport model, and a simulation of the instrumentation for pressure and flux trip. The initial conditions were normal rated power operation without automatic control. Only the Doppler and moderator coefficients of reactivity were used as feedback. The nominal values used for the main parameters in evaluating this accident are given in Table 15.2.8-1. The minimum CRA worth that satisfies the criterion for a shutdown margin of 1%  $\Delta k/k$  at the hot standby condition is used through the analysis.

##### 15.2.8.2.3 Results of Analysis

(See Section 15.2.8.4 for reanalysis of loss of feedwater event)

Both the steam line break and the feedwater line break result in reactor trip, with the feedwater line break having the greater DNB margin during the transient. Mass and energy discharged from the steam generator during a feedwater line break are released within the containment, while the steam line break effluent could be released to the atmosphere. The environmental consequences resulting from the loss of normal feedwater due to a feedwater line break between the first feedwater line upstream check valve and the steam generator produce results no worse than the steam line break accident presented in Subsection 15.4.4.

For a loss of feedwater accident due to a feedwater accident caused by a feedwater valve failure, feedwater pump failure, or feedwater line break upstream of the first feedwater line upstream check valve, the complete loss of all feedwater has been analyzed as this is a more conservative case. The sequence of events and the evaluation of consequences are as follows:

1. Termination of all feedwater results in a reduction in secondary system heat removal capability.
2. Auxiliary feedwater pumps (two turbine driven) are automatically started by SFRCS on loss of reactor coolant pumps or high main feedwater/steam generator reverse differential pressure (low steam generator level backs up these trips).



3. The SFRCS (reverse differential pressure, low steam generator pressure) closes the main steam isolation valves.
4. Increased Reactor Coolant System temperature and pressure result in a reactor trip.
5. Following closure of the turbine stop valves, secondary system steam is relieved through the steam generator code safety valves.
6. Eventually, thermal equilibrium is re-established; i.e., the heat removal rate (steam flow through the code safety valves) is equal to the heat input (core decay heat).
7. If electric power is available and if the SFRCS signal can be cleared, decay heat removal and cooldown of the Reactor Coolant System is then provided by opening the main steam isolation valves and by relieving steam to the condenser through the turbine bypass valves with feedwater being supplied by the auxiliary or startup feedwater system. If electric power is not available or if the SFRCS signal cannot be cleared, cooldown is accomplished by manually operating the atmospheric vent valves with the feedwater being supplied by the Auxiliary Feedwater System.

The Auxiliary Feed water System is designed to remove decay heat if the main feedwater system fails or is isolated from the steam generators. The auxiliary feedwater design requirements for a loss of main feedwater transient are more severe than the auxiliary feedwater requirements following other transients such as LOCA's or steam line breaks. Table 15.2.8-2 summarizes the Loss of Normal Feedwater Analysis.

The analyses show that an auxiliary feedwater flow rate of 600 gpm is sufficient to remove the decay heat and reduce the RCS temperatures to a level where the decay heat removal system can be placed into operation. These analyses assume that the auxiliary feed water flow is delivered to the steam generators within 40 seconds after the water level in the steam generator reaches a low level of 10 inches above lower tube sheet.

The Auxiliary Feedwater System instrumentation is discussed in Sections 7.3 and 7.4.

Figures 15.2.8-1a thru d shows steam generator collapsed level, hot leg temperature, hot leg pressure and pressurizer collapsed level for a loss of feedwater transient using a 600 gpm auxiliary feedwater flow. This analysis assumed that the SFRCS steam generator low level trip to initiate auxiliary feedwater occurs at 10 inches actual water level above the lower tubesheet.

The rupture of a feedwater line between the first feedwater line upstream check valve and the steam generator was analyzed with (1) no loss of offsite power, (2) a loss of offsite power at the beginning of the accident (i.e. rupture), and (3) a loss of offsite power at the time of reactor trip.

All cases were analyzed with the following common set of assumptions:

1. A blowdown rupture area of 1.7 ft<sup>2</sup> corresponding to two (2) fourteen inch (O.D.) feedwater pipes.
2. The reactor is initially operating at rated power (2772 MWt).

## Davis-Besse Unit 1 Updated Final Safety Analysis Report

3. The unaffected steam generator is isolated from the affected steam generator by the main steam line isolation valves. No credit for steam generator isolation by the turbine stop valves is assumed.
4. Loss of all feedwater to the unaffected steam generator at the time of rupture, i.e., all feedwater flow from the main feedwater pumps is assumed to be forced out of the break.
5. Failure of an auxiliary feedwater pump or valve opening on the line (from the unaffected steam generator) supplying steam to drive an auxiliary feedwater pump turbine.
6. The steam generators are operating with an unfouled inventory.

Assumptions (2) and (6) are conservative assumptions which result in the highest power level with the minimum full power inventory. This clean generator inventory results in less water initially available for cooling in this overheating event and a minimum generator inventory for available steam to run the auxiliary feedwater pump turbines. Assumption (5) represents the worst case single failure which could occur during the transient.

### Case 1

Based on these assumptions the rupture of a feedwater line between the first feedwater line upstream check valve and the steam generator, with offsite power available, results in the following sequence of events:

1. The rupture of a feedwater line causes an immediate two phase blowdown of the affected generator. Low pressure in the affected feedwater line (differential pressure sensor) at 1 second after rupture initiates closure of the main steam line isolation valves (5 seconds closing time assumed) thus isolating the unaffected steam generator on the steam side and initiating closure of the feedwater valves on the feedwater side.
2. The termination of all feedwater to the affected steam generator will result in a reduction in its heat removal capability.

The heat removal capability of the affected steam generator rapidly diminishes as the steam generator blows dry as evidenced by the rapid loss of inventory, mixture height and steam generator pressure (Figure 15.2.8-2).

The unaffected steam generator maintains steam pressure which increases following isolation resulting in steam relief through the main steam line safety valves (Figure 15.2.8-3).

3. The eventual result is an increase in the Reactor Coolant System temperature and pressure which continues until the reactor trips on high reactor system pressure at 8.2 seconds (including .6 sec. delay) in turn, initiating trip of the turbine generator.
4. Low steam line pressure (600 psia at 6.9 sec) in the affected steam generator initiates realignment of the auxiliary feedwater piping and pumps servicing the affected generator to take steam from the unaffected steam generator and start

feeding the unaffected steam generator. A conservative delay of 40 seconds was assumed.

5. The feedwater line rupture control logic assures isolation of the unaffected steam generator to assure an adequate inventory and steam pressure to run the remaining available auxiliary feedwater pump turbine. Auxiliary feedwater is thus available to the unaffected steam generator for long-term core decay heat removal.
6. At 35 seconds thermal equilibrium is re-established, i.e. the heat removal rate steam flow through the main steam safety valves, is equal to the heat input (core decay heat), as evidenced by the average moderator temperature (Figure 15.2.8-4).
7. Decay heat removal and cooldown of the Reactor Coolant System is then provided by steam relief to the atmosphere through the atmosphere vent valves with auxiliary feedwater being supplied to the unaffected steam generator.

A complete loss of feedwater reduces the heat removal capability of both steam generators. This results in an increase in the reactor system pressure which continues until the reactor trips. As the above sequence of events states, the accident under consideration also results in a reduction in the secondary system heat removal capability. The heat removal capability of the affected generator is completely lost as the generator blows dry.

The unaffected steam generator always maintains some heat removal capability. The result is a more rapid increase in reactor system temperature and pressure (Figures 15.2.8-4 and 15.2.8-5) than shown in Figure 15.2.8-1, thus the reactor trips sooner. Curves of reactor power and thermal power are presented in Figure 15.2.8-5 for the feedwater line break with offsite power available. A curve of hot channel minimum DNBR versus time is presented in Figure 15.2.8-4. A plot of steam generator mass and energy release rate out the break as a function of time is presented in Figure 15.2.8-6. Since the thermal power is less than 112% of a nominal power level of 2772 MWt throughout the transient, the reactor system pressure is less than core design limits and minimum hot channel DNBR is well above 1.3; there is no danger of core damage.

### Case 2

The case of a feedwater line rupture concurrent with a loss of offsite power at the time of rupture has been examined and was found to result in the following sequence of events.

1. Loss of offsite power causes an immediate loss of all 4 reactor coolant pumps.
2. The reactor trips on loss of power to the control rod drives.
3. The rupture of the feedwater line causes an immediate two phase blowdown of the affected generator. Low pressure in the affected feedwater line (differential pressure sensor) at 1 second after rupture initiates closure of the main steam line isolation valves (5 second closing time assumed) thus isolating the unaffected steam generator on the steam side and initiates closure of the feedwater valves on the feedwater side.
4. The affected steam generator will continuously depressurize while the unaffected steam generator pressure increases to the main steam safety valve setpoint to

remove decay heat. Turbine bypass valve relief capability is lost due to the loss of power to the condenser circulating pumps.

5. The blowdown of the affected steam generator will cause a reduction in the Reactor Coolant System temperature and pressure until most of its inventory is depleted. With the affected steam generator and no feedwater flow assumed to the unaffected steam generator, the Reactor Coolant System temperature will increase as shown in Figure 15.2.8-7.
6. Heatup of the Reactor Coolant System will continue until auxiliary feedwater is initiated. Low steam line pressure (600 psia at 6.9 sec) in the affected steam generator initiates realignment of the auxiliary feedwater piping and pumps servicing the affected generator to take steam from the unaffected steam generator and start feeding auxiliary feedwater to the unaffected steam generator. A conservative delay of 40 seconds was assumed.
7. The feedwater line rupture control logic assures isolation of the unaffected steam generator to assure an adequate inventory and steam pressure to run the remaining available auxiliary feedwater pump turbine. Auxiliary feedwater is thus available to the unaffected steam generator for long-term core decay heat removal.
8. After initiation of auxiliary feedwater, thermal equilibrium is re-established, i.e. the heat removal rate (steam flow through the main steam safety valves is equal to the heat input (core decay heat). Decay heat removal and cooldown of the reactor system is provided by steam relief through the safety valves with auxiliary feedwater being supplied to the unaffected steam generator.

The results during the critical period for this event are less severe than, but similar to the results for the station blackout, Subsection 15.2.9. The blowdown of the affected steam generator provides more cooling than the station blackout prior to reactor coolant heatup and initiation of auxiliary feedwater. Plots of total reactor power, thermal power and RC System pressure are given in Figure 15.2.8-8. Plots of Reactor Coolant System flow, minimum hot channel DNBR, and average moderator temperature are presented in Figure 15.2.8-7. Steam pressure in the affected and unaffected generators are presented as a function of time in Figure 15.2.8-9. Steam generator blowdown mass and energy release rates are given in Figure 15.2.8-10. Thus the concurrent loss of offsite power with the feedwater line rupture does not result in any core damage or otherwise adversely affect the Reactor Coolant System.

### Case 3

A feedwater line break with a loss of offsite power at reactor trip was analyzed and found to be identical to the analysis with offsite power available (Case 1) until reactor trip. After trip, the results are similar to the loss of offsite power at the time of rupture. In the analysis for loss of offsite power at trip, there is less inventory in the affected steam generator at trip which results in less reactor system cooling. Auxiliary feedwater is initiated in the same manner as Cases 1 and 2 to provide for long-term decay heat removal. Plots of total reactor power, thermal power, and Reactor Coolant System pressure are given in Figure 15.2.8-11. Plots of Reactor Coolant System flow, minimum hot channel DNBR, and average moderator temperature are presented in Figure 15.2.8-12. Steam pressure in the affected and unaffected steam generators are presented as a function of time in Figure 15.2.8-13. Steam generator blowdown mass and energy release rates are given in Figure 15.2.8-14. Thus, this event will not result in any core damage or otherwise adversely affect the reactor coolant system.

Since the thermal power does not exceed 112% of a nominal power level of 2772 MWt and the reactor coolant system pressure does not exceed design limits, the safety evaluation criteria are met.

### 15.2.8.3 Plant Changes and Effects

#### 15.2.8.3.1 Identification of Changes

In late 1970's the following modification was performed in the plant to change the location of SFRCS pressure taps for the reverse differential pressure sensors. Specifically the location of the pressure taps for the reverse differential pressure sensors was changed to either side of the last check valve in the main feedwater line to the steam generator. This change was made because of spurious SFRCS actuations with the previous pressure tap location.

#### 15.2.8.3.2 Effects of Plant Changes

The rupture of a feedwater line inside containment causes an immediate blowdown of the affected steam generator and diversion of feedwater flow from the unaffected steam generator to the break. This results in a rapid decrease in pressure in the affected steam generator due to blowdown through the break. Similarly, due to diversion of the feedwater flow through the break, the pressure upstream of the check valve connected to the unaffected steam generator also decreases. This causes a reverse flow across the check valve on the unaffected steam generator which results in its closure. This closed check valve results in a high reverse differential pressure across the check valve and activates SFRCS. There are four channels of SFRCS reverse differential pressure instrumentation and thereby the SFRCS actuation is assured under limiting single active failure conditions.

The Technical Specifications have been revised to permit a total response time of 6.5 seconds to fully close the MSIVs after the SFRCS reverse differential pressure trip setpoint is reached.

This plant modification does not have a significant impact on the feedwater line break analyses presented above because in the above analyses it is assumed that the auxiliary feedwater to the unaffected steam generator is aligned based on the low pressure SFRCS signal in the affected steam generator. The low pressure SFRCS signals are not affected by this plant modification. The mass and energy releases are bounded by steam line break events.

(See Section 15.2.8.4 for reanalysis of loss of feedwater event)

In 1988, reanalysis (Reference 46) for a loss of Normal Feedwater event was performed to support raising the normal operating pressurizer level to 220 inches. This analysis, including some additional minor model changes, indicated a maximum pressurizer level of approximately 410 inches at 185 seconds and a maximum pressurizer pressure of only 2543 psia. Although the analysis indicated a higher pressurizer level than that shown in USAR figure 15.2.8-1D (from Reference 45) the pressurizer maintained a steam bubble. Peak pressurizer pressure was not strongly affected, remaining well below the acceptance limit.

#### 15.2.8.4 Reanalysis of Loss of Feedwater (LOFW) Event

##### 15.2.8.4.1 Need for Reanalysis

The acceptance criteria that were used for the LOFW analysis presented in USAR Section 15.2.8.2.1, included the requirement to maintain the Reactor Coolant System (RCS) pressure below code pressure limits, but did not include the design goal of preventing the pressurizer from becoming water-solid during the event. Because of this, the original analysis did not include a verification of the worst-case Pressurizer level. A reanalysis was performed to calculate the worst case pressurizer level for this event and to correct the non-conservative assumptions in the previous analyses.

##### 15.2.8.4.2 Analysis Results

The LOFW event is characterized by a rapid decrease in feedwater flow, causing a decrease in primary to secondary heat transfer. This results in an increase in RCS temperature and pressure. The reactor trips on high RCS pressure, with turbine trip shortly after. Upon turbine trip, steam generator pressure increases, causing the main steam safety valves to lift. The steam generator inventory decreases until the auxiliary feedwater (AFW) is initiated on low-low steam generator level. The limiting single failure for this event is loss of one AFW pump, so that only one steam generator receives the AFW flow. The LOFW event is a more severe transient than a feedwater line break because for a feedwater line break AFW will be initiated much sooner, based on a steam generator to feedwater reverse DP signal.

The acceptance criteria for this event consistent with Section 15.2.8.2.1, are as follows:

1. No fuel damage shall occur. This is demonstrated by maintaining the core power less than 112% of core power (2772 MWt) for fuel DNB response.
2. RCS pressure shall not exceed code pressure (i.e., 110% of the design pressure or 2750 psig).

In addition the analysis should demonstrate that the following two design goals are met.

1. Pressurizer will not go water solid during the event to prevent water release through safety valves to ensure valve reliability.
2. The steam generator tube to shell average temperature difference (tubes hotter than shell) is less than 65F to ensure that the steam generator tube load does not exceed allowable values. (Note: A higher temperature difference may be allowed, but the usage factor on certain components may increase or the allowed number of transient cycles may be significantly reduced.)

This analysis was performed with the RELAP5/MOD2-B&W computer code, Version 25. RELAP5/MOD2-B&W has been approved by the NRC for use in loss of coolant accident (LOCA) and transient (non-LOCA) analyses for the B&W-designed operating plants. The code simulates both the reactor coolant system (RCS) and steam system operation. The reactor core model is based on a point kinetics solution with reactivity feedback for control rod assembly insertion, fuel temperature changes, and moderator density changes. The RCS model provides for heat transfer from the core, transport of the coolant to the steam generators, and heat transfer to the steam generators. The secondary model included a detailed model of the main steam system, including steam relief to the atmosphere through the main steam safety valves,

simulation of turbine stop valve and main steam isolation valve operation. The secondary model also includes the delivery of feedwater, both main and auxiliary, to the steam generators.

The methodology employed is outlined in Appendix A of Reference 56. The analysis results are presented in Ref 57. Pertinent initial conditions and boundary conditions for the reanalysis are listed in Table 15.2.8-3.

To demonstrate that all of the LOFW event acceptance criteria and design goals are met, two separate cases are required. The first case is for peak pressure and only considers the pressurizer code safety valves. It does not model any pressurizer spray or PORV operation. The second case is for peak pressurizer level, and it includes the PORV and pressurizer spray. The core power and SG tube-to-shell average temperature difference are evaluated for each case.

#### Peak RCS Pressure Case

In this case, the pressurizer sprays and PORV are not used in the analysis because actuation of the sprays or PORV would reduce the peak RCS pressure.

The loss of feed water causes a rapid decrease in primary to secondary heat transfer which causes RCS temperature and RCS pressure to rise quickly. Pressurizer pressure increases past the lift setpoint of the Pressurizer Safety Valves, but RCS pressure remains below the acceptance criterion of psig 2764.7 psia (2750 psig), with maximum RCS pressure reaching 2740.0 psia. The combination of reactor trip and PSV relief limits the RCS pressure but the PSVs cycle until the AFW boiloff coupled with MSSV lift is sufficient to remove decay heat and reactor coolant pump (RCP) heat. From that point on the RCS pressure and temperature decrease as the heat removal rate of the steam generators exceeds the heat input from the RCS.

With the termination of MFW and lift of the MSSV, SG inventory boils off and is not replenished by MFW, and SG level decreases rapidly. When, the low-low level setpoint is reached, AFW to one SG is initiated following the 40 second delay. The introduction of AFW to one SG increases the collapsed liquid level to 10 inches, where it remains until transient termination.

The maximum tube average to shell average temperature difference was 56.2 F at 250.0 seconds, which is within the design goal of 65 F. The analysis results showed that the pressurizer did not become water solid during the transient. Because the LOFW is a heatup event, and the moderator temperature coefficient is zero and the Doppler temperature coefficient is negative, the maximum power level of 102 percent power (of 2772 MWt) occurs at time zero. Therefore all acceptance criteria and design goals are met in this case.

The sequence of events for this case are summarized in Table 15.2.8-4.

#### Peak Pressurizer Level Case

The overall transient evolution of this case is very similar to the peak RCS pressure case. In this case, operation of pressurizer spray and the PORV is included to maximize the pressurizer level.

The loss of MFW causes a rapid decrease in primary to secondary heat transfer, which causes RCS temperature and RCS Pressure to rise quickly. Even though the pressurizer sprays are on and the PORV lifts, the pressurizer pressure increases past the lift setpoint of the PSVs. The

combination of reactor trip and pressure relief causes the RCS pressure to decrease, but the ongoing heat transfer deficit causes the PORV to cycle until the AFW boiloff, coupled with MSSV lift, is sufficient to remove decay heat and RCP heat. Then the RCS pressure and temperature decrease, as the heat removal rate of the steam generators is higher than the heat input from the RCS. The maximum RCS pressure is 2679.7 psia, which is below the acceptance criterion of 2764.7 psia.

Pressurizer level rises rapidly during the heatup due to expansion of the RCS inventory. Just as in the peak RCS pressure case, as AFW and MSSV relief begin to match decay heat and RCP heat, pressurizer level flattens out at approximately 225 seconds. Level then stays nearly flat until the contraction of RCS inventory (due to a decrease in RCS temperature) causes the level to drop. The analysis results showed that the pressurizer did not become water solid during the transient.

The analysis used an initial pressurizer level of 231.6" which includes an instrument uncertainty to the nominal pressurizer level during power operation. The pressurizer level value used in the analysis is higher than the 228" allowed by the plant Technical Specifications. Since prevention of water relief through safety valves is a goal for abnormal transient operation, rather than a safety limit, instrument uncertainty was not applied to the maximum value allowed by the Technical Specifications.

The maximum tube average to shell average temperature difference was approximately 56.5 F, which is within the design goal of 65 F. The maximum power level of 102 percent power (of 2772 MWt) occurs at time zero. Therefore, all acceptance criteria and design goals are met in this case. The sequence of events for this case are summarized in Table 15.2.8-5.

#### 15.2.8.5 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. For a loss of main feedwater event, that evaluation concluded that the secondary mass in the replacement Steam Generators will meet or exceed the value used in the existing analysis (lower mass causes worse results). Additionally, the heat transfer and the dryout time for the replacement Steam Generators are essentially the same as for the original Steam Generators. Therefore, the existing loss of main feedwater analyses remained bounding with the replacement Steam Generators installed.

For a main feedwater line break, the evaluation concluded that the secondary inventory in the replacement Steam Generators, combined with their improved AFW bypass performance, would reduce the severity of the event, relative to RCS pressurization and DNBR. Therefore, the existing main feedwater line break analyses remain applicable with the replacement Steam Generators installed.



TABLE 15.2.8-1

Loss of Normal Feedwater Accident Parameters

Doppler coefficient at rated power, BOL, (Dk/k)/ °F	-1.28 x 10 <sup>-5</sup>
Moderator coefficient at rated power, BOL (Dk/k)/ °F	+0.13 x 10 <sup>-4</sup>
Trip parameters	
Delay for high pressure trip, sec.	0.5
Delay for high flux trip, sec.	0.4
CRA travel time to 2/3 insertion, sec.	1.4

(See Section 15.2.8.4 for reanalysis of this event.)

TABLE 15.2.8-2

Summary of Loss of Normal Feedwater Analysis

Reactor trip, sec.	14
Auxiliary feedwater initiation, sec.	40
Maximum reactor coolant system pressure, psia	2590
Maximum thermal power,%	100 (of 2772 MWt)

(See Section 15.2.8.4 for reanalysis of this event.)

TABLE 15.2.8-3

Loss of Normal Feedwater Accident Parameters for the Reanalysis

Core Power	102% of 2772 MWt
Pressurizer Level	231.6 inches
Doppler coefficient at rated power, BOL (Dk/k)/ °F	-1.34 x 10 <sup>-5</sup>
Moderator Temperature Coefficient, BOL (Dk/k)/ °F	0.0
Trip Parameters	
High Pressure Reactor Trip Setpoint (Hot Leg Tap)	2400 psia
Delay for high pressure trip, sec.	0.6 second
CRA travel time to full insertion	2.3 seconds
AFW Parameters	
AFW Temperature	120 F
AFW Delay Time	40 seconds

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 15.2.8-4

Sequence of Events for Peak RCS Pressure Case

EVENT	Time, Sec
Event Starts	0
MFW at 0.0 lbm/sec	7
Reactor Trip on High Pressure	16.5
Rod Motion	17.1
Turbine Trip	17.1
PSV lift	19.1
MSSV lift	19.6
Peak RCS Pressure 2740.0 psia	19.8
Peak Secondary Pressure 1154.7 psia	21.5
Low-Low SG Level in SG A	60.8
Pressurizer Level Offscale High	100
AFW flow to SG B Initiated	100.8
Peak SG Tube to Shell DT, 56.2 F SG B	250
RCS Cooldown Begins, Max Tavg 615.4 F	265
Peak Pressurizer Level 39.2 feet	375
Termination	600

Note: Pressurizer is considered water solid at a pressurizer level of 41.9 feet.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 15.2.8-5

Sequence of Events for Peak Pressurizer Level Case

EVENT	Time, Sec
Event Starts	0
MFW at 0.0 lbm/sec	7
PZR Sprays on	11.5
Reactor Trip on High Pressure	16.7
Rod Motion	17.3
Turbine Trip	17.3
PORV Lift	17.8
PSV lift	19.1
Peak RCS Pressure 2679.7 psia	19.5
MSSV lift	19.8
Peak Secondary Pressure 1152.6 psia	21.7
Low-Low SG Level in SG A	60
Low-Low SG Level in SG B	65.8
Pressurizer Level Offscale High	92
AFW flow to SG B Initiated	100
Peak SG Tube to Shell DT, 56.5F SG B	220
RCS Cooldown Begins, Max Tavg 615.7 F	265
Peak Pressurizer Level 40.9 feet	340
Termination	600

Note: Pressurizer is considered water solid at a pressurizer level of 41.9 feet.

15.2.9 Loss of all AC Power to the Station Auxiliaries (Station Blackout)

15.2.9.1 Identification of Causes

The loss of all AC power accident is a hypothetical case in which all unit AC power is lost. The unit batteries are available to supply DC power. This event is very conservative for normal station operation since redundant quick starting emergency diesel generators are available to supply station essential loads. A dedicated alternate AC power source (Station Blackout Diesel Generator) is available to supply systems required for coping with a station blackout as defined in 10CFR50.2. The SBODG is not taken credit for in this analysis. However, even in the event of this hypothetical case, reactor trip will occur promptly upon loss of power without the assistance of the Reactor Protection System. System and decay heat can be removed with the Steam and Auxiliary Feedwater Systems to preclude fuel damage or excessive pressures in the Reactor Coolant System.

15.2.9.2 Analysis of Effects and Consequences

15.2.9.2.1 Safety Evaluation Criteria

The safety evaluation criteria for this accident are:

1. Fuel damage shall not occur.
2. The Reactor Coolant System pressure shall not exceed code pressure limits (2750 psig).

15.2.9.2.2 Methods of Analysis

A B&W analog hybrid computer code ("POWERTRAIN - General Hybrid Simulation for Reactor Coolant and Secondary System Transient Response," BAW-10070, July 1973) was used to determine the characteristics of this accident. The program simulates Reactor Coolant and Steam System operation. The reactor core model is based on a point kinetics solution with reactivity feedback for CRA insertion, Doppler and moderator temperature coefficients. A Reactor Coolant System model provides for heat transfer from the core to the steam generators with appropriate coolant loop transient times. The model includes a detailed description of the secondary steam side, including the flow and valve operation of the main steam and feedwater systems. Steam relief to the atmosphere through the atmospheric vent and safety valves is distinguished from steam relief through the turbine bypass valves. The model also permits simulation of turbine stop and governor valve operation, and feedwater pump and valve control.

15.2.9.2.3 Results of Analysis

The sequence of events and the evaluation of consequences for this accident are:

1. A loss of power results in gravity insertion of the CRA's, trip of the turbine stop valves and reactor coolant pumps, actuating the SFRCS and closing the main steam isolation valves.
2. After the turbine stop valves trip, excessive temperatures and pressures in the Reactor Coolant System are automatically prevented by excess steam relief through the main steam line safety valves (turbine bypass valve steam relief is lost due to the loss of power to the condenser circulating pumps). The steam relief

capability of the station is discussed in the description of the main steam system in Chapter 10. Excess steam is relieved until the Reactor Coolant System temperature is below the saturation temperature for the pressure corresponding to the lowest set-point of the steam safety valves. Thereafter, the main steam safety valves are used to remove decay and system heat until AC power is restored by the emergency diesels. Operationally it would normally be preferable to use manual control of the atmospheric vent valves, if available, to maintain secondary system pressure below the lift pressure of the main steam safety valves rather than allowing the reactor coolant system to beat up and the steam generators to repressurize enough to lift the main steam safety valves repeatedly. However, no credit was taken in this analysis for operation of the atmospheric vent valves.) Once AC power is restored, the operator can then reduce the reactor coolant temperature to 280°F and place the Decay Heat Removal System into operation to continue the cooldown to ambient conditions.

3. Following loss of power to the reactor coolant pumps, the Reactor Coolant System flow decays without the occurrence of fuel damage. The energy stored in the main turbine is not required to supply power to the reactor coolant pumps following the loss of power. Decay heat removal after coastdown of the reactor coolant pumps is provided by the natural circulation characteristics of the system. This capability is discussed in the Loss of Forced Reactor Coolant Flow evaluation (Subsection 15.2.5).
4. The turbine-driven auxiliary feed pumps provide feedwater to the steam generator by taking suction from the Condensate Storage Tanks and are driven by steam from either steam generator. The Auxiliary Feedwater System is discussed in Chapter 9. The manual controls and auxiliary systems for the auxiliary feed pump operate on DC power supplied from the station DC buses or emergency diesel generators.

In view of the foregoing sequence, the loss of all AC power as shown in Figure 15.2.9-1 does not result in excessive pressure in the Reactor Coolant System. Likewise the natural circulation characteristics of the Reactor Coolant System as shown in Figure 15.2.9-1 and tabulated in Table 15.2.5-4, will assure core decay heat removal and a minimum core DNBR greater than 1.30.

#### 15.2.9.2.4 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. Natural circulation performance in the replacement Steam Generators is improved due to the replacement Steam Generators having less AFW bypass flow and a small increase in heat transfer capability. Therefore, the existing analyses remain applicable with the replacement Steam Generators installed.

#### 15.2.10 Excessive Heat Removal Due to Feedwater System Malfunction

##### 15.2.10.1 Identification of Causes

Excessive heat removal from the Reactor Coolant System can result from a malfunction or inadvertent operator adjustment of the feedwater control system which causes a reduction in feedwater temperature or an excessive increase in feedwater flow. A reduction in feedwater

temperature would result if feedwater flow was diverted around the feedwater heaters without a corresponding reduction in feedwater flow. An increased feedwater flow would result if a feedwater control valve was opened to greater than its normal operating position.

During end-of-life rated power operation, excessive heat removal from the Reactor Coolant System will result in a maximum reactivity insertion to the core since the average reactor coolant temperature will decrease and the moderator temperature coefficient is most negative. Normally operator or ICS action would correct feedwater system malfunctions, however, such actions were not considered in the analysis of this accident. Only the low reactor coolant pressure and high neutron flux trip were used in the analysis to assure reactor protection.

#### 15.2.10.2 Analysis of Effects and Consequences

##### 15.2.10.2.1 Safety Evaluation Criteria

The safety evaluation criteria for this accident are:

1. Fuel damage shall not occur.
2. The Reactor Coolant System pressure shall not exceed code pressure limits.

##### 15.2.10.2.2 Method of Analysis

A B&W analog hybrid computer code ("POWER TRAIN - General Hybrid Simulation for Reactor Coolant and Secondary System Transient Response," BAW-10070, July 1973) was used to determine the characteristics of this accident. The program simulates reactor coolant and secondary steam system operation. The reactor core model is based on a point kinetics solution with reactivity feedback based on the Doppler and moderator temperature coefficients. A Reactor Coolant System model provides for heat transfer from the core to the steam generators with appropriate coolant loop transit times. The model includes a detailed description of the secondary steam side, including feedwater and steam flow and valve operation. The initial conditions for this accident are summarized in Table 15.2.10-1.

##### 15.2.10.2.3 Results of Analysis

Two cases were analyzed which could lead to excessive heat removal due to feedwater system malfunctions:

1. A reduction in feedwater temperature due to loss of a feedwater heater, or accidental opening of the feedwater bypass valve which diverts flow around the high pressure feedwater heaters.
2. An increase in feedwater flow due to full opening of a feedwater control valve with the reactor at rated power and at no load conditions.

##### Reduction in Feedwater Temperature:

The feedwater temperature can be reduced by 40°F due to the loss of a last stage heater. However, the maximum reduction in feedwater temperature at rated power occurs by opening of the bypass around the high pressure feedwater heaters for which an 85°F decrease in feedwater temperature can be realized. At any given load there is a balance in the energy exchanged between the primary and secondary side of the steam generators. Normally the ICS

will modify the total feedwater flow demand signal, as a function of the feedwater temperature error, to maintain this balance. This analysis, however, assumes the failure of the ICS to adjust the feedwater demand signal in response to the feedwater temperature reduction.

The reactor coolant temperature decreases during the transient and results in a power increase due to the negative moderator coefficient. Without temperature compensation of the feedwater flow the steam generator level rises to the high level limit where feedwater flow is reduced to prevent flooding of the steam generator. For the 85°F step drop in feedwater temperature the reactor will trip on high flux in approximately 30 seconds as shown in Figure 15.2.10-1. Reactor trip terminates the transient by limiting system temperatures and pressures and by ensuring that the DNBR remains above 1.3 as shown in Figure 15.2.10-1; thus fuel and Reactor Coolant System damage are prevented from occurring.

#### Increase in Feedwater Flow:

Opening a feedwater control valve with the reactor at rated power causes the total feedwater flow to overshoot. Normally the ICS reacts to adjust the total flow to its steady-state condition. This analysis, however, assumes the failure of the ICS to adjust the feedwater demand signal.

The reactor coolant temperature decreases due to increased subcooling in the steam generator, which results in a power increase due to the negative moderator coefficient. If no ICS or operator action is assumed in the rated power analysis, then the steam generators would be flooded just after reactor trip. Liquid entrainment would soon thereafter disable the main feedwater pump turbines, terminating the source of the transient. Additionally, the SFRCS high steam generator level trip terminates this transient by isolating the steam generator. However, these analyses do not take credit for this SFRCS trip.

To assess the potential for pressure buildup in the secondary system, the liquid relief capacity of the steam safety valves was calculated. The liquid relief capacity of these valves was based on the following conservative assumptions:

1. liquid flow calculated with isentropic expansion techniques.
2. flow corrected by factor of 0.9 to account for arbitrary conservatism imposed on all valves by ASME.
3. flow adjusted by discharge coefficient of 0.975.
4. flow based on saturated liquid relief conditions.

The liquid relief capacity of the combined safety valves per steam generator (9 valves with a total area of 0.93 ft<sup>2</sup>) was calculated to be about 15.8x10<sup>6</sup> lbm/hr. This capacity is more than twice as much as that required to handle the transient feedwater flow rate per steam generators of 6.7x10<sup>6</sup> lbm/hr prior to main feedwater termination. The two lowest setpoint safety valve banks (4 valves with a total area of 0.44 ft<sup>2</sup>) alone would accommodate the feedwater flow, thus limiting the secondary system pressure to 1085 psia. Due to the large saturated liquid relief capacity of these valves the secondary pressure would also continue to remain within ASME design limits during decay heat removal following feedwater termination.

As discussed in the transient evaluation and shown in Figure 15.2.10-1, the core is protected by reactor trip on high flux. In the no-load analysis, no ICS or operator action is required since, as shown in Figure 15.2.10-2, a new steady-state operation is established without reactor trip.

For the feedwater malfunction transient of a 15% increase in feedwater flow from 102% power (of 2772 MWt) the parameters versus time are shown in Figure 15.2.10-3. These parameters also represent the 85°F feedwater temperature decrease transient at 102% of a nominal power of 2772 MWt since both transients produce the same excessive heat removal from the RC system. In the case of the feedwater temperature malfunction, the feedwater flow rate remains constant at 102%, assuming the failure of the ICS to adjust the feedwater demand signal in response to the feedwater temperature reduction.

The effect of opening a feedwater control valve with the reactor at no load conditions and the turbine stop valves closed is a greater decrease in reactor coolant temperature due to steam generator subcooling than the rated power case. This transient thus results in a more severe power rise for end-of-life conditions. Figure 15.2.10-2 shows the major reactor coolant and secondary system variables with time. The core average temperature and the reactor system pressure decrease initially without causing reactor trip, then turn around due to the increase in secondary system pressure when the bypass capability of 25% steam flow is exceeded. The increase in secondary system pressure decreases the enthalpy of the steam thereby decreasing the energy rate of transfer capability which increases coolant temperature until thermal equilibrium between the primary and secondary side of the steam generators is achieved. Secondary pressure increases to the setpoint of the first steam safety valve bank where it is regulated until the operator can compensate for the malfunction. Since there is not a reactor trip and no core design limits are exceeded, there will be no resultant core or Reactor Coolant System damage and the safety evaluation criteria are met.

The Reactor Protection System provides DNBR protection against those reactor outlet temperatures and reactor coolant pressures which would cause a minimum "hot channel" DNBR less than 1.3 for 100% flow and a nominal power of 112% of 2772 MWt. Figure 15.2.10-2 indicates neutron power never exceeds 65% and the reactor outlet temperature corresponding to the minimum reactor coolant pressure during the transient (2000 psia) is much less than that required to cause a minimum DNBR less than 1.3 even at 112% power (of 2772 MWt). Therefore, there is no danger of exceeding a minimum "hot channel" DNBR of 1.3 for this transient. Furthermore, the RPS would initiate a reactor trip if conditions were to approach those which would cause a "hot channel" DNBR less than 1.3.

In the transient discussed above in which the main feedwater control valve is assumed to open with the reactor at no load, the analysis assumes that this failure admits a large excess flow of feedwater to the steam generator. Normally there is no steam being supplied to the main feedwater pump turbines and the main feedwater block valves are closed. Hence, any actual increase in feedwater flow would be small and the effect would be minimal. Thus, the analysis presented is extremely conservative.

#### 15.2.10.2.4 Effects of Plant Changes

The Technical Specifications have been revised to change the setpoints on the Main Steam Safety Valves (MSSVs) to allow setting 7 of the 9 MSSVs per steam generator to relieve at 1100 psig while keeping two valves set to relieve at 1050 psig. This means that the two MSSVs previously set at 1070 psig and the 3 MSSVs previously set at 1090 psig may now be set to relieve at 1100 psig instead. Thus, the post-trip secondary pressure spike may be slightly higher than before, but it will still remain less than 1155 psig (i.e., 110% of the system design pressure of 1050 psig). Also, the secondary system pressure during liquid relief discussed in Section 15.2.10.2.3 would be limited to 1115 psia rather than 1085 psia since it is based on the setpoints of the 4 lowest setpoint MSSVs per generator.



#### 15.2.10.2.5 Additional Analyses

As part of analyses performed more recently to permit a more negative moderator coefficient (Reference 31), it was demonstrated that the Excessive Heat Removal event, when initiated from Hot Full Power (HFP) conditions, was bounded by the results of the Steam Line Break event. Further, it was shown that the moderator coefficient assumed for the Excessive Heat Removal event at no load Hot Zero Power (HZP) conditions ( $-3.0 \times 10^{-4} \Delta k/k/^\circ F$ ), was bounded by the temperature coefficient (combination of moderator and Doppler coefficients) assumed for the Steam Line Break event at HZP and colder conditions ( $-3.1 \times 10^{-4} \Delta k/k/^\circ F$ ). Therefore, the negative moderator coefficient limits developed from the Steam Line Break event results are bounding for the Excessive Heat Removal event.

#### 15.2.10.2.6 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. That evaluation concluded that the replacement Steam Generator's small increase in RCS flow, small increase in steam generator tube heat transfer area, and small decrease in thermal conductivity of the tubes result in no net impact on the plant's response to a feedwater system malfunction. Therefore, the existing analyses remain applicable with the replacement Steam Generators installed.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 15.2.10-1

Excessive Heat Removal  
Accident Parameters

Doppler coefficient, EOL, $\Delta k/k/^\circ\text{F}$	$-1.45 \times 10^{-5}$
Moderator coefficient, EOL, $\Delta k/k/^\circ\text{F}$	$-3.0 \times 10^{-4}$
High flux trip, %	112 (of 2772 MWt)
High flux trip delay time, sec.	0.4
Maximum decrease in feedwater temperature, $^\circ\text{F}$	85
Maximum feedwater flow, %	115

15.2.11 Excessive Load Increase

15.2.11.1 Identification of Causes

The excessive load increase accident is defined as a sudden increase in secondary-side steam flow causing a mismatch between the reactor core power production and the steam generator heat demand. This accident could result from the inadvertent opening of a steam relief or turbine bypass valve by the operator or an equipment malfunction such as a pressure regulator failure. The steam conversion system adjusts to load increases within the limits of its automatic control operation as discussed in Chapter 10. When load increases cannot be accommodated, the Reactor Protection System will trip the reactor on low reactor coolant pressure or high neutron flux.

15.2.11.2 Analysis of Effects and Consequences

The consequences of this accident will be less severe than, but similar to those discussed in the Steam Line Break Analysis (Section 15.4.4).

15.2.11.2.1 Safety Evaluation Criteria

The safety evaluation criteria for this accident are:

1. The core shall remain intact for effective core cooling.
2. Loss of reactor coolant boundary pressure integrity resulting from steam generator tube failure due to the loss of secondary side pressure and resultant temperature gradients shall not occur.
3. Resultant doses shall not exceed 10CFR100 guideline values.

15.2.11.2.2 Results of Analysis

An excessive load increase which cannot be accommodated by the automatic control operation of the steam conversion system is analogous to a steam line break.

The inadvertent opening of a steam relief or turbine bypass valve by the operator or an equipment malfunction such as a pressure regulator failure will cause a sudden decrease in the secondary system pressure. The reduction in steam pressure is accompanied by an increase in the steam flow through the steam generator which decreases the RC system temperature and pressure. The steam conversion system will adjust to load increases within its automatic control operation as discussed in Chapter 10. However, if the load increase is not within its automatic control operation, the RC system temperature and pressure will continue to decrease until the Reactor Protection System trips the reactor on low reactor coolant pressure or high neutron flux.

Reactor trip is followed by turbine trip and initiates minimum feedwater level control to provide for core decay heat removal. In any event, the most conservative excessive heat removal accident postulated will be bounded by the steam line break accident presented in Subsection 15.4.4, thereby ensuring core safety.

15.2.11.2.3 Comparison of Excessive Load Increase Accident to Inadvertent Opening of Pressurizer Safety Valve Accident

The following comparison was prepared in response to NRC Question 15.2.16 in the FSAR.

The inadvertent opening of a pressurizer safety valve will necessarily provide saturated conditions within the Reactor Coolant System. This is due to a decrease in the Reactor Coolant System pressure, with only slight variation in the Reactor Coolant System temperature. By comparison the excessive load increase accident maintains the RC system in a subcooled state, since the RC system is not only depressurized but cooled down as well.

After the RC system reaches saturation, both the RC temperature and pressure will continue to decrease resulting in reactor trip on low RC pressure. Following reactor trip the Reactor Coolant System continues to depressurize initiating the High Pressure Injection (HPI) System.

Initially, very low quality steam is relieved through the pressurizer safety valve. This results in a flow rate through the valve which exceeds the makeup capability of the HPI system (one pump operating). Thus, even after actuation of the HPI system the Reactor Coolant System will continue to depressurize until high quality steam is relieved through the pressurizer safety valve. With the system depressurized, HPI flow will be capable of matching high quality steam flow, thus terminating the transient.

Inadvertent opening of a pressurizer safety valve does not result in any adverse core or reactivity effects. By comparison, the excessive load increase accident represents the more severe transient from a reactivity standpoint, and a less severe transient with respect to core effects.

A detailed analysis of the transient resulting from the inadvertent opening of a pressurizer safety valve for a 205-FA plant is given in the Topical Report BAW-10099, "Babcock & Wilcox Anticipated Transients Without Scram Analysis." Since it is an ATWS event, the accident described in the report represents a conservative account of the sequence of events and of the results of the same accident when assuming reactor trip capability. The analysis shows that the core is covered throughout the transient because of the HPI action. Cladding temperature is maintained near the saturation temperature of the water because of plentiful water flow and sufficient inventory to cover the core with solid water. Peak primary pressure reaches 2662 psia, while the maximum building pressure of 10.6 psig is well below the DBA pressure limit. The reactor was brought subcritical quickly.

Although the hydrodynamic and containment analyses were performed for a 205-FA plant, it is concluded that the results are valid also for Davis-Besse 1 because of the similarity of design between the two plants and the ease with which the accident is controlled. The conclusion that the referenced analysis represents a conservative description of the inadvertent opening of a pressurizer safety valve for DB-1 is supported by the results of the 0.05-ft<sup>2</sup> small break analysis, including reactor trip, for DB-1 (BAW-10075A, Rev 1). That break size is larger than the 0.03-ft<sup>2</sup> safety valve size on the pressurizer of the 205-FA plant in the ATWS analysis. Furthermore, the 0.05-ft<sup>2</sup> break is assumed to be in the cold leg, which gives more severe results than if the break were in the hot leg for simulation of the stuck-open pressurizer valve. Also for this case, the analysis shows that the temperature of the cladding never exceeds the saturation temperature of the water.

#### 15.2.11.2.4 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. That evaluation concluded that the plant's response to an excessive load increase remains bounded by the main steam line break analysis, as documented above. Therefore, the existing analyses remain applicable with the replacement Steam Generators installed.

#### 15.2.12 Anticipated Variations in the Reactivity of the Reactor

##### 15.2.12.1 Identification of Causes

During normal operation of the reactor, the overall reactivity of the core changes because of fuel depletion, xenon burnout, and changes in fission product poison concentration and burnable poison depletion. These reactivity changes, if left uncompensated, could cause the operating limits to be exceeded. In all cases, however, the Reactor Protection System prevents the safety limits from being exceeded. No core or Reactor Coolant System boundary damage occurs from these conditions.

The ICS senses any reactivity change in the reactor during normal operation. Depending on the direction of the reactivity change, the reactor power increases or decreases. Correspondingly, the average temperature of the Reactor Coolant System increases or decreases, and the Automatic Reactor Control System acts to restore reactor power to the power demand level and to re-establish this temperature at its setpoint. If manual corrective action is not taken, or if the Automatic Control System malfunctions, then the Reactor Coolant System's average temperature changes to compensate for the reactivity disturbance. In the analysis it is assumed that the secondary system follows the temperature changes in the Reactor Coolant System. Even though ICS action is available to prevent or mitigate this accident, the accident analysis was done without ICS action.

##### 15.2.12.2 Analysis of Effects and Consequences

###### 15.2.12.2.1 Safety Evaluation Criteria

The safety evaluation criteria for this accident are:

1. The rate of reactivity addition shall be much less than the minimum rate at which the operator can compensate for the addition.
2. The rate of temperature change shall be much less than the minimum rate at which the ICS can compensate for the change.

###### 15.2.12.2.2 Methods of Analysis

A B&W digital computer code ("KAPP4-Digital Computer Program for Solution of Reactor Kinetics and Primary system Pressure Response," BAW-10068, June 1973) was used to determine the characteristics of this accident based on the maximum rate of reactivity change from fuel depletion or fission product poison concentration. Included were a complete point kinetics model and an average fuel rod model. The initial conditions were normal rated power operation without automatic control. The Doppler and moderator coefficients of reactivity were used as feedback. The values of the principal parameters used in this analysis are listed in Table 15.2.12-1.

#### 15.2.12.2.3 Results of Analysis

Table 15.2.12-2 summarizes the reactivity changes and the corresponding change in the average moderator temperature for each uncompensated reactivity disturbance for four-, three-, and two-reactor coolant pump operation. The reactivity addition rates reflect differences in the maximum allowable power for four-, three-, and two-reactor coolant pump operation, while the rates of average temperature change due to uncompensated reactivity changes reflect both the power differences and differences in reactor coolant flowrates.

The reactivity addition rates and rates of temperature change due to uncompensated fuel depletion with three- and two-reactor coolant pump operation are not shown since they are bounded by the corresponding rates associated with xenon buildup.

For xenon buildup and xenon burnout, Table 15.2.12-2 shows that the rates of RC system temperature change are consistent for two-, three- and four-pump operation. The maximum reactivity rates for four-pump operation are very close to the values for two- and three-pump operation. The rates for two- and three-pump operation are so relatively small that a conservative value was calculated and used for both cases. While analysis results for two reactor coolant pumps are reported, power operation with only two reactor coolant pumps running is not allowed by Davis-Besse Operating License Condition C.3.a.

The reactivity changes due to fuel depletion, xenon buildup, and xenon burnout are extremely slow and allow time for the operator or the Integrated Control System to detect and compensate for the change. Normally these operating reactivity changes can be handled by the automatic control system, which would insert or withdraw control rod assemblies to maintain the correct power level and thus limit the reactor coolant system temperature change.

#### 15.2.12.2.4 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. That evaluation concluded that steam generator performance does not impact this accident in any way. Therefore, the existing analyses remain applicable with the replacement Steam Generators installed.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 15.2.12-1

Uncompensated Operating Reactivity  
Change Parameters

Doppler coefficient at rated power, ( $\Delta k/k$ ) / °F	-1.28 x 10 <sup>-5</sup>
Moderator coefficient at rated power, ( $\Delta k/k$ ) / °F	+0.13 x 10 <sup>-4</sup>

TABLE 15.2.12-2

Uncompensated Reactivity Disturbances

<u>Cause</u>	<u>Maximum reactivity rate, (<math>\Delta k/k</math>) / sec</u>	<u>Rate of average temperature change, °F / sec</u>
Fuel depletion		
For four pumps	-2.95 x 10 <sup>-9</sup>	-0.0000067
Xenon buildup		
For four pumps	-3.82 x 10 <sup>-7</sup>	-0.00083
For three pumps	-3.18 x 10 <sup>-7</sup>	-0.0012
For two pumps	-3.18 x 10 <sup>-7</sup>	-0.002
Xenon burnout		
For four pumps	+1.53 x 10 <sup>-5</sup>	+0.034
For three pumps	+1.90 x 10 <sup>-5</sup>	+0.073
For two pumps	+1.90 x 10 <sup>-5</sup>	+0.120

### 15.2.13 Failure of Regulating Instrumentation

#### 15.2.13.1 Accident Analysis

The failure of regulating instrumentation is the basis of many of the accidents analyzed. A malfunction of components in the ICS and CRA drive control system would not result in an accident worse than the uncontrolled CRA group withdrawal from subcritical (Subsection 15.2.1) or rated power (Subsection 15.2.2) conditions. A failure of regulating instrumentation would not result in a greater power-coolant mismatch than those described by the pump startup (Subsection 15.2.6) or the loss of coolant flow (Subsection 15.2.5) accidents. The failure of regulating instrumentation in the secondary system would not result in an accident worse than the load increase (Subsection 15.2.11) or loss of feedwater (Subsection 15.2.8) accidents.

Since this accident is bounded by the results of other accidents, detailed analyses have not been performed for every system from a failure point of view. However, at the design stage, each system that has a possible effect on the safety of the plant is reviewed to determine if the failure or maloperation of that system will adversely affect the safety of the plant.

In addition to these reviews, it is a routine safety analysis assumption that unless an action is "guaranteed" by the protection system it does not occur. Therefore, no credit is taken for runbacks, interlocks, etc.

"Guarantee" refers to the assurance provided in the design of such safety-related features as the Reactor Protection System and the engineered safety features through redundancy, physical separation, seismic considerations, etc., that, if challenged, these systems would perform their intended function. Specific system design criteria for these systems are summarized in Table 7.1-1. These criteria are also discussed in greater detail in the design basis for safety-related features in Chapter 7.

#### 15.2.13.2 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. That evaluation concluded that steam generator performance does not impact this accident in any way. Therefore, the existing analyses remain applicable with the replacement Steam Generators installed.

### 15.2.14 External Causes

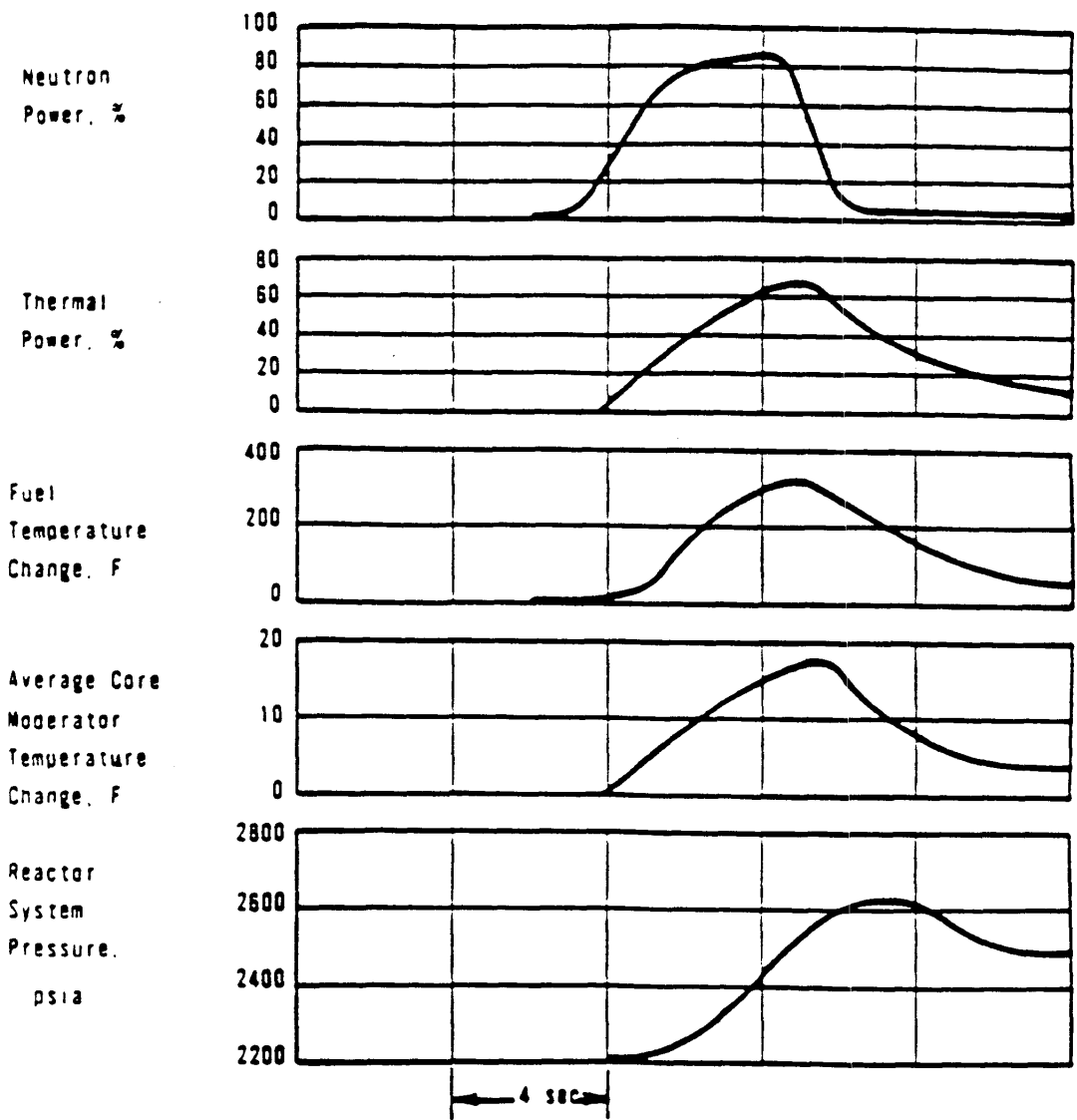
#### 15.2.14.1 Accident Analysis

Storms and earthquakes have been considered in the design of the reactor plant as illustrated by the responses to the General Design Criteria in Section 3.1 and Chapter 6.

#### 15.2.14.2 Impact of Replacement Steam Generators

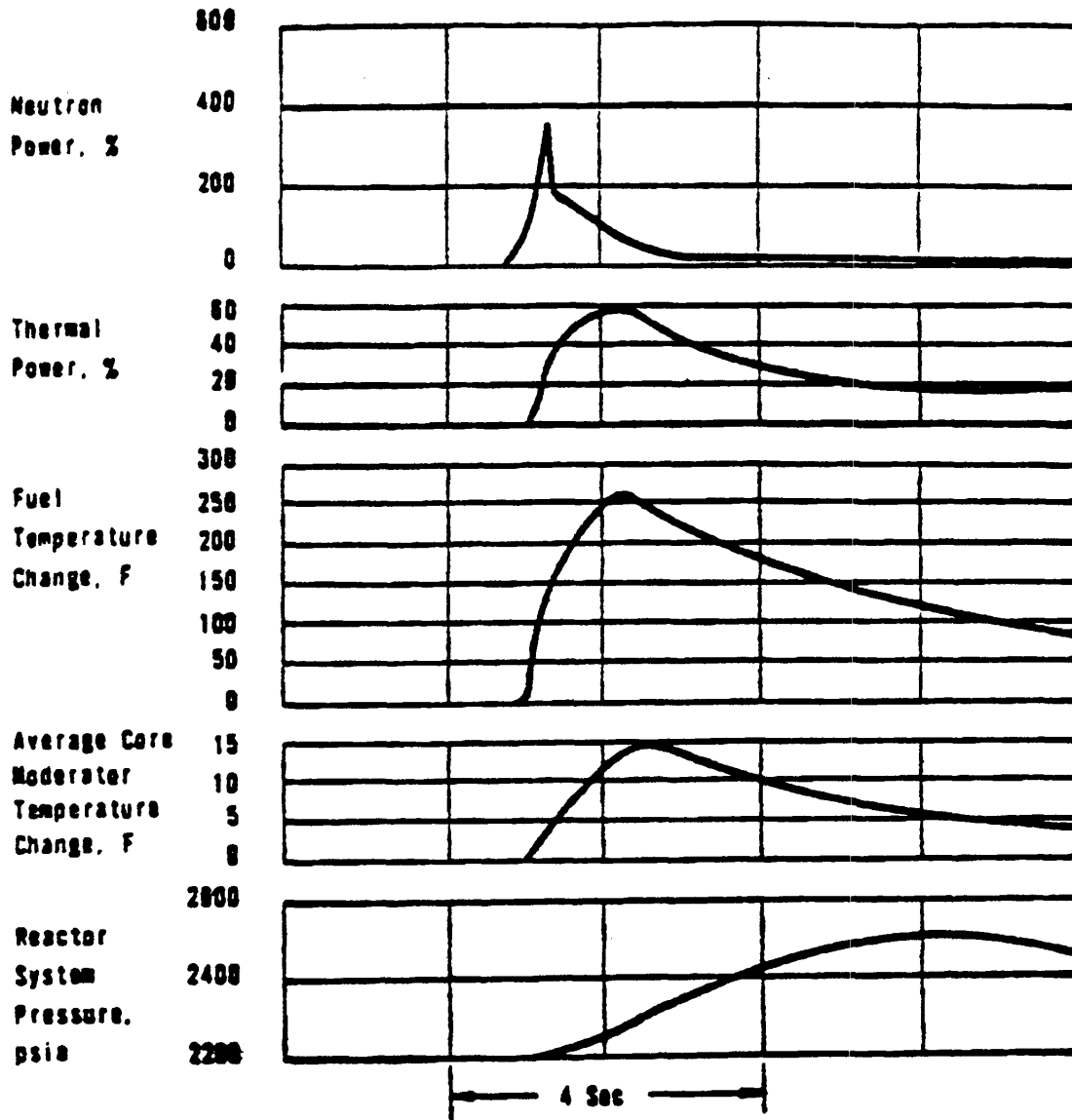
As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. That evaluation concluded that steam generator performance does not impact this accident in any way. Therefore, the existing analyses remain applicable with the replacement Steam Generators installed.





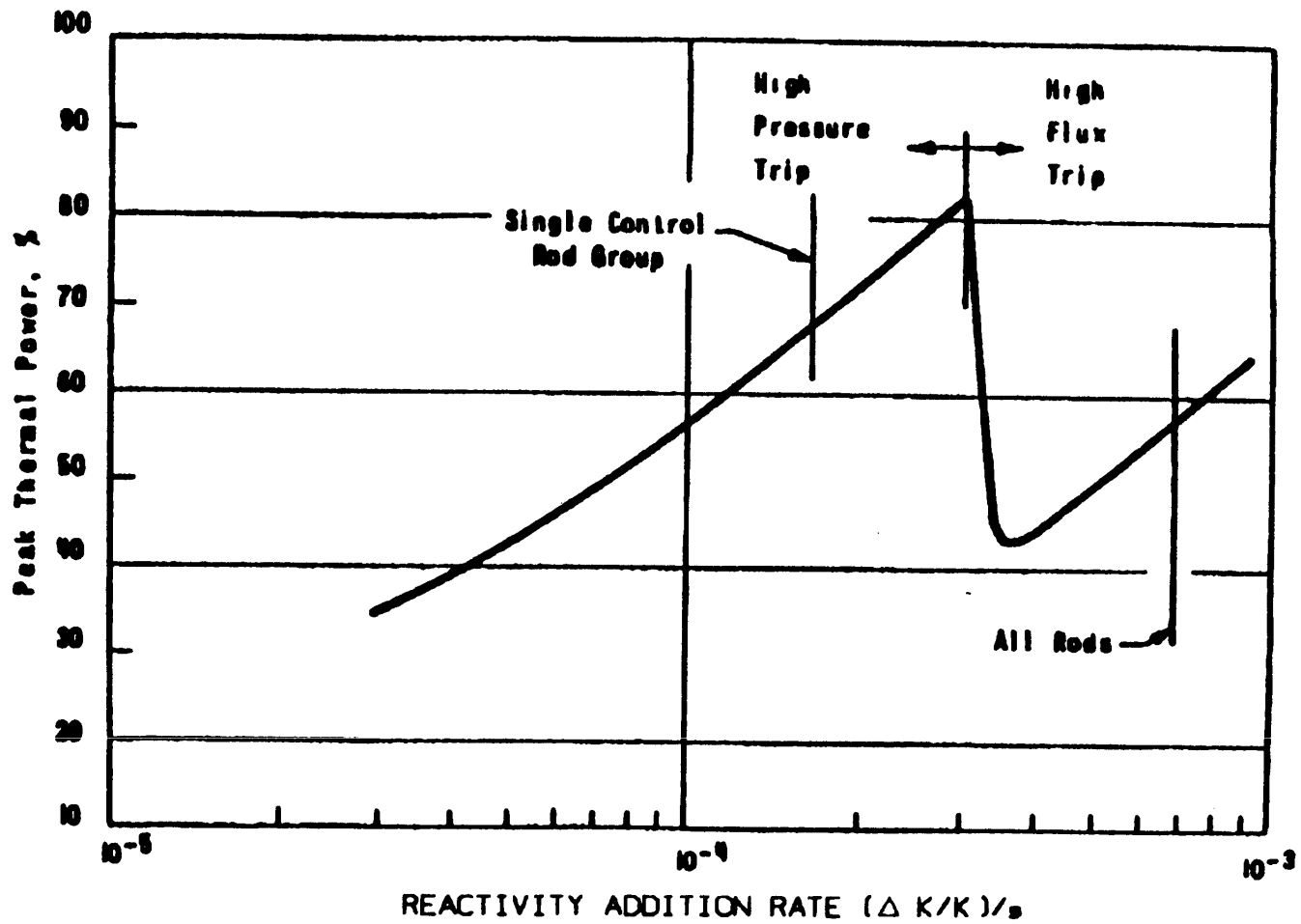
DAVIS-BESSE NUCLEAR POWER STATION  
 STARTUP ACCIDENT FROM  $10^{-9}$  RATED POWER  
 FOR A REACTIVITY ADDITION RATE  
 OF  $1.65 \times 10^{-4}$  ( $\Delta K/K$ )/S;  
 HIGH PRESSURE REACTOR TRIP IS ACTUATED  
 FIGURE 15.2.1-1

REVISION 12  
 JULY 1990



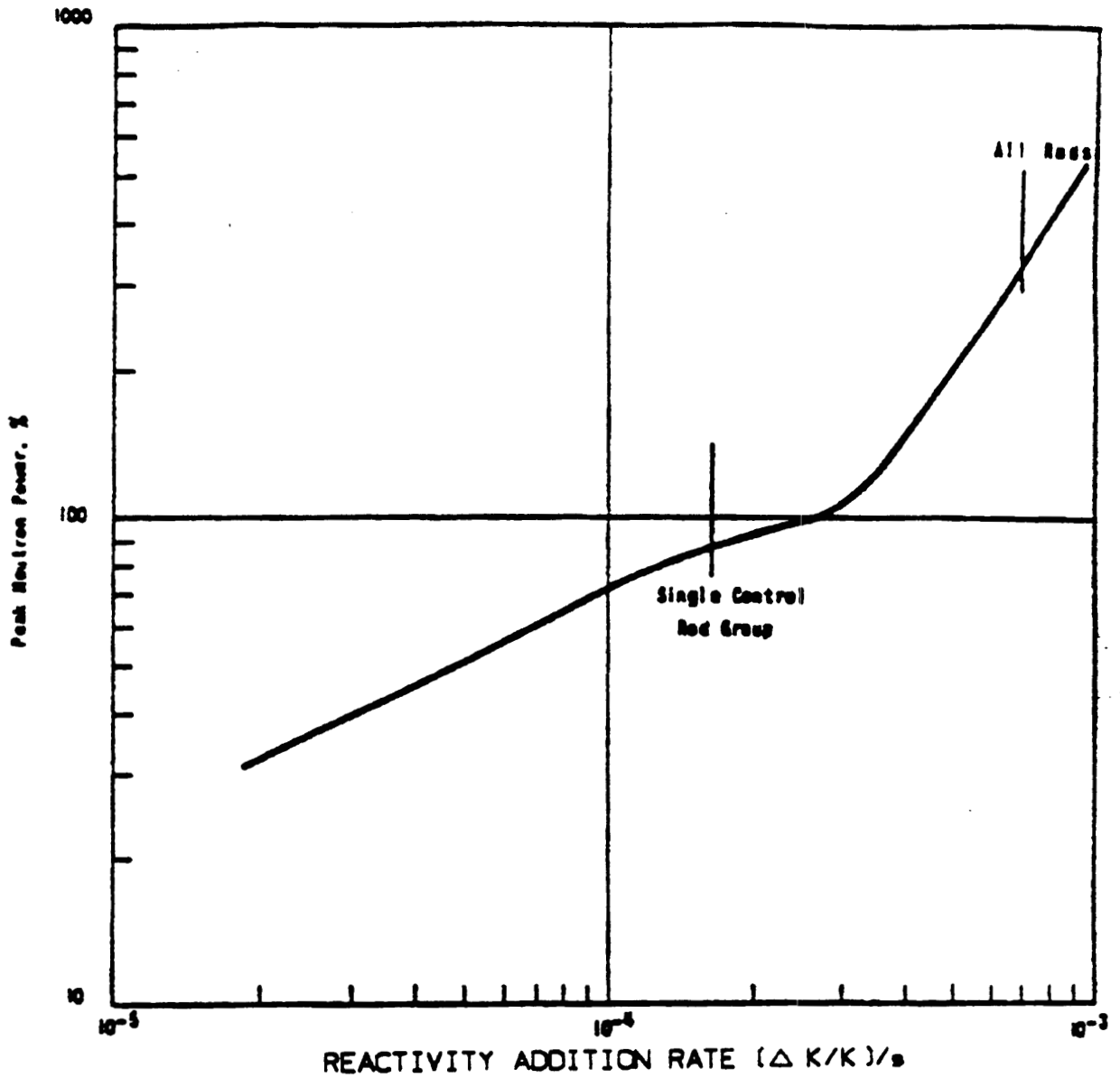
DAVIS-BESSE NUCLEAR POWER STATION  
 STARTUP ACCIDENT FROM 10<sup>-9</sup> RATED POWER  
 FOR A REACTIVITY ADDITION RATE OF  
 $7.19 \times 10^{-4} (\Delta K/K)/s$   
 (SIMULTANEOUS WITHDRAWAL OF ALL CRA'S);  
 HIGH FLUX REACTOR TRIP IS ACTUATED  
 FIGURE 15.2.1-2

REVISION 9  
 JULY 1989



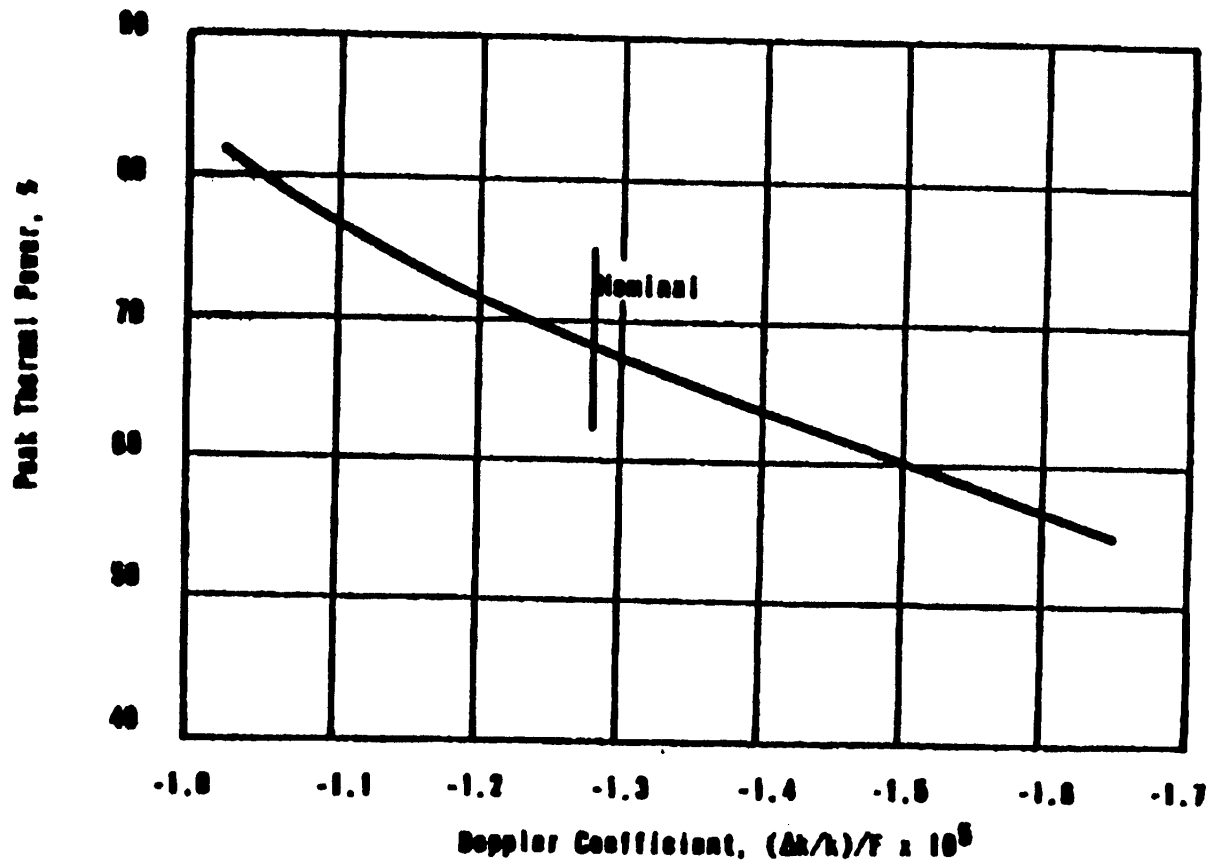
DAVIS-BESSE NUCLEAR POWER STATION  
 PEAK THERMAL POWER VERSUS REACTIVITY ADDITION RATE  
 FOR A STARTUP ACCIDENT FROM 10<sup>-9</sup> RATED POWER  
 FIGURE 15.2.1-3

REVISION 9  
 JULY 1989

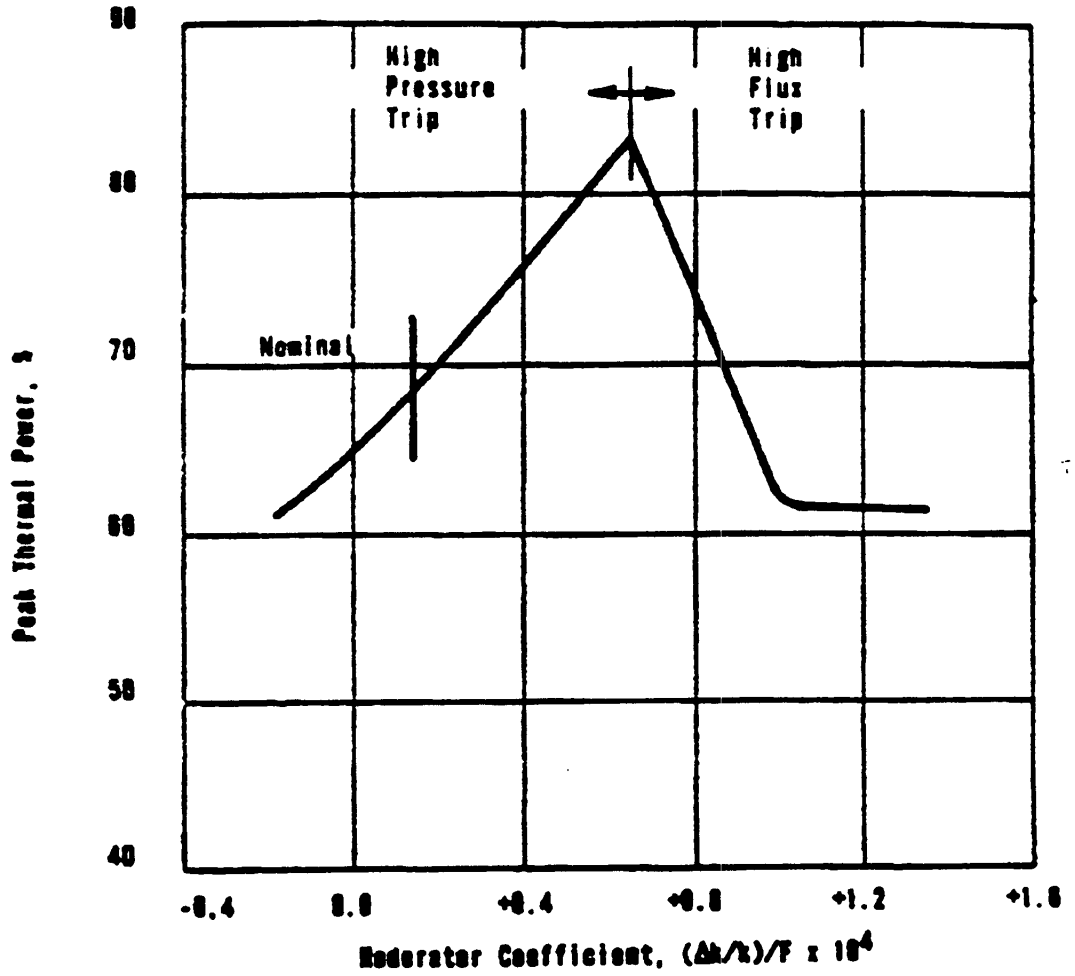


DAVIS-BESSE NUCLEAR POWER STATION  
 PEAK NEUTRON POWER VERSUS REACTIVITY ADDITION RATE  
 FOR A STARTUP ACCIDENT FROM  
 $10^{-9}$  RATED POWER  
 FIGURE 15.2.1-4

REVISION 9  
 JULY 1989

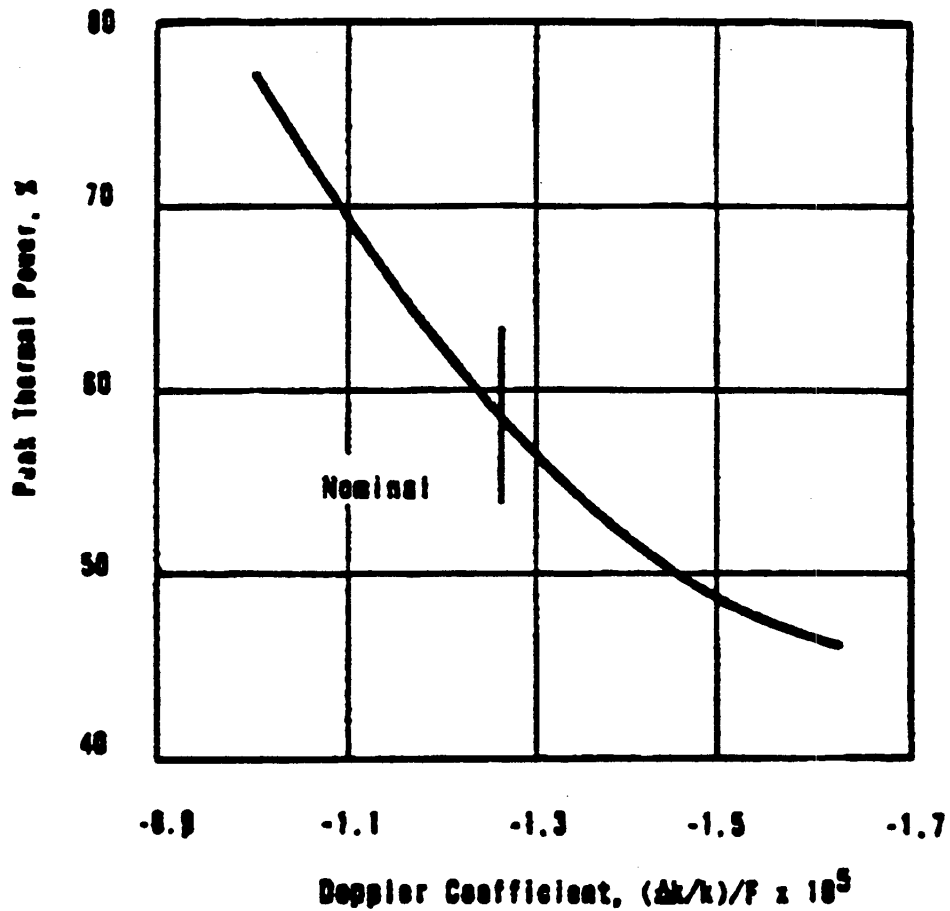


DAVIS-BESSE NUCLEAR POWER STATION  
 PEAK THERMAL POWER VERSUS DOPPLER COEFFICIENT  
 FOR A STARTUP ACCIDENT WITH A CONSTANT  
 REACTIVITY ADDITION RATE OF  $1.65 \times 10^{-4} (\Delta K/K)/s$   
 FROM  $10^{-9}$  RATED POWER  
 FIGURE 15.2.1-5



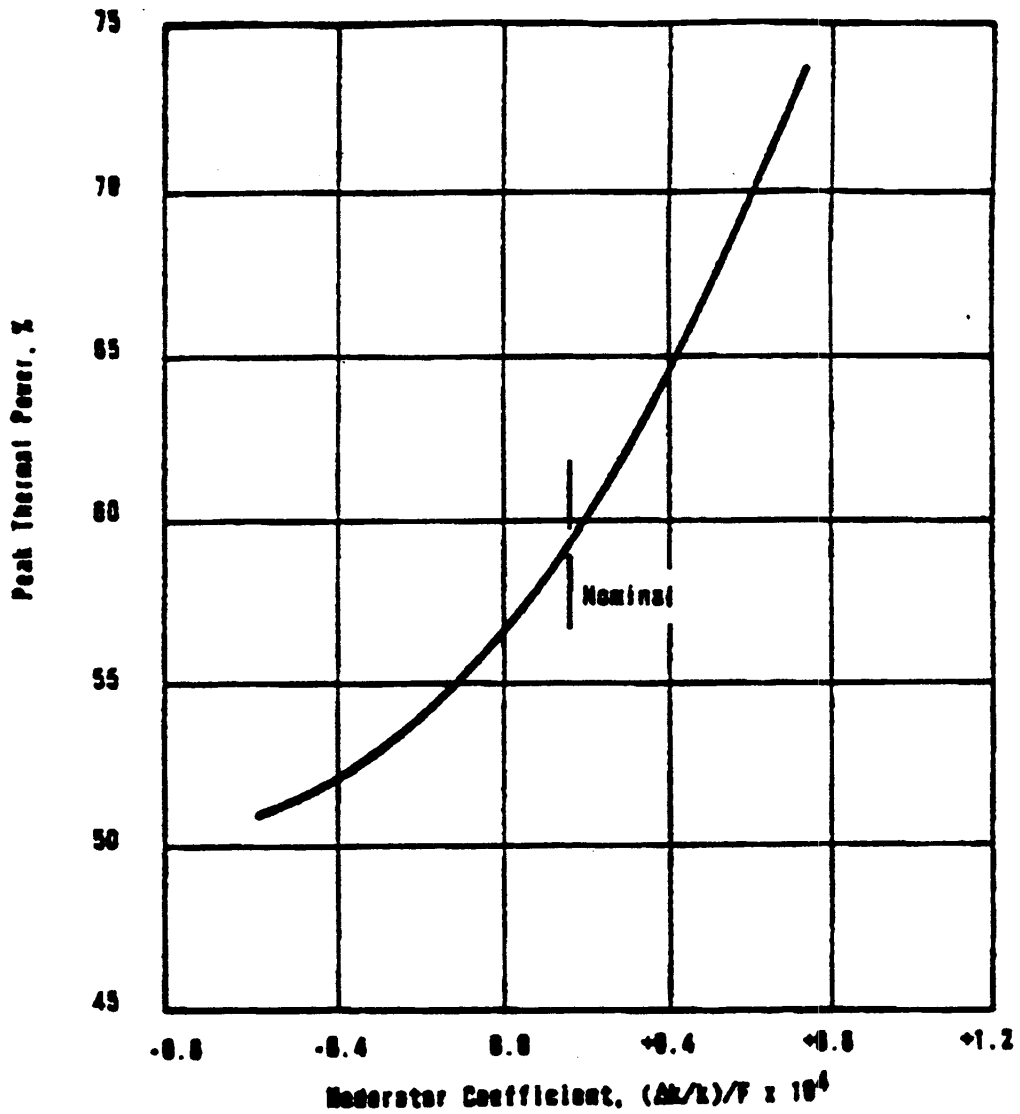
DAVIS-BESSE NUCLEAR POWER STATION  
 PEAK THERMAL VERSUS MODERATOR COEFFICIENT  
 FOR A STARTUP ACCIDENT WITH A CONSTANT  
 REACTIVITY ADDITION RATE OF  
 $1.65 \times 10^{-4} (\Delta k/k)/s$  FROM  $10^{-9}$  RATED POWER  
 FIGURE 15.2.1-6

REVISION 9  
 JULY 1989



DAVIS-BESSE NUCLEAR POWER STATION  
 PEAK THERMAL POWER VERSUS DOPPLER COEFFICIENT  
 FOR A STARTUP ACCIDENT WITH A REACTIVITY  
 ADDITION RATE OF  $7.19 \times 10^{-4}$   $(\Delta K/K)/s$ ,  
 (SIMULTANEOUS WITHDRAWAL OF ALL CRA'S):  
 FROM  $10^{-9}$  RATED POWER  
 FIGURE 15.2.1-7

REVISION 9  
 JULY 1989



DAVIS-BESSE NUCLEAR POWER STATION  
 PEAK THERMAL POWER VERSUS MODERATOR COEFFICIENT  
 FOR A STARTUP ACCIDENT WITH A REACTIVITY ADDITION  
 RATE OF  $7.19 \times 10^{-4} (\Delta K/K)/s$  (SIMULTANEOUS WITHDRAWAL  
 OF ALL CRA'S) FROM  $10^{-9}$  RATED POWER  
 FIGURE 15.2.1-8

REVISION 9  
 JULY 1989



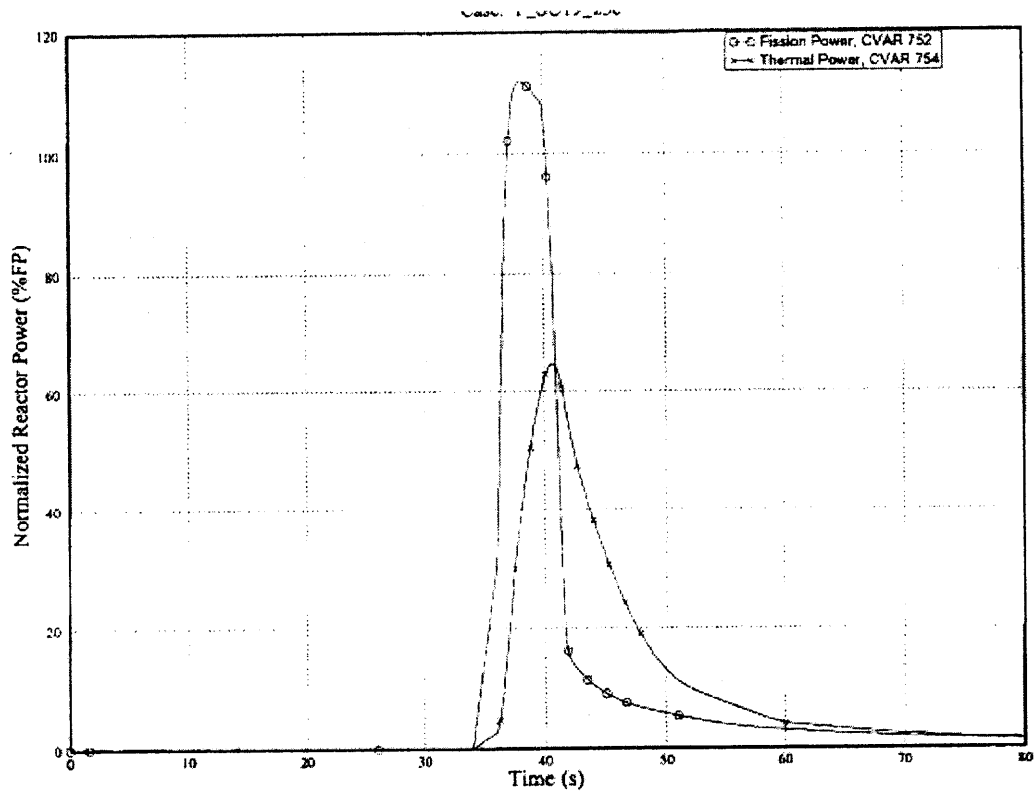


Figure 15.2.1-9

Power vs. Time – Worst Case Startup Accident

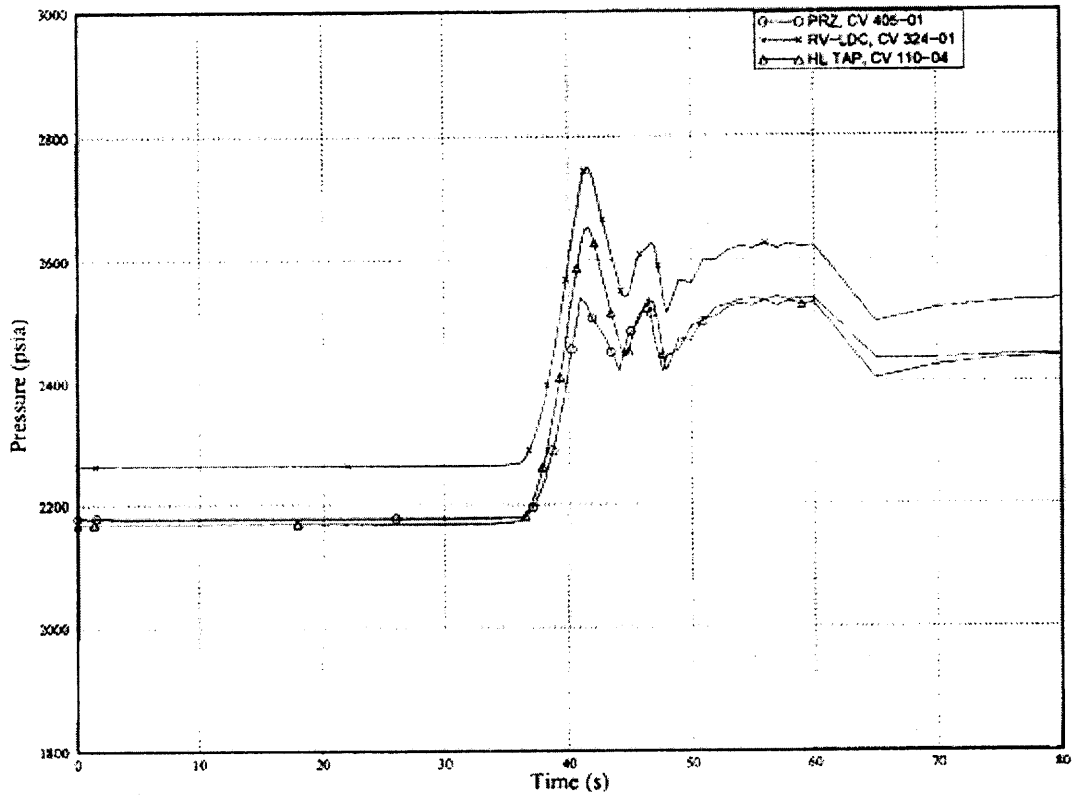
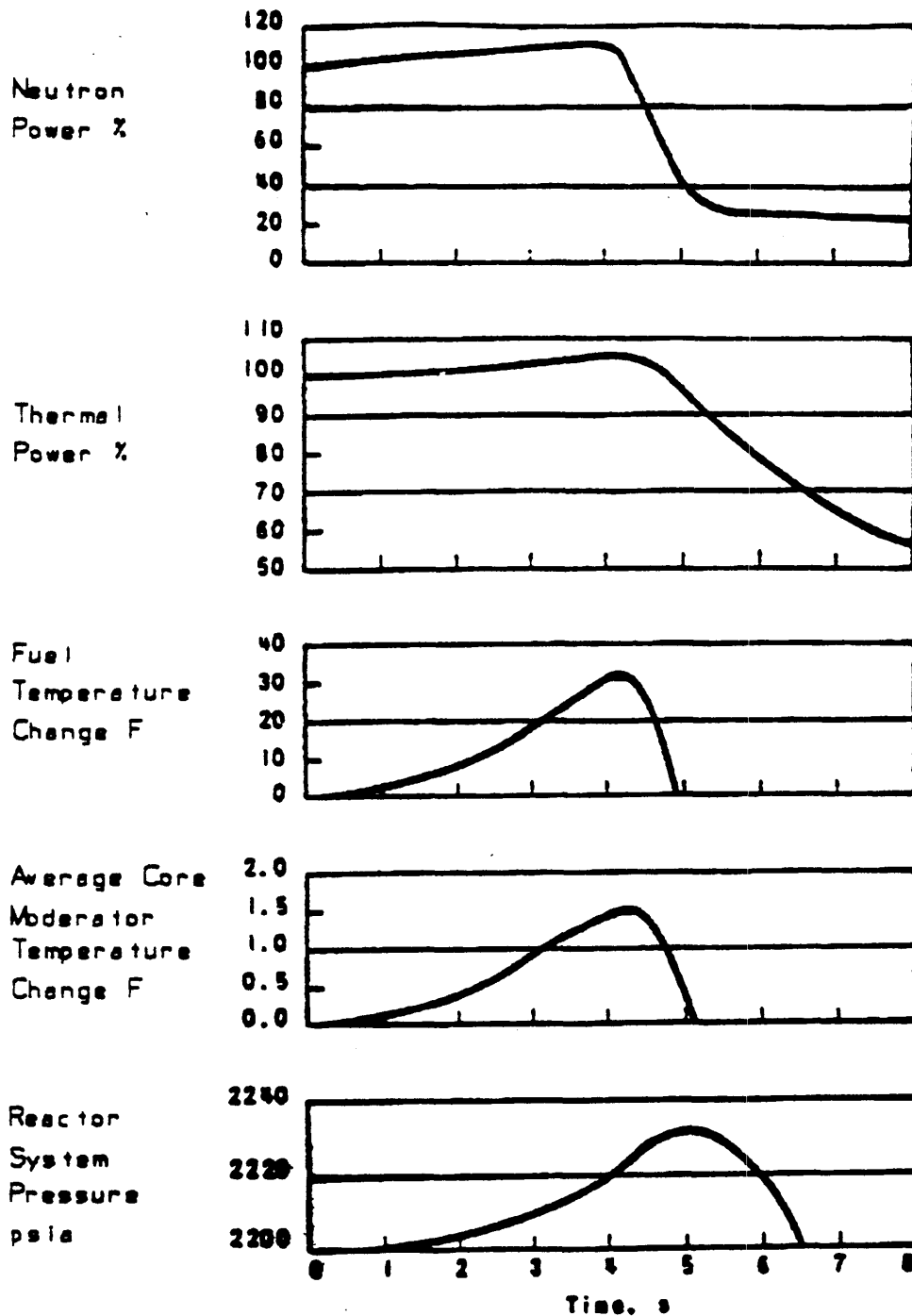


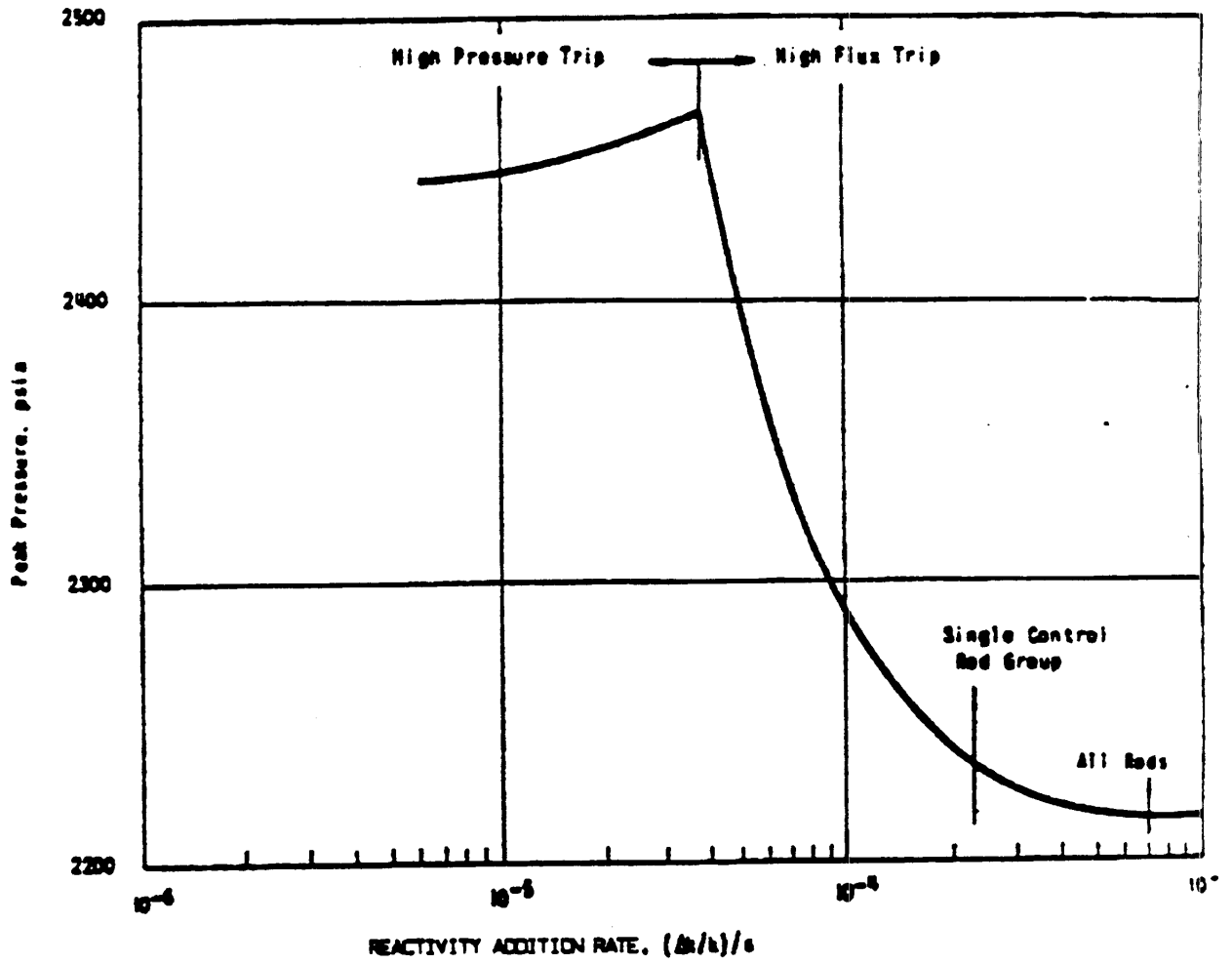
Figure 15.2.1-10

Pressure vs. Time – Worst Case Startup Accident



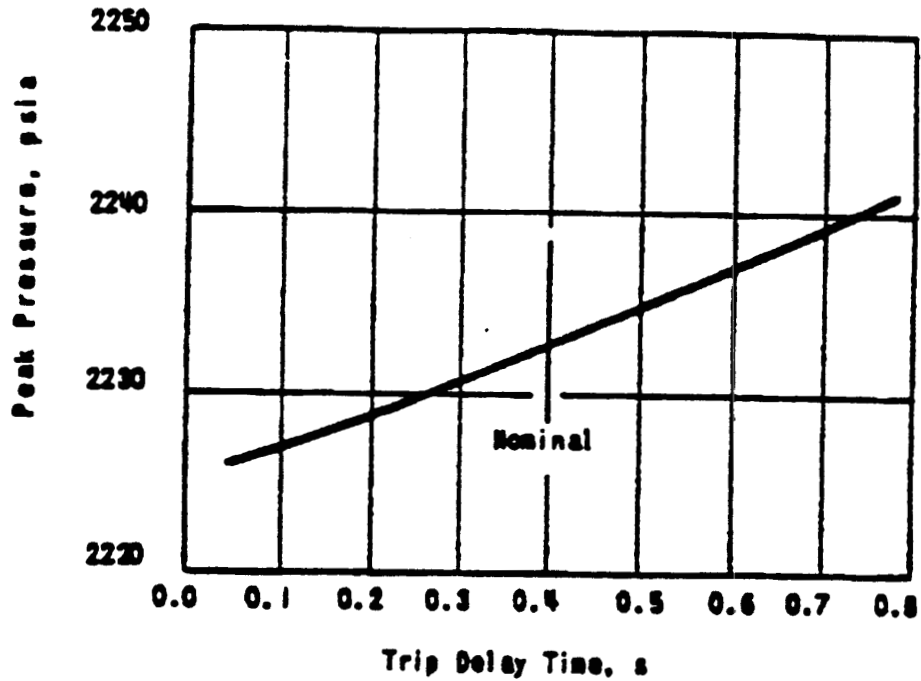
DAVIS-BESSE NUCLEAR POWER STATION  
 CRA WITHDRAWAL ACCIDENT FROM RATED POWER  
 FOR A REACTIVITY ADDITION RATE OF  $2.3 \times 10^{-4}$  ( $\Delta K/K$ )/s;  
 HIGH FLUX REACTOR TRIP IS ACTUATED  
 FIGURE 15.2.2-1

REVISION 9  
 JULY 1989



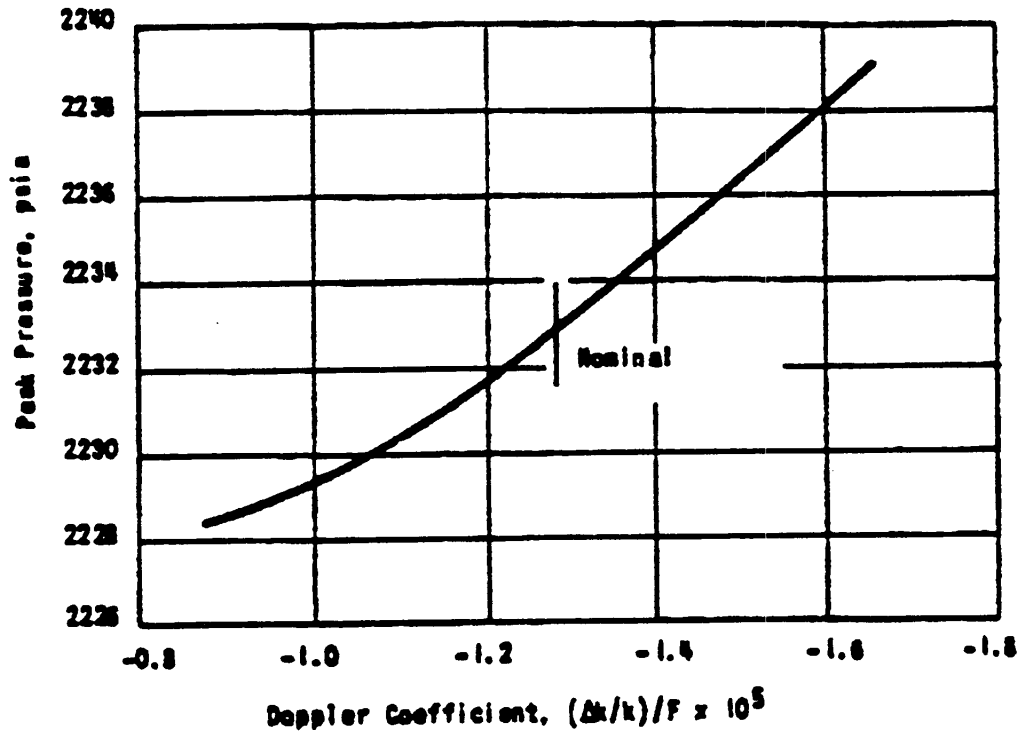
DAVIS-BESSE NUCLEAR POWER STATION  
 PEAK PRESSURE VERSUS REACTIVITY ADDITION RATE FOR  
 A CRA WITHDRAWAL ACCIDENT FROM RATED POWER  
 FIGURE 15.2.2-2

REVISION 9  
 JULY 1989



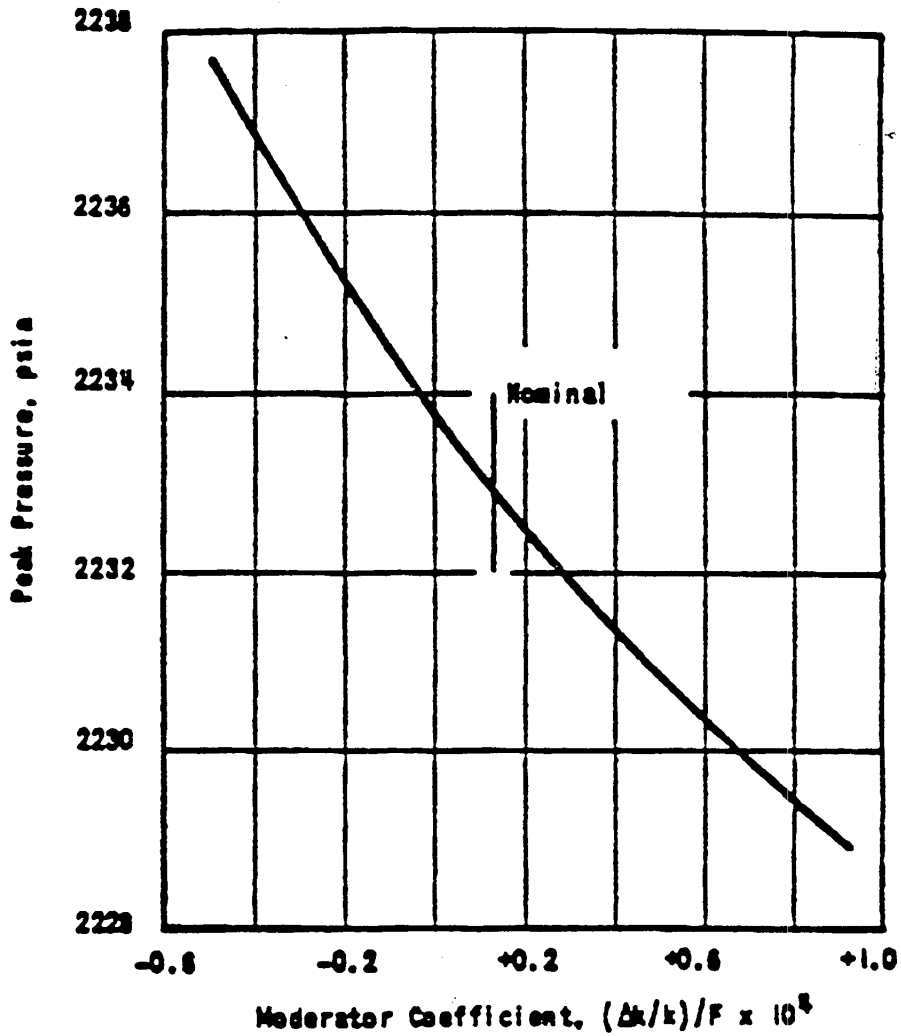
DAVIS-BESSE NUCLEAR POWER STATION  
 PEAK PRESSURE VERSUS TRIP DELAY TIME FOR A CRA  
 WITHDRAWAL ACCIDENT FROM RATED POWER  
 WITH A CONSTANT REACTIVITY ADDITION RATE OF  
 $2.3 \times 10^{-4} (\Delta K/K)/s$   
 FIGURE 15.2.2-3

REVISION 9  
 JULY 1989



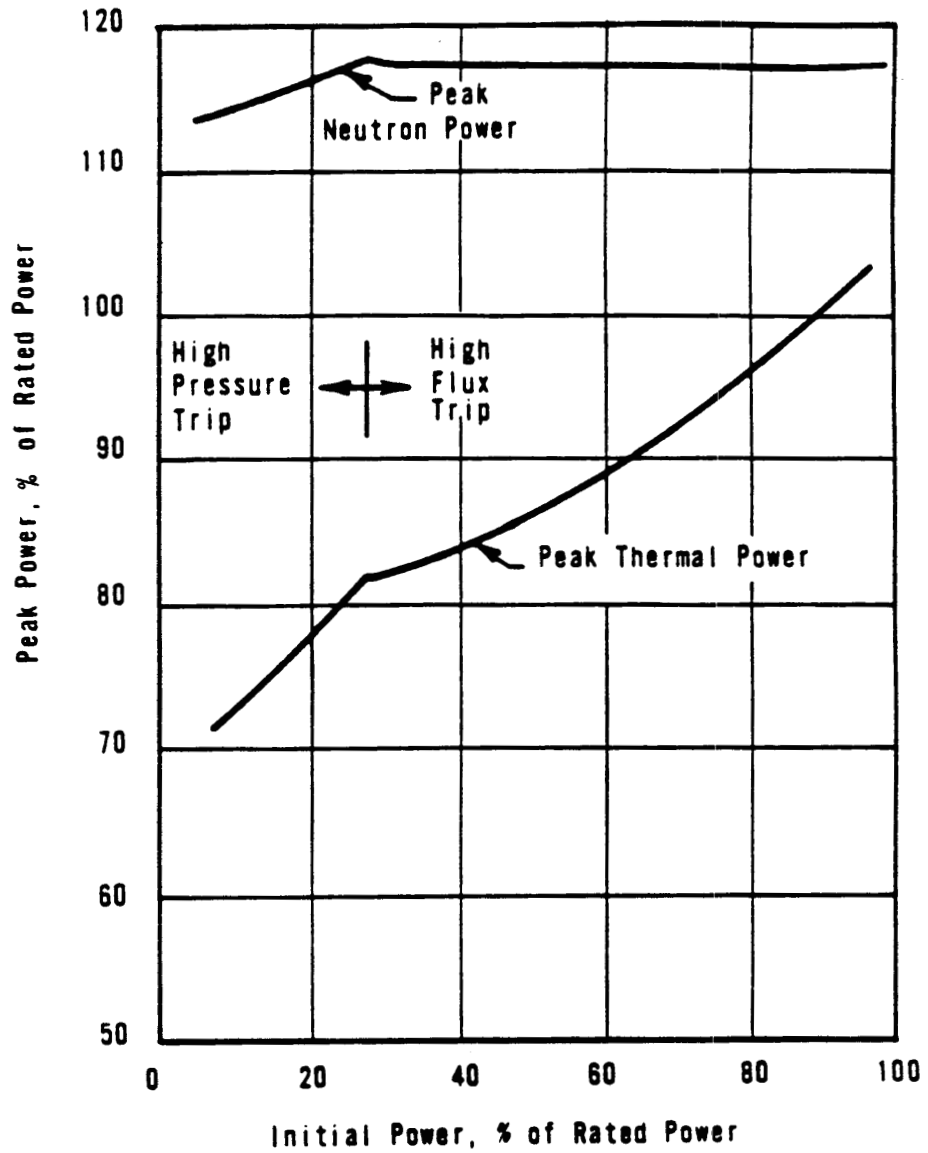
DAVIS-BESSE NUCLEAR POWER STATION  
 PEAK PRESSURE VERSUS DOPPLER COEFFICIENT  
 FOR A CRA WITHDRAWAL ACCIDENT FROM  
 RATED POWER WITH A CONSTANT REACTIVITY  
 ADDITION RATE OF  $2.3 \times 10^{-4} (\Delta K/K)/s$   
 FIGURE 15.2.2-4

REVISION 9  
 JULY 1989



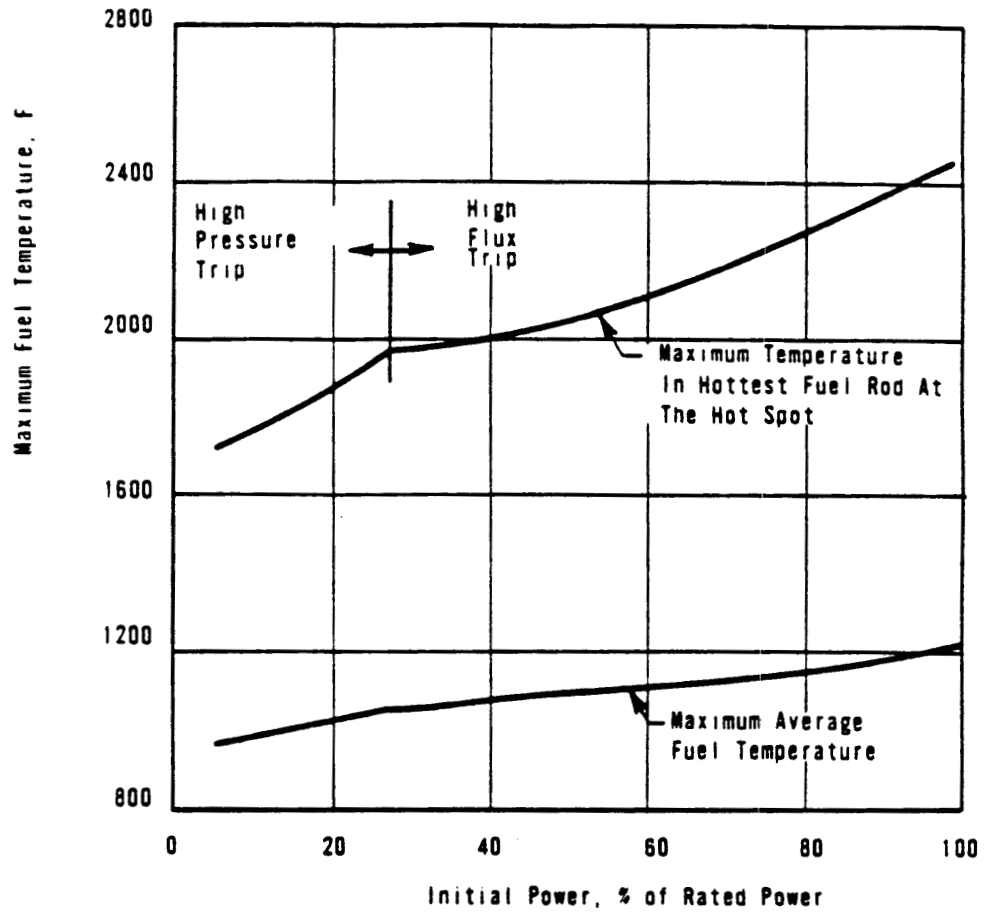
DAVIS-BESSE NUCLEAR POWER STATION  
 PEAK PRESSURE VERSUS MODERATOR COEFFICIENT  
 FOR A CRA WITHDRAWAL ACCIDENT FROM RATED  
 POWER WITH A CONSTANT REACTIVITY ADDITION RATE  
 OF  $2.3 \times 10^{-4} (\Delta K/K)/s$   
 FIGURE 15.2.2-5

REVISION 12  
 JULY 1990

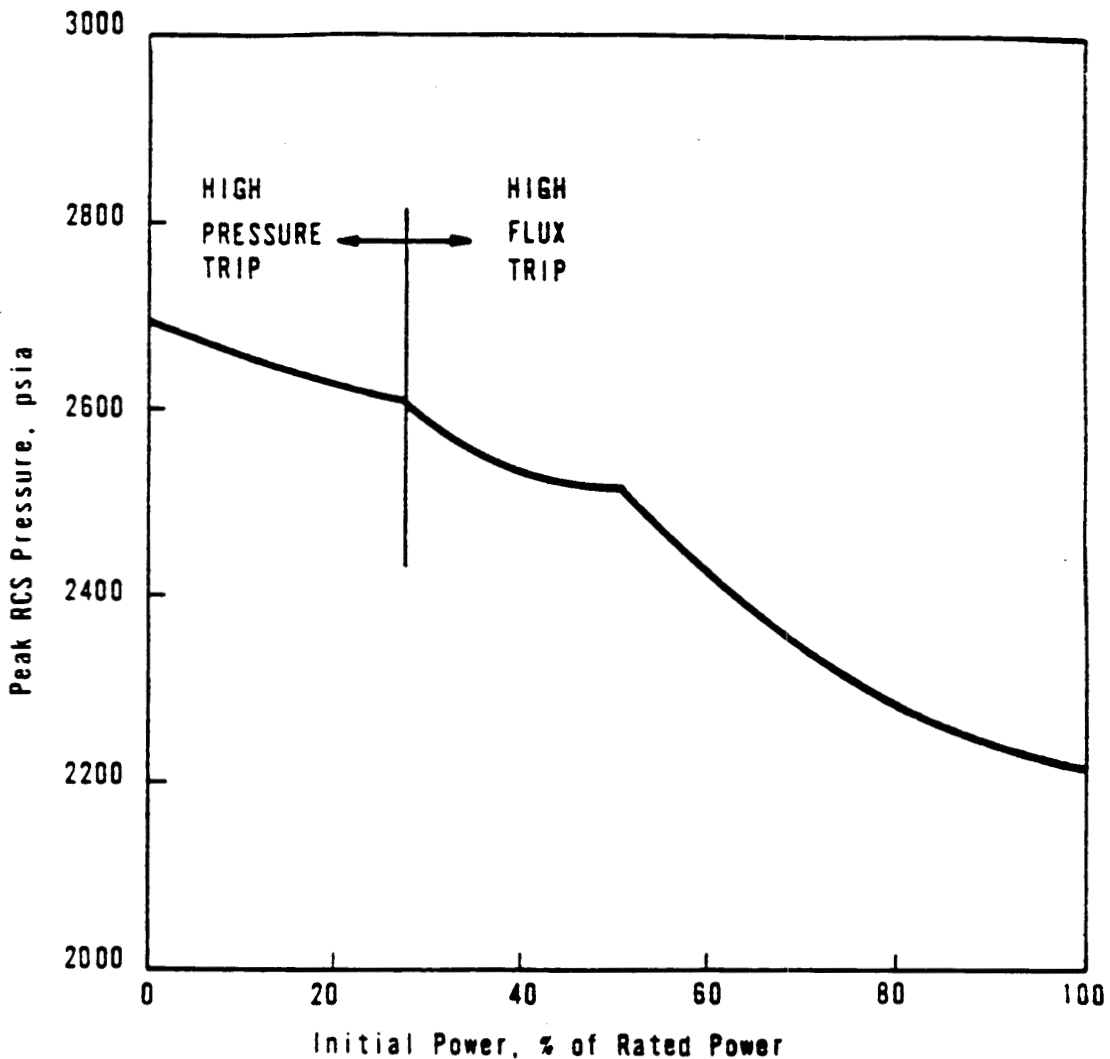


**DAVIS-BESSE NUCLEAR POWER STATION  
 MAXIMUM NEUTRON AND THERMAL POWER FOR AN  
 ALL-CRA WITHDRAWAL ACCIDENT FROM  
 VARIOUS INITIAL POWER LEVELS  
 FIGURE 15.2.2-6**





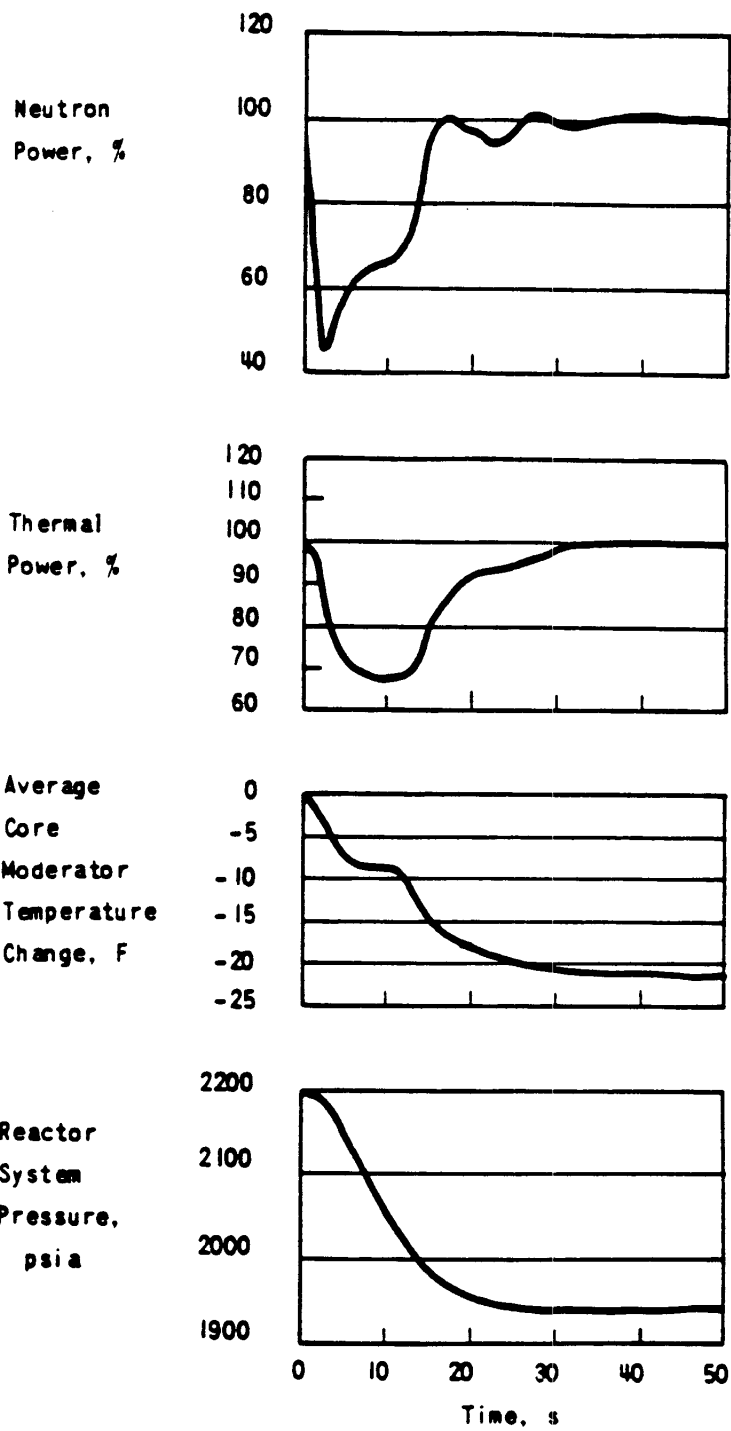
**DAVIS-BESSE NUCLEAR POWER STATION  
 PEAK FUEL TEMPERATURE IN AVERAGE ROD AND HOT  
 SPOT FOR AN ALL-CRA WITHDRAWAL ACCIDENT  
 FROM VARIOUS INITIAL POWER LEVELS  
 FIGURE 15.2.2-7**



DAVIS-BESSE NUCLEAR POWER STATION  
 PEAK REACTOR COOLANT PRESSURE  
 VERSUS POWER FOR ALL CRA GROUP  
 WITHDRAWAL

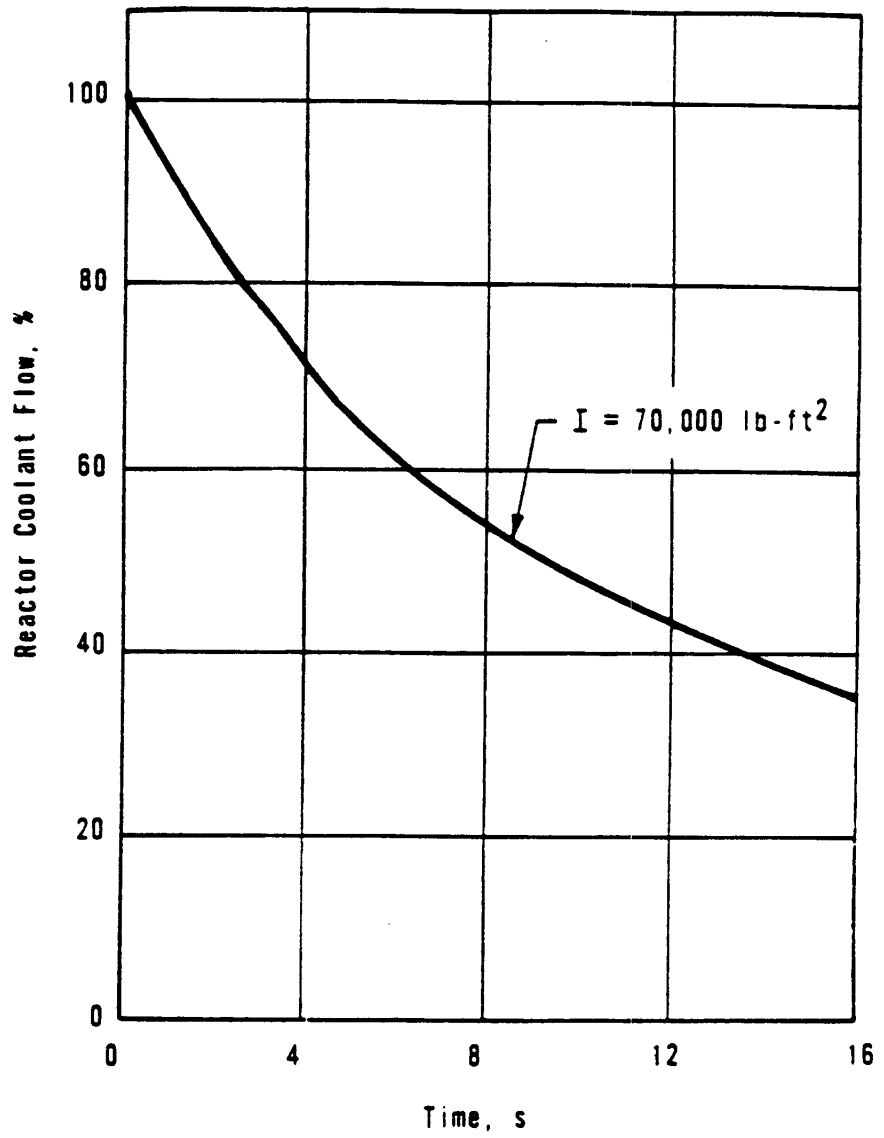
FIGURE 15.2.2-8

REVISION 0  
 JULY 1982



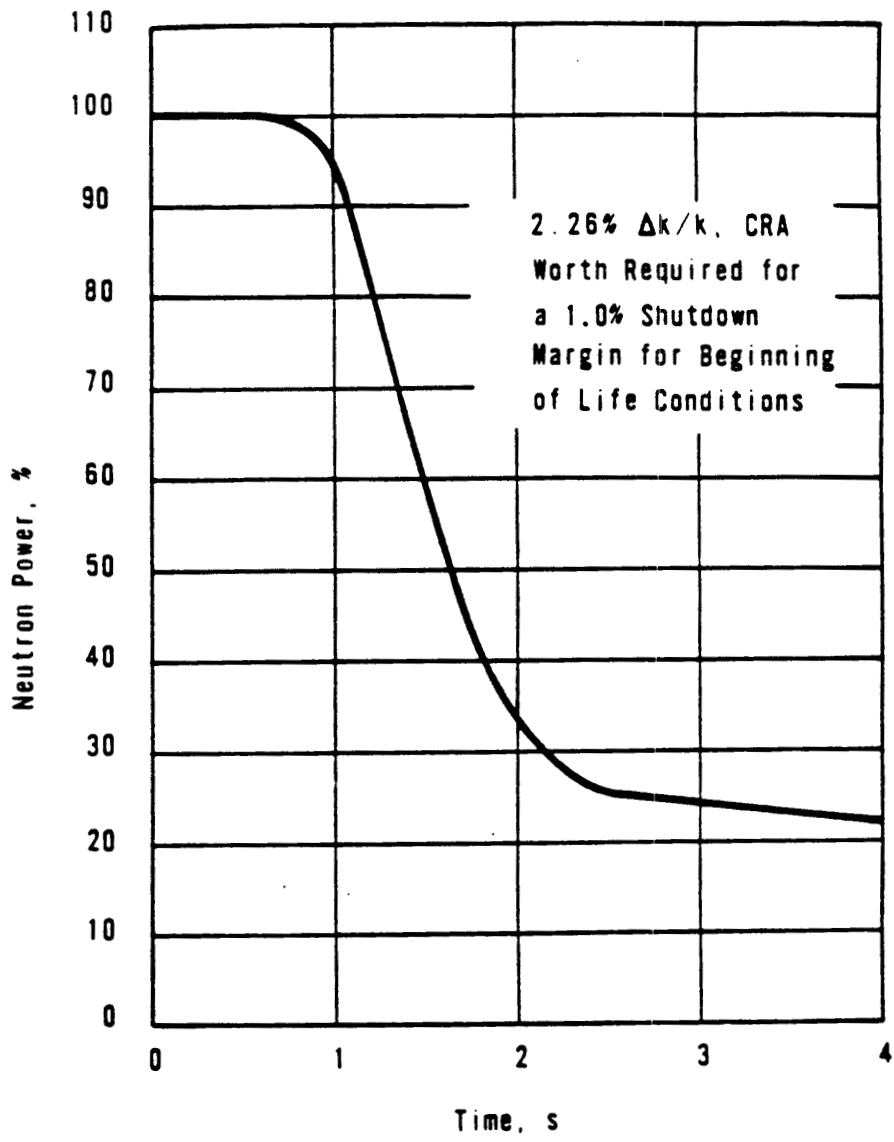
DAVIS-BESSE NUCLEAR POWER STATION  
 0.65%  $\Delta k/k$  CRA DROP FROM RATED  
 POWER AT EOL CONDITION  
 FIGURE 15.2.3-1

REVISION 0  
 JULY 1982



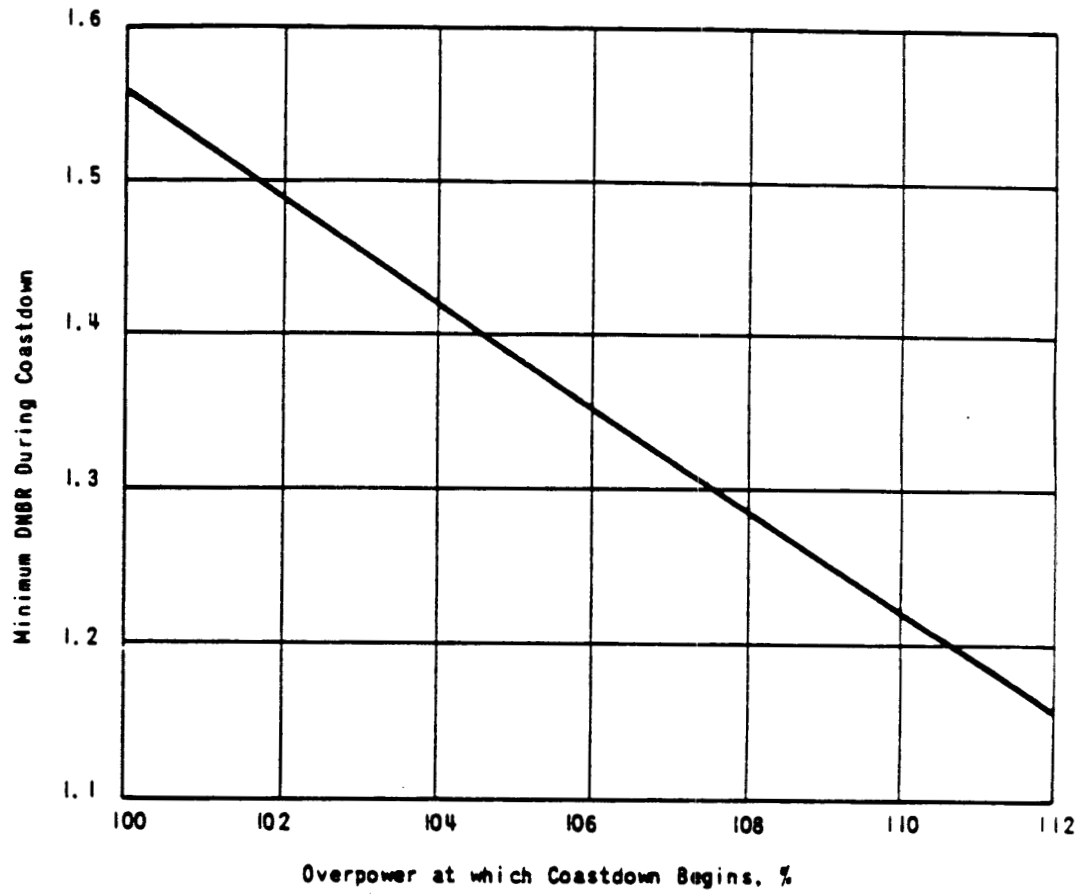
DAVIS-BESSE NUCLEAR POWER STATION  
PERCENT REACTOR COOLANT FLOW AS A FUNCTION  
OF TIME AFTER LOSS OF PUMP POWER  
FIGURE 15.2.5-1

REVISION 0  
JULY 1982



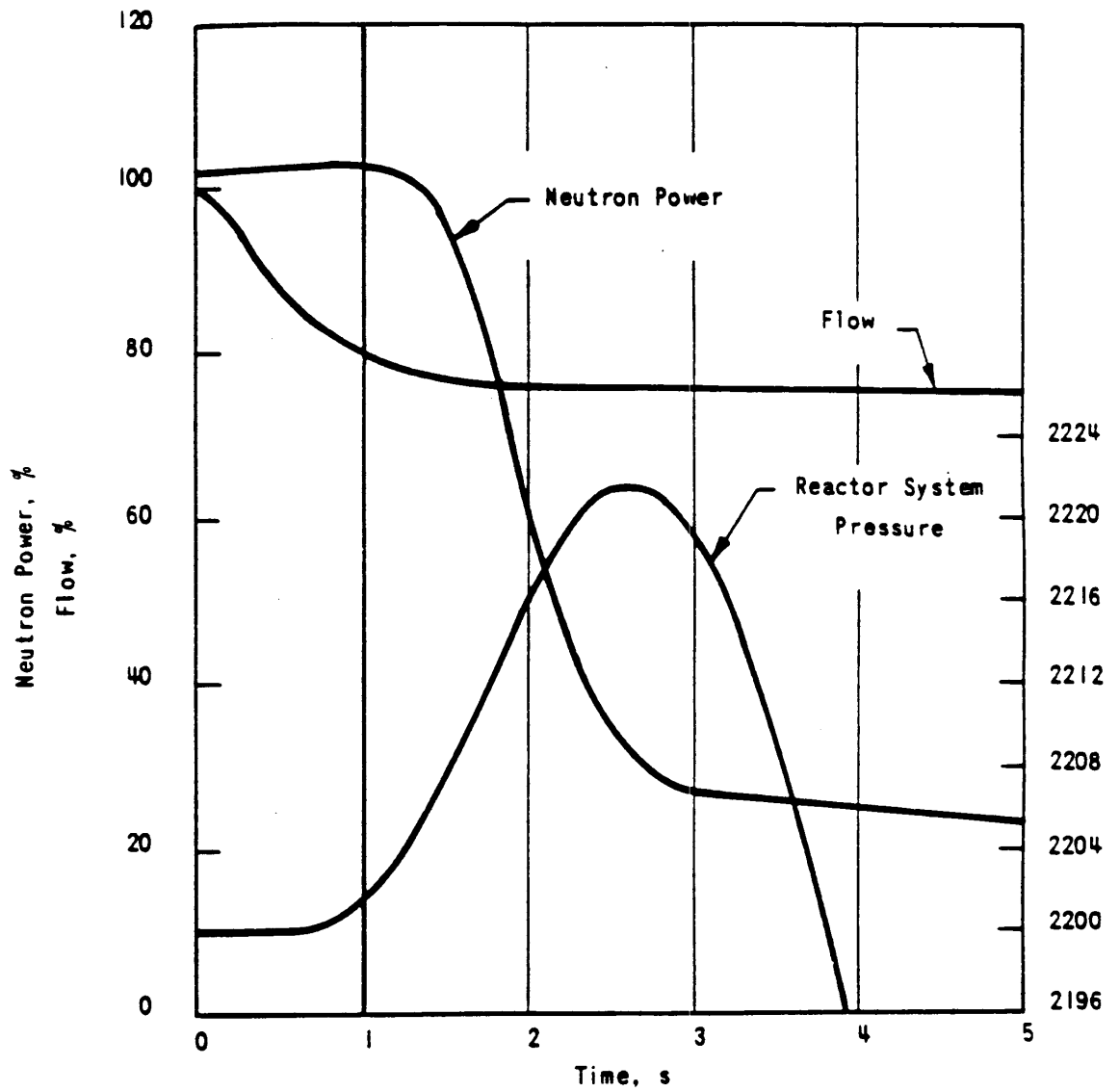
DAVIS-BESSE NUCLEAR POWER STATION  
 PERCENT NEUTRON POWER VERSUS TIME  
 FOLLOWING REACTOR TRIP  
 FIGURE 15.2.5-2

REVISION 0  
 JULY 1982



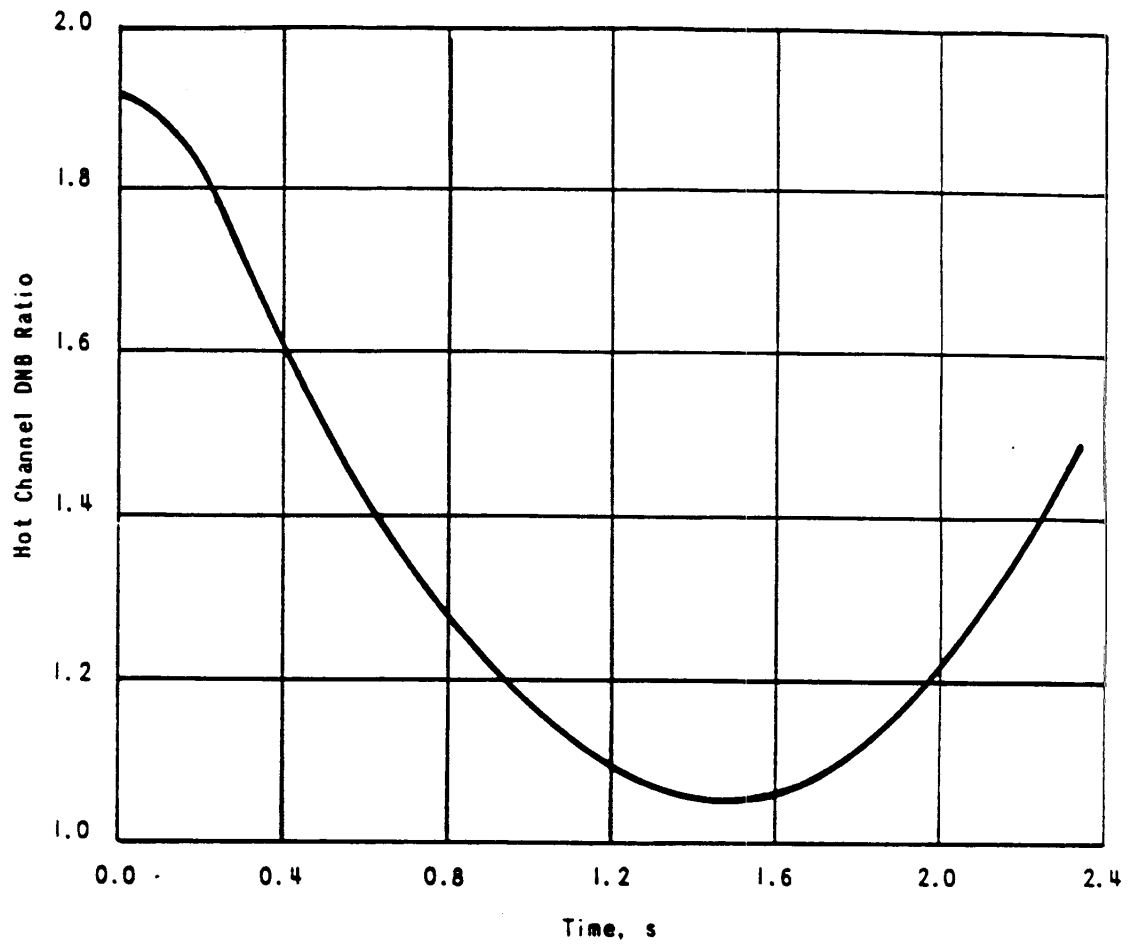
DAVIS-BESSE NUCLEAR POWER STATION  
 MINIMUM DNBR WHICH OCCURS DURING A FOUR PUMP  
 COASTDOWN FROM VARIOUS INITIAL  
 POWER LEVELS  
 FIGURE 15.2.5-3

REVISION 0  
 JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION  
 NEUTRON POWER, FLOW, AND REACTOR SYSTEM  
 PRESSURE FOR A LOCKED ROTOR ACCIDENT,  
 BOL PARAMETERS  
 FIGURE 15.2.5-4

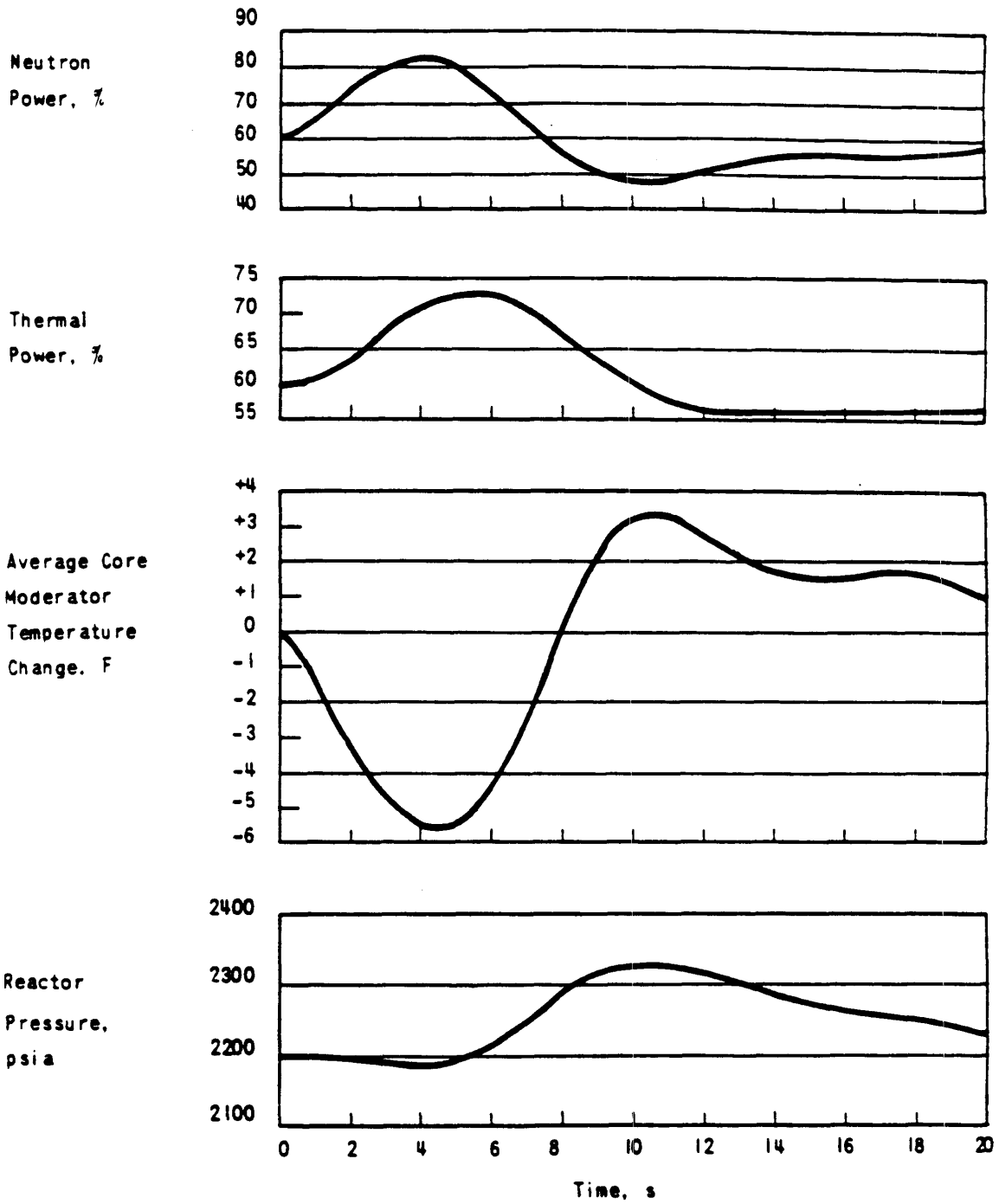
REVISION 0  
 JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION  
DNB RATIO VERSUS TIME FOR LOCKED ROTOR  
ACCIDENT FROM 102% OF RATED POWER  
FIGURE 15.2.5-5

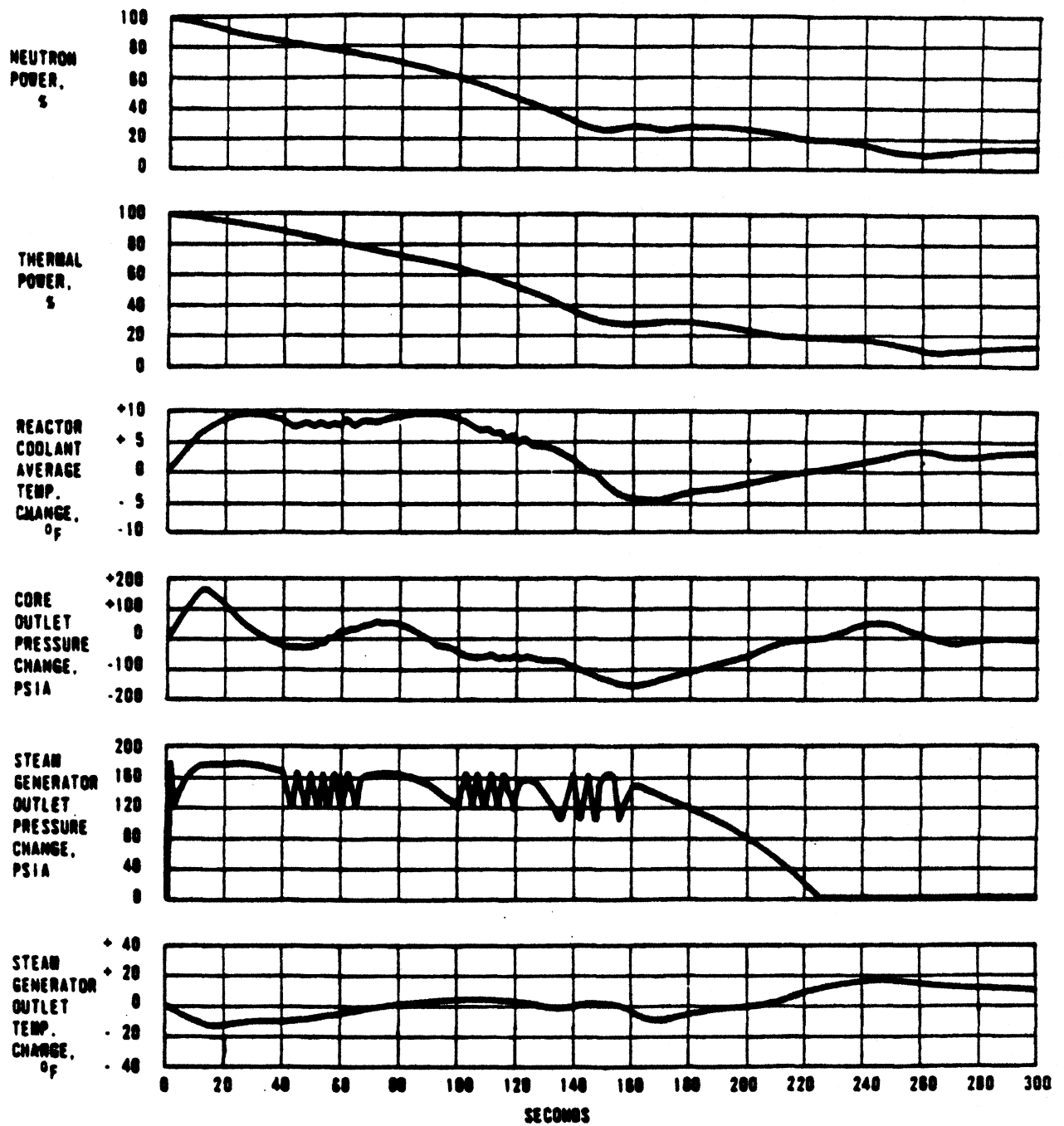
REVISION 0  
JULY 1982





DAVIS-BESSE NUCLEAR POWER STATION  
 TWO PUMP STARTUP FROM 60% POWER AND 49% FLOW  
 FIGURE 15.2.6-1

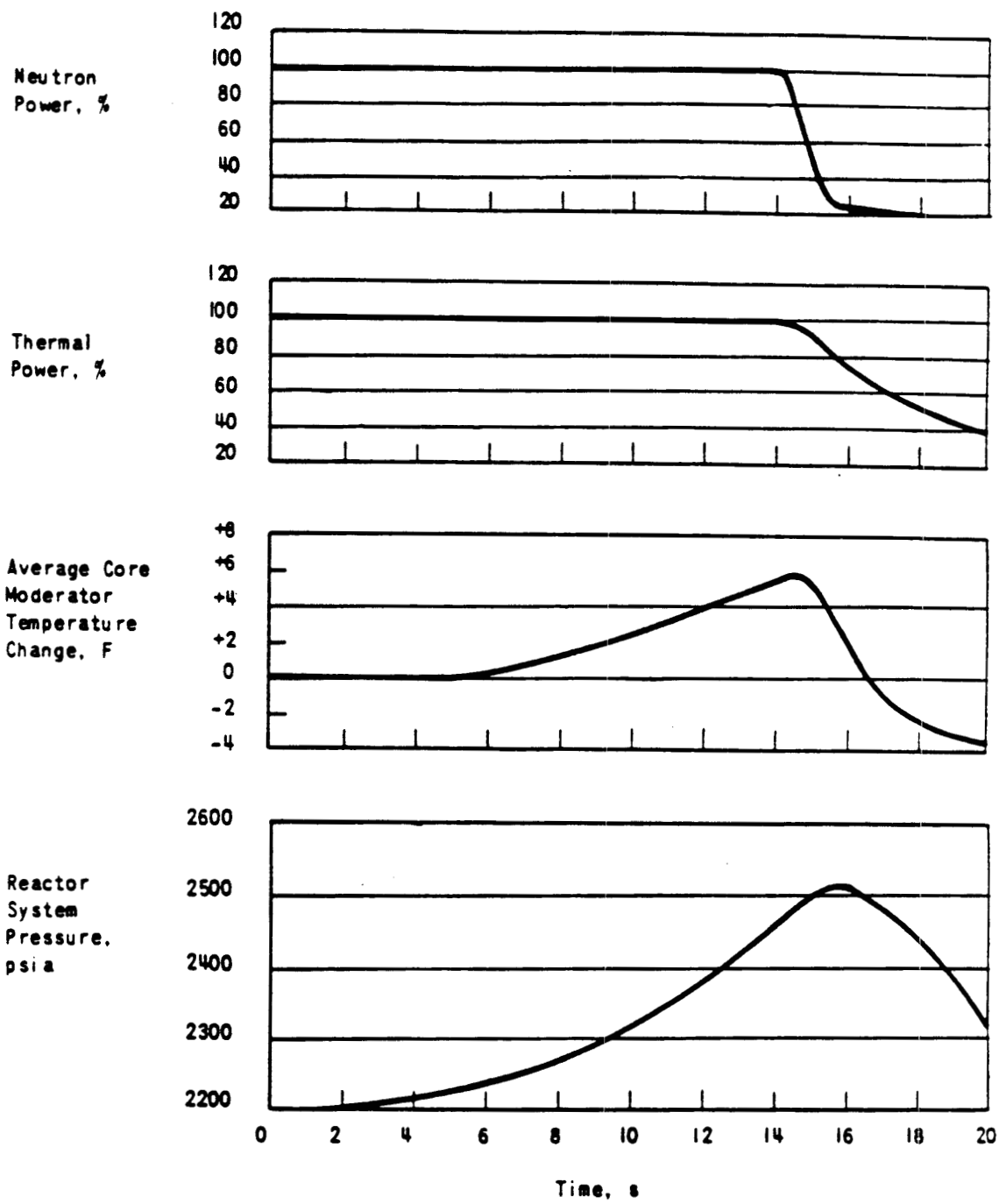
REVISION 0  
 JULY 1982



Note: Figure 15.2.7-1 represents the original design of the plant. This is the ICS runback to the steam generator low level limit. Analysis for the turbine trip when the runback is not successful is given in Reference 67.

**DAVIS-BESSE NUCLEAR POWER STATION  
LOSS OF EXTERNAL ELECTRICAL LOAD  
AT RATED POWER WITH AUTOMATIC  
POWER RUN BACK  
FIGURE 15.2.7-1**

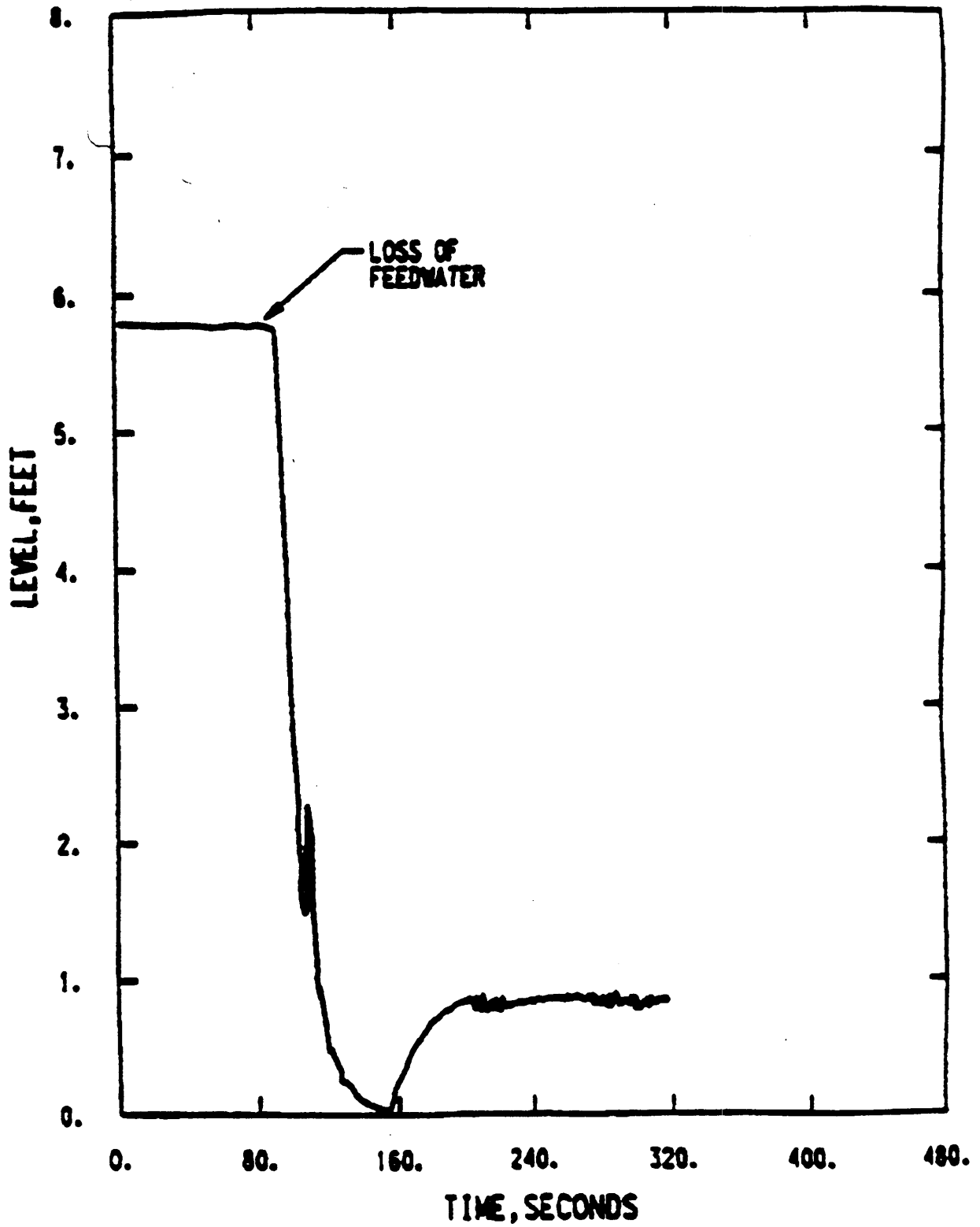
Revision 30  
October 2014



DAVIS-BESSE NUCLEAR POWER STATION  
 LOSS OF ALL FEEDWATER FROM RATED POWER

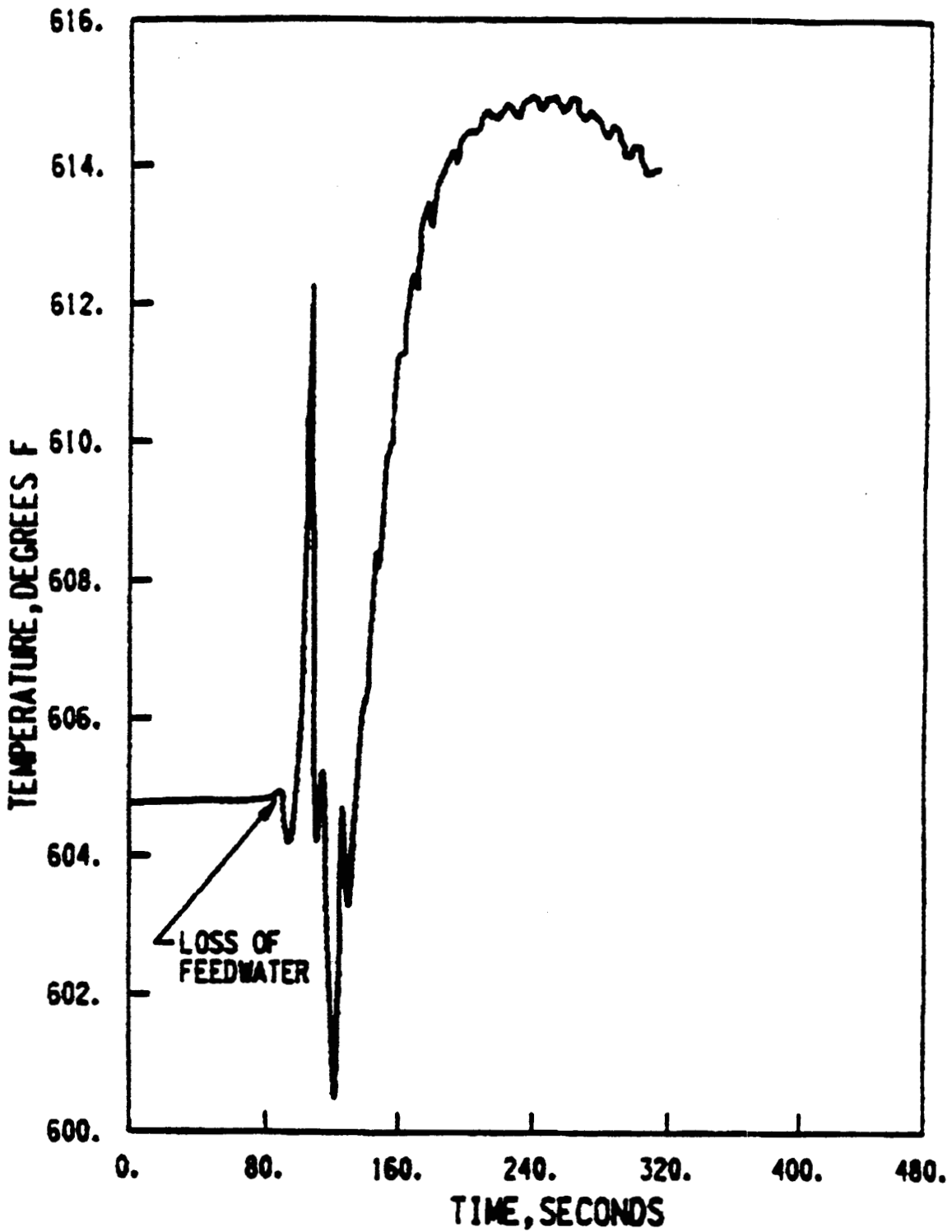
FIGURE 15.2.8-1

REVISION 0  
 JULY 1982



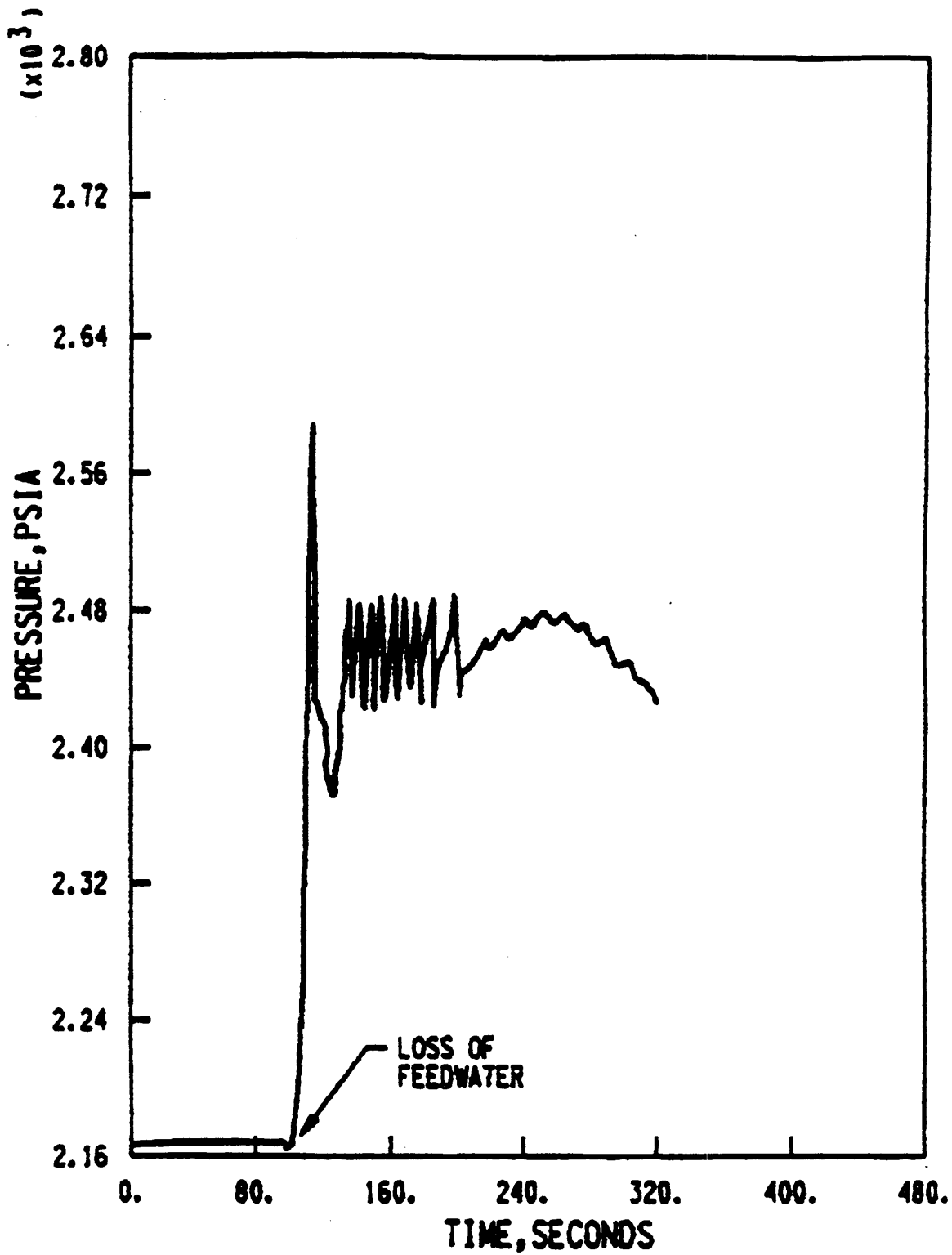
DAVIS-BESSE NUCLEAR POWER STATION  
STEAM GENERATOR COLLAPSED LEVEL (LOOP 1)  
FIGURE 15.2.8-1A

REVISION 9  
JULY 1989



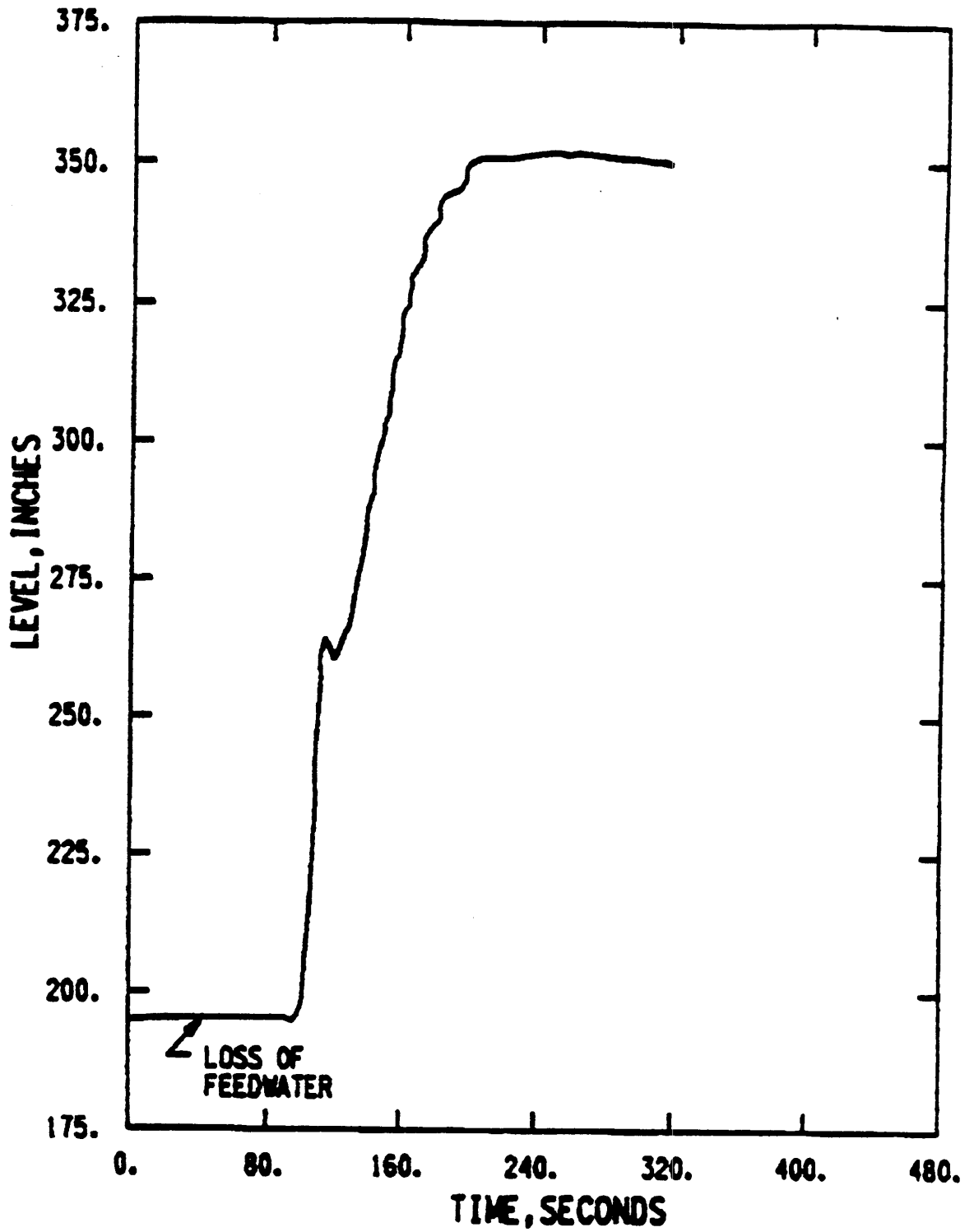
DAVIS-BESSE NUCLEAR POWER STATION  
HOT LEG TEMPERATURE (LOOP 1)  
FIGURE 15.2.8-1B

REVISION 9  
JULY 1989



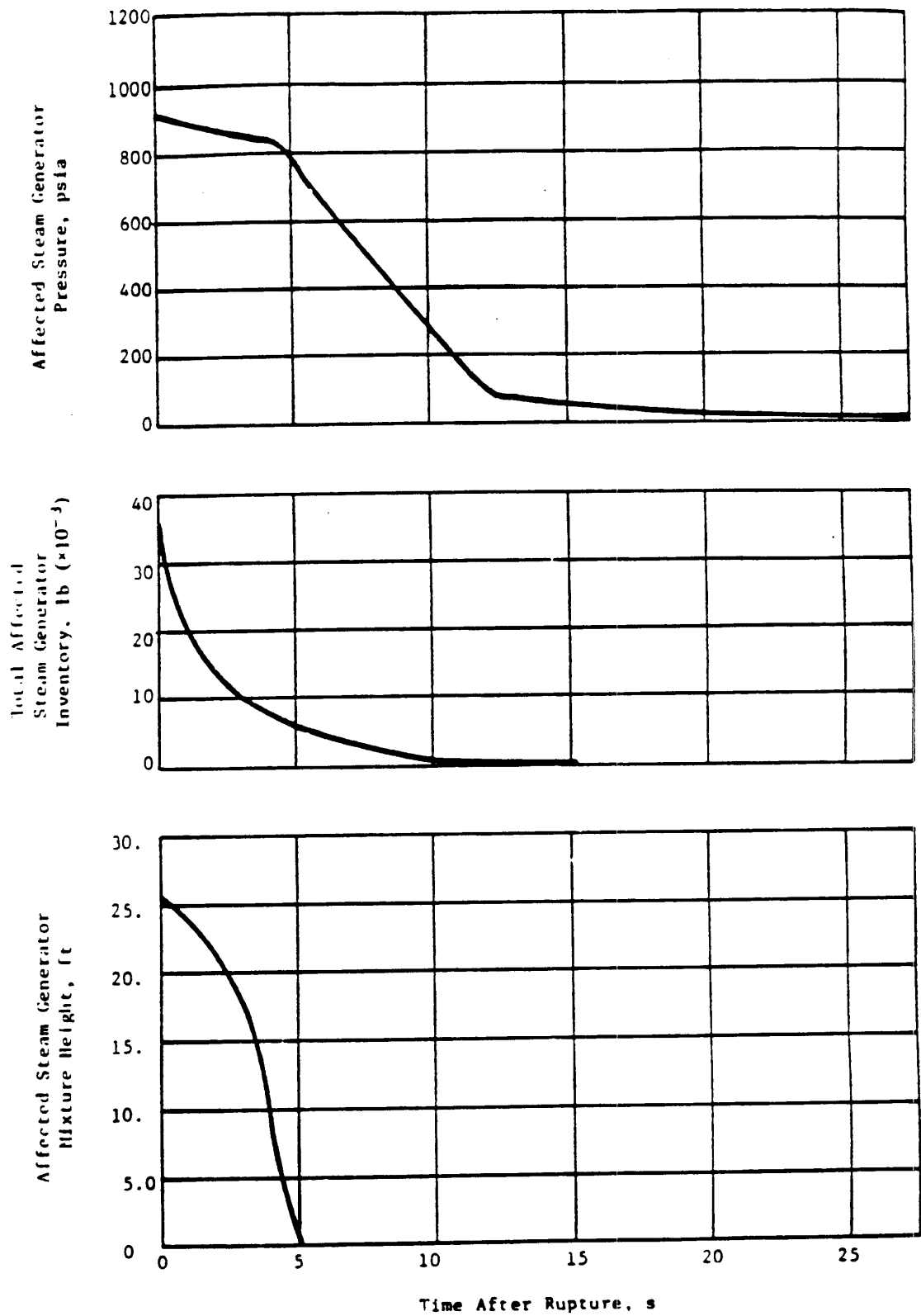
DAVIS-BESSE NUCLEAR POWER STATION  
HOT LEG PRESSURE (LOOP 1)  
FIGURE 15.2.8-1C

REVISION 9  
JULY 1989



DAVIS-BESSE NUCLEAR POWER STATION  
PRESSURIZER COLLAPSED LIQUID LEVEL  
FIGURE 15.2.8-1D

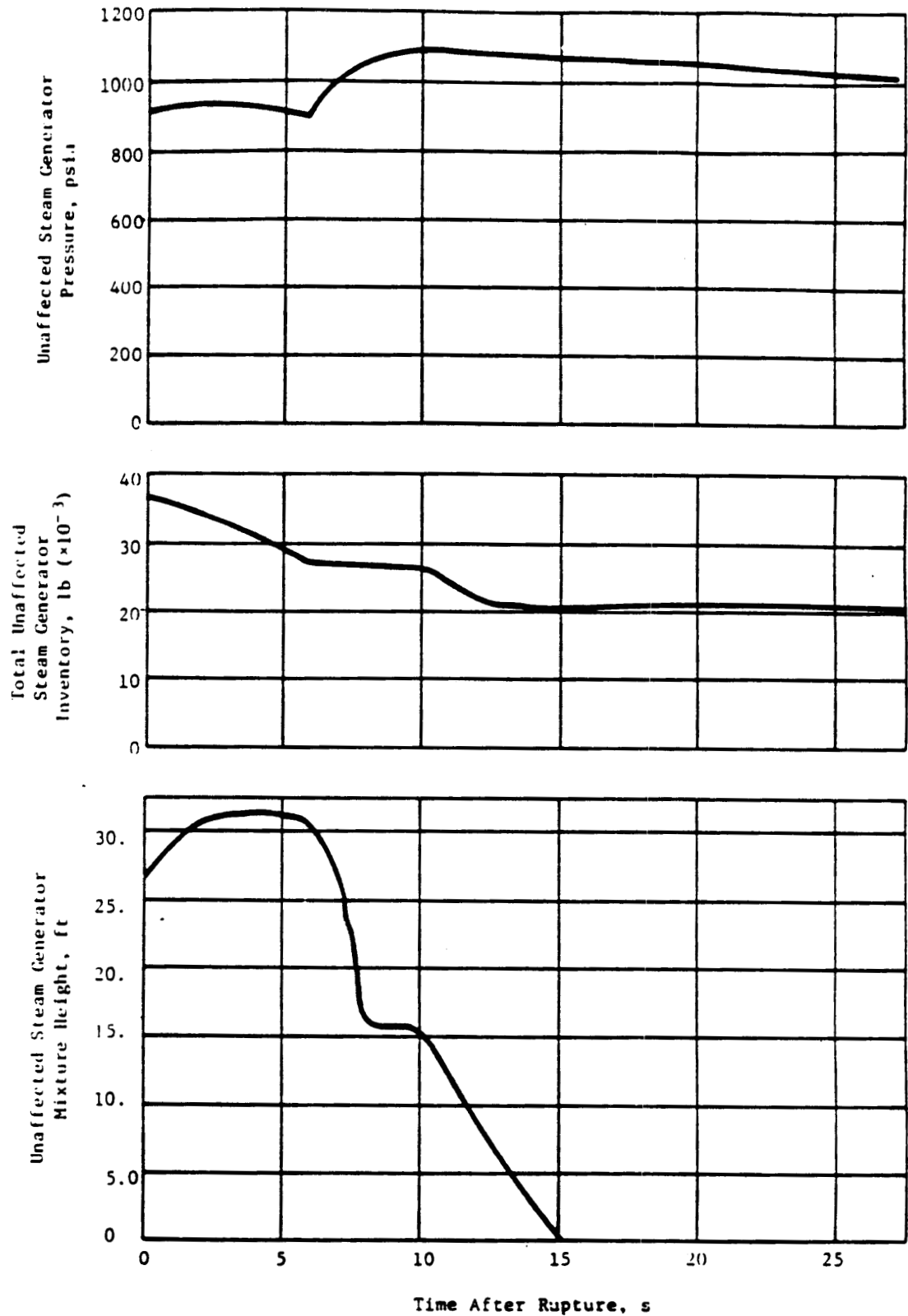
REVISION 9  
JULY 1989



DAVIS-BESSE NUCLEAR POWER STATION  
 FEEDWATER LINE BREAK WITH  
 OFFSITE POWER AVAILABLE  
 FIGURE 15.2.8-2

REVISION C  
 JULY 1982

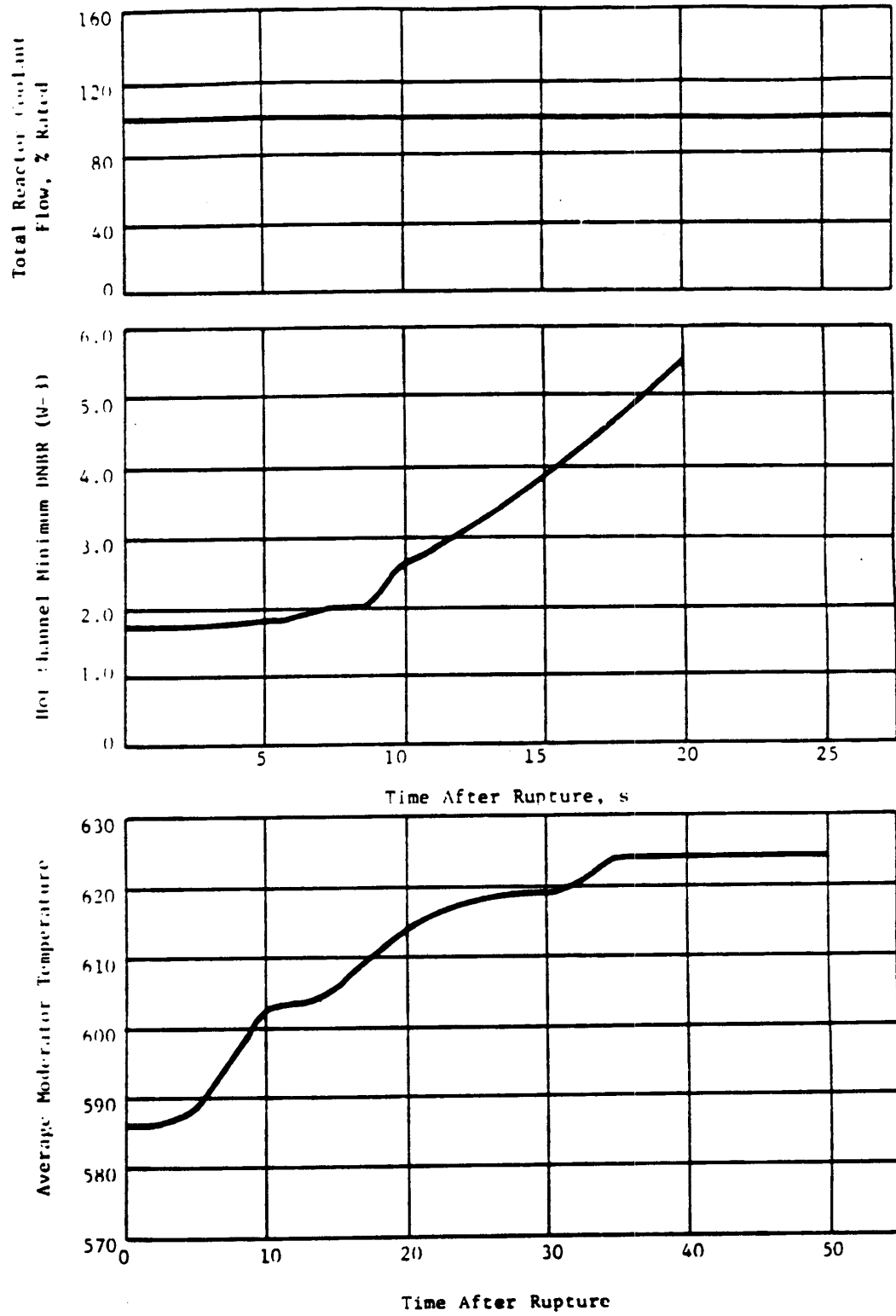




DAVIS-BESSE NUCLEAR POWER STATION  
 FEEDWATER LINE BREAK WITH  
 OFFSITE POWER AVAILABLE

FIGURE 15.2.8-3

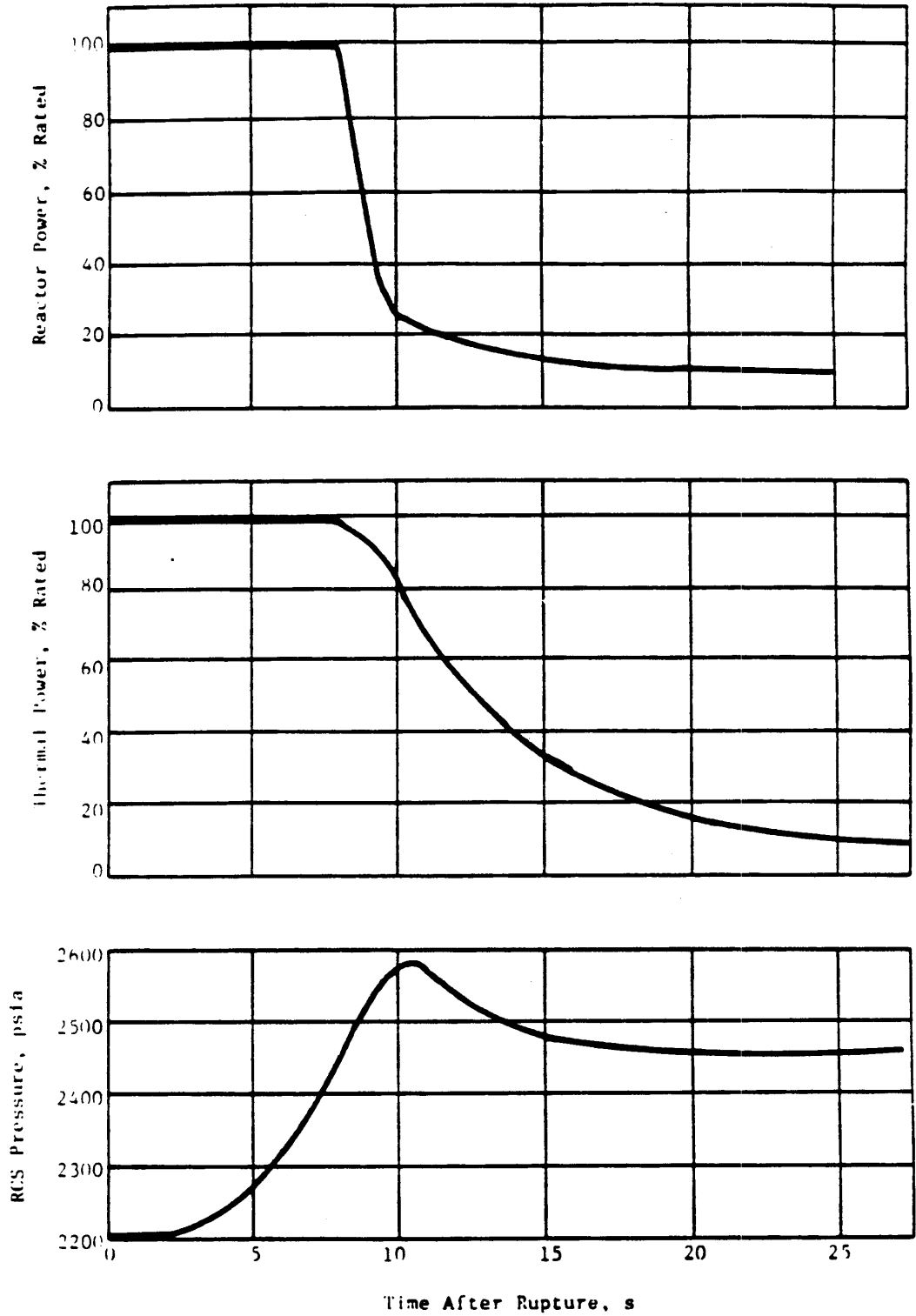
REVISION 0  
 JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION  
 FEEDWATER LINE BREAK WITH  
 OFFSITE POWER AVAILABLE

FIGURE 15.2.8-4

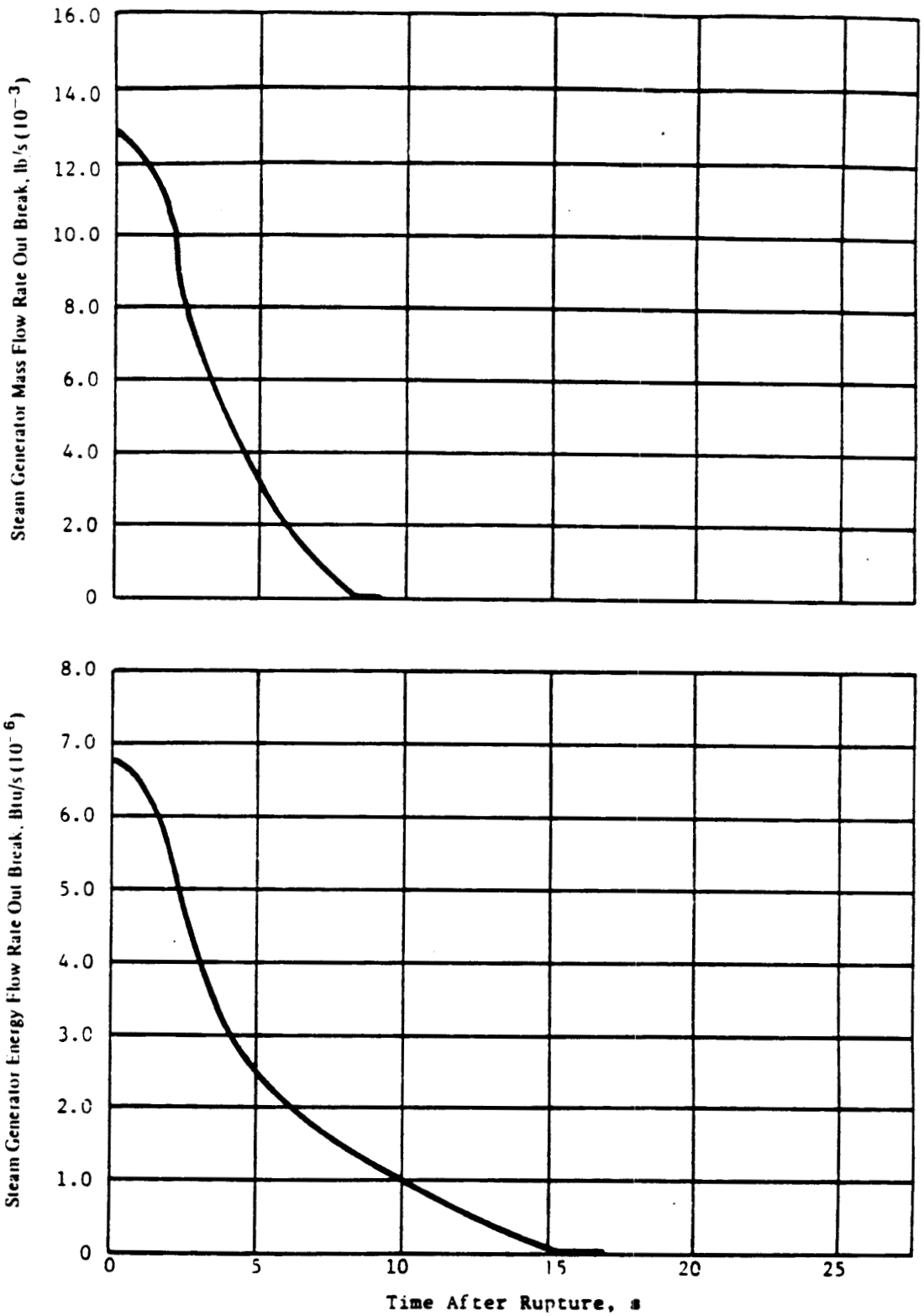
REVISION 0  
 JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION  
 FEEDWATER LINE RUPTURE WITH  
 OFFSITE POWER AVAILABLE-CASE 1

FIGURE 15.2.8-5

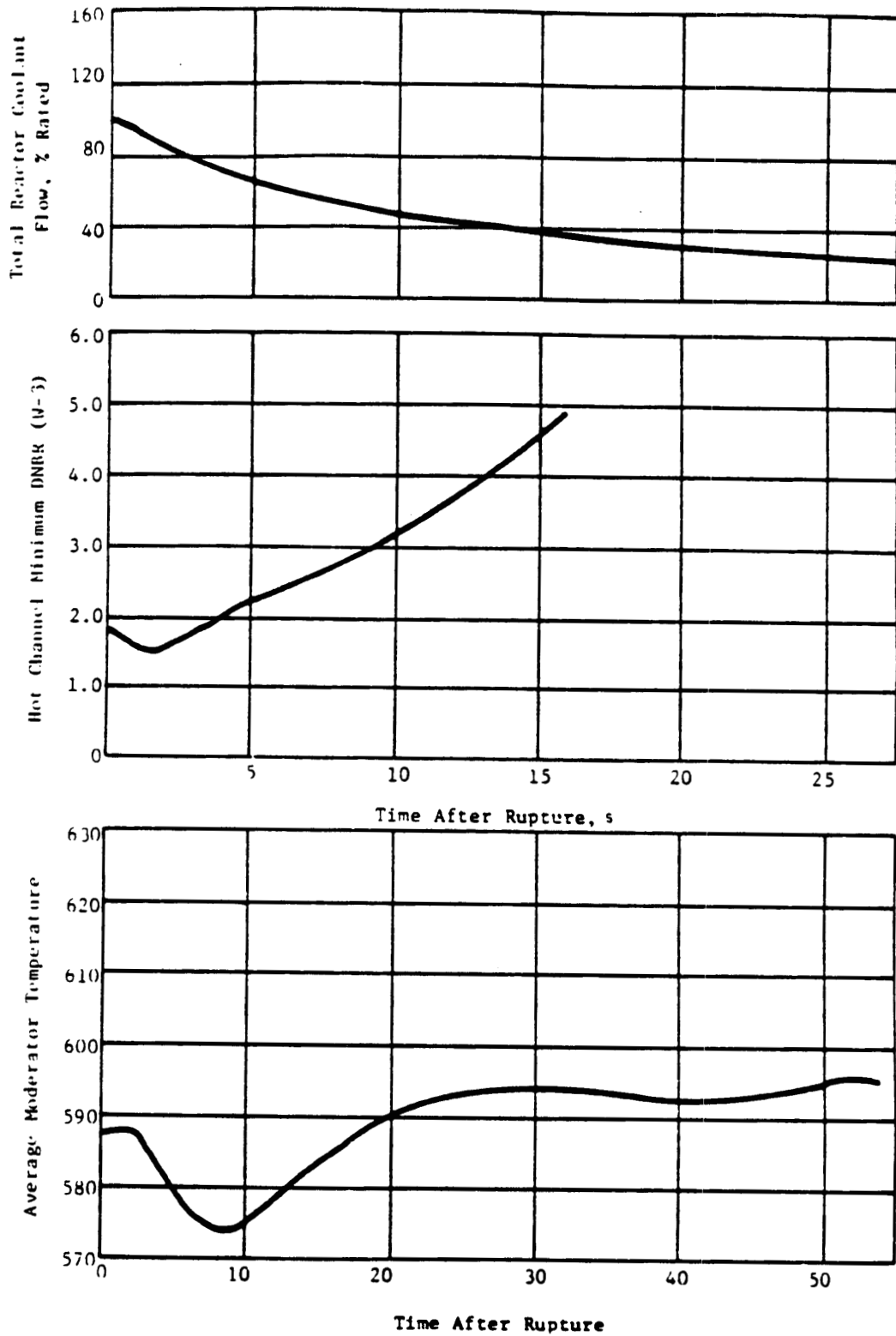
REVISION 0  
 JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION  
 FEEDWATER LINE BREAK WITH  
 OFFSITE POWER AVAILABLE

FIGURE 15.2.8-6

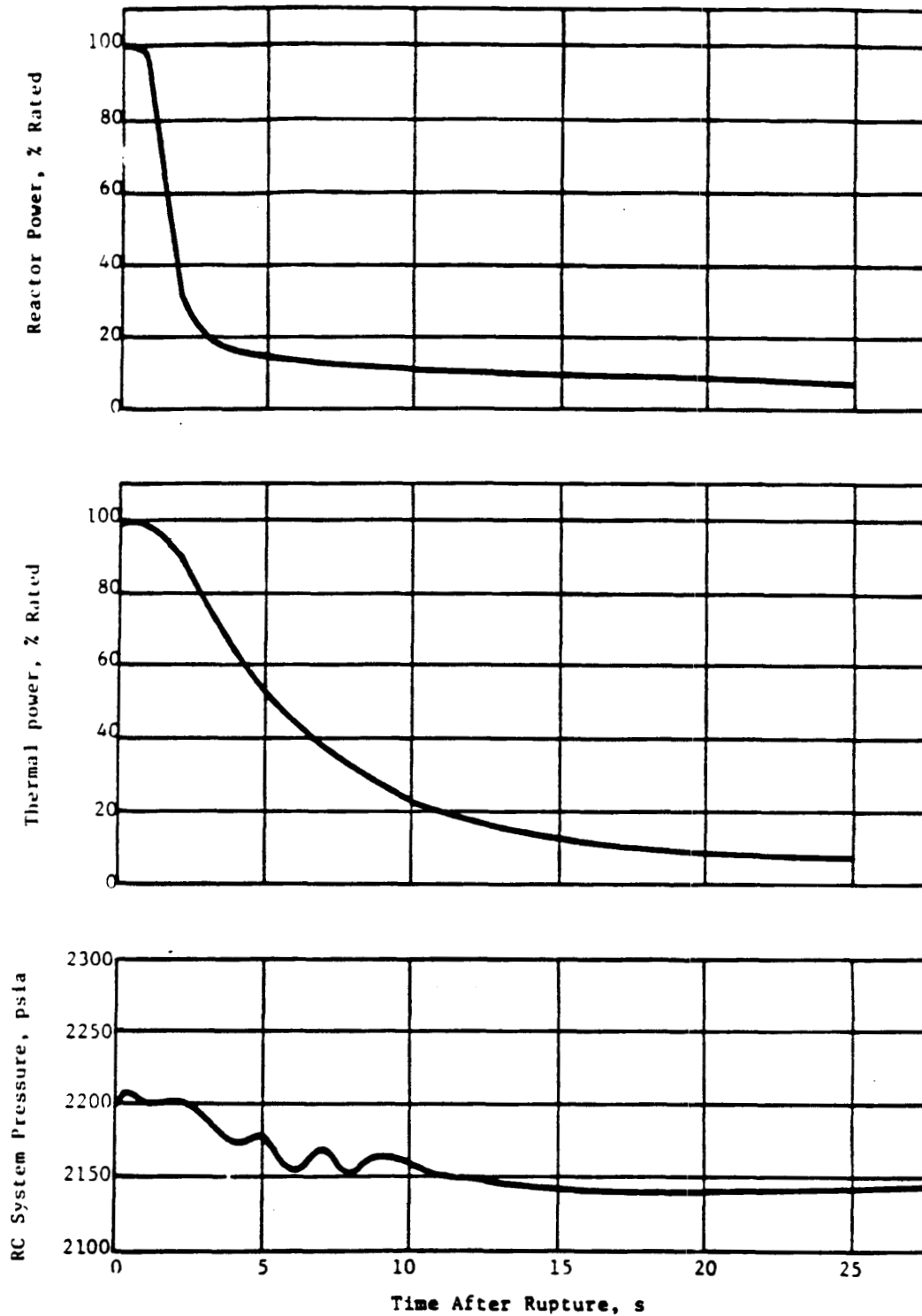
REVISION 0  
 JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION  
 FEEDWATER LINE BREAK WITH LOSS  
 OF OFFSITE POWER AT RUPTURE

FIGURE 15.2.8-7

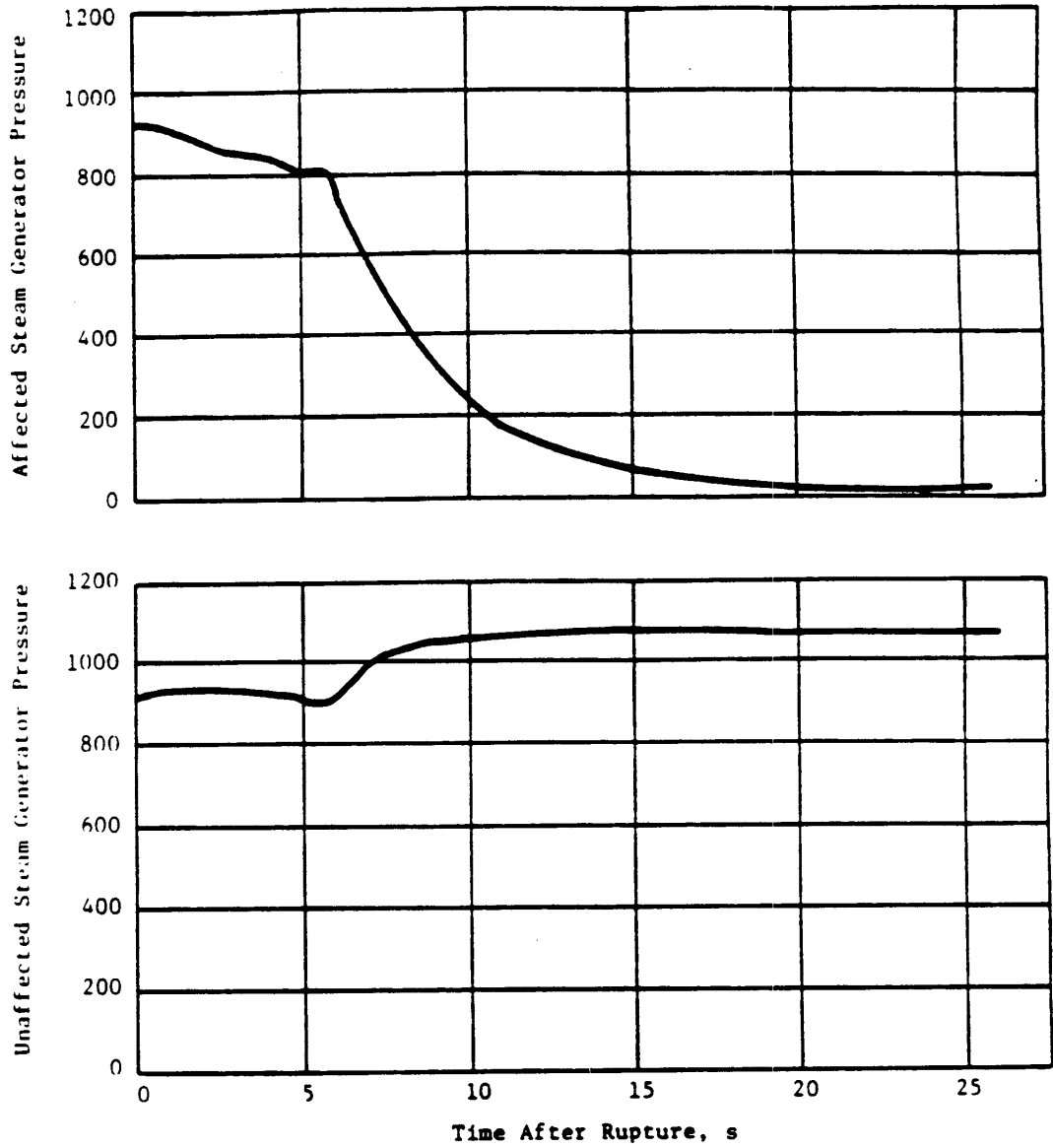
REVISION 0  
 JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION  
 FEEDWATER LINE RUPTURE WITH LOSS  
 OF OFFSITE POWER AT RUPTURE  
 CASE 11

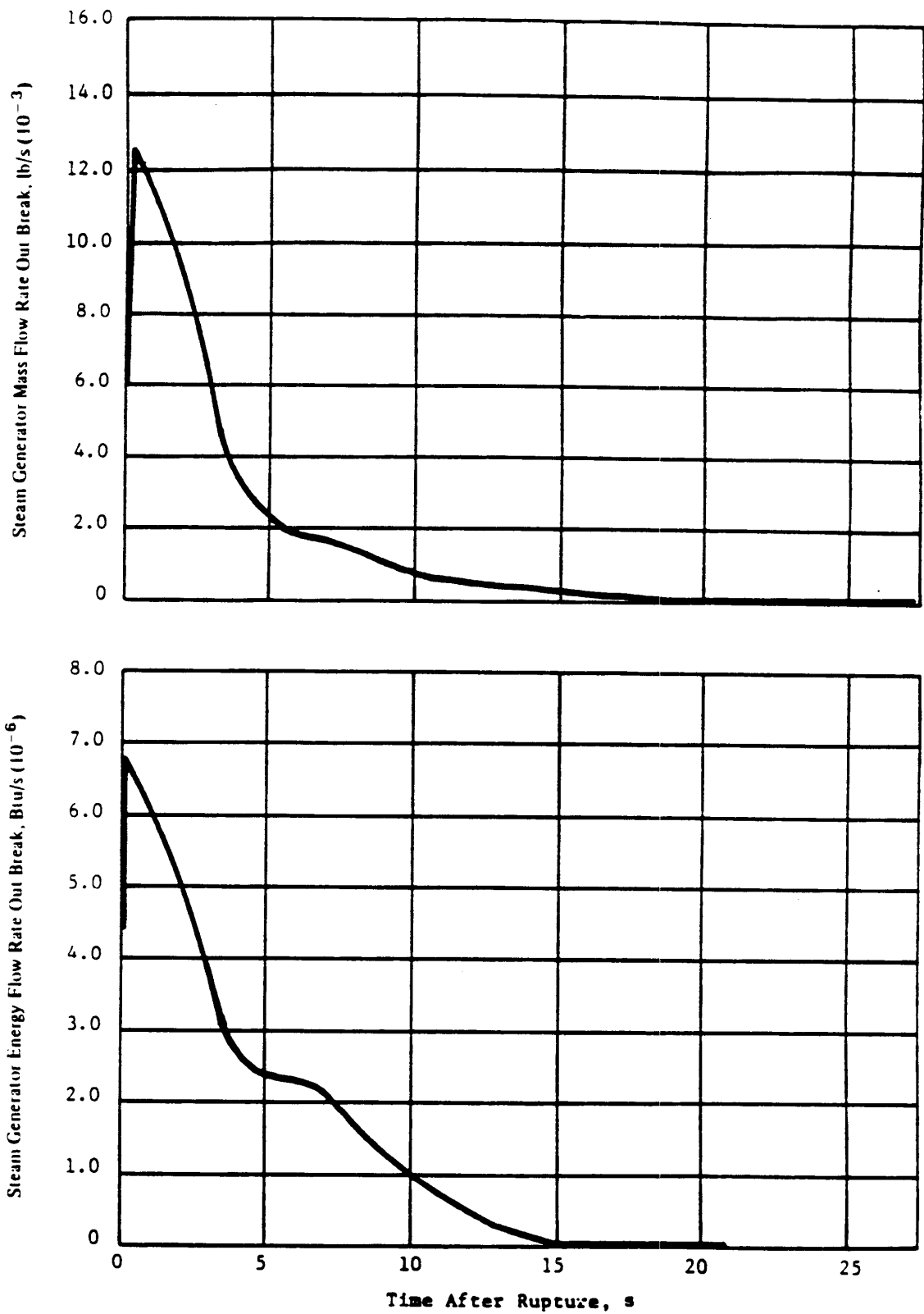
FIGURE 15.2.8-8

REVISION 0  
 JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION  
 FEEDWATER LINE BREAK WITH LOSS  
 OF OFFSITE POWER AT RUPTURE

FIGURE 15.2.8-9

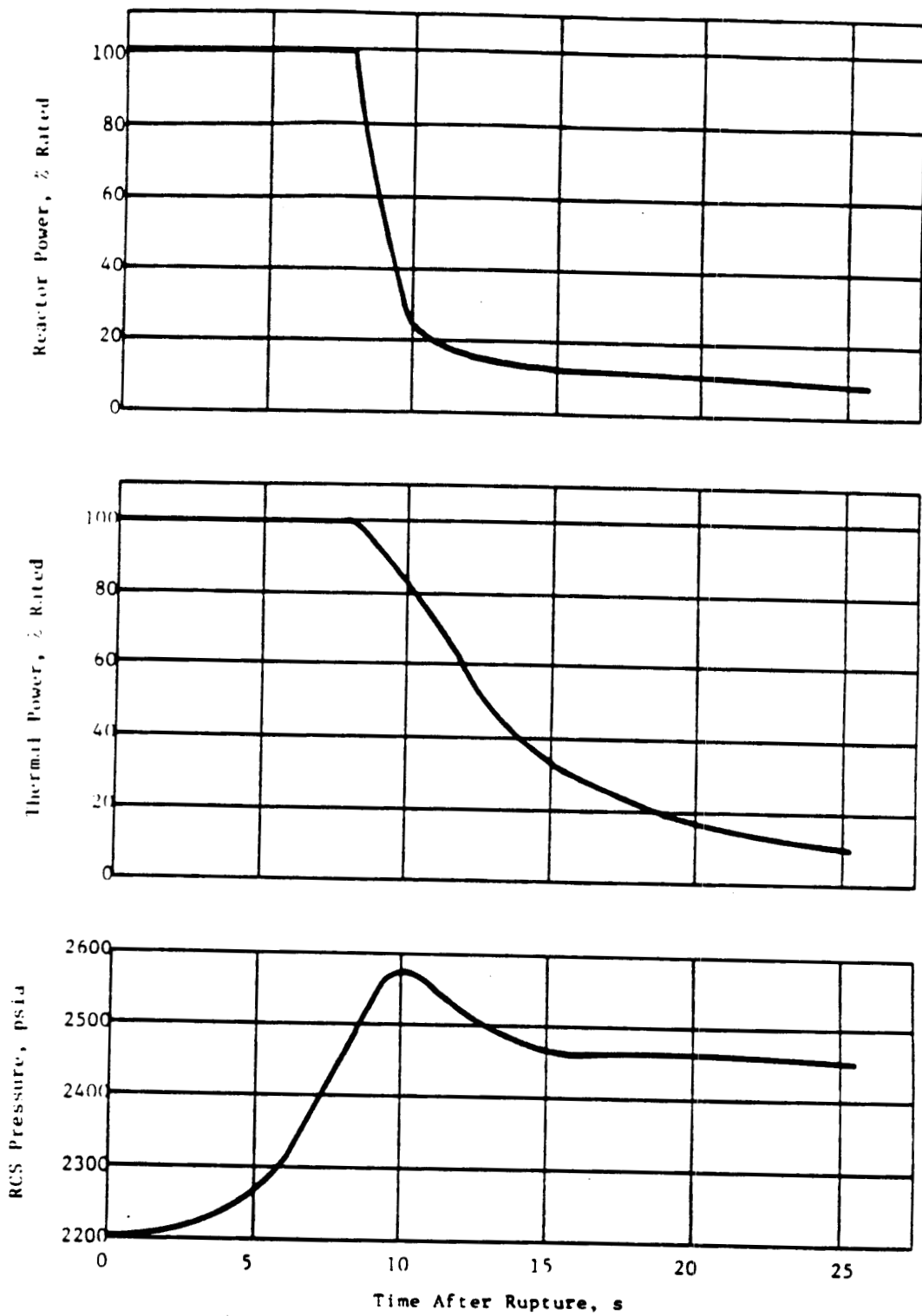


DAVIS-BESSE NUCLEAR POWER STATION  
 FEEDWATER LINE BREAK WITH LOSS  
 OF OFFSITE POWER AT RUPTURE

FIGURE 15.2.8-10

REVISION 0  
 JULY 1982

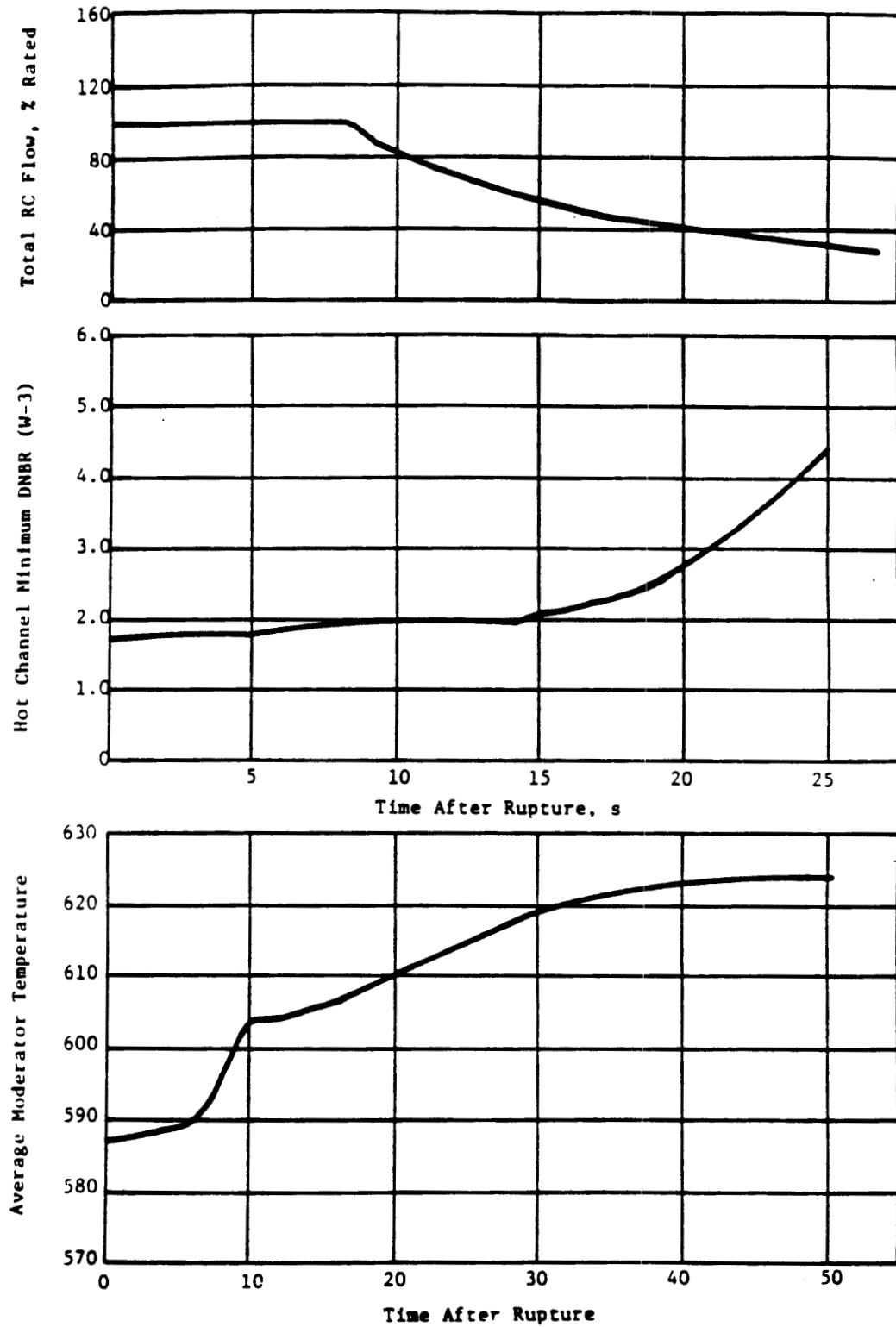




DAVIS-BESSE NUCLEAR POWER STATION  
 FEEDWATER LINE RUPTURE WITH LOSS OF  
 OFFSITE POWER AT TRIP-CASE III

FIGURE 15.2.8-11

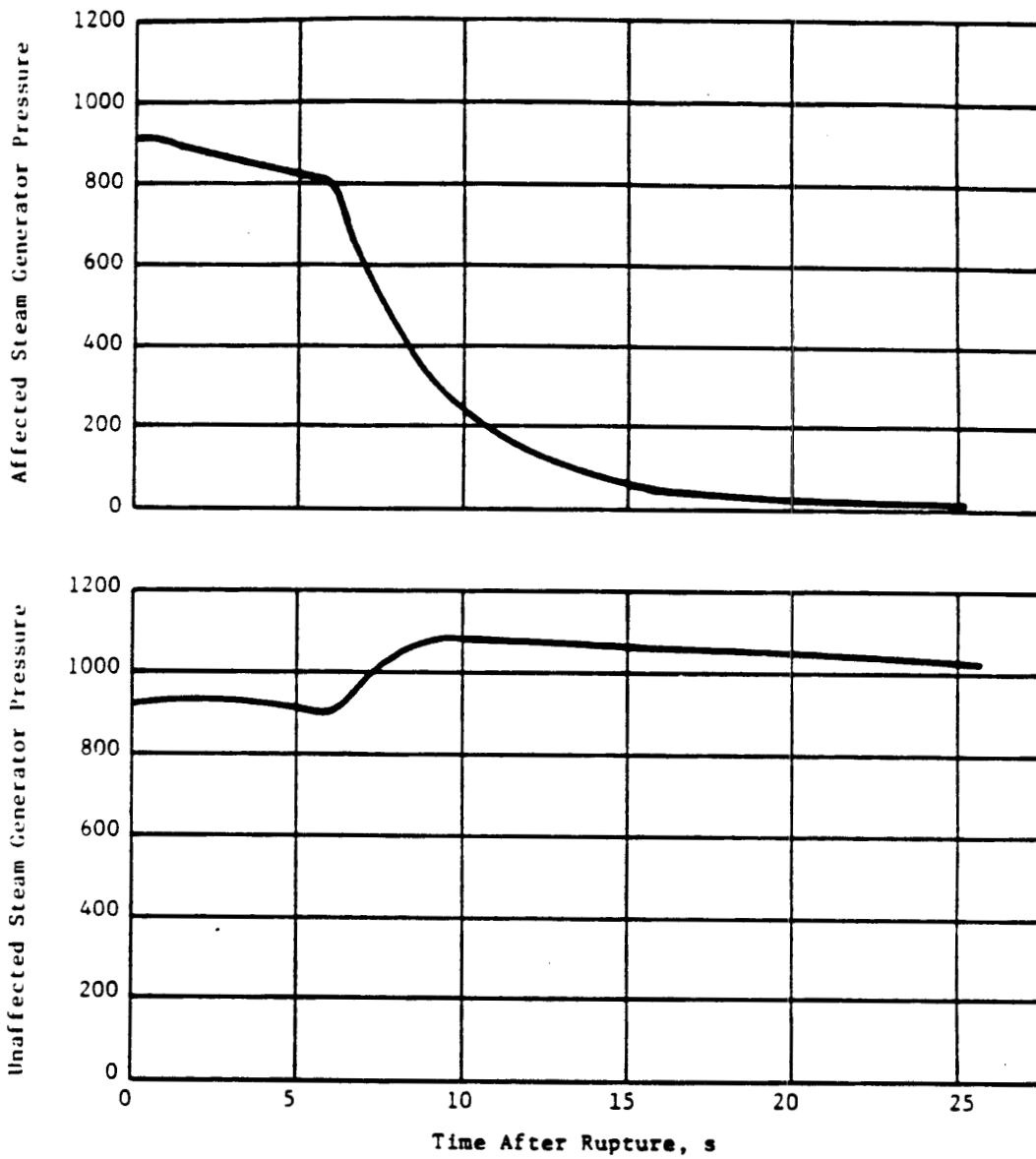
REVISION 0  
 JULY 1982



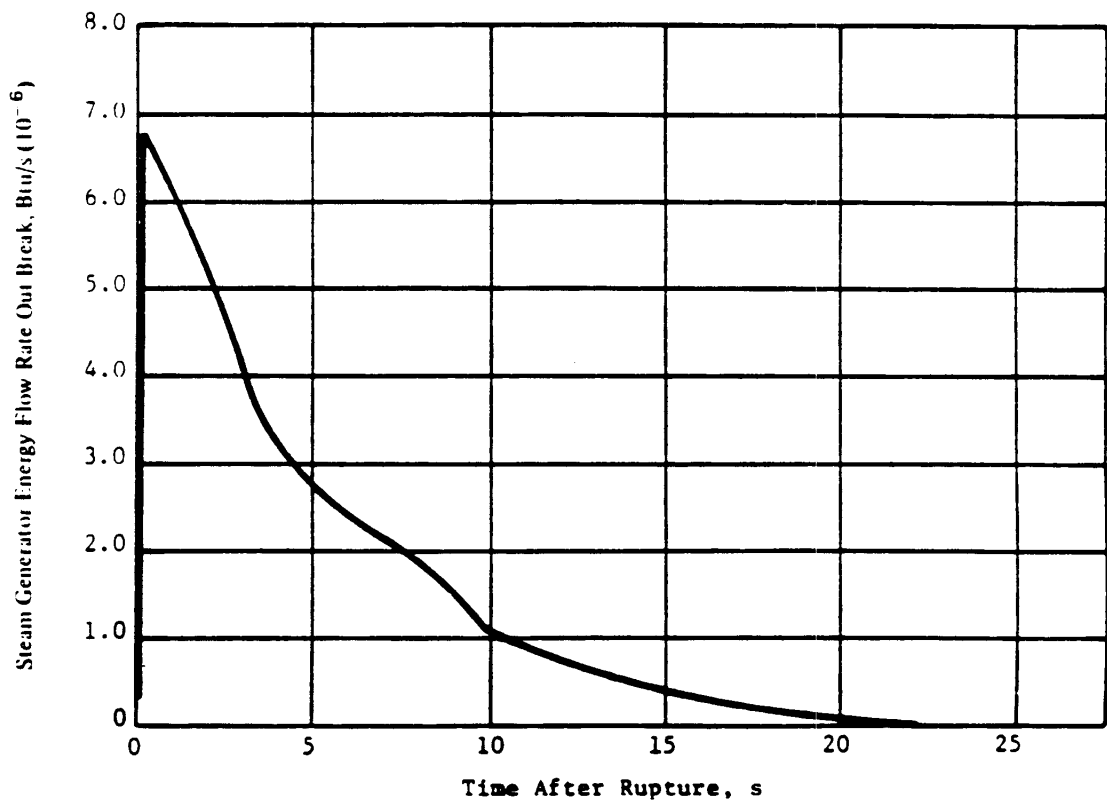
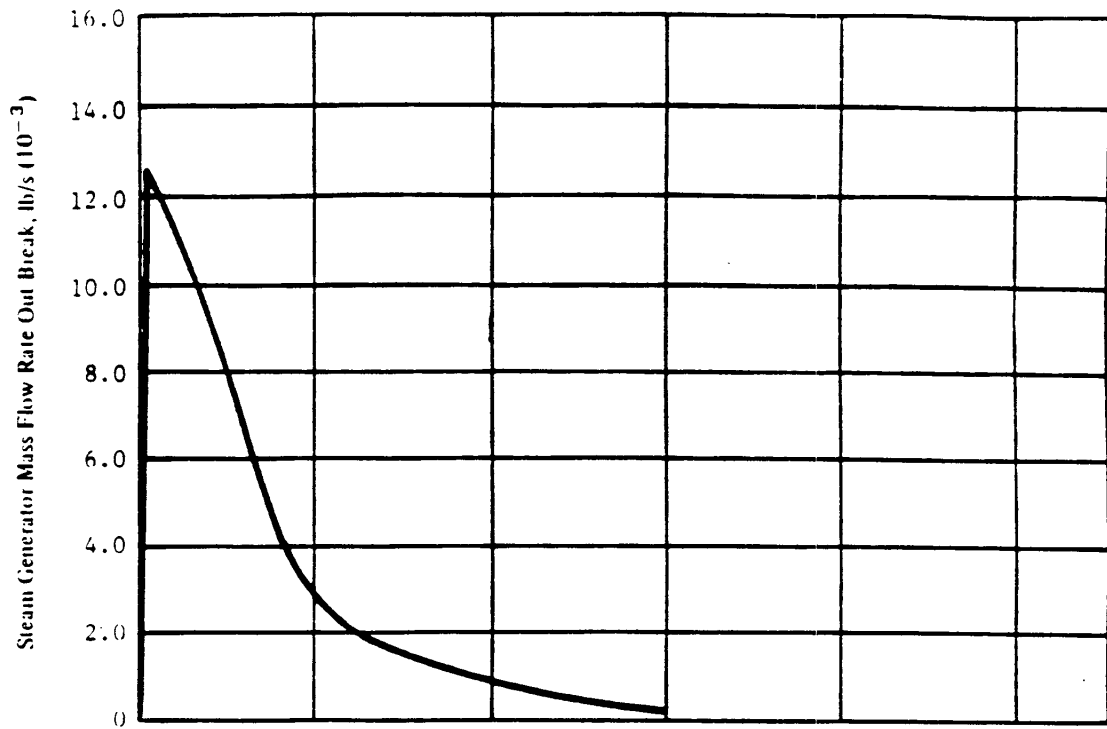
DAVIS-BESSE NUCLEAR POWER STATION  
 FEEDWATER LINE BREAK WITH LOSS OF  
 OFFSITE POWER AT TRIP

FIGURE 15.2.8-12

REVISION 0  
 JULY 1982



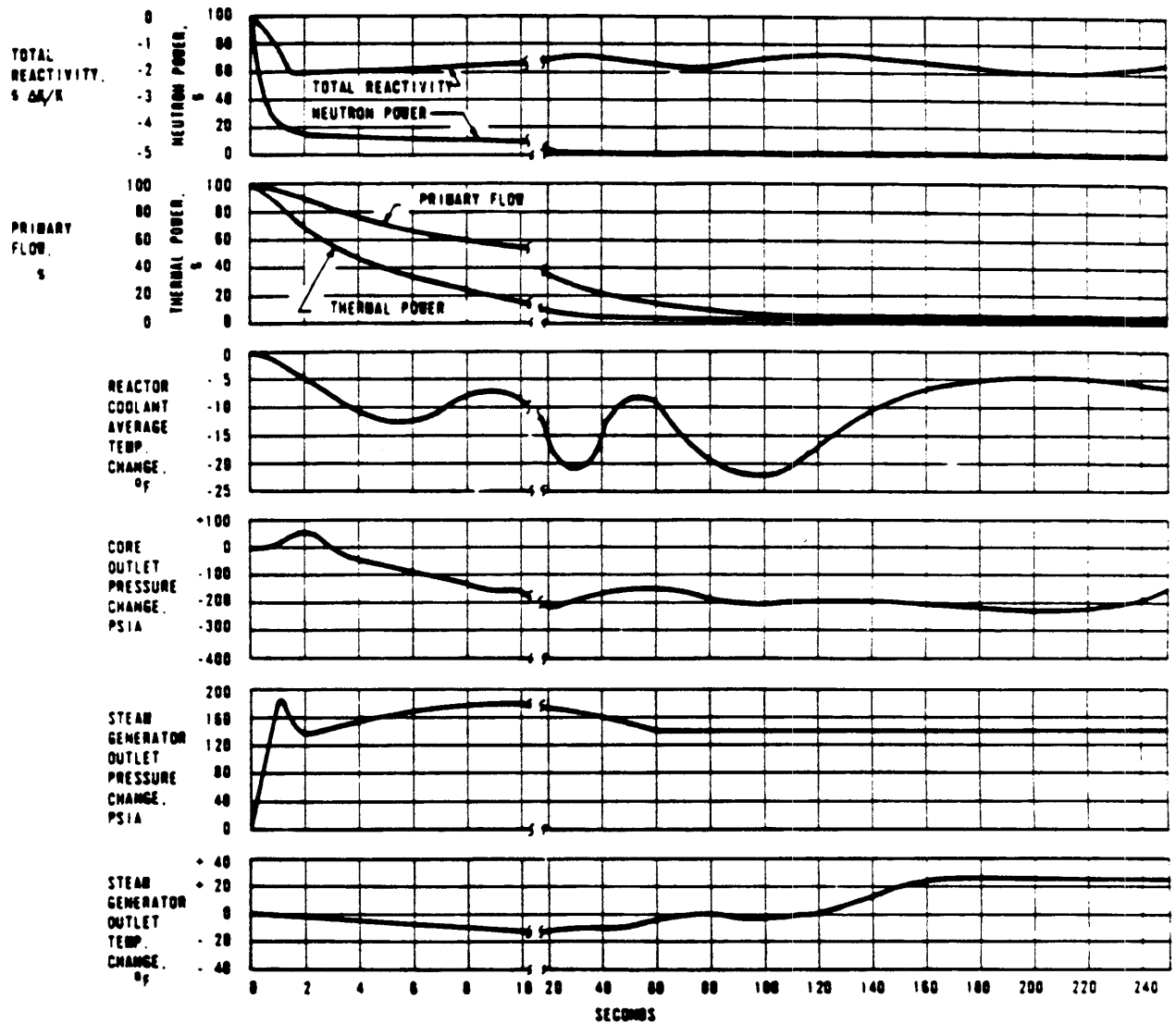
DAVIS-BESSE NUCLEAR POWER STATION  
 FEEDWATER LINE BREAK WITH LOSS  
 OF OFFSITE POWER AT TRIP  
 FIGURE 15.2.8-13



DAVIS-BESSE NUCLEAR POWER STATION  
 FEEDWATER LINE BREAK WITH LOSS  
 OF OFFSITE POWER AT TRIP

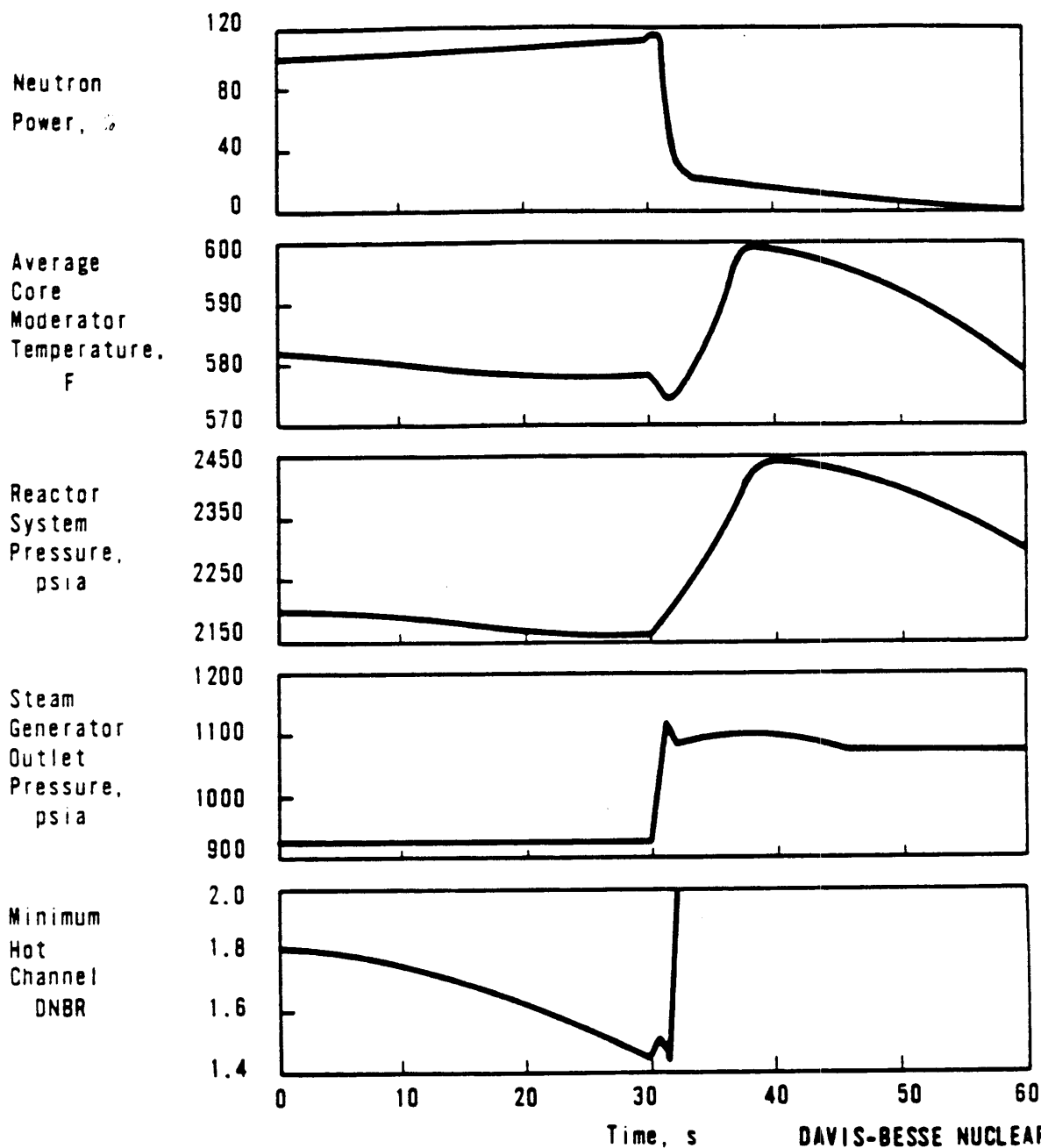
FIGURE 15.2.8-14

REVISION 0  
 JULY 1982



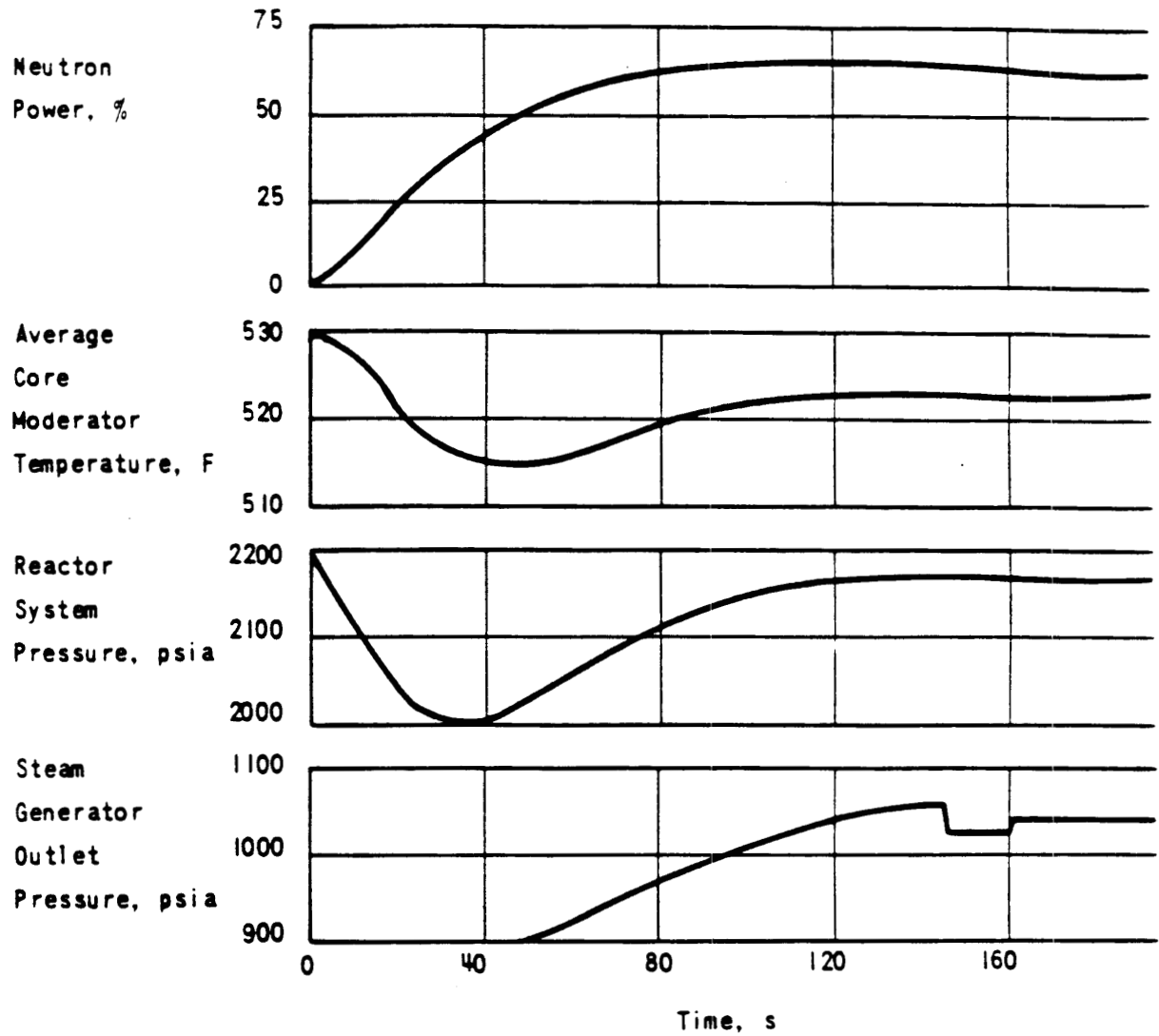
DAVIS-BESSE NUCLEAR POWER STATION  
 LOSS OF A.C. POWER WHILE POWER  
 OPERATING AT RATED POWER  
 FIGURE 15.2.9-1

REVISION 0  
 JULY 1982



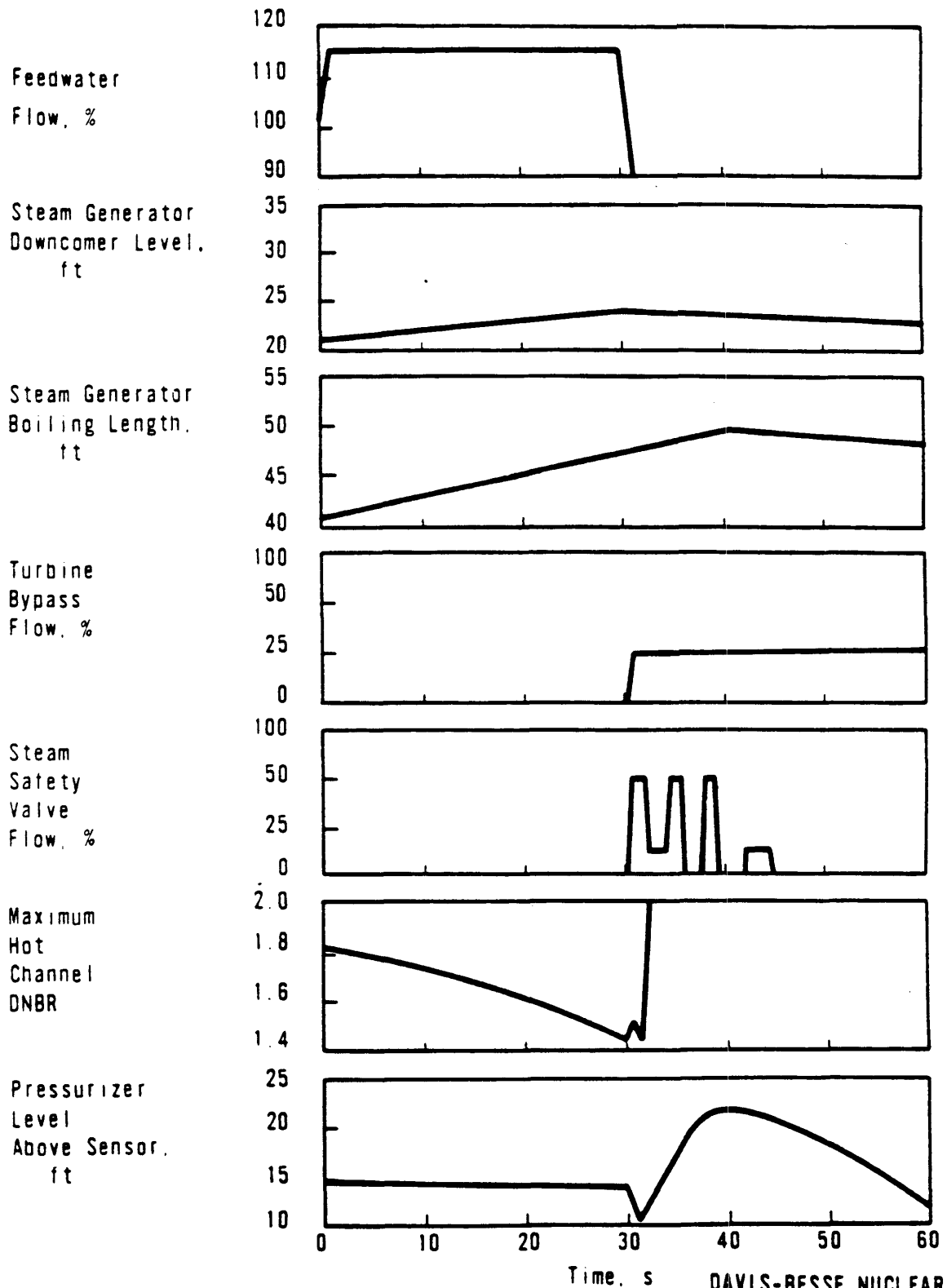
DAVIS-BESSE NUCLEAR POWER STATION  
 RESPONSE OF REACTOR COOLANT SYSTEM  
 TO FEEDWATER TEMPERATURE DECREASE  
 FIGURE 15.2.10-1

REVISION 0  
 JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION  
 RESPONSE OF REACTOR COOLANT SYSTEM TO FEEDWATER  
 FLOW INCREASE TO NO LOAD CONDITION  
 FIGURE 15.2.10-2

REVISION 0  
 JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION  
 TECO-14 EXCESSIVE HEAT REMOVAL  
 DUE TO 115% FW FLOW

FIGURE 15.2.10-3

REVISION 0  
 JULY 1982



15.3 CLASS 2 – EVENTS LEADING TO SMALL TO MODERATE RADIOACTIVE RELEASES AT EXCLUSION AREA BOUNDARY

Class 2 events are off-design operational transients or accidents which may result in the following:

- a. Fuel failure in excess of those expected during normal operation.
- b. A breach of the fuel cladding (which leads to fission product release) or of the Reactor Coolant System boundary.
- c. Operation of engineered safety features or the use of the containment to limit the consequences of a transient.
- d. Offsite radiation exposures in excess of the limits permitted during normal operation.

The consequences of Class 2 events are not of such severity as the required interruption or restriction of public use of areas beyond the station exclusion area boundary. Furthermore, these events do not in themselves lead to the occurrence of the more serious Class 3 events. Table 15.3-1 summarizes the accidents categorized as Class 2 events.

TABLE 15.3-1

Class 2 Events

Event	Analysis assumptions	Effect
Loss of reactor coolant from small ruptured pipes or from cracks in large pipes which actuates emergency core cooling	Reactor coolant leakage through a spectrum of areas smaller than for the design basis LOCA is considered. Environmental effects are based on the release of all the gap activity from the fuel.	The accident results are discussed in Chapter 6. Environmental effects are discussed in Subsection 15.4.6.
Minor secondary pipe break	The rupture of a steam line of small area is considered. The reactor is assumed to be operating with 1% defective fuel and 1 gpm steam generator tube leakage. Reactor coolant leakage into the steam generator continues until the Reactor Coolant System is cooled down and depressurized to ambient conditions.	The consequences of this accident and its environmental effects are discussed in Subsection 15.4.4.
Inadvertent loading of a fuel assembly into an improper position	Fuel assemblies are loaded into improper core positions. Also fuel assemblies with incorrect fuel enrichments are loaded into their normal core positions.	Conditions which could produce power maldistributions and the core protection against a maldistribution of power is presented in Subsection 4.3.4.3.

15.3.1 Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates Emergency Core Cooling

15.3.1.1 Accident Analysis

The small break analysis is reported in Reference 51. This report, in accordance with the RELAP5-based LOCA evaluation model (BAW-10192PA) presents the results of the spectrum of cold leg breaks from 0.01 ft<sup>2</sup> in cross-sectional area up to and including the full double-end break of the cold leg piping. Double-ended breaks of the HPI and CFT lines were analyzed. Hot leg breaks were not included in the analysis. Sensitivity studies provided with the EM concluded that since their location would prevent a direct loss of the emergency injection fluid out the break, all of the ECC fluid injected by the Core Flooding Tanks, the HPI pump, and the LPI pump must enter the core before being lost out the break.

With SG heat transfer available, the consequences of the small break transient decrease with decreasing break size. Depending on the break location and imposed boundary conditions, a break area can be identified for which the HPI or normal makeup system is capable of matching the leak rate ensuring an orderly shutdown. For example, the leak rate resulting from the rupture of a 3/4" schedule 160 instrument line (0.002 ft<sup>2</sup>) is matched by the normal makeup system by 1000 seconds without a complete loss of the pressurizer liquid level. The pressure at 1000 seconds is approximately 7 psi above the HPI actuation setpoint. For larger break areas, the HPI system will be actuated during the transient and will supply borated water to the Reactor Coolant System at a sufficient rate to maintain continuous core cooling. Most break sizes result in a calculated core mixture level below the top of the core, resulting in a temperature excursion. The peak clad temperatures, however, were less than 1800°F and all of the acceptance criteria of 10CFR50.46 were met. Other nonradiological aspects of this accident are discussed in Chapter 6.

The environmental consequences of this accident would be less than the environmental consequences discussed as a part of Subsection 15.4.6, Major Rupture of Pipes Containing Reactor Coolant Up to and Including Double-Ended Rupture of the Largest Pipe in the Reactor Coolant System (Loss-of-Coolant Accident). Discussion of the methods of detecting small leaks and the time required to evaluate the occurrence is found in Subsection 5.2.4.

15.3.1.2 Effects of Plant Changes

To accommodate a twenty-four (24) month cycle the SFAS RCS Low Pressure Trip analytical setpoint was revised to 1515 psia. This setpoint was included in the analyses contained in Reference 51. The analyses demonstrated that the accident acceptance criteria will be met.

15.3.1.3 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 65) to determine the impact of the replacement Steam Generators on the analyses of record. That evaluation concluded that the existing small break analysis remains applicable with the replacement Steam Generators installed.

### 15.3.2 Minor Secondary System Pipe Break

#### 15.3.2.1 Accident Analysis

This accident is defined as the rupture of any steam line in the secondary system of small area. This accident is discussed as a part of Subsection 15.4.4, Steam Line Break.

#### 15.3.2.2 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. The net effects of the replacement Steam Generator's secondary mass limit, its slightly greater heat transfer capacity, and smaller steam outlet nozzles result in the minor secondary system pipe break remaining bounded by the double-ended Main Steam Line Break. Therefore, the existing analyses remain applicable with the replacement Steam Generators installed.

### 15.3.3 Inadvertent Loading of a Fuel Assembly Into an Improper Position

#### 15.3.3.1 Identification of Causes

The arrangement of assemblies with different fuel enrichments in the core will determine the power distribution of the core during normal operation. The loading of fuel assemblies into improper core positions or the incorrect preparation of the fuel assembly enrichment could alter the power distribution of the core.

The following fuel misloadings are considered in this accident analysis:

- a. Misloading a fuel pellet with an incorrect enrichment in a fuel rod.
- b. Misloading a fuel rod with an incorrect enrichment in a fuel assembly.
- c. Misloading a fuel assembly with an incorrect enrichment into the core.

Enrichment errors in fuel pellets or pins beyond normal tolerances will result in local power shapes which vary from those calculated with nominal enrichments. Assembly enrichment errors or loading errors may cause gross power shapes which are peaked in excess of reference design values. The Incore Instrumentation System is designed to monitor assembly power distributions as discussed in Section 7.8, and is capable of detecting assembly misplacement. Fuel pellet and pin enrichment loading errors in excess of manufacturing tolerances are prevented by extensive loading controls and procedures. One such manufacturing process to assure that fuel pellets have been properly loaded, is by in-process gamma-scanning. Also, gadolinia rods within an assembly are loaded with approved templates that identify their location. Gross fuel assembly misplacement in the core is prevented by administrative loading procedures and the prominent display of identification markings on each fuel assembly upper end fitting. During fuel loading, these identification numbers are compared to the loading diagram by at least two persons working independently.

Following each refueling, an incore power distribution is taken during startup testing and compared to calculated power distributions. Gross fuel assembly misplacement would be detected by the incore detectors during this phase by the fact that a radial power tilt is present or developing. Similarly, the out-of-core detectors will indicate quadrant tilt conditions.

### 15.3.3.2 Analysis

Section 4.3 presents the core protection analysis for the accident.

Power distributions resulting from enrichment loading errors in pellets, rods, and assemblies have been analyzed. The thermal-hydraulic conditions resulting from the perturbed power shapes have been determined and compared to design values. The enrichments analyzed are conservative and are the greatest possible enrichments.

The following cases have been analyzed:

#### Case 1

A 3.40 wt% Uranium-235 fuel pellet was loaded in the center of a 2.70 wt% U-235 fuel rod. The nuclear analysis was performed using a one-dimensional axial representation of the fuel rod.

#### Case 2

A 3.40 wt % <sup>235</sup>U fuel rod was loaded into the high flux region of a 2.70 wt % assembly. The nuclear analysis was performed in two dimensions.

#### Case 3A

The center assembly of an equilibrium fuel cycle core was replaced by a 3.40 wt % assembly. This was an enrichment increase of 0.55 wt % <sup>235</sup>U.

#### Case 3B

An equilibrium cycle symmetrical assembly (near the outer edge of the core) was replaced by a 3.4 wt % <sup>235</sup>U assembly. This was an enrichment increase of 0.55 wt % <sup>235</sup>U.

The power distributions from cases 3A and 3B were obtained from a two-dimensional, x-y plane, PDQO7 analysis.

#### Results

The power distributions for case 3A are presented in Figure 15.3.3-1. Only power peaks in the central region of the core are appreciably altered by a misloaded center assembly.

Similarly, misplacement of the center assembly by a higher enriched assembly does not cause a radial power tilt. The maximum radial-local power peak occurs in the center assembly, which is a detector assembly. The incore instrumentation would detect an assembly power increase of this magnitude.

The power distribution for case 3B is presented in Figure 15.3.3-2. A significant power tilt results, which would be detected by the incore instrumentation. Misplaced assemblies in other core orientations would introduce radial power tilts which would be more easily detected than this case.

The thermal analysis of cases 1 and 2 (misloaded pellet and pin) resulted in localized DNBR reductions which are limited to the misloaded pellet or pin.

Conclusion

Strict administrative controls will prevent enrichment errors during fuel fabrication and during fuel loading. In the unlikely event that gross core loading errors occur, the incore instrumentation is designed to detect it.

15.3.3.3 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. That evaluation concluded that steam generator performance does not impact this accident in any way. Therefore, the existing analyses remain applicable with the replacement Steam Generators installed.

NOMINAL	1.51	1.11	1.40	1.11	1.03	1.42	1.12	0.83
MISLOADED	1.98	1.20	1.45	1.19	1.02	1.41	1.11	0.83
		1.29	1.63	1.48	1.05	1.14	0.92	0.83
		1.33	1.68	1.50	1.05	1.13	0.92	0.83
			1.58	1.25	1.25	1.00	1.07	0.80
			1.60	1.28	1.25	1.00	1.07	0.80
				1.00	1.06	1.18	1.03	
				.90	1.03	1.17	1.03	
					1.25	1.27	1.03	
					1.16	1.24	1.02	
						1.14		
						1.12		

xxx	NOMINAL POWER DISTRIBL
xxx	MISLOADED ASSEMBLY POW-
	DISTRIBUTION

**DAVIS-BESSE NUCLEAR POWER STATION  
RADIAL X LOCAL ASSEMBLY POWER  
DISTRIBUTION - CASE 3A**

FIGURE 15.3.3-1

REVISION 0  
JULY 1982

				1.11	1.11	1.06	1.11	1.11						
				.76	.78	.79	.78	.77						
		.96	1.17	1.46	1.01	1.23	1.01	1.46	1.17	.96				
		.98	.99	1.01	.91	1.09	.92	1.06	1.04	1.04				
	.95	1.02	1.16	1.05	1.18	1.30	1.18	1.05	1.16	1.02	.95			
	1.06	1.20	1.10	.93	1.13	1.38	1.15	.99	1.20	1.30	1.19			
.96	1.02	1.00	.95	1.14	1.00	0.99	1.00	1.14	.95	1.00	1.02	.96		
.98	1.19	1.17	1.01	1.20	1.02	1.02	1.06	1.26	1.08	1.31	1.34	1.11		
1.17	1.16	.95	.88	1.01	1.21	1.01	1.21	1.01	.88	.95	1.16	1.17		
.98	1.09	1.01	.95	1.13	1.43	1.15	1.48	1.27	.98	1.12	1.26	1.12		
1.11	1.46	1.05	1.14	1.01	1.26	1.35	1.26	1.35	1.26	1.01	1.14	1.05	1.46	1.11
.74	1.00	.92	1.19	1.19	1.51	1.57	1.38	1.63	1.62	1.24	1.36	1.07	1.23	.92
1.11	1.01	1.18	1.00	1.21	1.35	1.26	1.19	1.26	1.35	1.21	1.00	1.18	1.01	1.11
.76	.87	1.10	1.01	1.41	1.56	1.24	1.11	1.31	1.68	1.58	1.28	1.47	1.16	.99
1.06	1.23	1.30	0.99	1.01	1.26	1.19	1.55	1.19	1.26	1.01	0.99	1.30	1.23	1.06
.76	1.04	1.33	.99	1.10	1.35	1.07	1.50	1.14	1.45	1.17	1.29	1.93	1.44	.99
1.11	1.01	1.18	1.00	1.21	1.35	1.26	1.19	1.26	1.35	1.21	1.00	1.18	1.01	1.11
.76	.87	1.10	1.01	1.41	1.56	1.24	1.11	1.31	1.68	1.58	1.28	1.47	1.16	.99
1.11	1.46	1.05	1.14	1.01	1.26	1.35	1.26	1.35	1.26	1.01	1.14	1.05	1.46	1.11
.74	1.00	.92	1.19	1.19	1.51	1.57	1.38	1.63	1.62	1.24	1.36	1.07	1.23	.92
1.17	1.16	.95	.88	1.01	1.21	1.01	1.21	1.01	.88	.95	1.16	1.17		
.98	1.09	1.01	.95	1.13	1.43	1.15	1.48	1.27	.98	1.12	1.26	1.12		
.96	1.02	1.00	.95	1.14	1.00	0.99	1.00	1.14	.95	1.00	1.02	.96		
.98	1.19	1.17	1.01	1.20	1.02	1.02	1.06	1.26	1.08	1.31	1.34	1.11		
.95	1.02	1.16	1.05	1.18	1.30	1.18	1.05	1.16	1.02	.95				
1.06	1.20	1.10	.93	1.13	1.38	1.15	.99	1.20	1.30	1.19				
.96	1.17	1.46	1.01	1.23	1.01	1.46	1.17	.96						
.98	.99	1.01	.91	1.09	.92	1.06	1.04	1.04						
				1.11	1.11	1.06	1.11	1.11						
				.76	.78	.79	.78	.77						

xxx NOMINAL POWER DISTRIBUTION  
xxx MISLOADED ASSEMBLY POWER DISTRIBUTION

DAVIS-BESSE NUCLEAR POWER STATION  
 RADIAL X LOCAL ASSEMBLY POWER  
 DISTRIBUTION - CASE 38

FIGURE 15.3.3-2

REVISION 0  
 JULY 1982



15.4 CLASS 3 - DESIGN BASIS ACCIDENTS

Class 3 events are accidents of very low probability, but are postulated because the conservatively calculated potential offsite doses resulting from these accidents is significant. This will have a bearing on the design and performance of the station to ensure that fission product release to the station environment will not result in undue risk to the public health and safety. These postulated accidents may require operation of engineered safety features. Potential offsite doses resulting from design basis accidents must be less than the guideline values given in 10CFR100. Table 15.4-1 summarizes the accidents categorized as Class 3 events.

TABLE 15.4-1

Class 3 Events

Event	Analysis assumptions	Effect
Waste gas tank rupture	A tank is assumed to contain the gaseous activity evolved from degassing all of the reactor coolant following operation with 1% defective fuel.	Environmental results are shown in Table 15.4.1-1.
Steam generator tube rupture	The reactor has been operating with 1% defective fuel and 1-gpm steam generator tube leakage. Following rupture of the steam generator tube, isolation of the affected generator is not achieved until the Reactor Coolant System is cooled down and depressurized below the lowest pressure set point on the main steam safety valves.	Reactor trips on low Reactor Coolant System pressure. Environmental effects are described in Table 15.4.2-3.
CRA ejection accident	All fuel rods that experience DNB are assumed to release their total gap activity to the reactor coolant (following operation with 1% defective fuel).	Reactor trip occurs on high flux or high pressure. Some fuel clad failure. Table 15.4.3-6 presents environmental effects.
Steam line break	The reactor has been operating with 1% defective fuel and 1 gpm steam Generator tube leakage. Reactor coolant leakage into the steam generator continues until the Reactor Coolant System is cooled down and depressurized to ambient conditions.	Reactor trips following a large rupture on high flux or low coolant pressure. Environmental effects are indicated in Table 15.4.4-4.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 15.4-1 (Continued)

Class 3 Events

Event	Analysis assumptions	Effect
Break in instrument line or lines from primary system that penetrate containment	Double-ended rupture of 2-1/2 in. letdown line outside Reactor Building. The reactor has been operating with 1% defective fuel.	System isolates and reactor trips on low pressure. Flashing of coolant and partial release of activity. Table 15.4.5-2 presents environmental effects.
Loss-of-coolant accident	Environmental effects are based on the release of all the gap activity with the reactor operating with 1% defective fuel. Release of 100% noble gases, 50% iodine, and 1% solid fission products considered as maximum hypothetical accident.	See Table 15.4.6-1 for environmental effects. Table 15.4.6-2 presents environmental effects of maximum hypothetical accident.
Fuel handling accident	Gap activity is released from 56 fuel rods in one assembly while in spent fuel storage pool.	See Tables 15.4.7-2a and 15.4.7-3 for environmental effects.

#### 15.4.1 Waste Gas Decay Tank Rupture

##### 15.4.1.1 Identification of Causes

The waste gas decay tank is used in the radioactive waste disposal system to store radioactive gaseous waste from the station until such time that the radioactive decay renders the gas safe for release to the site environment. Rupture of a waste gas tank would result in the premature release of its radioactive contents to the station ventilation system and to the atmosphere through the station vent. Although it is not considered credible, this accident was analyzed in order to evaluate the resultant dose at the site boundary.

##### 15.4.1.2 Analysis of Effects and Consequences

###### 15.4.1.2.1 Safety Evaluation Criteria

The safety evaluation criteria for this accident is that resultant Exclusion Area Boundary and Low Population Zone doses shall not exceed a small fraction of the 10CFR100 limits and the Control Room doses shall not exceed the limits of General Design Criteria 19.

Beginning with cycle 5, the fuel cycle length was extended to 18 months. The plant Technical Specifications limit the RCS activity to a value which is significantly less than the iodine activity associated with 1% failed fuel, which was assumed in the original evaluation of this accident given in section 15.4.1.2.2. Therefore, the Waste Gas Decay Tank activities presented in Table 15.4.1-2 bound the activities associated with any fuel cycle. For this reason no additional evaluation was performed for this accident to support extended 24 month fuel cycles.

###### 15.4.1.2.2 Methods of Analysis

A waste gas tank is assumed to contain all noble gases in one reactor coolant volume at the end of the third cycle and the iodine from one reactor coolant volume after a DF of  $10^5$ . This DF is a conservative addition of 100 for the purification demineralizers and  $10^4$  for the degasifier. Operation with 1 percent defective fuel is assumed. In addition, the following assumptions are made:

- a. The accident duration is 2 hours, that is, 99.9 percent of all airborne activity is released over two hours.
- b. The Control Room Ventilation System is isolated upon receipt of a high radiation signal in the Auxiliary Building exhaust stack (station vent). Isolation requires a maximum of 11 seconds (including 6 seconds maximum for instrument response). The air delivery rate for the five seconds after the fan is shut down was calculated using the following assumptions:
  1. One supply fan with its related return fan is operating.
  2. The supply and return fans are de-energized simultaneously.
  3. The system dampers are fully open during the 5 second time interval.

## Davis-Besse Unit 1 Updated Final Safety Analysis Report

- c. The tank is assumed to rupture and release its contents to the Auxiliary Building. As the release is vented to the atmosphere by the Emergency Ventilation System, it passes through charcoal filters that remove iodine with an efficiency of 95 percent. However, no iodine filtering is credited.

The results are as follows:

### Air Delivery Rates Following Fan Shutdown

<u>Shutdown Time Intervals</u>	<u>Flow Rate (cfm)</u>
Running cfm	21,920
One Second	20,824
Two Seconds	19,471
Three Seconds	17,839
Four Seconds	16,440
Five Seconds	15,227

(NOTE: A conservative flowrate of 22,000 cfm for 0 seconds to 15 seconds was analyzed.)

4. The dispersion factors for the fuel-handling accident and waste gas tank rupture are equal since the release point is the same for both accidents.

The release point (station vent) is 160 feet horizontal distance from the control room intake and 180 feet vertical distance.

#### 15.4.1.2.3 Results of Analysis

The rupture of a waste gas decay tank would release the entire contents of the tank to the auxiliary building atmosphere. The Auxiliary Building is ventilated and discharged to the station vent. In the analysis, however, the activity is assumed to be released from the waste gas decay tank to the atmosphere over a two-hour time period. Table 15.4.1-2 lists the isotopic release to the atmosphere.

Atmospheric dilution for the site and low population zone boundary doses is calculated using the 2-hour dispersion factor developed in Section 2.3. The two-hour integrated doses at the exclusion area boundary and the 30-day doses at the outer boundary of the low-population zone, as shown in Table 15.4.1-1, are well below the limits of the 10CFR100 guideline. The Control Room doses provided in Table 15.4.1-1 are well below the limits of General Design Criteria 19.

See Section 15.4.1.2.1 for the evaluation to support extended fuel cycles.

#### 15.4.1.2.4 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. That evaluation concluded that steam generator performance does not impact this accident in any way. Therefore, the existing analyses remain applicable with the replacement Steam Generators installed.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 15.4.1-1<sup>(1)</sup>

Resultant Doses From Waste Gas Tank Rupture

	Exclusion Area Boundary <u>0 to 2 hours</u>	Low Population Zone <u>0 to 30 days</u>
Thyroid dose, rem	$2.20 \times 10^{-3}$	$1.14 \times 10^{-4}$
Whole-body dose, rem	0.317	$1.65 \times 10^{-2}$
	Operator in Control Room <u>0 to 2 hours</u>	
Thyroid dose, rem	$1.72 \times 10^{-2}$	
Beta-skin dose, rem	2.45	
Total body gamma, rem	$8.09 \times 10^{-2}$	

<sup>(1)</sup>See Section 15.4.1.2.1 for the evaluation to support extended fuel cycles.

TABLE 15.4.1-2<sup>(1)</sup>

Activity Released Due to Waste Gas Tank Rupture (Ci)

I-131	$1.70 \times 10^{-2}$
I-132	$1.59 \times 10^{-2}$
I-133	$1.52 \times 10^{-2}$
I-134	$1.79 \times 10^{-3}$
I-135	$7.29 \times 10^{-3}$
Xe-131m	$8.15 \times 10^2$
Xe-133m	$1.04 \times 10^3$
Xe-133	$9.48 \times 10^4$
Xe-135m	$1.17 \times 10^2$
Xe-135	$2.45 \times 10^3$
Xe-137	0
Xe-138	$1.64 \times 10^2$
Kr-83m	$1.33 \times 10^2$
Kr-85m	$6.34 \times 10^2$
Kr-85	$7.05 \times 10^3$
Kr-87	$3.43 \times 10^2$
Kr-88	$1.04 \times 10^3$
Kr-89	0

<sup>(1)</sup>See Section 15.4.1.2.1 for the evaluation to support extended fuel cycles.

## 15.4.2 Steam Generator Tube Rupture

### 15.4.2.1 Identification of Causes

The environmental effects associated with the complete severance of a steam generator tube are evaluated. For this occurrence, activity contained in the reactor coolant would be released to the secondary system. Some of the radioactive noble gases and iodine would be released to the atmosphere through the condenser air removal system and the steam line safety valves.

### 15.4.2.2 Accident Analysis

#### 15.4.2.2.1 Safety Evaluation Criteria

The safety evaluation criteria for this accident are:

- a. Resultant doses shall not exceed 10CFR100 limits.
- b. Additional loss of reactor coolant boundary integrity shall not occur due to a loss of secondary side pressure and resultant temperature gradients.

Beginning with cycle 5, the fuel cycle length was extended to 18 months. The plant Technical Specifications limit the RCS activity to a value which is significantly less than the iodine activity associated with 1% failed fuel, which was assumed in the original evaluation of this accident given in section 15.4.2.2.4. Therefore the resultant doses presented in Table 15.4.2-3 bound the radiation doses for any fuel cycle (References 54 and 55). The plant Technical Specification limits are based on the NRC evaluation as documented in the NRC SER. For this reason no additional evaluation was performed for this accident to support extended 24 month fuel cycles.

#### 15.4.2.2.2 Methods of Analysis

In analyzing the consequences of this failure, the following sequence of events is assumed to occur (input parameters are shown in Table 15.4.2-1 and results are summarized in Table 15.4.2-2).

- a. Reactor Coolant System Response
  1. A double-ended rupture of one steam generator tube occurs with unrestricted discharge from each end to the secondary side of the steam generator.
  2. The initial leak rate exceeds the normal makeup to the Reactor Coolant System, and system pressure decreases. No initial operator action is assumed, and a low Reactor Coolant System pressure trip will occur.
  3. After reactor trip, the Reactor Coolant System pressure continues to decrease until high pressure injection is automatically actuated. The capacity of high pressure injection is sufficient to compensate for the leakage and thereafter, it is assumed that the operator has properly diagnosed the problem and takes action by initiating a reactor coolant system cooldown and depressurization. After the Reactor Coolant System reaches 500°F and 1065 psia the affected steam generator is isolated.

## Davis-Besse Unit 1 Updated Final Safety Analysis Report

4. Following isolation of the affected steam generator, the cooldown is continued at 100°F/hr with the unaffected steam generator until the Reactor Coolant System temperature reaches 280°F. Thereafter, cooldown to ambient conditions is continued using the Decay Heat Removal System.
- b. Secondary System Response
1. Following reactor trip, the turbine stop valves close, and under normal conditions, the ICS changes the pressure setpoints on the turbine bypass system from 920 psig to 995 psig on the turbine bypass valves to the condenser. However, it has been assumed in this analysis that this function has failed.
  2. Following closure of the turbine stop valves, the secondary system pressure will increase, causing the turbine bypass valves and the main steam line safety valves to open. Steam relief directly to the atmosphere will continue until secondary system pressure drops below the main steam line safety valve setpoint.
  3. Thereafter, the turbine bypass valves will continue to relieve steam to the condenser. The operator initiates Reactor Coolant System cooldown and depressurization by further opening the turbine bypass valve on the unaffected steam generator.
  4. When the Reactor Coolant System pressure has fallen below the 1050 psig low steam safety valve setpoint, the operator closes and latches the turbine bypass valve to the condenser to complete final isolation of the affected steam generator.

The distinguishing characteristic of this event is the buildup of activity levels in the secondary steam system. The condenser offgas monitor will detect an increase in secondary steam system noble gas release to the station vent and alert the operator via an alarm when the activity level exceeds normal operating limits. This alarm, coupled with dropping RC pressure and pressurizer level indications, provides sufficient information for the operator to diagnose the occurrence of this accident in comparison to other possible events. In addition, N-16 radiation monitors located on each steam line will provide the operator with rapid identification of the affected steam generator.

The method used to calculate all coolant activities is described in detail in Chapter 11.

### 15.4.2.2.3 Results of Analysis

The results of this accident are summarized in Tables 15.4.2-2 and 15.4.2-3. The analysis shows that the consequences of this accident are within the established criteria stated in Subsection 15.4.2.2.1. A steam generator tube failure concurrent with partial loss of coolant flow produces effects less severe than the rated power condition described above. This is due to the less severe cooldown transient that occurs.

See Section 15.4.2.2.1 for evaluation to support extended fuel cycles.



#### 15.4.2.2.4 Environmental Consequences

During the venting time of the affected steam generator, it is conservatively assumed that all fission products leaking from the Reactor Coolant System go directly to the atmosphere. Prior to the tube rupture, the unit is assumed to have been operated with a 1 gpm tube leak and 1% defective fuel rods. Volatile activity that reaches the condenser is released to the atmosphere after passing through the condenser air removal system. An iodine partition coefficient of  $10^4$  is assumed between the liquid and vapor phases in the condenser (references 1 and 2) during this accident.

Individual isotopic activities which enter the secondary system in the reactor coolant during this accident are listed in Appendix 15A, Table 15A-4. The doses presented in Table 15.4.2-3 are conservatively calculated assuming that all the iodine and noble gas activity contained in the reactor coolant is released to the affected steam generator (435 gpm for 34 minutes) and subsequently released directly to the environment (i.e. credit for iodine partitioning is not considered). The result and doses are within the 10CFR100 guidelines.

See Section 15.4.2.2.1 for the evaluation to support extended fuel cycles.

#### 15.4.2.2.5 Consequences of Less Severe Ruptures

A rupture of a steam generator tube which results in a leak rate equal to the primary makeup capability will be detected by N-16 detectors in the main steam line headers in less than 15 seconds, setting off an alarm in the control room. Upon receipt of the alarm signal in the control room, the operator initiates station shutdown.

Site boundary thyroid and whole body doses are necessarily lower than those listed in Table 15.4.2-3, since the activity released from the primary to the secondary is far less.

See Section 15.4.2.2.1 for the evaluation to support extended fuel cycles.

#### 15.4.2.2.6 Effects of Plant Changes

##### 15.4.2.2.6.1 Twenty-Four Month Fuel Cycles

To support the change to twenty-four month operating cycles, it was necessary to revise the SFAS RCS Low Pressure trip analytical setpoint to 1515 psia. Table 15.4.2-1 lists the High Pressure Injection setpoint (which corresponds to the SFAS RCS Low Pressure trip) as 1600 psig. This change in the trip setpoint that actuates HPI does not affect the results of the analysis nor the environmental consequences of the accident.

The water assumed to exit the steam generator until isolation occurs is held at a constant radioactive concentration throughout the analysis. No credit is taken for the dilution of the RCS fluid that would occur when HPI begins to enter. Since the time to isolation of the steam generator is not affected, the same amount of radiation is released to the environment. Therefore, no change in the environmental consequences occurs.

##### 15.4.2.2.6.2 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. That evaluation concluded that since the replacement Steam Generator's tube inside

## Davis-Besse Unit 1 Updated Final Safety Analysis Report

diameter and length are unchanged, the leak flow rate is unaffected. Therefore, the existing analyses remain applicable with the replacement Steam Generators installed.

TABLE 15.4.2-1<sup>(1)</sup>

Steam Generator Tube Failure Parameters

Initial tube leak rate in affected steam generator, gpm	435
Leak rate in unaffected generator, gpm	1
Normal makeup rate, gpm	70
High pressure injection setpoint, psig	1600 <sup>(2)</sup>
Assumed defective fuel, %	1

<sup>(1)</sup>See Section 15.4.2.2.1 for the evaluation to support extended fuel cycles.

<sup>(2)</sup>See Section 15.4.2.2.6 for revised setpoint.

TABLE 15.4.2-2

Summary of Steam Generator Tube Failure Analysis

Low pressure trip occurs at	8 min.
High pressure injection automatically starts at	12 min. <sup>(1)</sup>
Operator takes action by initiating Reactor Coolant System cooldown and depressurization at	20 min.
Final isolation of affected steam generator is achieved at	34 min.
Initiation of Decay Heat Removal System is achieved at	184 min.
Volume of injection water required to compensate for reactor coolant leakage prior to affected steam generator isolation	1978 ft <sup>3</sup>
Steam venting time to the atmosphere from affected steam Generator	30 sec.
Steam vented to the atmosphere from affected steam Generator	18,667 lb.
Steam venting time from unaffected steam generator to atmosphere (through condenser)	145 min.
Total steam vented to atmosphere (through condenser) for unaffected steam generator prior to actuation of Decay Heat Removal System	286,000 lb.

<sup>(1)</sup>See Section 15.4.2.2.6 for evaluation of revised SFAS Low Pressure trip setpoint.

TABLE 15.4.2-3<sup>(1)</sup>

Resultant Doses From Steam Generator Tube Rupture

	Exclusion area boundary <u>0-2 hours</u>	LPZ boundary <u>0-30 days</u>
Thyroid dose, Rem	27.1	1.41
Whole body dose, Rem	0.23	0.012

<sup>(1)</sup>See Section 15.4.2.2.1 for the evaluation to support extended fuel cycles.

### 15.4.3 CRA Ejection Accident

#### 15.4.3.1 Identification of Causes

Reactivity excursions initiated by uncontrolled CRA withdrawal (Section 15.2) were shown to be safely terminated without harming the reactor core or the integrity of the Reactor Coolant System. In order for reactivity to be added to the core at a more rapid rate, physical failure of a pressure barrier component in the Control Rod Drive Assembly must occur. Such a failure could cause a pressure differential to act on a Control Rod Assembly and rapidly eject the CRA from the core region. The power excursion due to the rapid increase in reactivity is limited by the Doppler effect and terminated by reactor protection system trips. No operator action is required.

Since Control Rod Assemblies are used to control load variations and boron dilution is used to compensate for fuel depletion, only a few Control Rod Assemblies are inserted (some only partially) at rated power level. Thus, the severity of a CRA ejection accident is inherently limited because the amount of reactivity available in the form of CRA worth is relatively small.

#### Accident Bases:

Using an analytical method based on diffusion theory, the worth of the most reactive Control Rod Assembly in each CRA group was determined for different Control Rod Assembly configurations. The maximum CRA worths and other important parameters used in the study are shown in Table 15.4.3-1. The tripped CRA worth corresponds to the minimum worth available with the maximum-worth CRA stuck out at BOL and EOL.

The severity of the CRA ejection accident depends on the worth of the ejected CRA and the reactor power level. The Control Rod Assembly group of greatest worth is the first in the entire CRA pattern to be withdrawn. The maximum worth of a CRA in this group can be as high as 1.3 percent  $\Delta k/k$ , but a CRA would have this worth only when the reactor is subcritical. The details of the Control Rod Assembly worth calculations are presented in Chapter 4, and the methods of selecting the number of CRA's in each group are presented in Chapter 7.

When the reactor is subcritical, the boron concentration is maintained at a level that ensures that the reactor is at least 1% subcritical with the CRA of greatest worth fully withdrawn from the core. Thus, a CRA ejection will not cause a nuclear excursion when the reactor is subcritical and all the other CRA's are in the core.

A detailed analysis has been performed at power (2772 MWt) and zero power for CRA worths from 0.2 to 0.7 percent  $\Delta k/k$ .

A maximum CRA worth of 0.65 percent  $\Delta k/k$  at power (2772 MWt) has been considered as a limiting value to demonstrate the inherent ability of the system to safely terminate this postulated transient.

A CRA must be fully inserted in the core to have the greatest reactivity worth value. Assuming that the failure occurs so that the pressure barrier no longer offers any restriction to the ejection and that there is no viscous drag force limiting the rate of ejection, the CRA travel time to the top of the active region of the core is calculated to be 0.176 second. Since most of the reactivity is added during the central 75% of this travel, only this distance is used in the analysis, resulting in an ejection time of 0.15 second for the analysis.

Fuel Rod Damage:

The consequences of a CRA ejection accident depend largely on the rate at which the thermal energy resulting from the nuclear excursion is released to the coolant. If the fuel rods remain intact while the excursion is being terminated by the negative Doppler coefficient and by reactor trip, then the energy release rate is limited by a relatively low surface-to-volume ratio for heat transfer. The energy stored in the fuel rods will then be gradually released to the coolant (over a period of several seconds) at a rate that poses no threat to the integrity of the Reactor Coolant System. However, if the magnitude of the nuclear excursion is so great that the fuel rod cladding does not remain intact, then both fuel and cladding may be dispersed into the coolant to such an extent that the heat transfer rate increases significantly.

Power excursions caused by reactivity disturbances of the order of magnitude occurring in CRA ejection accidents could lead to three potential modes of fuel rod failure. Failure by the first mode occurs when internal pressures developed in the fuel rod are insufficient to cause cladding rupture, but subsequent heat transfer fuel to cladding raises the temperature of the cladding and weakens it until local failure occurs. Departure from nucleate boiling (DNB) usually accompanies and contributes to this mode of failure, and little or no fuel melting would be expected. In this mode of failure, fuel fragmentation is usually only minor, and any dispersal of fuel to the coolant would occur very gradually; system contamination would be the worst probable consequence.

The second mode of failure occurs when significant fuel melting causes a rapid increase in internal fuel rod pressure which, combined with a loss of cladding strength at higher temperatures, causes the fuel rod cladding to rupture (ref. 6). Some fuel vaporization may occur, contributing to the pressure buildup. Considerable fragmentation and dispersal of the fuel would be expected in this mode.

The third and most serious mode of fuel rod failure is the occurrence of extensive fuel melting and subsequent vaporization due to a very large and rapid reactivity transient in which there is insufficient time for heat to be transferred from the fuel to the cladding. In this mode, destructive internal pressures can be generated without increasing cladding temperatures significantly.

In evaluating the effects of these modes of failure, two failure thresholds are considered. The first, associated with a gradual and usually minor cladding failure, may be defined approximately by the minimum heat flux for DNB at the cladding surface. The second failure threshold, defined as the enthalpy threshold for prompt fuel failure with significant fragmentation and dispersal of fuel and cladding into the coolant, is used to describe the energy required to cause failure by either the second or third mode of failure.

A correlation of the results of different experiments conducted on Zircaloy-2-clad UO<sub>2</sub> fuel rods at TREAT (ref. 7) has been interpreted by the experimenters to show a threshold at 280 cal/g of fission energy input. That is, below this value the fuel rod can be expected to remain intact, and above this value fragmentation can be expected. The enthalpy corresponding to the melting point of UO<sub>2</sub> is about 260 cal/g, (ref. 8) and the heat of fusion is at least 78 cal/g (ref. 9). Thus, the 280 cal/g represents a condition in which only part of the fuel is molten. Also of interest as a probable indication of the degree and rapidity of fuel and cladding dispersal are the measurement of pressure rise rates in the autoclave in the TREAT experiments (ref. 7). Preliminary analysis indicates that there is only a modest pressure rise up to an energy input of 400 cal/g. Above 500 cal/g, however, there is a very definite pressure pulse. Thus, between 400 and 500 cal/g there is a transition, which probably corresponds to the change from the second to the third failure mode discussed previously.

A fuel failure threshold of 280 cal/g, at the pellet radius corresponding to the average temperature of the hottest fuel pellet, has been used in this study to define the extent of fuel failure.

In computing the average enthalpy of the hottest fuel pellet during the excursion for the rated power (2772 MWt) cases, it is assumed that no heat is transferred from the fuel rod between the time the accident is initiated and the time when the neutron power returns to the rated power level (2772 MWt). For the zero-power cases, the enthalpy increase was based on the peak value of the average fuel temperature. In all cases the average enthalpy rise from the integrated energy or the fuel temperature traces is multiplied by the maximum peaking factor to obtain the enthalpy increase in the hottest fuel pellet.

The latest correlation of the ANL TREAT (ref. 7) data from the meltdown experiments on Zircaloy-2-clad UO<sub>2</sub> fuel rods shows the threshold for the zirconium-water reaction to be 210 to 220 cal/g energy input. A conservative threshold value of 210 cal/g is used in this study.

In calculating the volume of the core experiencing burnout in a given CRA ejection accident, it is assumed that any DNB condition results in burnout for each rod where the DNB occurs. DNB in a CRA ejection transient is assumed to occur whenever the peak thermal power of a given fuel rod exceeds the peak at steady-state conditions that could result in a DNB, which in turn is assumed to occur for a DNBR of 1.3 using the W-3 correlation.

In determining the environmental consequences from this accident, an even more conservative approach is taken in computing the extent of DNB experienced in the core. All fuel rods that undergo DNB to any extent are assumed to experience cladding failure with subsequent release of all the gap activity. Actually, most of the fuel rods will recover from DNB, and no fission product release will occur. The fuel rods that experience DNB at BOL are assumed to have EOL gap activities.

#### 15.4.3.2 Accident Analysis

##### 15.4.3.2.1 Safety Evaluation Criteria

The safety evaluation criteria for this assumed accident are:

- a. The effects of a Control Rod Assembly ejection accident shall not further violate the Reactor Coolant System integrity.
- b. The resultant doses shall not exceed 10CFR100 limits.

##### 15.4.3.2.2 Methods of Analysis

A B&W digital computer program has been used to analyze the CRA ejection accident. This program agrees to within a few percent in all cases with CHICKIN (reference 10). The core heat transfer model allows for up to 30 radial mesh points in the fuel and clad, and the mesh size can be different in the two regions. The model accounts for the gap conductivity and film coefficient of heat transfer. Reactivity feedback is calculated in each mesh point and in the coolant and is weighted for inclusion in the kinetics simulation. The thermal properties are input separately for each mesh point but remain constant with time. The loop model includes a simulation of the steam generator which can have a nonlinear heat demand input on the secondary side. Trip action is initiated on high or low Reactor Coolant System pressure or on high neutron flux.

Decay heat can be taken into account as well. This code was used to calculate the neutron and thermal power, integrated energy, reactivity components, pressure, and fuel rod and loop temperatures. Six delayed neutron groups are considered. The control rod trip is represented by a multi-segment curve of reactivity insertion during trip versus time, obtained by combining the actual CRA worth curve with a CRA velocity curve. Nominal values for the various nuclear and physical parameters used as inputs are listed in Table 15.4.3-2.

As a check on the point kinetics calculation, the CRA ejection accident was also analyzed for a limited number of cases in support of the Technical Specification CRA worth using the two-dimensional, space-and-time dependent TWIGL digital computer program (reference 11). The point kinetics model assumes that the flux shape remains constant during a transient. This flux shape contains peaking factors which reflect unusual CRA patterns such as the flux adjacent to a position where a high worth CRA has been removed. Therefore, these point kinetics peaking factors are much higher than any that would actually occur in the core during normal operation. The purpose of using an exact space-time calculation is to find the flux shape during a transient. However, a transient wherein a CRA is ejected from the core must necessarily start with a flux shape that is depressed in the region of the ejected CRA. In fact, the higher the worth of the CRA, the more severe becomes the depression. This flux depression also causes a fuel temperature depression. When the CRA is ejected from this position, the flux quickly assumes a shape that shows some local peaking.

However, when this "exact" peaking is applied to a region initially at depressed fuel temperatures, as it is in the case of the regions adjacent to the ejected CRA, the resultant energy deposited in these regions causes a lower peak temperature and peak thermal power than does applying an arbitrary maximum peaking factor to an undepressed peak power region. The results from TWIGL were used to calculate the maximum total energy deposited in each region of the core following a CRA ejection; the highest energy is reported in Table 15.4.3-3. The result is that the hottest region simulated in the TWIGL code actually undergoes a less severe transient than the hottest fuel rod assumed in the point kinetics model. As seen in Table 15.4.3-3 this result is uniformly true for all CRA worths.

#### 15.4.3.2.3 Results of Analysis

##### Zero Power Level:

The nominal BOL and EOL CRA ejection analysis was performed at  $10^{-3}$  of power (2772 MWt), and the results can be seen in Table 15.4.3-4. No DNB and no fuel damage would result from these transients.

A sensitivity analysis has been performed around these two cases in which the Doppler and moderator coefficients, trip delay time, and CRA worth were varied. Figure 15.4.3-1 shows the peak neutron power as a function of ejected CRA worth from 0.2 to 0.7 percent  $\Delta k/k$ . The curve shows two distinct parts corresponding to worths less than and values near to and greater than  $\beta$ . Figure 15.4.3-2 shows the corresponding results for the peak thermal power. It is seen that for CRA worth values near prompt critical, the period is small enough to carry the transient through the high neutron flux trip. For lower values the pressure trip is relied on. No DNB occurs for any of these parameter variations.

Figure 15.4.3-3 shows that the peak enthalpy in the fuel for the CRA worths in the range being evaluated never exceeds 80 cal/g. Therefore, no threshold for damage is approached.



Figures 15.4.3-4 and 15.4.3-5 show the peak neutron and thermal power as a function of Doppler coefficient from  $-0.8$  to  $-1.9 \times 10^{-5} (\Delta k/k)/^{\circ}\text{F}$ . It is seen that the variation is relatively small. Similar results are shown in Figures 15.4.3-6 and 15.4.3-7 for the variation of the moderator coefficient from  $-4.0 \times 10^{-4}$  to  $2.0 \times 10^{-4} (\Delta k/k)/^{\circ}\text{F}$ . The slope of the curve for  $10^{-3}$  power (2772 MWt) at BOL is the greatest slope of any of the four curves because this case relies on the pressure trip, which makes it a longer transient. Figure 15.4.3-8 shows the effect of the trip delay time on the peak thermal power. It is seen that there is very little effect.

#### Rated Power:

An analysis was performed for a BOL CRA ejection at power (2772 MWt). The results of this analysis are shown in Table 15.4.3-4. A sensitivity study was made around this case and around the same CRA worth at EOL. Figures 15.4.3-1 through 15.4.3-8 show these results.

As seen in Figure 15.4.3-2, the peak thermal power shows relatively little change with increased CRA worth. The peak neutron power in Figure 15.4.3-1 does show a marked change with increased worths, but the thermal effect is small because the transients are rapidly terminated by the Doppler effect. As further evidence of this small thermal effect, the peak fuel enthalpies are given in Figure 15.4.3-3. The threshold for the zirconium-water reaction is not reached until values of BOL and EOL ejected CRA worths are above any that are considered feasible. The effects on the peak neutron and thermal powers of varying the Doppler and moderator coefficients and trip delay time are shown in Figures 15.4.3-4 and 15.4.3-8.

The results of the DNB calculation for BOL are shown in Figure 15.4.3-9. For the BOL analysis, ejection of the maximum CRA worth of 0.65 percent  $\Delta k/k$  at rated power (2772 MWt) results in 45% of the pins in DNB.

#### 15.4.3.2.4 Energy Required to Produce Further Reactor Coolant System Damage

The reactor vessel has been analyzed to estimate the margin that exists between the CRA worths assumed for the calculated CRA ejection accident transients and those worths that could initiate reactor coolant system failure. The pressure vessel material is SA-533 Grade-B steel. Table 15.4.3-5 lists the values used in this analysis. The radial deformation assumed to represent failure of the vessel is 50% of the total elongation, or 0.13 in./in.

To calculate the weight of an explosive charge required to reach 50% elongation, the vessel was simulated by a single cylinder with the same OD as the actual vessel, but with an increased thickness to account for the thermal shield and core barrel.

Using the formula for the equivalent vessel, the required weight of explosive charge was calculated. The results indicate that 1410 pounds of TNT would strain the mid-meridian ring to the 50%  $\mu$ , i.e., 0.13 in./in. The 1410 pounds of TNT have an energy equivalent of  $6.74 \times 10^8$  cal.

Ejected CRA worths higher than those reported in the preceding sections were analyzed to estimate the transient required to obtain an energy release equivalent to 1410 pounds of TNT. These cases were evaluated to find the amounts of fuel melting and zirconium-water reaction. Using the conservative assumption that all the fuel that exceeds the melting threshold is fragmented, dispersed into the coolant, and quenched to the coolant average temperature, a total thermal energy release can be determined. The conversion of this energy release to an equivalent deformation energy is dependent upon the duration of the release. TNT has an energy release in microseconds and a deformation conversion efficiency of about 50%. The

energy generated during a reactor transient from the zirconium-water reaction and a molten fuel dispersal is in the range from milliseconds to seconds. Thus, the conversion efficiency to deformation energy would be considerably less and is assumed to be 1/5 that of TNT (reference 13). Using these figures, the reactor vessel's capability is  $3.37 \times 10^8$  cal, and under the foregoing assumptions, a reactivity addition of 1.52 percent  $\Delta k/k$  is required to release energy necessary to cause deformation of the vessel.

#### 15.4.3.2.5 Conclusions

The hypothetical CRA ejection accident has been investigated in detail at two different initial reactor power levels: nominal power (2772 MWt) and zero power; both BOL and EOL conditions were considered. The results of the analysis prove that the reactivity transient resulting from this accident will be limited by the Doppler effect and terminated by the Reactor Protection System with no serious core damage or additional loss of the coolant system integrity. Furthermore, it has been shown that an ejected CRA worth greater than 1.52 percent  $\Delta k/k$  would be required to cause a pressure pulse, due to prompt dispersal of fragmented fuel and zirconium-water reaction, of sufficient magnitude to cause rupture of the pressure vessel, whereas the maximum CRA worth shown in Table 15.4.3-1 is about a factor of 2 less.

As a result of the postulated pressure housing failure associated with the accident (Subsection 15.4.3.1), the reactor coolant is lost from the system. The rate of mass and energy input to the Containment Vessel is considerably lower than that subsequently reported for the smallest rupture size considered in the loss-of-coolant analysis (Chapter 6). The maximum diameter hole size resulting from a CRA ejection is approximately 2.76 inches. This lower rate of energy input results in a much lower containment vessel pressure than those obtained for any rupture sizes considered in this loss-of-coolant accident.

##### 15.4.3.2.5.1 Partial Coolant Flow Condition

For partial flow operation, two ejected CRA worths were analyzed at nominal (2772 MWt) and zero power.

The results of the 0.65% ejected CRA worth case show peak thermal power values of 96% and 126% for two and three pumps, respectively. Calculations for percent pins in DNB and peak enthalpy of hottest fuel rod were made and show that the CRA ejection protection criteria was not exceeded. The worst case was for three pumps and its results showed 6% of the fuel pins were in DNB and the peak enthalpy of the hottest fuel rod was 194 cal/gm.

##### 15.4.3.2.6 Environmental Consequences

The environmental consequences of this accident are calculated by conservatively assuming that all fuel rods undergoing DNB release all of their gap activity to the reactor coolant. Just the activity in the gap is released from the fuel assembly since only the DNB limits are exceeded, and the worst possible consequence of exceeding DNB limits is possible cladding defects. The fuel rods in DNB are calculated for the ejection of the maximum CRA worth at BOL from nominal power (2772 MWt). Subsequently this gap activity and the activity in the reactor coolant from operation with 1% defective fuel is released.

After the CRA ejection occurs (causing a 2.76-inch Reactor Coolant System rupture to the Containment Vessel), the reactor coolant will undergo a subcooled expansion for approximately 50 seconds. As the Reactor Coolant System continues to depressurize, the reactor coolant will be at saturation temperature corresponding to the reactor coolant pressure. When the Reactor

Coolant System pressure falls below the setpoint of the turbine bypass valves, the secondary system can be isolated. Moody leak flow rate tables were used to determine the reactor coolant to secondary system leakage during the accident. Using these tables and nominal power (2772 MWt) conditions, a leak flow area was calculated for the 1 gpm leak rate. This area and the Moody tables were then used to determine the reactor coolant to secondary leakage during the Reactor Coolant System depressurization. It is conservatively assumed that choked flow in the leak flow area did not occur, and that the steam generator pressure was low enough to allow critical flow during the accident. This yields a maximum leak flow rate. It is also conservatively assumed that the friction factor associated with the leak flow area was equal to 1.0, and that the discharge coefficient was equal to 1.0. Using these assumptions, it is calculated that 5 gallons is released to the atmosphere from the condenser. A gas-to-liquid partition factor of  $10^{-4}$  is assumed for the iodine in the condenser, (refs. 1 and 2) but the noble gases are assumed to be released directly to the atmosphere.

All reactor coolant that is not released to the secondary system is released to the Containment Vessel. Fifty percent of the iodine released to the Containment Vessel is assumed to plate out.

Fission product activities for this accident are calculated using the methods discussed in Chapter 11. Doses resulting from this accident were evaluated using the environmental models and dose rate calculational methods discussed in the section on the loss-of-coolant accident. Table 15.4.3-6 shows the resulting thyroid and whole body doses for a 2-hour exposure at the exclusion distance and for a 30-day exposure at the low population distance, which include the dose contribution due to the activity released to the atmosphere via the secondary system and that released via containment vessel leakage. Activity released due to normal operation within the Technical Specification Limits were not considered in this accident analysis and are considered to be negligible. The doses resulting from the accident are well below the guideline values of 10CFR100.

#### 15.4.3.2.7 Additional Analyses

##### Moderator Coefficient Evaluation

As part of the above analyses, sensitivity studies were performed with moderator coefficients as negative as  $-4.0 \times 10^{-4} \Delta k/k/^\circ F$  at both Hot Full Power (HFP) and Hot Zero Power (HZP) conditions (Reference 31). As expected, these studies demonstrated that the Control Rod Ejection event is less severe as the moderator coefficient becomes more negative (see Figures 15.4.3-6 and 15.4.3-7). Therefore, a Control Rod Ejection event occurring with a moderator coefficient of  $-4.0 \times 10^{-4} \Delta k/k/^\circ F$  at either HFP and HZP conditions will continue to meet the Safety Evaluation Criteria of Section 15.4.3.2.1.

##### Control Rod Ejection Re-analysis

A re-analysis of the Control Rod Ejection Accident (CREA) was performed (See reference 40) using USAR methodology inputs and verified with new methodology, by use of the RELAP5 code. The RELAP5 methodology was within 3% of the USAR's old KAPP code methodology in calculating the resulting effective Full-Power-Seconds (FPS) for the CREA. Using the acceptance criterion of a fuel enthalpy of 210 cal/gm (threshold value for Zirconium-water reaction) the maximum total peak would be 3.43, which is higher than the currently allowed design total peak of 2.96. Therefore, the CREA re-analysis bounds the design peak assumption. BWFC also verified that the 45% of fuel rod failures originally calculated as the consequence of the CREA and assumed in the offsite dose calculation is bounding. This is so because the original calculations were done with a very conservative point (0-D) kinetics model

in combination with an adiabatic fuel enthalpy calculation. A 3-D re-analysis with a stronger ejected rod assumption (0.8%  $\Delta k/k$ ) and a more realistic, non-adiabatic model yielded results of approximately 20% of the rods in DNB. The use of a higher total peaking factor (3.43) would result in a proportionately higher number of pins in DNB, but would be well below the 45% fuel rod failures assumed for the radiological dose calculations.

Extended Fuel Cycles

Additional evaluations were performed to support the extended fuel cycle. The assumptions used in the new evaluations are more conservative than the assumptions given in USAR section 15.4.3.2.6, which discusses the environmental consequences due to a control rod ejection accident. The original analysis provided in this section takes credit for the availability of the condenser to reduce the iodine releases (using a gas-liquid partition factor of 10,000) from the secondary side to the environment. This assumption is not as conservative as the NRC's SER because off-site power may not be available following this accident. That is, the evaluations performed by the NRC assumed that off-site power is not available for this accident.

The following assumptions, used in the new evaluations, are more conservative than the assumptions given in the USAR section 15.4.3.2.6.

1. The fuel rod gap activity is assumed to be 10% of the iodine and noble gas activity in the fuel.
2. No credit for iodine partitioning at the condenser is assumed. All the iodine and noble gas activity released to the secondary side is released to the environment.

These two assumptions are consistent with Regulatory Guide 1.77 and the NRC Safety Evaluation Report for Davis-Besse.

Furthermore, it is conservatively assumed that 100% of noble gas gap activity and 50% of the iodine gap activity from the fuel rods reaching DNB are available for release from the containment simultaneous with the rod ejection accident. This assumption is conservative because the size of the opening in the RCS due to the ejected rod is very small (2.76 inch diameter) and considerable time would be required to release all the activity to the containment.

Using the above assumptions, the total dose via both release (i.e., steam generators and containment) pathways are as follows:

	<u>Thyroid</u>	<u>Whole Body</u>
Exclusion Area Boundary dose (0-2 hr.)	47 Rem	0.3 Rem
Low Population Zone dose (30 days)	5 Rem	0.03 Rem

The USAR acceptance criterion for this accident's calculated doses requires that they be less than 10CFR100 guidelines. The acceptance criterion for both the NRC SER and the Standard Review Plan (SRP) requires that the calculated doses (at the exclusion area boundary and the Low Population Zone) be well within the guidelines of 10CFR100 (i.e., 25% of 10CFR100 guidelines, or 75 Rem Thyroid, 6 Rem Whole Body Dose). Since the calculated doses are less than 25% of 10CFR100 guidelines, the USAR, the NRC SER, as well as the SRP acceptance criteria are satisfied.

Reanalysis of Control Rod Assembly Ejection Accident for Mark-B-HTP Fuel Assemblies

Implementation of the Mark-B-HTP fuel assembly design commenced with fuel cycle 15. The BHTP critical heat flux correlation is utilized to predict DNBR for the Mark-B-HTP fuel assemblies. This correlation is applicable to fuel assemblies containing the M5™ HTP spacer grids. Because of the fuel assembly design change, a re-analysis of the Control Rod Assembly (CRA) Ejection Accident was performed to incorporate the BHTP critical heat flux correlation. The reanalysis was performed in accordance with the safety criteria and analysis methodology defined and/or referenced by Reference 23.

The acceptance criteria for the CRA ejection accident are, (1) a fuel enthalpy of less than 210 cal/gm (threshold value for zirconium-water reaction) and, (2) a total core fuel rod failure of less than 45 percent. A fuel rod failure rate of 45 percent of the total number of fuel rods in the core is assumed by the offsite dose calculation.

A NEMO-K (3-D kinetics model) reanalysis was performed using the design total peak of 2.97, an ejected rod worth of 0.65 percent  $\Delta k/k$ , and a non-adiabatic enthalpy model. Results of the reanalysis indicate that approximately 41 percent of the rods achieve DNB and the fuel enthalpy is less than 210 cal/gm during the accident. Therefore, the acceptance criteria of this accident are satisfied.

15.4.3.2.8 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. The steam generators play only a minor role in this event, supporting the cooldown and depressurization of the Reactor Coolant System. The evaluation concluded that the replacement Steam Generator design is essentially identical to the original Steam Generator design for these purposes. Therefore, the existing analyses remain applicable with the replacement Steam Generators installed.

TABLE 15.4.3-1

Control Rod Assembly Ejection Accident Parameters

Maximum worth of ejected CRA, % $\Delta k/k$	0.65
CRA ejection time, sec.	0.15
Rated power level, MWt	2772 (See Note 1)
Reactor trip delay time	
High flux trip, sec.	0.4
High-pressure trip, sec.	0.6
CRA drive trip time to 2/3 insertion, sec.	1.4

(1) The reanalysis described in USAR Section 15.4.3.2.7 and in Reference 40 was based on 102% of 2772 MWt.

TABLE 15.4.3-2

Nominal Values of Input Parameters for CRA Ejection Accident Analysis (See Note 2, 4)

	<u>BOL</u>	<u>EOL</u>
Delayed neutron fraction, $B_{eff}$	0.00689	0.00516
Neutron lifetime, msec.	34.6	33.0
Moderator coefficient, $(\Delta k/k)/^{\circ}F$	$0.13 \times 10^{-4}$	$-3.0 \times 10^{-4}$ (see Note 1)
Doppler coefficient, $(\Delta k/k)/^{\circ}F$	$-1.28 \times 10^{-5}$	$-1.45 \times 10^{-5}$
Reactor coolant inlet temperature, $^{\circ}F$	555.2	555.2
Initial system pressure, psia	2200	2200
Total nuclear peaking factor, $F_q$	2.89	2.89 (see Note 3)
Average fuel temperature of average pellet, $^{\circ}F$	1200	1304
Average fuel temperature of hottest pellet, $^{\circ}F$	2400	2490

- (1) Sensitivity studies have shown that a moderator coefficient of  $-4.0 \times 10^{-4} \Delta k/k/^{\circ}F$  at End of Life (EOL) yields acceptable results (see Section 15.4.3.2.7).
- (2) Reference 40 lists the nominal values of Input Parameters for the re-analysis of the CRA Ejection Accident.
- (3) A total nuclear peaking factor of 3.43 has been shown in a re-analysis of the Control Rod Ejection Accident (reference 40) to yield acceptable results (See Section 15.4.3.2.7)
- (4) Input parameters associated with an EOC  $T_{AVE}$  reduction maneuver are analyzed and verified acceptable and results are included in the associated cycle reload report. (Reference USAR Appendix 4B)

TABLE 15.4.3-3

Comparison of Space-Dependent and Point Kinetics Results  
Of Fuel Enthalpy (based on 2772 MWt)

	Ejected CRA worth <u>% <math>\Delta k/k</math></u>	<u>Peak-to-average values</u>		<u>Fuel enthalpy cal / g</u>	
		<u>TWIGL</u>	<u>Point kinetics</u>	<u>TWIGL</u>	<u>Point kinetics</u>
BOL rated power	0.38	3.04	3.24	125	150
	0.83	2.67	3.24	174	225
BOL zero power	0.56	4.1	3.24	38	60
	0.83	4.4	3.24	48	71

TABLE 15.4.3-4

Summary of Control Rod Assembly Ejection Accident Analysis (based on 2772 MWt)

Initial power level, <u>% rated power</u>	Ejected CRA worth, <u>% <math>\Delta k/k</math></u>	<u>Peak power, % rated power</u>	
		<u>Neutron</u>	<u>Thermal</u>
0.1 (BOL)	0.65	76	63
0.1 (EOL)	0.65	982	41
0.1 (BOL)	1.0	6,128	156
0.1 (EOL)	1.0	13,612	149
100.0 (BOL)	0.65	702	165
100.0 (EOL)	0.65	1,545	148

Percent of fuel rods in DNB due to ejection of  
a 0.65 %  $\Delta k/k$  CRA worth at 100% power  
BOL, %

45

Reactor coolant to secondary leakage during  
reactor coolant system depressurization,  
gallons

5

NOTE: The reanalysis described in USAR Section 15.4.3.2.7 and in Reference 40 was based on 102% of 2772 MWt.

TABLE 15.4.3-5

Reactor Vessel Parameters

Vessel temperature, °F	600
Yield strength (0.2% offset), psi	55,000
Ultimate strength, psi	80,000
Ultimate strain ( $\epsilon_u$ ), %	26
Strain energy ( $E_s$ ) per unit volume up to Strain equal to 1/2 ultimate strain, In.-lb/in. <sup>3</sup>	8,000
Strain energy ( $E_s$ ) per unit volume up to Ultimate strain, in.-lb/in. <sup>3</sup>	17,000
Equivalent pressure vessel dimensions	
OD, in.	188.25
ID, in.	166.69
Thickness, in.	10.78

The expression <sup>(12)</sup> used for the weight of explosive required to strain the vessel a given amount is

0.811

$$W = \left[ \frac{1.407E_e (3.41 + 0.117R_i / t) (R_e^2 - R_i^2) 1.85}{10^5 w^{-0.85} (1.47 + 0.0373R_i / t)^{0.15} (R_i) 0.15} \right]$$

where

- W = charge weight (TNT or Pentolite), lb
- w = weight density of vessel material, lb/ft<sup>3</sup>,
- R<sup>i</sup> = initial internal radius of vessel, ft,
- R<sup>e</sup> = initial external radius of vessel, ft,
- t = initial wall thickness of vessel wall, ft,
- E = wall strain energy, in.-lb/in.<sup>3</sup>.



TABLE 15.4.3-6

Resultant Doses From a CRA Ejection Accident<sup>(1)</sup>

	Exclusion area boundary <u>0-2 hours</u>	LPZ boundary <u>0-30 days</u>
Thyroid dose, Rem	1.36	0.254
Whole body dose, Rem	$1.14 \times 10^{-2}$	$4.75 \times 10^{-3}$

<sup>(1)</sup>See Section 15.4.3.2.7 to support extended fuel cycles.

#### 15.4.4 Steam Line Break

##### 15.4.4.1 Identification of Causes

The loss of secondary coolant due to a failure of a steam line between the steam generator and the turbine causes a decrease in steam pressure and thus places a demand on the control system for increased feedwater flow. Increased feedwater flow, accompanied by steam flow through the turbine stop valves and the break, lowers the average reactor coolant temperature and pressure. The reactor trips on low pressure or high flux, depending on the break size (see subsection 15.4.4.2.3.1). The operation of the Emergency Core Cooling System provides effective core cooling and the ultimate shutdown of the core through its boron addition.

Analyses have been performed to determine the effects and consequences of a loss of secondary coolant due to a double-ended steam line rupture between the containment vessel and the Main Steam Isolation Valves, since this location will maximize the radiation release to offsite. However, to evaluate the effect on the containment vessel, it was then assumed the energy associated with this accident was released to the containment vessel.

##### 15.4.4.2 Accident Analysis

###### 15.4.4.2.1 Safety Evaluation Criteria

The safety evaluation criteria for this accident are:

- a. The core shall remain intact for effective core cooling.
- b. Loss of reactor coolant boundary pressure integrity resulting from steam generator tube failure due to the loss of secondary side pressure and resultant temperature gradients shall not occur.
- c. Resultant doses shall not exceed 10CFR100 guideline values.

###### 15.4.4.2.2 Methods of Analysis

The rate of reactor system cooling following a steam line break accident is a function of the steam generator water inventory available for cooling. The unfouled inventory as a function of power is shown in Table 15.4.4-1. The largest inventory, at rated power, results in the greatest mass available for cooling. Thus, the fouled steam generator inventory as shown in Table 15.4.4-2 was used in this analysis.

Other conservative assumptions used in the analysis of the steam line failure accident are as follows:

- a. A minimum tripped CRA worth with the maximum worth CRA stuck out of the core of 3.5%  $\Delta k/k$  was used. This worth accounts for the moderator deficit, Doppler deficit and reduction in CRA worth to produce a 1%  $\Delta k/k$  subcritical margin at hot shutdown end-of-life core conditions. CRA worths greater than this minimum tripped CRA worth are shown in Chapter 4 to always be available even with the highest worth CRA stuck out.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

- b. Conservative end-of-life Doppler and moderator coefficients were used. The large negative values of these coefficients produce the greatest reactivity insertion due to the Reactor Coolant System cooldown resulting from the accident.
- c. The reactor is assumed to be operating at 102% (of 2772 MWt) power before the accident. Other parameters used in the analysis are summarized in Tables 15.4.4-1 and 15.4.4-2.
- d. Loss of off-site power was not assumed.
- e. No operator action is required to mitigate the accident to meet the acceptance criteria.
- f. The systems required to function during the transient are as follows:

<u>Function</u>	<u>System</u>
Reactor trip	RPS
SG isolation/turbine trip (main steam line and main feedwater lines)	SFRCS
Auxiliary feedwater initiation	SFRCS
High Pressure injection initiation	SFAS

- g. The steam line rupture accident was analyzed assuming a complete double-ended rupture of the largest steam line. As indicated in Table 15.4.4-1 the analysis was based on the rupture of a 33.9" ID pipe (36" OD). The results shown in Figure 15.4.4-1 are consistent with this pipe diameter.
- h. Credit is taken for the turbine stop valve closure. This provides a more reliable redundant means for main steam isolation of the unaffected steam generator. Stop valves of the stem-sealed type have been used on 50 mW and larger General Electric steam turbines since 1948. Over 300 valve-years of service on nuclear turbine stop valves have been accumulated without a known failure to close. Based on experience through December 1972, G.E. has predicted a valve sticking rate of 0.26 failures per million hours at a 50 percent confidence.

Incipient sticking conditions have been found only on high temperature fossil-fuel units and have been due to the accumulation of an oxide layer in the stem and bushing. Oxidation is not experienced at the relatively low temperature of water cooled nuclear reactor applications. In addition, G.E. turbine stop valves are not subject to failure due to silica build-up between the stems and bushings, since the stem-sealed design precludes the transport of steam-carried impurities back along the stem. Further, there is a reduced tendency for carryover from a once through steam generator. Periodic full-closure testing of the stop valves will disclose any sticking conditions, so that a shutdown could be made to make the necessary correction.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

- i. The low steam generator pressure SFRCS trip results in MSIV closure within 6 seconds.

Each main steam isolation valve is designed with the capability of closing within 5 seconds with steam flow in the normal direction and a differential pressure across one valve of 910 psi. This psi differential to the atmosphere is the maximum that would occur under rupture conditions.

The expected pressure drop across a main steam isolation valve is tabulated as follows:

<u>% power</u>	<u><math>\Delta P</math>, psi</u>
100	6.9
75	3.5
50	1.6

The maximum allowable pressure drop is 8 psig.

- j. The main steam line check or non-return valve is located in the turbine building and is not essential for safe shutdown of the plant.

The integrity of the non-return check valves due to a main steam line break upstream of the valve is addressed in subsection 10.3.3.

The non-return check valves are not required to function during accident conditions as supported by the steam line break evaluation of this section. Furthermore, the closure of the turbine stop valves will provide the same degree of mitigation of the blowdown of the unaffected SG.

- k. In the event of a main steam line rupture, it is required that the main feedwater stop valves, control valve and the startup valve which is on the bypass line of the control valve be closed. The closure of the control valve and startup valve is to backup the stop valve to insure that main feedwater is isolated. The need for the closure of the main feedwater line together with the closure of main steam isolation valve is to effectively isolate the affected steam generator.

The feedwater stop valve is Q-listed, while the feedwater control valve and startup control valve are not Q-listed.

The stop valve is designed to close in 17 seconds to achieve containment vessel isolation. [Note: 17 seconds was used in this analysis and is not the actual valve stroke time.]

The feedwater control valves and startup control valves have been designed to provide a rapid reduction in feedwater flow following a reactor trip. It is required that the flow reduction to the steam generators be approximately 3 to 4 percent of full flow within 6 to 10 seconds. To accomplish this criteria, the feedwater control valve is designed to close in a maximum of 7 seconds, and the startup control valve is designed to close in a maximum of 12 seconds.

The FLASH 2 digital computer program was used to determine the characteristics of this accident. The multinode model included a detailed description of both the Reactor Coolant System and the steam generator.

The model provides simulation of most of the secondary system valves including the main and startup feedwater valve, auxiliary feedwater valves, turbine bypass valves, code safety valves, main steam isolation valves, and turbine stop valves.

The model also includes energy balances for the principal steam generator components, the entire Reactor Coolant System (core, loops, and steam generator), and the pressurizer (with both mass and energy transfer). The reactor kinetics, trip logic and action, and a fuel pin simulation with Doppler and moderator temperature feedback are also features of the model.

#### 15.4.4.2.3 Results of Analysis

##### 15.4.4.2.3.1 Minor secondary pipe break:

Minor steam line breaks include all leak areas up to and including the largest steam line other than the main steam line.

A steam line rupture of small area causes a slow decrease in steam pressure. The reactor power will increase with decreasing average reactor coolant temperature as a result of the negative moderator coefficient. The ICS will then cause Control Rod Assembly insertion in an attempt to limit reactor power to 102% of 2772 MWt. A reactor trip occurs due to low reactor coolant pressure or high neutron flux. The high flux and low RC pressure trip functions ensure core protection over the entire steam line break spectrum. The time from rupture to reactor trip as a function of steam line break size for DB-1 is given in Figure 15.4.4-5 for break areas from 1.0 to 5.4 ft<sup>2</sup>. The reactor will trip on the shorter of the two trip times shown. Therefore, for break areas larger than about 1.75 ft<sup>2</sup>, the reactor trips on low reactor coolant pressure; and for break areas smaller than about 1.75 ft<sup>2</sup>, the reactor trips on high flux. Following reactor trip and turbine trip, the turbine stop valves close. The steam generator in the steam loop associated with the rupture blows dry after steam and feedwater isolation on low steam pressure. Decay heat is removed by the unaffected steam generator by steam flow through the turbine bypass system to the condenser. If condenser vacuum is lost, decay heat will be removed by steam relief through atmospheric vent valves and safety valves.

The results of the analysis for the maximum break size at nominal power (2772 MWt) are similar to those discussed above, however, the maximum break size represents the worst condition for a steam line rupture accident.

##### 15.4.4.2.3.2 Double-ended main steam line break:

Following a double-ended main steam line rupture between the steam generator and main steam isolation valve, both steam generators will blow down causing a Reactor Coolant System cooldown and depressurization. The reactor trips on low reactor coolant pressure. Low steam pressure (600 psia) initiates closure of the main steam isolation valves and closure of the main feedwater isolation (block) valves, startup, and main control valves on both steam generators.

The steam generator with the assumed break will continue to blow down after the main steam isolation valves and feedwater valves are closed. The steam pressure in the unaffected steam generator will increase allowing auxiliary feedwater flow to be initiated.

After feedwater isolation is achieved, the affected steam generator blows dry and auxiliary feedwater flows to the unaffected steam generator. With closure of the main steam isolation valves, the unaffected steam generator side will repressurize and open the code safety valves, allowing decay heat removal from the reactor system to continue with auxiliary feedwater flow and steam relief through the code safety valves (nine valves per steam line).

The High Pressure Injection System will be actuated during the cooldown period associated with a steam line failure. This system supplies borated water to the Reactor Coolant System to increase the shutdown margin when cooling down below 550°F. During the controlled cooling to atmospheric pressure the addition of boron to the reactor coolant will prevent criticality at lower temperatures. At temperatures above 550°F, no credit is taken for the negative reactivity inserted due to the HPI injection; credit is taken, however, for both the pressure effect and the cooling effect due to the HPI injection.

Low RC system pressure 1515 psia is the principal analytical setpoint for SFAS initiation following a steam line break both inside and outside containment. In the steam line break analysis described in this subsection (15.4.4.2.3.2) and the analysis described in subsection 15.4.4.2.6.6, the HPI system actuation is needed for RC system makeup and is initiated on low RC system pressure in both cases. The low RC system pressure actuation setpoint is reached about 10 seconds after rupture. The conservatism in the delay for the HPI system is discussed in subsection 15.4.4.2.6.6, item 6.

Figures 15.4.4-1 through 15.4.4-3 show the response of the Reactor Coolant System for a double-ended main steam line rupture. Initially, both steam generators blow down until a low reactor coolant pressure trip occurs. The reactor coolant temperature leaving the unaffected steam generator increases after the main steam isolation valves close as a result of the pressure recovery and a reduction of the feedwater flow. The temperature of the coolant leaving the affected steam generator decreases until the unit has blown dry, at which time the temperature approaches the inlet temperature. Since the unaffected steam generator main steam isolation valve is closed and the steam generator with the rupture is dry, the Reactor Coolant System temperature can only be lowered as a result of the steam flow from the isolated steam generator through the code safety valves. Eventually, thermal equilibrium is re-established; i.e., the heat removal rate (steam flow through the code safety valves) is equal to the heat input (core decay heat).

The maximum cooling rate occurs during the first 10 seconds of blowdown, with no resultant return to criticality and a minimum DNBR of 1.42 (W-3) (Table 15.4.4-3); therefore, no fuel damage will occur. There is no danger of the hot channel DNBR exceeding the minimum value of 1.3 for this transient since the reactor trips almost instantaneously while the RC flow remains at rated flow. Table 15.4.4-4 shows the resulting thyroid and whole body doses for a 2-hour exposure at the exclusion distance and for a 30-day exposure at the low population distance.

During the first minute following the break, the average tube temperature in the affected steam generator remains above the shell temperature. Since thermal equilibrium is established, the average reactor coolant temperature will remain near the saturated temperature corresponding to the pressure at which the main steam safety valves or atmospheric vent valve (after blocking SFRCS) is set. Therefore, the tube-to-shell temperature difference will approach zero except the tubes wetted by Auxiliary Feedwater Spray. For tubes wetted by Auxiliary Feedwater Spray the tube tensile load is 3379 pounds. The resultant tube stresses will remain less than the stresses corresponding to the design pressure and temperature conditions of the tubes as discussed in Chapter 5.

15.4.4.2.3.3 Containment Vessel pressure:

The resultant mass and energy releases for the fouled steam generator are shown in Table 15.4.4-2. The resultant increase in the containment pressure due to a 5.4 ft<sup>2</sup> steam line break is 21.4 psi. The temperature response of the containment vapor region to a main steam line break is shown in Figure 15.4.4-4 along with the surface temperature response of the hottest structural heat sink in the containment. Note that, while the containment vapor temperature exceeds the 264°F equipment qualification temperature, the heat sink (which is a thin steel slab) reaches a maximum surface temperature of only 220.4°F. This behavior is characteristic of equipment exposed to short-term temperature transients in a superheated vapor atmosphere. The major process tending to heat the equipment is the condensing heat transfer mechanism. The total heat transfer rate to equipment and structures can be described by the following relationship:

$$q = h_{\text{cond}} (T_{\text{sat}} - T_w) + h_{\text{conv}} (T_v - T_w) \quad (1)$$

where

$q$  = surface heat transfer rate

$h_{\text{cond}}$  = condensing heat transfer coefficient

$h_{\text{conv}}$  = convective heat transfer coefficient

$T_{\text{sat}}$  = steam saturation temperature at containment atmosphere  
steam partial pressure

$T_w$  = equipment surface temperature

$T_v$  = containment vapor temperature

The first term of Equation (1) becomes identically zero for  $T_w > T_{\text{sat}}$  since condensation heat transfer can occur only if the condensable vapor in the region of the condensing surface can be cooled below its saturation temperature.

The maximum value of  $T_{\text{sat}}$  that occurs in the transient described in Figure 15.4.4-4 is 224 degree°F. For surface temperatures above this maximum saturation temperature, only the second term of Equation (1) can act to Transfer heat to equipment. Since  $h_{\text{conv}}$  is generally only 1 to 2 percent of the value of  $h_{\text{cond}}$ , only long exposures to elevated superheated vapor temperatures can bring the equipment temperature above the maximum value of  $T_{\text{sat}}$ . Thus, for short-term superheated vapor transients such as those encountered in main steam line breaks, a practical equipment and structure maximum temperature is the saturation temperature at containment atmosphere steam pressure.

(A detailed discussion of this phenomenon can be found in Bechtel Topical Report BN-TOP-3, Rev. 3, submitted to the NRC in August 1975.)

#### 15.4.4.2.4 Environmental Consequences

The environmental consequences from this accident are calculated by assuming that:

- a. The unit has been operating with a 1-gpm steam generator tube leak in the affected steam generator.
- b. The unit has been operating with 1% defective fuel rods.
- c. The steam line break occurs between the Containment Vessel and the main steam isolation valve. All other rupture locations would result in lower doses.
- d. Reactor coolant leakage into the steam generator continues for 9.0 hours until the Reactor Coolant System cools down and the pressure differential disappears. A total of 540 gallons of reactor coolant is assumed to be released to the atmosphere.

The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater and the reactor coolant leakage. The iodine, primarily resulting from reactor coolant leakage in the cooldown period following the steam line break, is assumed to be released directly to the atmosphere. Based on these assumptions, the resultant doses from this accident are given in Table 15.4.4-4.

Beginning with cycle 5, the fuel cycle length was extended to 18 months. The plant Technical Specifications limit the RCS activity to a value which is significantly less than the iodine activity associated with 1% failed fuel, which was assumed in the original evaluation of this accident. Therefore the resultant doses presented in Table 15.4.4-4 and 15.4.4-4a bound the radiation doses for any fuel cycle. The plant Technical Specification limits for the RCS activity are based on the NRC evaluation as documented in the NRC SER. For this reason no additional evaluation was performed for this accident to support extended 24 month fuel cycles.

#### 15.4.4.2.5 Conclusions

This analysis has shown that the reactor trips and remains subcritical, the integrity of the steam generator is maintained, and the environmental doses are within acceptable limits.

#### 15.4.4.2.6 Additional Analyses

##### 15.4.4.2.6.1 Control Room Habitability

There are main steam lines that pass in close proximity to the control room and mechanical equipment room housing the control room ventilation equipment. All the walls of the mechanical equipment room have been designed to withstand the effects of a steam-line rupture. Access into the room has been provided with pressure-tight doors to keep out the steam atmosphere. Access into the control room is from the Turbine Building. The first access door, which is in close proximity to the main steam lines, opens into the elevator lobby. Two additional doors in series must be passed through before entry into the control room. The Turbine Building has a very large volume; therefore, no significant pressure buildup will result due to a steam-line rupture.



In addition, in order to prevent damage to the control room wall due to the whipping of either main steam line or jet impingement after a postulated rupture, the wall thickness has been increased by 12 inches. Direct jet impingement on the control room door from a break in the main steam line is impossible. Therefore, it is not considered credible that an appreciable amount of steam atmosphere can enter the control room to endanger habitability or safe shutdown of the station.

Area radiation monitors are provided for the control room which continuously give the background radiation level. In case of any abnormal increase in the background level, the operator can manually isolate the normal ventilation system and start the Control Room Emergency Ventilation System if needed. The control room doses following a main steam line break accident are less than those for the loss-of-coolant accident presented in Table 15.4.6-2.

#### 15.4.4.2.6.2 Partial Coolant Flow

The most severe steam line break occurs at rated power. For partial flow the cooldown will be slower, and the effects of steam line break less severe.

#### 15.4.4.2.6.3 Steam Line Break (inside containment) Dual S/G Blowdown

Any postulated single failure which results in the opening of a nonactuated atmospheric vent valve in the steam generator not supplying the broken steam line represents a passive failure in addition to the steam line rupture and is considered to be in excess of design requirements. However, an analysis of a double-ended rupture upstream of the main steam isolation valve has been performed assuming the failure of an atmospheric vent valve to close after actuation on the steam line connected to the unaffected steam generator. The extended opening of any one steam relief valve would not be expected to significantly alter the core thermal conditions presented above, since the relief capacities of individual valves have been sized to prevent such possibility.

With the unaffected steam generator isolated, ten valves are capable of relieving steam. One valve is an atmospheric vent valve having a capacity of 5% rated steam flow. The other nine are safety valves; seven of these safety valves are relieving approximately 7% of full power each; the remaining two are of a lesser capacity. The most severe postulated active failure would be the failure of the atmospheric vent valve to reseal after actuation. However, to demonstrate the capability of the system to accept an even larger steam relief, failure of a code safety valve to reseal will be assumed for this analysis.

The steam line break with dual SG blowdown analysis originally contained in the USAR contained an error with respect to calculated reactor subcriticality. A reanalysis (ref. 21) was performed with the TRAP2 computer code utilizing updated analytical techniques and assumptions, and is summarized as follows:

The major distinguishing assumption of this steam line break analysis is the failure of a main steam safety valve (MSSV) to reseal, remaining fully open. Other assumed parameters are listed in Table 15.4.4-6.

Following the double-ended rupture of the main steam line, low steam pressure (600 psia) on the affected steam line actuates SFRCS. This initiates closure of the main steam isolation valves (MSIVs), closure of the main feedwater stop valves, closure of the turbine stop valves, and initiates auxiliary feedwater to the unaffected steam generator (refer to Subsection 7.4.1). In the analyses, the SFRCS trip of the turbine causes a reactor trip via ARTS. At the time of the

break, main feedwater is assumed to instantly go to run-out flow, until reactor trip. After main feedwater isolation the affected steam generator blows dry.

After MSIV closure on the unaffected steam generator, pressure will rise causing opening of the MSSVs. Although this is when the assumed MSSV failure would occur, for simplicity and conservatism the valve is failed open at break initiation.

Figures 15.4.4-6 through 15.4.4-8 show the system response to this accident. The SFRCS-initiated ARTS trip (including delay time) occurs at 0.77 seconds after rupture. After closure of the MSIVs, unaffected cold leg temperatures increase as associated SG secondary pressure increases. The temperature of the affected cold legs decreases until the SG has blown dry at which time it increases to approximately the hot leg temperature. After affected SG blow down, and with auxiliary feedwater established to the unaffected SG, a steady RCS cooldown rate occurs as steam flow out the stuck open MSSV continues. Although steam flow rates (dependent on SG pressure) will decrease as core decay heat decreases, for conservatism, a constant bounding RCS cooldown rate was assumed.

With respect to minimum DNBR the reanalysis performed did not specifically recalculate DNBR ratios. The original analysis performed concluded minimum DNBR values occurred within five seconds of the steam line break, and were well above the limit of 1.3. Initial conditions and assumptions for the reanalysis are essentially identical during initial portions of the transient, and DNBR will be greater during the subsequent cooldown portion of the transient. As such, the previous conclusion regarding DNBR remains valid. Since this is the case, a plot of the heat flux as a function of the time during the transient is not included. The maximum fuel temperature occurs at the time of the rupture, and fuel temperature continually decreases throughout the transient.

During the initial portion of the transient shown, the minimum subcritical margin is 0.569%  $\Delta$  k/k calculated at 34 seconds after the break. Additionally, conservative extrapolation of TRAP2 results was performed to determine if continued MSSV cooling will cause an eventual return to criticality. The conservative extrapolation assumed continued AFW flow to the unaffected SG and operator action to throttle HPI flow to prevent complete filling of the primary system. The extrapolation estimated a return to criticality at approximately 28 minutes after the break.

Review of reanalysis results and comparison with other USAR analyzed steam line breaks indicate that all acceptance criteria for this accident are met. Namely, the reactor trips and remains subcritical, the integrity of the steam generator is maintained, and the environmental doses are within acceptable limits. Note that while conservative extrapolation indicates a return to criticality at 28 minutes after the break, the key assumption of continued AFW flow for 28 minutes is well beyond design-basis required limitations on operator actions, and, as such, these results indicate sufficient time exists to mitigate a main steam line break with concurrent failure of a MSSV on the unaffected steam generator.

#### 15.4.4.2.6.4 Minimum Reactivity Margin Evaluation

The minimum reactivity margins are shown in Table 15.4.4-5 for the following five main steam line break situations:

Case I – 102% of 2772 MWt. Break is inside containment (36" line). No offsite power.

Case II – 102% of 2772 MWt. Break is inside containment (36" line). Offsite power is available.

Case III – 102% of 2772 MWt. Break is outside containment and upstream of isolation valves. No offsite power.

Case IV – 102% of 2772 MWt. Break is outside containment and downstream of isolation valves. Offsite power is available.

Case V – Hot standby or low power operation.

Case II represents the steam line break situation presented in subsection 15.4.4.2.3. To readily assess the resultant effect of Case I on minimum reactivity margin, Case II was reanalyzed with loss of offsite power assumed to occur at reactor trip. As shown in Table 15.4.4-5, loss of offsite power would not significantly effect the result. Case III is identical to Case I since the only effect of the steam lines would be to slow down the Case I transient. The minimum subcritical margin for Case IV will be the same as that for Case II if a single failure of the main steam line isolation valve on the affected steam generator is assumed. Otherwise, the subcritical margin for Case IV will be greater than that for Case II since the length of the blowdown period is reduced for the affected steam generator. The subcritical margin for Case V will be larger than that for Case II since in both of these situations, the steam generator inventories are considerably lower than Case II (about 20,000 lbm, compared to 62,500 lbm). Thus, the overcooling of the primary system is reduced and the minimum subcritical margin will be increased.

In all cases analyzed, single failures have been used that lead to increased overcooling of the primary system. Single failures in the feedwater system and in the main steam system have been considered. The failure of the main feedwater isolation valve and of the turbine stop valve have been postulated in the analysis of the steam line break accident, Subsection 15.4.4.2.3. Both failures have been conservatively assumed to happen simultaneously. Failure of a steam relief valve has been studied in Subsection 15.4.4.2.6.3.

#### 15.4.4.2.6.5 Comparison of Controlling Parameters With and Without Offsite Power

A steam line rupture accompanied by a loss of offsite power is shown to have a subcritical margin slightly less than the case where offsite power is available (Table 15.4.4-5). Immediately upon loss of offsite power the reactor and the reactor coolant pumps trip. The decrease in the reactor coolant flow will inhibit the normal initial rapid cooling of the primary system (with offsite power) thus, decreasing the potential of a return to power. An immediate reactor trip from the loss of offsite power as opposed to the finite time required to trip the reactor on high flux or low pressure with offsite power, increases the potential for a return to power because the core has not increased its stored energy prior to reactor trip. An immediate reactor trip with loss of offsite power also results in less mass release since the feedwater control valves close with reactor trip. Therefore, the variation in the controlling parameters resulting from a steam line rupture with loss of offsite power will be similar to those parameters shown in Figure 15.4.4-1.

#### 15.4.4.2.6.6 Steam Line Break Concurrent with Operator Error Allowing Continued Feedwater

The occurrence of a steam line break accident concurrent with an operator error allowing feedwater to be admitted to the affected steam generator has also been investigated using the following assumptions:

1. The plant was initially operating at nominal power (2772 MWt).

Davis-Besse Unit 1 Updated Final Safety Analysis Report

2. Operator error allowing a conservative main feedwater pump runout to 135 percent of rated feedwater flow to the affected generator was assumed. This conservative approach bounds an operator error which would allow auxiliary feedwater to be admitted to the affected steam generator.
3. The maximum negative moderator coefficient corresponding to end-of-life conditions of the equilibrium cycle was used, although this is an instantaneous value occurring only at the last moment of operation before refueling.
4. The minimum tripped CRA worth corresponding to the Technical Specification limit for the minimum shutdown margin was used; this was a very conservative approach.
5. The maximum-worth CRA was assumed to stick out, although the combining of this assumption with 2 through 4 above is clearly an unrealistic approach.
6. The increased capacity of the High Pressure Injection System with decreased Reactor Coolant System pressure has been neglected; design flow rates have been used throughout the transient. Also, the High Pressure Injection System was assumed to have 35-second delay after the steam line break. The conservatism of this assumption is demonstrated by the sequence of events listed below:

Case I - Loss of offsite power at the instant of the break

<u>Sequence of Events</u>	<u>Elapsed Time</u>
Steam line break/loss of offsite power	0 sec
Diesel starts	0.5 sec
SFAS setpoint reached	10 sec
Diesel up to speed	10.5 sec
SFAS time delay	15 sec
Diesel sequence step 2/HPI pumps starts	20 sec
HPI pump accelerates to speed/HPI pump discharge valve opens	<u>30 sec</u>
Total Elapsed Time	30 sec
(Time after SFAS setpoint reached)	(20 sec)

## Davis-Besse Unit 1 Updated Final Safety Analysis Report

### Case II - No loss of offsite power

<u>Sequence of Events</u>	<u>Elapsed Time</u>
Steam line break	0 sec
SFAS setpoint reached	10 sec
SFAS time delay/HPI pump starts	15 sec
HPI pump accelerates to speed/HPI pump discharge valve opens	<u>25 sec</u>
Total Elapsed Time (Time after SFAS setpoint reached)	25 sec (15 sec)

7. The boron injection is assumed to be perfectly mixed with all the reactor coolant before entering the core, although the injection occurs at the reactor vessel inlet and so would have the highest concentration in the core region.
8. Perfect heat transfer is assumed in the affected steam generator after the initial part of the transient; that is, the time constant for heat transfer is zero with no stored energy accounted for.

The steam line rupture causes an increase in the heat transfer from the reactor coolant to the feedwater. As Figures 15.4.4-1 through 15.4.4-3 show, this initiates a cooldown of the Reactor Coolant System, such that the reactor trips on low pressure at about 1.13 sec after the rupture (includes a total trip delay of 0.6 second). A main steam pressure reduction to 600 psig trip point initiates an isolation signal that actuates valves isolating both the steam side and the feedwater side of both steam generators. For the cooldown part of the calculations, it is assumed that the main feedwater flow (at 135 percent of rated flow) continues to the affected steam generator. With the above assumptions, the resulting coolant system temperature decrease causes high pressure injection actuation at 35 seconds after the steam line break. This injection of boron will keep the core subcritical during cooldown below 550°F.

#### 15.4.4.2.6.7 Moderator Coefficient Evaluation

Although the Steam Line Break Event is initiated from Hot Full Power (HFP) conditions, it immediately produces a reactor trip which results in the reactor being at least one percent shutdown when Hot Zero Power (HZP) (532°F) conditions are reached. Therefore, since it is the continuing cooldown below HZP conditions that is of concern, the value of moderator coefficient at HZP and colder conditions will determine the reactor response to the Steam Line Break.

Although the original (double-ended rupture of 36 inch pipe diameter) Steam Line Break analyses for Davis-Besse Unit 1 used a constant moderator coefficient of  $-3.0 \times 10^{-4} \Delta k/k/^\circ F$  over all temperatures, later analyses (Reference 31) performed using the TRAP2 computer code instead employed a more realistic reactivity-versus-moderator temperature model, which accounted for both moderator and Doppler effects. Evaluating this reactivity-versus-moderator temperature function produced an average temperature coefficient (combination of moderator and Doppler coefficients) of  $-3.1 \times 10^{-4} \Delta k/k/^\circ F$  over the range of temperatures from HZP down to the lowest RCS temperature that occurred during the Steam Line Break transient. In other words, if a constant temperature coefficient of  $-3.1 \times 10^{-4} \Delta k/k/^\circ F$  were used for all temperatures

below HZP, the resulting reactivity insertion would be identical to that actually calculated in the TRAP2 analyses. Further, since the Doppler coefficient used in the Steam Line Break event was  $-0.177 \times 10^{-4} \Delta k/k/^\circ F$ , the resulting moderator coefficient of  $-2.923 \times 10^{-4} \Delta k/k/^\circ F$  is bounded by the constant value of  $-3.0 \times 10^{-4} \Delta k/k/^\circ F$  assumed in the original analysis. Therefore, the later analyses using TRAP2 are bounded, in terms of reactor response, by the original Steam Line Break analysis, and the temperature coefficient of  $-3.1 \times 10^{-4} \Delta k/k/^\circ F$  for HZP conditions and colder can be considered to be bounding for the Steam Line Break event.

It should be noted that this value of temperature coefficient assumes temperatures of HZP and colder and also assumes that all control rods are fully inserted in the core with the exception of the maximum worth stuck rod, which is fully withdrawn. Limiting moderator coefficients for HFP all rods out conditions may be determined from this temperature coefficient of  $-3.1 \times 10^{-4} \Delta k/k/^\circ F$ , but must account for the differences in temperatures and rod configuration.

#### 15.4.4.2.6.8 Reanalysis of Steam Line Break in Containment

The Containment Vessel's response during a Main Steam Line Break was reanalyzed with mass and energy release data generated by RELAP5/MOD2-B&W computer code. The reanalysis maximized the mass and energy release from the break for the purpose of predicting a conservative peak temperature and peak pressure of the Containment Vessel. Additional details are provided in Section 6.2.1.3.2.

#### 15.4.4.3 Plant Changes and Effects

##### 15.4.4.3.1 Post June 9, 1985 Loss of Feedwater Event

Following the June 9, 1985 loss of feedwater event additional analyses were performed to demonstrate that the Auxiliary Feedwater (AFW) will not be isolated to both steam generators under assumed limiting single failure conditions following a main steam line break between the steam generator and the associated MSIV. These analyses assumed that the SFRCS low pressure trip on the affected steam generator (steam generator with the break) will trip the turbine and align the auxiliary feed water to the unaffected steam generator. Two cases were analyzed:

Case A - Turbine Stop Valves (TSVs) close within 1 second of initial SFRCS trip.

Case B - TSVs fail to close (single failure) and MSIVs close within 6 seconds of initial trip.

These analyses show that (Figures 15.4.4-9, 15.4.4-10) the unaffected steam generator pressure would not drop below the SFRCS low pressure trip setpoint provided that the TSVs close in 1 second. Consequently, the SFRCS will not isolate the unaffected steam generator. The Case B results show that the unaffected steam generator pressure could fall below 600 psid due to failure of the TSVs to close. Since failure of TSV to close represents a single failure, no additional failure (e.g., failure of the AFW isolation valve) needs to be postulated.

It is noted that in order to improve overall reliability, the SFRCS logic was modified such that low pressure in one steam generator would continue to isolate that steam generator, and initiate auxiliary feedwater to other steam generator. A subsequent low pressure trip on the other steam generator is blocked and will not have any effect on the AFW system. If the pressure on the first steam generator recovers above 600 psig, the SFRCS will respond according to

conditions in the second steam generator. If both steam generators recover, then SFRCS will respond based upon other plant conditions. Thus the potential for the total loss of AFW to the steam generators under multiple failure conditions is significantly reduced.

#### 15.4.4.3.2 Lowering the RPS Low Pressure Trip Setpoint (Reference 37)

The limiting accident in Section 15.4.4 is the double-ended steam line break (MSLB). For this accident, the reactor trips on low RCS pressure. Consequently, the reduction in the RPS low pressure trip setpoint to 1900 psig would delay the time of reactor trip following the double-ended MSLB. Based upon the rate of depressurization of the RCS prior to the reactor trip, lowering the low pressure trip setpoint to 1900 psig would delay the reactor trip by approximately 1.0 second. This delay would have an insignificant effect upon the accident analyses related to RCS overcooling and containment pressurization as discussed below.

For a double-ended steam line break, the delay in reactor trip due to a reduced RCS low pressure trip setpoint does not have a significant impact upon the minimum sub-critical margin or the minimum moderator temperature. In actuality, a delayed reactor trip causes more energy to be added to the RCS, thereby minimizing the overcooling associated with the steam line break. Consequently, using the original RCS low pressure trip setpoint is conservative for analyzing overcooling effects.

A delayed reactor trip does have a slight effect upon MSLB mass and energy release data. Since isolation of the main steam and main feedwater valves is initiated by the Steam and Feedwater Rupture Control System (SFRCS) on low secondary side pressure, the low RCS pressure setpoint has no impact upon the integrated mass release from the secondary side of the steam generators and the Main Steam System. However, a delayed RCS low pressure trip setpoint does have a slight impact upon the integrated energy release associated with an MSLB.

For the design basis MSLB a 1.0 second delay in reactor trip would add approximately 2.7 MBTU of additional energy to the RCS. This additional energy would cause the RCS fluid temperature to be slightly higher than what is presently calculated for the design basis MSLB. The increase in RCS temperature, in turn, would cause a slightly larger  $\Delta T$  between the RCS fluid and steam generator secondary side fluid. This increase would result in greater energy addition to the secondary side. If it is conservatively assumed that all the additional reactor energy is added to the integrated energy released by the MSLB, the increase in integrated energy is less than 2% of the total energy associated with the design MSLB. This increase in energy would have a negligible impact upon containment peak pressure and temperature results for the MSLB. The peak containment pressure associated with the MSLB with the increased energy addition is 22.6 psig. The MSLB would still be enveloped by the 2A hot leg break as the design basis accident for the containment.

#### 15.4.4.3.3 Raising the Maximum Allowable Steam Generator Water Level

Additional analyses were performed in order to support Technical Specification Amendment 192 (Reference 41 and 42) which allows the maximum steam generator (SG) water level to be 96% of the Operate Range (OR). The primary concern associated with this change was the additional SG inventory available in MODE 3 which can be released during an accident condition. These analyses examined the consequences of this higher steam generator inventory on containment pressurization following a MSLB, environmental effects for line breaks outside containment, offsite radiological consequences and core criticality.

Containment Pressurization:

During operation in Modes 1 and 2, revised operating limits have been implemented which are based on main steam superheat and indicated SG Operate Range. For plant operation up to 96% OR, these limits maintain the SG inventory at or below that used in the 100 percent Full Power MSLB containment analyses described in Sections 6.2.1.3.2 and 15.4.4.2.3.

While the plant is in Mode 3, if a Main Feedwater Pump is capable of supplying water to the SG and the SFRCS Low Pressure Trip is bypassed, the SG inventory is limited to 50 inches Startup Range. This limits the amount of energy available for release to the containment to less than that released during a MSLB at 100% full power.

While the plant is in Mode 3, with the SFRCS Low Pressure Trip active, the inventory in the SGs may be increased to a level of 96% OR. If the SFRCS Low Pressure Trip is bypassed, but the possible flow to the SG is limited to that available from the Motor Driven Feed Pump, the inventory may be increased to a level of 74% OR. For these conditions, the mass and energy released will remain bounded by the 100% Full Power MSLB containment analyses, including 10 minutes of feed to the SG.

Line Breaks Outside Containment:

Plant operating limits ensure that the SG inventory in Modes 1 and 2 at 96% OR is bounded by the SG initial inventory assumed for the existing HELB analyses.

Line breaks outside containment were examined to determine if the environmental effects remain bounding despite the larger steam generator inventory available in Mode 3 at 96% OR. The line breaks examined included the MSLB, main feedwater line break, main steam to auxiliary feed pump turbines line break and the Steam Generator Blowdown System line break.

In all cases the mass and energy released were bounded by the existing analyses in Section 3.6, except for one case. In Mode 3, with an initial water level of 96% OR with the MDFP supplying the SGs and with the SFRCS Low Pressure Trip active, the mass of water released is more than that assumed in Section 3.6.2.7.1.5. However, the energy content of the steam exiting the break is always lower at any given time in the transient because of the Mode 3 conditions. As a result, the environmental effects are bounded by the existing analysis.

Reactivity:

A MSLB with increased inventory in the SGs results in rapid overcooling of the RCS, thereby adding positive reactivity to the reactor. Administrative controls have been added to ensure adequate shutdown margin is present to prevent the reactor from attaining criticality during any postulated MSLB.

In Modes 1 and 2, with the maximum SG water level of 96% OR, plant operating limits maintain the SG inventory to that assumed by the existing 100% full power MSLB analyses. Therefore the cooldown is unaffected by the SG level change in Modes 1 and 2.

During Mode 3, the largest cooldown is caused by a MSLB with the steam generators at 74% OR and the SFRCS Low Pressure Trip bypassed. The cooldown and positive reactivity insertion of this case bounds all other Mode 3 MSLB scenarios. To ensure the reactor remains subcritical in this event, administrative controls include the requirement to determine the boron concentration necessary to compensate for the calculated cooldown and procedural



requirements to establish the necessary boron concentration in the RCS prior to raising the SG level above low level limits.

When the plant is in MODE 4, the SGs can only induce a very limited cooldown of the RCS following any secondary side line breaks. Therefore, no additional reactivity requirements are needed. No specific feed pump requirements are needed for the same reason. The maximum SG inventory limit is provided to ensure the SGs remain capable of decay heat removal by maintaining a steam flow path.

The assumptions in Section 15.4.4.2.3 related to Departure from Nucleate Boiling Ratio (DNBR) are consistent with the inventory in the SGs for operation in Modes 1 and 2. In Modes 3 and 4, departure from nucleate boiling can not occur due to the very low heat flux in the reactor.

#### Radiological Consequences:

The radiological consequences of a MSLB bounds all other steam line breaks. The radiological consequences of a HSLB in Mode 3 with a maximum initial steam generator water level was greater than the values given in Section 15.4.4.2.4., but remained below the NRC acceptance criteria. The results of the dose analysis are listed in Table 15.4.4-4a.

#### 15.4.4.3.4 Lowering the SFAS RCS Low Pressure Trip Setpoint

The SFAS RCS Low Pressure Trip analytical setpoint, which initiates High Pressure Injection during a Steam Line Break (SLB), was revised to 1515 psia. All SLB analyses had used this value as an input assumption. Therefore, this change had no effect on the existing analyses.

#### 15.4.4.3.5 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. That evaluation concluded that the reduction in the replacement Steam Generator steam nozzle area offsets the slight increase in the heat transfer capacity such that the MSLB analysis for core response, presented above, remains applicable with the replacement Steam Generators installed.

TABLE 15.4.4-1

Steam Line Failure Parameters

Steam generator inventory (I) as a function of fractional power level (P) for powers greater than 15% of nominal power (2772 MWt)	$I = 41,200P + 13,800 \text{ lb}$
Maximum pipe size (ID), in.	33.9
Trip Variable	Low Pressure
Trip Delays Time, sec	0.6
Doppler coefficient (EOL), ( $\Delta k/k$ )/°F	$-1.77 \times 10^{-5}$ (see note 1)
Moderator coefficient (EOL), ( $\Delta k/k$ )/°F	$-3.00 \times 10^{-4}$ (see note 1)
Trip delay time (low pressure trip), sec	0.6
CRA movement time to 2/3 insertion during trip, sec	1.4

- (1) An equivalent average temperature coefficient (combination of moderator and Doppler coefficients) of  $-3.1 \times 10^{-4} \Delta k/k/^\circ\text{F}$  has also been used for temperatures at Hot Zero Power (HZP) and below (see Section 15.4.4.2.6.7), and this temperature coefficient is bounded by the values of moderator coefficient and Doppler coefficient shown in this table.

TABLE 15.4.4-2

Mass and Energy Releases for Building Pressure Analysis

	<u>Mass, lb</u>	<u>Energy, Btu</u>
Steam generator inventory (fouled)	62,500	$35.9 \times 10^6$
Feedwater flow to affected steam generator (includes flow until trip and a 17 sec main feedwater control valve closing time)	18,550	$8.2 \times 10^6$
Reactor coolant systems energy transferred	--	$65.7 \times 10^6$
Available mass in feedwater line between feedwater control valves and affected steam generator	37,800	$16.7 \times 10^6$
Steam flow from unaffected steam generator (until isolation)	71,950	$40.1 \times 10^6$
Total releases	190,800	$166.6 \times 10^6$

TABLE 15.4.4-3

Summary of Steam Line Failure Analysis

Minimum subcritical margin during transient, % $\Delta k/k$	0.69
Steam released to atmosphere from affected generator prior to feedwater isolation, lb	118,850
Steam released to atmosphere from unaffected steam generator prior to steam line isolation, lb	71,950
Reactor coolant to secondary leakage during reactor coolant system depressurization, lb	1,788
Minimum DNBR during transient	1.42

TABLE 15.4.4-4

Resultant Doses From a Steam Line Failure In Mode 1<sup>(1)</sup>

	<u>Exclusion area boundary 0-2 hr</u>	<u>LPZ boundary 0-30 days</u>
Thyroid dose (Rem)	0.79	0.041
Whole body dose (Rem)	$6.7 \times 10^{-3}$	$3.46 \times 10^{-4}$

<sup>(1)</sup>See 15.4.4.2.4 referenced re-analyses for the resultant doses from a steam line failure.

TABLE 15.4.4-4a

Resultant Doses From a Steam Line Failure  
In MODE 3 With SG Level at 96% Operate Range

	<u>Exclusion Area boundary 0-2 Hr.</u>	<u>LPZ boundary 0-30 Days</u>
Thyroid Dose (Rem)	0.951	0.063
Whole Body Dose (Rem)	$3.0 \times 10^{-3}$	$2.0 \times 10^{-4}$

TABLE 15.4.4-5

Minimum Reactivity Margins for Various Main  
Steam Line Break Situations

Case	Minimum Reactivity Margin, % $\Delta k/k$
I. 102% of 2772 MWt. No offsite power. Break inside containment.	0.62
II. 102% of 2772 MWt. With offsite power. Break inside containment.	0.69
III. 102% of 2772 MWt. No offsite power. Break outside containment but upstream of main isolation valve.	0.62
IV. 102% of 2772 MWt. With offsite power. Break outside containment but downstream of main steam isolation valve.	$\geq 0.69$
V. Hot standby or low power operation.	$>0.69$

TABLE 15.4.4-6

Steam Line Failure with Concurrent MSSV  
Failure Parameters

Initial Power	102% of 2772 MWt
Steam generator inventory	55,000 lbm
RPS high flux trip setpoint (with 0.4 second delay)	112% of 2772 MWt
SFAS low pressure trip setpoint (with 30 second HPI delay)	1515 psia
Doppler coefficient (EOL), ( $\Delta k/k$ )/ $^{\circ}$ F	$-1.77 \times 10^{-5}$ (see note 1)
Initial moderator coefficient (EOL), ( $\Delta k/k$ )/ $^{\circ}$ F	$-3.00 \times 10^{-4}$ (see note 1)
Initial tripped rod worth, ( $\Delta k/k$ )/ $^{\circ}$ F	$-3.5 \times 10^{-2}$
Initial boron concentration	16 ppm
SFRCS low steam line pressure setpoint	600 psia
Turbine Stop Valve closure delay after SFRCS	1 second (.5 sec. delay, .5 sec. ramp)
Main Steam Isolation Valve closure delay after SFRCS	6 seconds (1 sec. delay, .5 sec. ramp)
Main Feedwater Stop Valve closure delay after SFRCS	18 seconds (1 sec. delay, 17 sec. ramp)
Auxiliary Feedwater delay after SFRCS	13 seconds

- (1) An equivalent average temperature coefficient (combination of moderator and Doppler coefficients) of  $-3.1 \times 10^{-4} \Delta k/k/^{\circ}$ F has also been used for temperatures at Hot Zero Power (HZP) and below (see Section 15.4.4.2.6.7), and this temperature coefficient is bounded by the values of moderator and Doppler coefficient shown in this table.

#### 15.4.5 Break in Instrument Lines or Lines from Primary System That Penetrate Containment

##### 15.4.5.1 Identification of Causes

A break in fluid-bearing lines which penetrate the Containment Vessel could result in the release of radioactivity to the environment. There are no instrument lines connected to the Reactor Coolant System which penetrate the Containment Vessel. There are, however, other piping lines from the Reactor Coolant System to the Makeup and Purification System and the Decay Heat Removal System which do penetrate the Containment Vessel. Leakage through fluid penetrations not serving accident-consequence-limiting systems is minimized by a double barrier design so that no single, credible failure or malfunction of an active component will result in loss of isolation or intolerable leakage. The installed double barriers take the form of closed piping, both inside and outside the Containment Vessel, and various types of isolation valves.

The most severe pipe rupture with regard to radioactivity release during nonnal station operation occurs in the Makeup and Purification System. This would be a rupture of the letdown line just outside the Containment Vessel but upstream of the letdown control valves. The occurrence of a rupture at this point results in a loss of reactor coolant until the temperature switches on the outlet of letdown coolers close the redundant valves on the inlet to the coolers.

Additionally, the rupture outside containment could be isolated by an SFAS level 2 signal on low RCS pressure. An SFAS level 2 signal would close the letdown line containment isolation valves thereby terminating release of reactor coolant system fluid outside containment. Although both the redundant temperature switches and the SFAS isolation signal are available for isolation of the letdown system, the design analyzed in Subsection 15.4.5.2.3 uses the SFAS actuation on low reactor coolant system pressure to terminate release to the environment. This assumption represents a bounding analysis for total integrated doses by maximizing the blowdown duration.

##### 15.4.5.2 Analysis of Effects and Consequences

###### 15.4.5.2.1 Safety Evaluation Criteria

The safety evaluation criterion for this accident is that resultant doses shall not exceed 10CFR100 limits.

Beginning with cycle 5, the fuel cycle length was extended to 18 months. The plant Technical Specifications limit the RCS activity to a value which is significantly less than the iodine activity associated with 1% failed fuel, assumed in the evaluation of this accident presented in section 15.4.5.2.3, 15.4.5.3.1 and 15.4.5.3.2. Therefore the resultant doses presented in 15.4.5.2.3, 15.4.5.3.1 and 15.4.5.3.2 bound the radiation doses for any fuel cycle. The plant Technical Specification limits are based on the NRC evaluation as documented in the NRC SER. For this reason no additional evaluation was performed for this accident to support extended 24 month fuel cycles.

###### 15.4.5.2.2 Methods of Analysis

A digital computer program was used to determine loss of coolant characteristics of this accident. The multinode model included a detailed description of the Reactor Coolant System. The model provides mass, energy, and momentum balances for the Reactor Coolant system nodal arrangement.

This analysis assumed a complete severance of the 2-1/2 inch letdown line. No operator action was assumed. Coolant was assumed to flow out until the isolation valve was fully closed. Credit was not taken for the reduction in flow during the last seconds while the valve was closing. The normal makeup system was assumed to function which results in a slightly longer time to reach the low reactor coolant pressure setpoint and a correspondingly higher mass release.

#### 15.4.5.2.3 Environmental Consequences

Figure 15.4.5-1 shows the Reactor Coolant System pressure as a function of time. As can be seen from this figure, the time to 1600 psig is approximately 95 seconds. Adding on the 10 seconds for the valve closure time, coolant escapes for a total period of 105 seconds. The total mass of reactor coolant released is 7,955 pounds.

Assuming the reactor operated with 1 percent defective fuel cladding, the individual isotopic activities released to the Auxiliary Building are listed in Table 15.4.5-1. Since the building ventilation exhaust passes through activated charcoal adsorbers, a reduction factor of 20 is assumed for iodine, based on an adsorber efficiency of 95%.

Atmospheric dilution is calculated using the 2-hour dispersion factor developed in Section 2.3. The total integrated doses at the exclusion distance as shown in Table 15.4.5-2 are well below the limits of the 10CFR100 guideline.

#### 15.4.5.3 Effects of Plant Changes

##### 15.4.5.3.1 Lowering the RPS Low Pressure Reactor Trip Setpoint

Technical Specification Amendment 149 approved a reduction of the Reactor Protection System (RPS) Reactor Coolant System (RCS) low pressure trip to 1900 psig. This accident was reanalyzed (Reference 32) in support of a reduced RPS RCS Low Pressure trip. For the re-analysis the reactor trip is assumed to occur due to an RCS low pressure trip. As with the present USAR analysis, no credit was taken for isolation from the existing temperature switches. Coolant flow out of the break was assumed until closure of the letdown line isolation valve following an SFAS low RCS pressure actuation.

Following the break, the RCS depressurizes due to the loss of inventory to atmosphere. Based on the B&W analysis, the low RCS pressure condition was reached at approximately 130 seconds following break initiation. Following the reactor trip and assumed coincident loss of offsite power, the RCS continues to depressurize until an SFAS low RCS pressure condition occurred at approximately 250 seconds after break initiation. The SFAS trip at this time caused the letdown line isolation valve to close, thereby terminating the release of RCS inventory. Assuming a 10 second valve closure time, RCS coolant release occurred for approximately 260 seconds. The total mass of coolant released to the atmosphere by the break is 44,460 lb with an average enthalpy of 430 BTU/lb. The diesel generator and sequence delay times were assumed to be included in the 250 second duration of the transient (i.e., coincident reactor trip and loss of offsite power).

The radiation doses associated with the above mass release are calculated in Reference 33 using the following assumptions.

- All the noble gas activity in the reactor coolant discharged through the break is released to the environment.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

- The iodine activity contained in the portion of reactor coolant that flashes into steam is assumed to become airborne and is released to the environment.
- Although the EVS will be actuated by SFAS, the analysis does not take credit for EVS filters, or plateout of iodine on surfaces.
- The reactor coolant activity is based on activities given in USAR Section 15A for 1% failed fuel, which are considerably higher than the Technical Specification limit of 1  $\mu$  Ci/gm dose equivalent I-131.

The resultant doses due to the revised RPS RCS Low Pressure Trip are:

	<u>Exclusion Area Boundary</u>	<u>Low Population Zone</u>
Thyroid (Rem)	3.52	0.18
Whole Body (Rem)	0.03	0.002

The above doses are higher than those presently reported in the USAR Table 15.4.5-2. This is primarily attributed to the assumptions made in the re-analysis of not taking any credit for any iodine removal due to operation of EVS. Regardless, these results satisfy the Standard Review Plan (NUREG 0800) Section 15.6.2 acceptance criteria that doses be well below 10% of 10CFR100 guideline values.

15.4.5.3.2 Lowering the SFAS Low Pressure Trip Setpoint

In support of twenty four (24) month operating cycles, it was necessary to revise the SFAS RCS Low Pressure Trip Setpoint. The analytical value used in analyses is 1515 psia. Reducing this value from the previous 1600 psig (1585 psia) delays isolation of the letdown line break to 45.51 seconds. The additional mass and energy releases associated with the time delay were determined to be 7,104 lb and 3,054,720 BTU, respectively.

The values were then used in a radiation dose calculation to determine the increase in doses at the exclusion area boundary and the low population zone. The resultant doses due to a RPS RCS Low Pressure Setpoint of 1900 psig and an SFAS RCS Low Pressure setpoint of 1515 psia are:

	<u>Exclusion Area Boundary</u>	<u>Low Population Zone</u>
Thyroid (Rem)	4.83	0.25
Whole Body (Rem)	0.04	0.002

These does are higher than previously reported in Table 15.4.5-2. However, the results still satisfy the acceptance criteria of NRC Standard Review Plan (SRP) Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment," which requires that doses do not exceed a small fraction of 10CFR100 guideline values, that is, 2.5 Rem and 30 Rem for the whole-body and thyroid doses, respectively.



#### 15.4.5.3.3 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. That evaluation concluded that the replacement Steam Generator design differences do not affect the inputs or integrated leak flow used in the analysis of a letdown line break. Therefore, the existing analyses remain applicable with the replacement Steam Generators installed.

TABLE 15.4.5-1

Activity Released to Auxiliary Building<sup>(1)</sup>  
From Letdown Line Rupture

<u>Isotope</u>	<u>Activity (Ci)</u>
Kr-83m	1.58
Kr-85m	8.37
Kr-85	47.2
Kr-87	4.58
Kr-88	14.7
Xe-131m	11.8
Xe-133m	15.4
Xe-133	1330
Xe-135m	5.07
Xe-135	27.9
Xe-138	2.8
I-131	17.6
I-132	26.4
I-133	20.8
I-134	2.72
I-135	10.38

<sup>(1)</sup>See 15.4.5.2.1 and 15.4.5.3 referenced re-analyses for the activity released from letdown line rupture.

TABLE 15.4.5-2<sup>(1)</sup>

Resultant Doses From Letdown Line Rupture

	<u>Exclusion area boundary 0-2 hrs</u>	<u>LPZ boundary 0-30 days</u>
Thyroid dose (Rem)	0.123	6.37 x 10 <sup>-3</sup>
Whole body dose (Rem)	0.015	7.67 x 10 <sup>-4</sup>

<sup>(1)</sup>See 15.4.5.2.1 and 15.4.5.3 referenced re-analyses for the resultant doses from the letdown line rupture.

15.4.6 Major Rupture of Pipes Containing Reactor Coolant Up To and Including Double-Ended Rupture of the Largest Pipe in the Reactor Coolant System (Loss-of-Coolant Accident)

The computer model CRAFT (Model for Equilibrium LOCA Analysis) was originally used to describe this accident. All other methods and assumptions that were used are described in BAW-10034, Rev. 3 (May, 1972). Subsequent large break loss-of-coolant accident analyses performed in accordance with AEC ECCS "final acceptance criteria" were performed with methods and assumptions as described in BAW-10104, Rev. 5 (November 1988) and BAW-10105, Rev. 1 (July, 1975).

For Cycle 13 and onward, the large and small break LOCA spectrum was reanalyzed using the RELAP5/MOD2-B&W-based evaluation model (BAW-10192PA, July 1998). The analysis results are summarized in Reference 51 and demonstrate that the acceptance criteria of 10CFR50.46 are met. A discussion of the non-radiological aspects of this accident is provided in Chapter 6.

15.4.6.1 Accident Analysis

The nonradiological aspects of this accident are discussed in Chapter 6.

15.4.6.2 Safety Evaluation Criterion

The safety evaluation criterion for this accident is that resultant doses shall not exceed 10CFR100 guideline values.

15.4.6.3 Environmental Analysis of Loss-of-Coolant Accident

Safety injection is designed to prevent significant cladding melting in the event of a loss-of-coolant accident. The analysis in Chapter 6 demonstrates that safety injection will prevent cladding melting for a loss-of-coolant accident resulting from Reactor Coolant System ruptures ranging in size from small leaks to the complete severance of a hot-leg coolant pipe. Without cladding melting, only the radioactive material in the coolant at the time of the accident plus some gap activity is released to the Containment Vessel. The environmental consequences from a loss-of-reactor coolant accident are analyzed by assuming that 1 percent of the fuel rods are defective before the release of reactor coolant to the Containment Vessel. In addition to the coolant activity, the activity associated with the gap of all fuel rods is also assumed to be released. While perforation of fuel cladding will require some time, it is conservatively assumed that all gap activity in the fuel rods is released directly to the Containment Vessel at the time of the accident. Appendix 15A lists the maximum activity in the coolant in Table 15A-4 and the total fuel rod gap activity in Table 15A-2. The resultant doses due to the maximum break size LOCA evaluated using the guidelines of Safety Guide 4 are given in Table 15.4.6-1.

15.4.6.4 Maximum Hypothetical Accident

The analysis in Chapter 6 has demonstrated that, even in the event of a loss-of-coolant accident, no significant core melting will occur. However, to demonstrate in a still more conservative manner that the operation of the station does not present any undue hazard to the general public, a hypothetical accident involving a gross release of fission products is evaluated. No mechanism whereby such a release occurs is postulated, since this would require a

multitude of failures in the engineered safety features, which are provided to prevent such an occurrence.

Fission products are assumed to be released from the core as stated in TID-14844; namely, 100 percent of the noble gases, 50 percent of the halogens, and 1 percent of the solids. Of the iodine released, 50 percent is assumed to plate out and the other half is assumed to remain in the Containment Vessel atmosphere where it is available for leakage. Although the Containment Vessel leakage rate will decrease as the pressure decays, the leakage is assumed to remain constant at the design leak rate for the first 24 hours. Thereafter, the leak rate is assumed to be reduced to one-half the design leak rate and to remain at this value for the duration of the accident. These assumptions are consistent with Regulatory Guide 1.4.

The 2-hour thyroid and whole-body doses at the exclusion distance, the 30-day doses at the low-population-zone distance (evaluated using Safety Guide 4), and a Dose Reduction Factor (DRF) for containment sprays are used for boric acid spray credit for the 0-2 hours dose at the site boundary and at the LPZ. The dose to the whole body from the passing cloud has been calculated using the same meteorological conditions used in determining the thyroid dose. The dose values shown in Table 15.4.6-2 are less than 10CFR100 limits.

A complete description of the Containment Vessel Spray System is given in Subsections 6.2.2.2.2 and 6.2.3.

Spray removal rates used in the analysis were evaluated using the model described by Parsly in ORNL-TM-2412 Part VII.

For elemental iodine,  $\lambda = 0.24 \text{ hr}^{-1}$  for injection of water from the Borated Water Storage Tank (BWST), and  $\lambda = 0.163 \text{ hr}^{-1}$  for water from the Containment Vessel Emergency Sump. Spray credit during the recirculation phase was conservatively taken from the time that recirculation starts until a DF of 100 was attained. The pH of the recirculation water is rising from the boric acid pH of 4-5 to the neutral pH of 7 prior to the time of recirculation (Ref 59).

For particulate iodine,  $\lambda = 0.2 \text{ hr}^{-1}$  for injection of water from the Borated Water Storage Tank, and  $\lambda = 0.134$  after the start of recirculation. No credit is taken for spray removal of organic iodine.

The injection phase is assumed to terminate at 40 minutes, the minimum time at which the water from the BWST could be exhausted. After this time reduced iodine removal credit is taken for the recirculation phase. Spray removal credit is terminated at a DF of 100 for both elemental and particulate iodine.

This analysis assumed the TID 14844 release of 25 percent of the core iodines available for release from the containment. Beginning with cycle 5, the fuel cycle has been extended. The total activity in the fuel and fuel rod gaps is given in Table 15A-6. The 24 month fuel cycle is used in the evaluation of MHA related doses. Of this, in accordance with Regulatory Guide 1.4, 4 percent was conservatively assumed to be organic. Iodine removal from the Containment air is accomplished by spray removal, decay, or leakage to the annulus. The iodines removed by the spray are deposited in the sump solution. Thus, when the sump is used as a source of spray water, the partition coefficient is smaller than for clean spray, resulting in the smaller spray removal constant. Partition coefficients can be found in ORNL-TM-2412 Part IV. Only one of the two containment spray pumps was assumed to be working at any one time, although this would not be the case unless there was an equipment failure.

## Davis-Besse Unit 1 Updated Final Safety Analysis Report

The assumptions used in this analysis were from L. F. Parsly's analysis in ORNL-TM-2412 Part VII and are considered conservative. The report BNP-100 by Postma, Coleman, and Hillard reported experimental results much greater for boric acid sprays than those calculated here.

The Containment Spray System (Subsection 6.2.2.2.2) is completely independent of the Emergency Ventilation System (EVS) (Subsection 6.2.3.1). The spray system works inside the containment. The EVS collects leakage into the annular space between the Containment Vessel and the Shield Building.

The dose analysis incorporated the following additional assumptions:

- a. The annular region between the Containment Vessel and the Shield Building is at atmospheric pressure upon accident initiation; 13 minutes are required to obtain a negative pressure in that region. It is assumed that all activity escaping the Containment Vessel during that time is released directly to the atmosphere without benefit of filtration or mixing. After the negative pressure has been obtained, 3 percent of the leakage is direct to the environment and the other 97 of the leakage is collected by the Emergency Ventilation System and exhausted through 95 percent efficient HEPA filters and charcoal adsorbers. Ref (35)
- b. Control room personnel on duty at accident initiation remain on duty for 96 hours (4 days).
- c. The operator absorbing the highest dose is assumed to be on duty for the initial 4-day period and 40 percent of the time for the period of 4 to 30 days.
- d. The Control Room Ventilation System is isolated by closure of the normal ventilation system upon receipt of a high-pressure (4 psig) or high-radiation signal (per Amendment 221, SFAS no longer actuates on high containment radiation) from the Containment Vessel, a low reactor pressure, or a high-radiation signal in the station vent. This isolation requires a maximum of eleven seconds execution time (assuming 6 seconds maximum for instrument response). The radioactive concentration entering the control room will be nonexistent during this time since it takes in excess of 15 seconds for the radioactive cloud to travel from the containment to the control room intake.
- e. Ten minutes after the start of the accident, the Emergency Control Room Ventilation System is started in the intake and recirculation mode.
- f. Inleakage to the control room is 63 cfm for the first 10 minutes.
- g. A fresh air (filtered) intake of 300 cfm is taken at 10 minutes and continues to 30 days. This intake slightly pressurizes the control room. In addition, it is assumed that 10 cfm unfiltered air reaches the control room due to door openings during the 10 minute to 30 day period.
- h. A direct gamma dose is received through the walls from the Containment Vessel/Shield Building structure.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

- i. The dispersion factors were calculated by use of

$$\frac{X}{Q} = \frac{1}{C a \mu}$$

for the control room dose analysis.

where C = shape factor = 0.5

A = vertical cross-sectional area of the containment

$\mu$  = average wind speed.

The meteorological conditions were chosen to correspond with those stipulated in Safety Guide 4, Section C, Paragraph 3, with the wind speed adjusted to correspond to actual site conditions, and the area calculated at 2597.4 m<sup>2</sup>. The dispersion factors used in the analysis are:

0 to 24 hours     5.85 x 10<sup>-4</sup> sec/m<sup>3</sup>

1 to 4 days        2.51 x 10<sup>-4</sup> sec/m<sup>3</sup>

4 to 30 days      5.7 x 10<sup>-5</sup> sec/m<sup>3</sup>

The control room recirculation flow is 3300 cfm in the isolation mode and 3000 cfm in the emergency mode. In both cases, the air flows through charcoal filters that are 95 percent efficient for elemental, particulate, and organic materials.

- j. Deleted

Table 15A-6 gives the isotopic inventory in curies at the containment at the beginning of the accident. These inventories are defined as A<sub>0</sub> in the equation that follows.

Mathematical Model:

A<sub>0</sub> = initial activity released to containment at time zero (Ci)

A<sub>1</sub>(t) = initial activity with decay (Ci)

$$= A_0 e^{-\lambda_1 t}$$

A<sub>2</sub>(t) = activity in primary containment at any time (Ci)

$$= A_2(t_0) e^{-\lambda_2 t}$$

A<sub>4</sub>(t) = activity in Shield Building at any time (Ci)

$$= \frac{\lambda_1 A_2(t_0)}{\lambda_4 - \lambda_2} \left[ e^{-\lambda_2 t} - e^{-\lambda_4 t} \right] + \lambda_4(t_0) e^{-\lambda_4 t}$$

R(t) = release rate of activity to the atmosphere (Ci/sec)

=  $(F_1 \lambda_3) (A_4(t))$ , where  $F_1$  is the filter nonremoval fraction.

IAR(t) = total activity released to atmosphere over time interval (Ci)

$$= \frac{F_1 \lambda_3 \lambda_2 A_2(t_0)}{\lambda_4 - \lambda_2} \left[ \frac{1 - e^{-\lambda_2 t}}{\lambda_2} - \frac{1 - e^{-\lambda_4 t}}{\lambda_4} \right] + F_1 \lambda_3 A_4(t_0) \frac{1 - e^{-\lambda_4 t}}{\lambda_4} + \frac{A_0}{R}(t_0)$$

R<sub>CR</sub>(t) = intake rate of activity for the control room (Ci/sec)

=  $F_2 V X/Q R(t)$

$F_2$  = filter nonremoval fraction;  $V$  = intake rate m<sup>3</sup>/sec.

ACR(t) = activity in the control room at any time

$$= F_2 V X/Q \frac{F_1 \lambda_3 \lambda_1 A_2(t_0)}{\lambda_4 - \lambda_2} \cdot \left[ \frac{e^{-\lambda_2 t} - e^{-\lambda_7 t}}{\lambda_7 - \lambda_2} - \frac{e^{-\lambda_4 t} - e^{-\lambda_7 t}}{\lambda_7 - \lambda_4} \right]$$

$$+ F_2 V X/Q \cdot F_1 \lambda_3 A_4(t_0) \cdot \frac{e^{-\lambda_4 t} - e^{-\lambda_7 t}}{\lambda_7 - \lambda_4} + A_{CR}(t_0) e^{-\lambda_7 t}$$

IACR(t) = integrated control room activity (Ci-sec)

$$= (F_2 V X/Q) \frac{F_1 \lambda_3 \lambda_1 A_2(t_0)}{\lambda_4 - \lambda_2} \left[ \frac{1}{\lambda_7 - \lambda_2} \left[ \frac{1 - e^{-\lambda_2 t}}{\lambda_2} - \frac{1 - e^{-\lambda_7 t}}{\lambda_7} \right] - \frac{1}{\lambda_7 - \lambda_4} \left[ \frac{1 - e^{-\lambda_4 t}}{\lambda_4} - \frac{1 - e^{-\lambda_7 t}}{\lambda_7} \right] \right]$$

$$+ (F_2 V X/Q) (F_1 \lambda_3 A_4(t_0)) \frac{1}{\lambda_7 - \lambda_4} \left[ \left[ \frac{1 - e^{-\lambda_4 t}}{\lambda_4} \right] - \left[ \frac{1 - e^{-\lambda_7 t}}{\lambda_7} \right] \right] + A_{CR}(t_0) \frac{1 - e^{-\lambda_7 t}}{\lambda_7}$$

Davis-Besse Unit 1 Updated Final Safety Analysis Report

<u>Symbol</u>	<u>Definition</u>
$A_0$	= initial activity released to the containment (Ci)
$A_1(t)$	= initial activity with decay (Ci)
$A_2(t)$	= activity in primary containment (Ci)
$A_4(t)$	= activity in secondary containment (Ci)
$R(t)$	= release rate of activity from secondary (Ci/sec)
$IA_R(t)$	= total activity released over time (Ci)
$R_{cr}(t)$	= control room activity intake rate (Ci/sec)
$A_{cr}(t)$	= activity in the control room (Ci)
$IA_{cr}(t)$	= Integrated activity in the control room (Ci – sec)
$\lambda_r$	= primary decay rate
$\lambda_1$	= primary leak rate
$\lambda_{1R}$	= primary cleanup rate – recirculation + spray
$\lambda_2$	= total primary loss rate = $\lambda_r + \lambda_1 + \lambda_{1R}$
$\lambda_{2R}$	= secondary cleanup rate
$\lambda_3$	= secondary leak or purge rate
$\lambda_4$	= total secondary loss rate = $\lambda_{2R} + \lambda_3 + \lambda_g$
$\lambda_6$	= control room cleanup rate
$\lambda_{CR}$	= control room leak or purge rate = intake rate
$\lambda_7$	= total control room loss rate = $\lambda_6 + \lambda_{CR} + \lambda_r$



Davis-Besse Unit 1 Updated Final Safety Analysis Report

<u>Symbol</u>	<u>Definition</u>
DS-TH	= site boundary thyroid dose (rem)
DS- $\beta$	= site boundary beta skin dose (rem)
DCR-TH	= control room thyroid dose (rem)
DCR- $\beta$	= control room beta skin dose (rem)
DCF <sub>TH</sub>	= dose conversion factor thyroid $\frac{\text{rem}}{\text{Ci}}$
DCF <sub>WB</sub>	= dose conversion factor - whole body total $\frac{\text{rem}}{\text{Ci}} \cdot \frac{\text{m}^3}{\text{sec}}$
BR	= breathing rate ( $\text{m}^3/\text{sec}$ )
X/Q	= meteorological dispersion factor ( $\text{sec}/\text{m}^3$ )
DS- $\gamma$	= site boundary whole body gamma dose (rem)
DCR- $\gamma$	= control room whole body gamma dose (rem)
$\psi$	= concentration time interval for the cloud $\frac{\text{Ci} \cdot \text{sec}}{\text{m}^3}$
DCF <sub><math>\beta</math>-SKIN</sub>	= dose conversion factor-beta skin $\frac{\text{rem}}{\text{Ci}} \cdot \frac{\text{m}^3}{\text{sec}}$

Site Boundary and LPZ Boundary

Thyroid Dose

$$D_{S-TH} = I_{AR} \cdot DCF_{TH} \cdot B.R. \cdot \frac{X}{Q}$$

$$\text{rem} = \text{Ci} \frac{\text{rem}}{\text{Ci}} \cdot \frac{\text{m}^3}{\text{sec}} \cdot \frac{\text{sec}}{\text{m}^3}$$

## Davis-Besse Unit 1 Updated Final Safety Analysis Report

### Whole Body Dose

$$D_{S-\gamma} = [0.25 \bar{E}_{\gamma}] \psi = DCF_{\gamma} \cdot \frac{X}{Q} IA_R$$

$$\text{rem} = \frac{\text{rem}}{\text{Ci}} \cdot \frac{\text{m}^3}{\text{sec}} \cdot \frac{\text{Ci sec}}{\text{m}^3} = \frac{\text{rem}}{\text{Ci}} \cdot \frac{\text{m}^3}{\text{sec}} \cdot \frac{\text{sec}}{\text{m}^3} \cdot \text{Ci}$$

$$D_{S-\beta} = [0.23 \bar{E}_{\beta}] \psi = DCF_{\beta \text{ skin}} \frac{X}{Q} \cdot IA_R$$

### Control Room Doses

#### Thyroid Dose

$$D_{CR-TH} = IA_{CR} \cdot DCF_{TH} \cdot B.R. \cdot \frac{1}{\text{vol}}$$

$$\text{rem} = \text{Ci-sec} \frac{\text{rem}}{\text{Ci}} \cdot \frac{\text{m}^3}{\text{sec}} \cdot \frac{1}{\text{m}^3}$$

#### Whole Body Dose

$$D_{CR-\gamma} = 0.25 \bar{E}_{\gamma} \cdot IA_{CR} \cdot \frac{1}{\text{vol}} = DCF_{\gamma} \cdot IA_{CR} \cdot \frac{1}{\text{vol}}$$

$$\text{rem} = \frac{\text{rem}}{\text{Ci}} \cdot \frac{\text{m}^3}{\text{sec}} \cdot \text{Ci-sec} \frac{1}{\text{m}^3} = \frac{\text{rem}}{\text{Ci}} \cdot \frac{\text{m}^3}{\text{sec}} \cdot \text{Ci-sec} \cdot \frac{1}{\text{m}^3}$$

vol = control room volume (m<sup>3</sup>)

$$D_{CR-\beta} = 0.23 \bar{E}_{\beta} \cdot IA_{CR} \cdot \frac{1}{\text{vol}} = DCF_{\beta \text{ skin}} \cdot IA_{CR} \cdot \frac{1}{\text{vol}}$$

To calculate the activity released to the atmosphere over any time period,  $IA_R(t)$  or the integrated control room activity,  $IA_{CR}(t)$  over any time period, equations for  $IA_R(t)$  and  $IA_{CR}(t)$  may be evaluated over incremental time periods within which all parameters are constant, and the activities from these increments summed. Time =  $t_0$  refers to the start of each time increment, and time =  $t$  is the length of the time increment.

For example, for a 10-second release where the filter nonremoval fraction,  $F_1 = 1$  for the first three seconds and  $F_1 = 0.5$  for the next seven seconds.

$$\begin{aligned}
 IA_R(t) &= IA_R(0 \text{ to } 3 \text{ sec}) + IA_R(3 \text{ sec to } 10 \text{ sec}) \\
 &= \frac{(1)(\lambda_3 \lambda_1 A_2(t_0 = 0))}{\lambda_4 - \lambda_2} \left[ \left[ \frac{1 - e^{(-\lambda_2)(3)}}{\lambda_2} \right] - \left[ \frac{1 - e^{(-\lambda_4)(3)}}{\lambda_4} \right] \right] \\
 &\quad + (1)(\lambda_3 A_4(t_0 = 0)) \frac{1 - e^{(-\lambda_4)(3)}}{\lambda_4} \\
 &\quad + \frac{0.5 \lambda_3 \lambda_1 A_2(t_0 = 3)}{\lambda_4 - \lambda_2} \left[ \left[ \frac{1 - e^{(-\lambda_2)(7)}}{\lambda_2} \right] - \left[ \frac{1 - e^{(-\lambda_4)(7)}}{\lambda_4} \right] \right] \\
 &\quad + (0.5)(\lambda_3 A_4)(t_0 = 3) \frac{1 - e^{(-\lambda_4)(7)}}{\lambda_4}
 \end{aligned}$$

#### 15.4.6.5 Effects of Engineered Safety Features Leakage During the Maximum Hypothetical Accident

An additional source of fission product leakage during the maximum hypothetical accident can occur from leakage of the engineered safety features external to the containment vessel during the recirculation phase for long-term core cooling. A single failure analysis on engineered safety features is given in Tables 6.2-21 and 6.3-6.

It is assumed that the water being recirculated from the Containment Vessel Emergency Sump through the external system piping contains 50 percent of the core saturation inventory. This is the entire amount of iodine released from the Reactor Coolant System. The assumption that all the iodine escaping from the Reactor Coolant System is absorbed by the water in the Containment Vessel is conservative since much of the iodine released from the fuel will be plated out on the vessel's walls. It is assumed that all the iodine contained in water that flashes is released to the Auxiliary Building atmosphere. The peak sump temperature during sump recirculation is 248°F. As the cooldown continues, the sump temperature decreases with time. This results in a corresponding reduction in flashing fraction. Although the temperature drops below 212°F within 24 hours, it is conservatively assumed that the flashing fraction is 0.041 (using the peak sump temperature of 251°F) for the first 24 hours. After 24 hours, the sump temperature is less than 212°F, and the partition factor for iodine is 0.01. The activity is assumed to be released through the 95 percent efficient HEPA filters and charcoal adsorbers of the Emergency Ventilation System to the station vent. Atmospheric dilution is calculated using the dispersion factors developed in Section 2.3.

The peak sump temperature determined by the Containment Vessel analysis described by Section 6.2.1.3 is slightly different than the evaluated peak temperature of 251°F. This

## Davis-Besse Unit 1 Updated Final Safety Analysis Report

difference was evaluated to have a negligible impact on the dose analysis (see Reference 6.5-42).

The total leakage outside containment was determined. Plant procedures limit total combined measured leakage during normal plant operation from both trains of ECCS to 40 gph. 40 gph was used in the calculation of doses from ECCS leakage (Reference 60). In Table 15.4.6-2, the "Resultant Doses from ESF Leakage" are included in the "Resultant Doses from MHA."

The leakage and the resultant thyroid dose at the exclusion distance and Low Population Zone (LPZ) are shown in Table 15.4.6-2. Based on the assumptions above, the leakage outside the containment gives a negligible contribution to the exclusion area boundary and LPZ boundary doses.

Technical Specifications requires that the low pressure injection system and containment spray system be included in a program to reduce leakage as low as practical. The program includes periodic visual inspection requirements and integrated leak test requirements. This ensures that the exclusion area boundary and LPZ doses are below the 10CFR100 guideline values.

TABLE 15.4.6-1<sup>(1)</sup>

Resultant Doses From Maximum Break Size LOCA

	<u>Exclusion Area Boundary 0 to 2 hours</u>	<u>LPZ Boundary 0 to 30 days</u>
Thyroid dose (Rem)	41.0	3.25
Whole-body dose (Rem)	1.03	0.128

<sup>(1)</sup>The environmental consequences for a LOCA, discussed in USAR Section 15.4.6.3, are bounded by the analyses provided in USAR Section 15.4.6.4 for a MHA.

TABLE 15.4.6-2

Resultant Doses From MHA

	<u>Exclusion Area Boundary 0 to 2 hours</u>	<u>LPZ Boundary 0 to 30 days</u>
Thyroid dose (rem)	234	25.2
Whole-body dose (rem)	3.5	0.4

Control Room Operator Doses (rem)

	<u>Thyroid</u>	<u><math>\beta</math> - Skin</u>	<u><math>\lambda</math> -Whole Body</u>
Inside the control room <sup>(1)</sup>	18.9	11.9	0.55
Direct gamma dose (Ref 61)	0	0	0.57
TOTALS	18.9	11.9	1.12

Resultant Doses from ESF Leakage

Assumptions: (Reference 60)

Liquid leakage, gph	40
Leakage which flashes, gph	1.6
Two-hour thyroid dose at exclusion area boundary, rem	2
Thirty-day thyroid dose at LPZ, rem	0.8

(1) The operator dose “inside the control room” includes an ECCS leakage term (References 58 and 60).

#### 15.4.6.6 Control Room Habitability

An analysis of release of noxious gas (chlorine) within the station and its effects on the control room is presented in Subsection 15.4.8. (Note: Chlorine Tank cars are no longer on site.) Control Room habitability following a main steam line break accident is discussed in Section 15.4.4.2.5.

Area radiation monitors are provided for the control room which continuously give the background radiation level. In case of any abnormal increase in the background level, the operator can manually isolate the normal ventilation system and start the Control Room Emergency Ventilation System if needed. The control room doses following a loss-of-coolant accident are presented in Table 15.4.6-2.

#### 15.4.6.7 Partial Loop Flow LOCA

Power operation with less than four reactor coolant pumps in operation is allowed by plant technical specifications. However, with less than four pumps in operation, the core power level must be reduced. Large break LOCA analyses are performed to determine allowable rod position limits in this configuration. The full-power LOCA linear heat rate limit is typically preserved during partial pump operation. Although the core power level is reduced, the initial stored energy in the fuel pellet (fuel average temperature) remains the same as with the full power case. At reduced power, a positive moderator temperature coefficient is also allowed. The purpose of the analysis is to demonstrate that the full power and full flow case bounds partial pump operation with reduced core power but with the same stored energy in the fuel and a positive MTC.

There are 5 possible break configurations at the pump discharge for partial loop operation:

1. 3-pump operation
  - a. break in idle pump discharge
  - b. break in active pump discharge of loop with the idle pump
  - c. break in pump discharge of loop with two active pumps
2. 2-pump operation, one idle pump in each loop
  - a. break in active pump discharge
  - b. break in idle pump discharge

Power operation with only two reactor coolant pumps running is not allowed by Davis-Besse License Condition C.3.a.

Partial pump (3-pump) LOCA analyses were performed with the RELAP5/MOD2-B&W-based LOCA evaluation model that is described in BAW-10192PA. Various sensitivity studies are performed in the evaluation model that generically demonstrates that the model provides an appropriate and conservative converged solution. In accordance with the SER on BAW-10192PA, plant-specific analyses for partial pump operation are required to be performed to demonstrate that partial pump operation is bounded. The partial pump analyses are performed at the maximum allowed linear heat rate limit from Reference 51 and a conservative

(positive) MTC of +1.0 pcm/°F. The LOCA analyses are based on a nominal core power level of 2966 MWt and includes a 2% heat balance error, e.g., the full power case was run at 102% and the partial pump case at 77% power. The results of the partial pump cases are summarized in Reference 51. The partial pump cases show that the peak clad temperature for the full power full flow case is higher and thus bounds the results for partial pump operation provided a linear heat rate penalty is applied to the full-power limit. The penalty is a function of power level and a maximum of 0.2 kW/ft.

#### 15.4.6.8 Additional Analyses and Related Plant Modifications

##### 15.4.6.8.1 DELETED

##### 15.4.6.8.2 Positive Moderator Temperature Coefficient (MTC) Analysis

The large break LOCA analyses documented in Reference 51 contain sensitivity studies to determine the allowable moderator temperature coefficient (MTC) as a function of the core power level with all of the reactor coolant pumps operating. The analysis was based on a nominal core power level of 2966 MWt. The results concluded that the MTC must be negative for core power levels above 80%, (2372.8 MWt) i.e., the allowable MTC is +9.0 pcm/°F at hot zero power and 0 pcm/°F at 80% power and above. The results are reported based on nominal core power, but the analysis included a heat balance error of 2% of full power.

##### 15.4.6.8.3 DELETED

##### 15.4.6.8.4 DELETED

##### 15.4.6.8.5 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. That evaluation documented that this event addresses only the doses resulting from a maximum hypothetical event. It also documented that the dose calculations do not directly depend on transient nuclear steam supply conditions. This hypothetical accident involves a gross release of fission products. No mechanism whereby such a release occurs is postulated. The design and performance differences of the replacement steam generators have no impact on this postulated gross release of fission products. Therefore, the replacement steam generators do not affect the analysis of record for the maximum hypothetical event.



#### 15.4.7 Fuel-Handling Accident

##### 15.4.7.1 Identification of Causes

Individual spent fuel assemblies are handled entirely under water (see Section 9.1.6.2 for handling of dry shielded canisters containing spent fuel assemblies). Before refueling, the reactor coolant and the refueling canal water above the reactor are increased in boron concentration so that, with all control rods removed, the  $k_{\text{eff}}$  of a core is no greater than 0.99. In the spent fuel storage pool, the fuel assemblies are stored under water in storage racks that have Boral neutron absorbing material to prevent criticality. Under these conditions, a criticality accident during refueling is not considered credible. Mechanical damage to the fuel assemblies during transfer operations is possible but improbable. The mechanical damage type of accident is considered the maximum potential source of activity release during refueling operations.

A dry fuel storage cask drop accident is not required to be postulated and the radiological consequences do not need to be evaluated when using the single-failure-proof main hoist on the 130 ton spent fuel cask crane in the Auxiliary Building as part of a single-failure-proof handling system for cask handling operations. A dry fuel storage cask drop analysis which applied to the drop of an OS197 transfer cask containing a 24P dry shielded canister (DSC) was previously provided in Section 15.4.7.2.5.3. Since a single-failure-proof handling system will be used for all future lifts of all transfer casks, the cask drop accident is no longer required to be postulated. Therefore, Section 15.4.7.2.5.3 has been removed and associated calculations have been made historical. Evaluations of potential accidents related to onsite dry fuel storage facility operations are discussed in the Davis-Besse Site Certified Safety Analysis Report (CSAR) for the AREVA / TN NUHOMS-24P Dry Shielded Canisters (DSCs), and in the AREVA / TN Standardized NUHOMS Updated Final Safety Analysis Report (UFSAR) NUH-003, Appendix U, for the 32PTH1 DSCs.

##### 15.4.7.2 Accident Analysis - Accident Outside Containment

###### 15.4.7.2.1 Safety Evaluation Criterion

The safety evaluation criterion for this accident is that resultant doses shall not exceed 10CFR100 guideline values.

In support of 18 month cycle operation, re-analyses of the Fuel Handling Accident (FHA) outside containment were performed (reference 30) based on three 450 EFPD cycles. Dose calculations were performed using more conservative source terms. Results of the evaluations showed that the offsite radiological dose for this accident was below the acceptance criterion value in the Standard Review Plan (NUREG 0800). This evaluation criterion was approved by the NRC via approval of the Cycle 6 Reload Report. The fuel handling accidents have been re-evaluated in USAR sections 15.4.7.2.5.1 and 15.4.7.3.1 for a maximum fuel assembly burnup of 60,000 MWD/MTU. That burnup limit is not changed for the extended fuel cycle.

###### 15.4.7.2.2 Methods of Analysis

The assumptions and guidelines of Safety Guide 25 were used in this analysis. For convenience, the major assumptions made for this analysis are shown in Table 15.4.7-1. The reactor is assumed to have been shut down for 72 hours, which is the minimum time for Reactor Coolant System cooldown, reactor closure head removal, and removal of the first fuel assembly. It is further assumed that the entire outer row of fuel rods in the assembly, 56 of 208, suffers

mechanical damage to the cladding. Since the fuel pellets are cold, only the gap activity is released. The fuel rod gap activity is calculated using the escape rate coefficients and calculational methods discussed in Section 11.1.

See 15.4.7.2.1 referenced re-analyses for a description of the methods of analysis performed to support extended fuel cycles.

#### 15.4.7.2.3 Results of Analysis

The gases released from the fuel assembly pass upward through the spent fuel storage pool water prior to reaching the fuel-handling-area atmosphere. Although there is experimental evidence that a portion of the noble gases will remain in the water, no retention of noble gases is assumed. In experiments whereby air/steam mixtures were bubbled through a water pond, Diffey, et al. (reference 3) demonstrated decontamination factors of about 1000 for Iodine. Similar results for iodine were demonstrated by Barthoux, et al. (reference 4) and predicted by Eggleton (reference 5). Based conservatively on these references, 99 percent of the iodine released from the fuel assembly is assumed to remain in the water. The fuel-handling area is ventilated and is discharged through Emergency Ventilation Systems 95 percent efficient HEPA filters and charcoal adsorbers to the station vent.

See 15.4.7.2.1 referenced re-analyses for results of calculations performed to support 18 month operation.

See 15.4.7.2.5 for analyses to support fuel enrichments to 5.0 wt% U-235 and fuel burnups to 60,000 megawatt days per metric ton (MWD/MTU).

The Spent Fuel Pool has been re-racked to provide 1624 storage locations. Based on the physical and analytical design of the racks, it was determined no changes to the Fuel Handling Accident were required.

#### 15.4.7.2.4 Environmental Consequences

The activity is assumed to be released over two hours from the station vent. Atmospheric dilution (for site and LPZ boundary) is calculated using the two-hour atmospheric dispersion coefficient developed in Section 2.3. Table 15.4.7-2 gives the total integrated dose at the exclusion distance for the whole body and the thyroid gland. These results are less than 10CFR100 guideline values.

See 15.4.7.2.5.1 for analyses for environmental consequences involving fuel enrichments up to 5.0 wt% U-235 and fuel burnups up to 60,000 megawatt days per metric ton (MWD/MTU) and applicability to extended fuel cycles.

#### 15.4.7.2.5 Additional Analyses

The following analyses supersede all previously documented USAR fuel handling accident analyses for an outside containment fuel handling accident.

##### 15.4.7.2.5.1 Effects of Extended Fuel Cycles, Fuel Burnup and Increased Fuel Enrichments

In order to evaluate the effects of extended fuel burnup and increased fuel enrichment on the consequences of the fuel handling accident outside containment, the source term and offsite dose calculations were reformed assuming that fuel assembly-average burnups could be as

## Davis-Besse Unit 1 Updated Final Safety Analysis Report

great as 60,000 MWD/MTU and that initial fuel enrichments could be as great as 5.0 weight percent uranium-235 (wt% U-235). Analysis assumptions are summarized in Table 15.4.7-1a.

All assumptions were made in accordance with Nuclear Regulatory Commission 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 (Safety Guide 25)," dated March 25, 1972).

For the extended fuel cycles, additional source term analyses were performed as described in Section 15.A.7.0 of Appendix 15A. Those analyses maintained the maximum fuel assembly burnup limit of 60,000 MWD/MTU and did not affect previously analyzed environmental consequences of the fuel handling accident.

### 15.4.7.2.5.2 Results

References 38 and 48 contain a detailed description of these fuel handling accident analyses. Source terms were determined to be extremely weak functions of fuel enrichment, with lower enrichments in general producing larger offsite doses than higher enrichments. Table 15.4.7-6 contains the source term results for fuel assemblies having initial enrichments of 3.0 and 5.0 wt% U-235. The values in Table 15.4.7-6 are fuel assembly-average fission product activities, and do not represent atmospheric release activities, as is the case in Table 15.4.7-3.

The offsite doses for these fuel handling accident outside containment analyses are shown in Table 15.4.7-2a. In summary, these results are applicable to the extended fuel cycle as well, since the pertinent maximum fuel assembly burnup limit is maintained. They meet the acceptance criteria provided in NUREG-0800 and, therefore, are well within the dose guidelines set forth in 10CFR100.

### 15.4.7.2.5.3 DELETED

#### 15.4.7.2.6 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. That evaluation concluded that steam generator performance does not impact this accident in any way. Therefore, the existing analyses remain applicable with the replacement Steam Generators installed.

TABLE 15.4.7-1<sup>(1)</sup>

Fuel-Handling Accident Parameters Outside Containment

Fuel burnup, full power days	1017
Power level for the assembly during operation, MWt	27.9
Filter efficiency for iodine removal, %	95
Atmospheric dispersion at exclusion distance, s/m <sup>3</sup> , assuming ground release	1.9 x 10 <sup>-4</sup>
The gap activity of the highest power fuel assembly is given in Table 15A-3, and the release by isotope is given in Table 15.4.7-3.	
Accident duration, hr	2

Bases:

- a. 56 fuel pins fail, releasing gap activity.
- b. 99 percent iodine remains in water.
- c. 72-hour decay
- d. Hottest assembly
- e. The control room ventilation system is isolated upon receipt of a high-radiation signal from the station vent. The isolation requires a maximum of ten seconds from attainment of the setpoint to closure of damper, during which time the intake rate of outside air is 10,960 cfm.
- f. Control room inleakage of outside air is 1 cfm.
- g. The release point (station vent) is 160 feet horizontal distance and 180 feet vertical distance.

<sup>(1)</sup>See 15.4.7.2.5 referenced re-analyses for the analysis assumptions.

Table 15.4.7-1a

Fuel Handling Accident Assumptions - Outside Containment

The following assumptions were used in the analysis that determined the doses listed in Table 15.4.7-2a:

1. The accident occurs at 72 hours following reactor shutdown.
2. Noble gas and iodine gas activities are based on Regulatory Guide 1.25.
3. The failure of 56 fuel pins.
4. A time-averaged radial peaking factor on an assembly basis of 1.4 is utilized.
5. The gap activity in the damaged fuel assembly is assumed to be released to the pool. All the noble gas activities that are released to the pool are assumed to escape from the pool; one percent of the iodine activities that are released to the pool are assumed to escape from the pool.
6. An instantaneous release (very high escape rate) from the Spent Fuel Pool is assumed to ensure that all the activity coming out of the pool is released to the environment in a short time.
7. Atmospheric dispersion factor (X/Q) at site boundary is  $1.9 \times 10^{-4} \text{ sec/m}^3$  and at LPZ it is  $9.9 \times 10^{-6} \text{ sec/m}^3$ .
8. Emergency Ventilation System charcoal filter efficiency for iodine removal is 95 percent.
9. The air intake of the Control Room's normal HVAC is automatically isolated upon receipt of a high radiation signal from the Station Vent radiation monitors.

Note: The assumptions utilized for the FHA outside the containment are identical to those for the FHA inside the containment with the exception of Items (3) and (8).

TABLE 15.4.7-2<sup>(1)</sup>

Resultant Doses From Fuel-Handling Accident Outside Containment

	<u>Exclusion Area Boundary 0 to 2 hours</u>	<u>LPZ Boundary 0 to 30 days</u>
Thyroid dose (rem)	0.106	$5.58 \times 10^{-3}$
Whole body dose (rem)	0.106	$5.59 \times 10^{-3}$
	<u>Control Room 0 to 2 hours</u>	
Thyroid dose (rem)	0.116	
$\beta$ -skin dose (rem)	0.024	
Total body gamma dose (rem)	$5.53 \times 10^{-3}$	

<sup>(1)</sup>See 15.4.7.2.5 referenced re-analyses for resultant doses.

TABLE 15.4.7-2a

Resultant Doses From Fuel Handling Accident Outside Containment - Extended Fuel Burnup (60,000 MWD/MTU)

	<u>Exclusion Area Boundary 0 to 2 hours</u>	<u>LPZ Boundary 0 to 30 days</u>
Thyroid Dose (rem)	0.85	$4.4 \times 10^{-2}$
Whole Body Dose (rem)	0.15	$8.0 \times 10^{-3}$

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 15.4.7-3<sup>(1)</sup>

Activity Released to the Atmosphere Due to the Postulated  
Fuel-Handling Accident Outside Containment (Ci)

I-131	$2.18 \times 10^1$
I-132	$1.47 \times 10^{-9}$
I-133	$5.52 \times 10^{-1}$
I-134	$3.66 \times 10^{-26}$
I-135	$1.09 \times 10^{-3}$
Xe-131m	$1.40 \times 10^2$
Xe-133m	$7.91 \times 10^1$
Xe-133	$1.19 \times 10^4$
Xe-135m	0
Xe-135	$1.52 \times 10^2$
Xe-137	0
Xe-138	$1.82 \times 10^{-73}$
Kr-83m	$5.00 \times 10^{-11}$
Kr-85m	$1.27 \times 10^{-3}$
Kr-85	$1.23 \times 10^3$
Kr-87	$4.60 \times 10^{-16}$
Kr-88	$3.45 \times 10^{-6}$
Kr-89	0

<sup>(1)</sup>See 15.4.7.2.5 referenced re-analyses for activity released to atmosphere.



### 15.4.7.3 Accident Analysis - Accident Inside Containment

#### 15.4.7.3.1 Safety Evaluation Criterion

The safety evaluation criterion for this accident is that resultant doses shall not exceed 10CFR100 guideline values.

In support of 18 month cycle operation, re-analyses of the Fuel Handling Accident (FHA) inside containment were performed (reference 30) based on three 450 EFPD cycles. Dose calculations were performed using more conservative source terms. Results of the evaluations showed that offsite radiological dose for this accident was below the acceptance criterion value in the current NRC Standard Review Plan (NUREG 0800). This evaluation criterion was approved by the NRC via approval of the Cycle 6 Reload Report.

The fuel handling accidents have been evaluated in USAR sections 15.4.7.2.5.1 and 15.4.7.3.4.1 for a maximum burnup of 60,000 MWD/MTU. That burnup limit is not changed for the extended fuel cycle.

#### 15.4.7.3.2 Analysis

The following assumptions were used in the analysis:

- a. The accident occurs at 72 hours following reactor shutdown.
- b. Noble gas and iodine gap activities are based on Regulatory Guide 1.25.
- c. One entire assembly is considered damaged.
- d. A time-averaged radial peaking factor on an assembly basis of 1.4 is utilized.
- e. The gap activity in the damaged fuel assembly is assumed to be released to the pool. All the noble gas activities that are released to the pool are assumed to escape from the pool; one percent of the iodine activities that are released to the pool are assumed to escape from the pool.
- f. Containment isolation is not assumed.
- g. An instantaneous release (very high escape rate) from the containment is assumed to ensure that all the activity coming out of the pool is released to the environment in a short time (see Table 15.4.7-5).
- h. Atmospheric dispersion factor (X/Q) at site boundary is  $1.9 \times 10^{-4}$  sec/m<sup>3</sup> and at LPZ it is  $9.9 \times 10^{-6}$  sec/m<sup>3</sup>.
- i. No filtration is assumed.

See 15.4.7.3.4.1 referenced re-analyses for a description of extended fuel cycle re-analysis and assumptions used.

See 15.4.7.3.4.1 for analyses supporting fuel enrichments up to 5.0 wt% U-235 and fuel burnups up to 60,000 megawatt days per metric ton (MWD/MTU).

#### 15.4.7.3.3 Environmental Consequences

The activity is assumed to be released over 2 hours. Atmospheric dilution for site boundary and LPZ is calculated using the 2-hour atmospheric dispersion coefficient developed in Section 2.3. Table 15.4.7-4 gives the calculated doses. These results are well within the 10CFR100 guideline values.

See 15.4.7.3.4.1 referenced re-analyses for the extended fuel cycle re-analysis of environmental consequences.

See 15.4.7.3.4.1 for analyses for environmental consequences involving fuel enrichments up to 5.0 wt% U-235 and fuel burnups up to 60,000 megawatt days per metric ton (MWD/MTU).

#### 15.4.7.3.4 Additional Analyses

The following analyses supersede all previously documented USAR fuel handling accident analyses for an inside containment fuel handling accident.

##### 15.4.7.3.4.1 Effects of Extended Fuel Cycle, Fuel Burnup and Increased Fuel Enrichments

In order to evaluate the effects of extended fuel burnup and increased fuel enrichment on the consequences of the fuel handling accident inside containment, the source term and offsite dose calculations were reperformed, assuming that fuel assembly-average burnups could be as great as 60,000 MWD/MTU and that initial fuel enrichments could be as great as 5.0 weight percent uranium-235 (wt% U-235).

All assumptions were made in accordance with Nuclear Regulatory Commission 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 (Safety Guide 25)," dated March 25, 1972).

For the extended fuel cycles, additional source term analyses were performed as described in Section 15.A.7.0 of Appendix 15A. Those analyses maintained the maximum fuel assembly burnup limit of 60,000 MWD/MTU and did not affect previously analyzed environmental consequences of the fuel handling accident.

##### 15.4.7.3.4.2 Results

Reference 38 contains a detailed description of these fuel handling accident analyses. Source terms were determined to be extremely weak functions of fuel enrichment, with lower enrichments in general producing larger offsite doses than higher enrichments. Table 15.4.7-6 contains the source term results for fuel assemblies having initial enrichments of 3.0 and 5.0 wt% U-235. The values in Table 15.4.7-6 are fuel assembly-average fission product activities, and do not represent atmospheric release activities, as is the case in Table 15.4.7-5.

The offsite doses for these fuel handling accident inside containment analyses are shown in Table 15.4.7-4a. In summary, these results meet the acceptance criteria provided in NUREG-0800 and, therefore, are well within the dose guidelines set forth in 10CFR100.

#### 15.4.7.3.4.3 Control Room Dose Analysis

Additional analysis of the fuel handling accident inside containment was completed (Reference 44). This analysis calculated the control room dose for a fuel handling accident inside containment using the following assumptions:

1. The accident occurs 72 hours after reactor shutdown.
2. The noble gas and iodine fuel activities are based on the 3 wt% enrichment U-235 for an extended burnup of 60,000 MWD/MTU and are consistent with Table 15.4.7-6.
3. Noble gas and iodine release fractions are consistent with RG 1.25: 10% of fuel assembly Xe, Kr and I except for Kr-85 which is 30%.
4. All fuel pins (208) of one assembly are assumed to release their activity instantaneously to the pool.
5. All the noble gases and 1% of the iodine are released from the pool to the containment. The activity is assumed to be released from containment to the atmosphere over 2 hours. No credit is taken for containment isolation. The 2 hour atmospheric dispersion coefficient for the control room is  $5.85 \times 10^{-4}$  sec/m<sup>3</sup>. No filtration is assumed prior to atmospheric release.
6. The control room normal HVAC air intake, which is more than 160 feet from the release point, is isolated prior to the release from containment reaching it. The HVAC air intake is automatically isolated upon receipt of a high radiation signal from the station vent.
7. Conservative values of Control Room volume (50,000 ft<sup>3</sup>) and Control Room unfiltered inleakage (59 cfm) were utilized.
8. No credit is taken for the control room emergency ventilation system to remove iodine activity leaked into the control room.
9. Normal control room HVAC is established 2 hours following initiation of the accident.

Table 15.4.7-4a gives the results of the control room dose calculation. The doses are well within the control room dose acceptance criteria as given in GDC 19 of 10CFR50 Appendix A.

#### 15.4.7.3.5 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. That evaluation concluded that steam generator performance does not impact this accident in any way. Therefore, the existing analyses remain applicable with the replacement Steam Generators installed.

TABLE 15.4.7-4<sup>(1)</sup>

Resultant Doses From Fuel-Handling Accident Inside Containment

	<u>Exclusion Area Boundary</u>	<u>LPZ Boundary</u>
Thyroid dose (rem)	44.7	2.33
Whole body dose (rem)	0.17	8.86x10 <sup>-3</sup>

<sup>(1)</sup>See 15.4.7.3.4 referenced re-analyses for resultant doses from the FHA inside containment.

TABLE 15.4.7-4a

Resultant Doses From Fuel Handling Accident Inside Containment - Extended Fuel Burnup (60,000 MWD/MTU)

	<u>Exclusion Area Boundary 0 to 2 hours</u>	<u>LPZ Boundary 0 to 30 days</u>
Thyroid Dose (rem)	62.6	3.26
Whole Body Dose (rem)	0.55	3.0x10 <sup>-2</sup>

Control Room<sup>(1)</sup>  
0 to 2 hours

Thyroid Dose (rem)	17.6
Whole Body Dose (rem)	0.10

<sup>(1)</sup>See 15.4.7.3.4.3 referenced reanalyses from the FHA inside containment.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 15.4.7-5<sup>(1)</sup>

Activity Released to the Atmosphere Due to the Postulated  
Fuel-Handling Accident Inside Containment (Ci)

I-131	$4.26 \times 10^2$
I-132	$3.11 \times 10^{-7}$
I-133	$1.18 \times 10^2$
I-134	$1.47 \times 10^{-22}$
I-135	$6.86 \times 10^{-1}$
Xe-131m	$3.50 \times 10^2$
Xe-133m	$1.28 \times 10^3$
Xe-133	$8.55 \times 10^4$
Xe-135m	0
Xe-135	$5.07 \times 10^2$
Xe-137	0
Xe-138	$5.67 \times 10^{-70}$
Kr-83m	$2.43 \times 10^{-8}$
Kr-85m	$2.93 \times 10^{-1}$
Kr-85	$2.48 \times 10^3$
Kr-87	$3.70 \times 10^{-13}$
Kr-88	$1.23 \times 10^{-3}$
Kr-89	0

<sup>(1)</sup>See 15.4.7.3.4 referenced re-analyses for the activity released from the FHA inside containment.

TABLE 15.4.7-6

Fuel Assembly Fission Product Activities (Curies)  
For Extended Fuel Burnup (60,000 MWD/HTU) (1)

<u>Isotope</u>	<u>Enrichment = 3 wt% U-235</u>	<u>Enrichment = 5 wt% U-235</u>
I-131	5.86E+05	5.81E+05
I-132	5.58E+05	5.56E+05
I-133	1.29E+05	1.30E+05
I-134	1.05E-18	1.07E-18
I-135	7.02E+02	7.03E+02
Xe-131m	8.32E+03	8.18E+03
Xe-133m	2.56E+04	2.55E+04
Xe-133	1.12E+06	1.12E+06
Xe-135m	1.13E+02	1.13E+02
Xe-135	1.35E+04	1.35E+04
Xe-138	0	0
Kr-83m	2.49E-04	2.62E-04
Kr-85m	1.71E+00	1.85E+00
Kr-85	8.18E+03	9.53E+03
Kr-87	1.87E-12	2.06E-12
Kr-88	6.59E-03	7.29E-03

(1) Corrected to include Regulatory Guide 1.25 Radial Peaking Factor and Radioactive Decay for 72 hours.

15.4.8 Effects of Toxic Material Release on the Control Room

In accordance with Sections 2.2 and 6.4, toxic materials are not stored in volumes which would affect control room habitability.

15.4.8.1 Identification of Causes

(Note: Chlorine Tank Cars are no longer on site).

Chlorine is the only toxic material stored on site which could affect control room operation.

Chlorine from a 30-ton railroad car is used to supply the station chlorination system. The major chlorination equipment, which includes the evaporators and dispensers, is located in the Water Treatment Building. One-and-one-half-inch supply piping from the car to this building is in a covered trench, except for passage over the cooling tower channel, where the pipe is encapsulated in a larger pipe.

15.4.8.2 Accident Analysis

The supply car (top of rail El. 585 ft) is approximately 560 feet from the air intake to the control room (El. 654 ft). The closest distance to this intake from the supply pipe is about 380 feet. Major assumptions relating to an accident are as follows:

a. Meteorology:

Pasquill F, wind one meter per second (same as for short-term accident diffusion estimates, Subsection 2.3.4).

b. Tank Car Accident:

A complete rupture is assumed in this case.

c. Supply Line Accident:

A maximum flow rate of 3.89 pounds per second is equivalent to the maximum flow permitted by the two excess-flow valves in the tank car. At this flow rate, the excess-flow valves are designed to close. (Reference: Chlorine Manual, The Chlorine Institute, Inc.)

d. All doors in the control room are closed. Free volume of control room is 7618 cubic feet.

e. Two chlorine detectors with independent essential power supplies are located in the control room fresh air intake vent and two detectors with independent essential power supplies, located at the chlorine tank car, actuate the normal ventilation system intake and exhaust dampers to close so that supply air volumetric flow rate linearly decreases with time.

f. Initial control room chlorine concentration is zero when chlorine detectors are actuated.

## Davis-Besse Unit 1 Updated Final Safety Analysis Report

- g. Effective height of intake above grade is one-half the actual intake height. This assumption was made because the size of the station structures may cause the wind to carry a ground-level chlorine release up and over the building, and this assumption attempts to include funneling effects.
- h. Full-face, self-contained breathing apparatuses will be provided at readily accessible locations in the control room.
- i. Flow into and out of the control room is that supplied by ventilation blowers.
- j. System initial design flow for control and other rooms is 21,920 cfm. Initial air flow to and from the control room is 4,100 cfm.
- k. Control room atmosphere is homogeneous.
- l. The inleakage rate of contaminated air to the control room after isolation is 25 cfm.
- m. The net free volume of the rooms surrounding the control room is 104,234 ft<sup>3</sup>.
- n. All these rooms are isolated during the accident.
- o. In the tank car rupture case, it was assumed that 25 percent of the total chlorine initially flashes to vapor.
- p. In the pipeline break case, it was assumed that 25 percent of the flow flashes as it flows out.
- q. A spill area of 1650 ft<sup>2</sup> was used (reference 16).



Control Room Concentration Model:

a. Atmospheric Diffusion:

1. The diffusion equation for a continuous ground-level release is: (reference 14)

$$X = \frac{Q_c}{\pi \sigma_y \sigma_z u} \exp \left[ -1/2 \left[ \frac{y}{\sigma_y} \right]^2 \right] \exp \left[ -1/2 \left[ \frac{z}{\sigma_z} \right]^2 \right] \quad (1)$$

where  $X$  = the short-term concentration,  $g/m^3$

$Q_c$  = amount of chlorine as continuous release,  $g/sec$

$u$  = wind speed,  $m/sec$

$\sigma_y$  = horizontal standard deviation of the plume,  $m$

$\sigma_z$  = vertical standard deviation of the plume,  $m$

2. The diffusion equation for an instantaneous (puff) ground-level release with a finite initial volume (reference 15) is:

$$X = \frac{Q_1}{7.87 [\sigma_{x,y}^2 + \sigma_1^2] [\sigma_z^2 + \sigma_1^2]^{1/2}} \exp \left[ -1/2 \left[ \frac{x^2}{\sigma_x^2 + \sigma_1^2} + \frac{y^2}{\sigma_y^2 + \sigma_1^2} + \frac{z^2}{\sigma_z^2 + \sigma_1^2} \right] \right] \quad (2)$$

where  $X$  = concentration at coordinates  $x, y, z$  from the center of the puff,  $g/m^3$

$Q_1$  = puff release quantity,  $g$

$\sigma_x, \sigma_y, \sigma_z$  = standard deviations of the gas concentrate in the horizontal alongwind, horizontal crosswind, and vertical crosswind directions, respectively (assume  $\sigma_x = \sigma_y$ ),  $m$

$7.87 = 2^{1/2} \pi^{3/2}$

## Davis-Besse Unit 1 Updated Final Safety Analysis Report

$\sigma_1$  = initial standard deviation of the puff, m

$$= \frac{Q_1}{7.87X_0} \text{ where } X_0 \text{ is the density of}$$

chlorine at standard conditions, g/m<sup>3</sup>

The variation of unit concentration at a specific stationary receptor is determined by evaluating X in the exponential term in Equation (2) as follows:

$$X = D = ut$$

where D is the source - receptor distance, m

u is the wind speed, m/sec

t is the time after release, sec

### b. Dilution Inside Control Room:

The differential equation used to describe chlorine concentration, C, inside the control room is

$$\frac{dC}{dt} = \frac{Q}{V} (X - C)$$

where Q = flow rate of air, cfm

- i. before damper closed - normal intake rate
- ii. while damper is closing - linearly decreasing
- iii. after damper is closed - inleakage rate

V = net free volume of control room, ft<sup>3</sup>

X = outside concentration of both puff and continuous release, ppm

- i. before damper is closed - concentration at damper
- iii. after damper is closed - concentration at subject rooms

### 15.4.8.3 Results of the Analysis

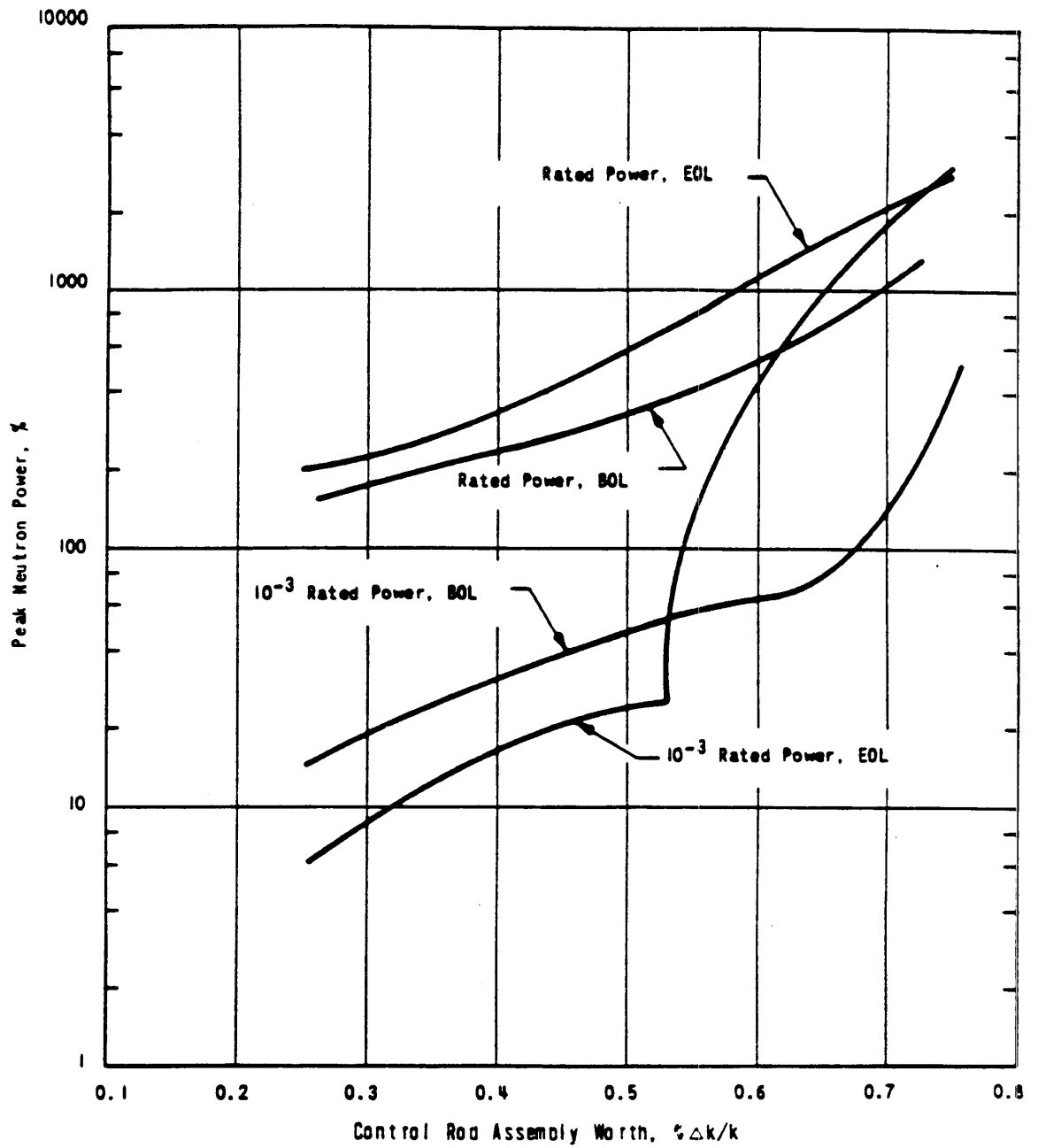
The following assumptions were made in the analysis of the chlorine-release accident. No credit was taken for the channeling of the cold and dense chlorine gas cloud around buildings and along ditches. The chlorine detectors were located at the chlorine tank car for the tank car accident and at the air intake for the pipe break and were assumed to have a response time of five seconds at a set point of 5 ppm. The transport time from the air intake to the isolation damper was determined to be five seconds, and a five-second closing time of the isolation dampers was used.

A 25-cfm inleakage of contaminated air was considered after the isolation dampers were closed. For the tank car accident, the chlorine concentration inside the control room when the isolation dampers closed was 0.0 ppm, and the maximum chlorine concentration reached was 16.0 ppm after approximately 14 hours. This was assuming that the control room ventilation was not restarted after the chlorine cloud had passed. The operators in the control room have well over two minutes to put on self-contained breathing apparatus. Five ppm is not reached until approximately 1.9 hours after the chlorine spill. This is in accordance with Regulatory Guide 1.95 (ref. 17.)

For the pipeline break case, the maximum chlorine concentration reached inside the control room is 4.02 ppm. Plots of the chlorine concentration in the control room versus time are presented in Figures 15.4.8-1 and 15.4.8-2.

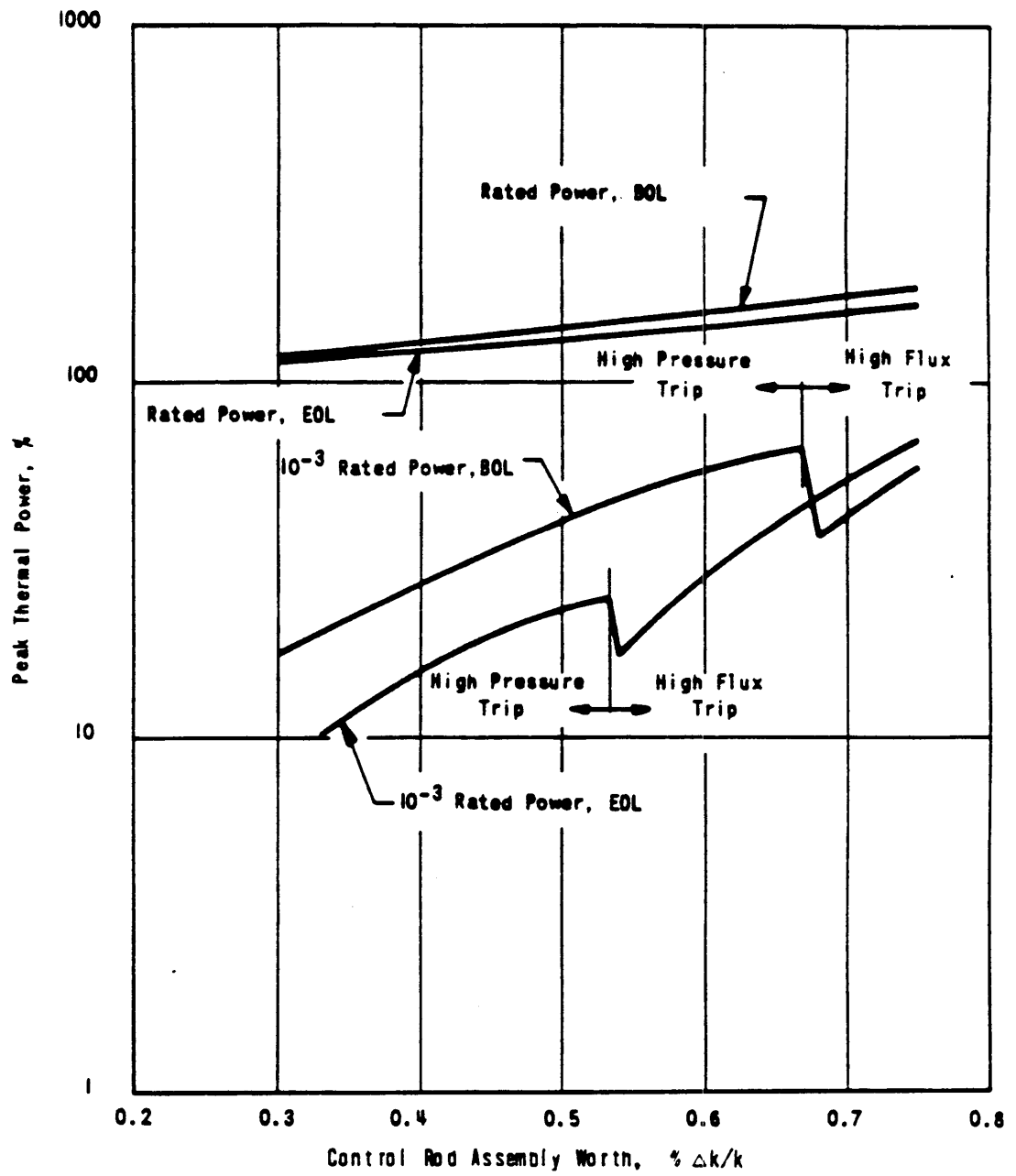
#### 15.4.8.3.1 Impact of Replacement Steam Generators

As part of the Steam Generator Replacement Project, AREVA performed an evaluation (reference 64) to determine the impact of the replacement Steam Generators on the analyses of record. That evaluation concluded that steam generator performance does not impact this accident in any way. Therefore, the existing analyses remain applicable with the replacement Steam Generators installed.



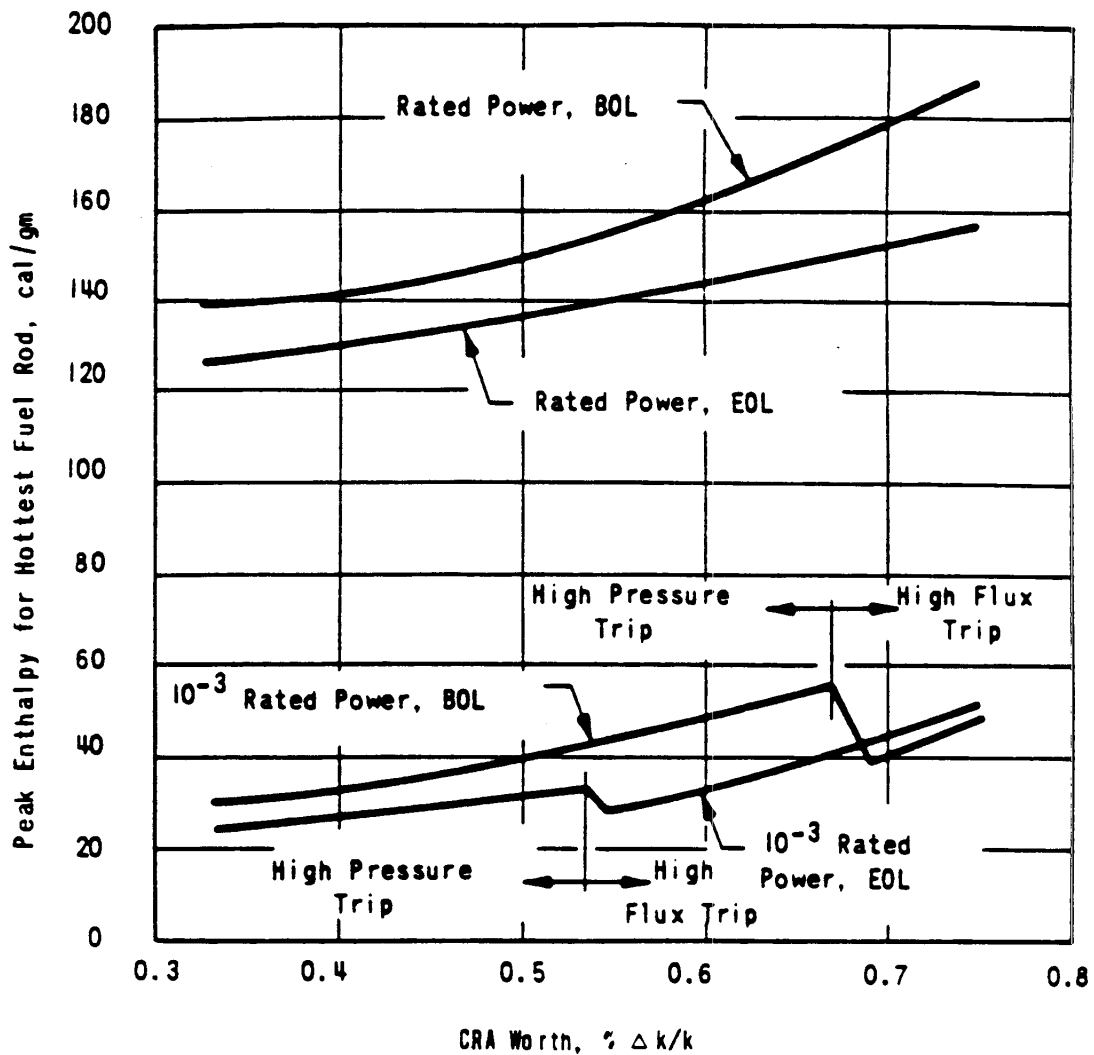
DAVIS-BESSE NUCLEAR POWER STATION  
 PEAK NEUTRON POWER AS A FUNCTION OF  
 EJECTED CRA WORTH  
 FIGURE 15.4.3-1

REVISION 0  
 JULY 1982



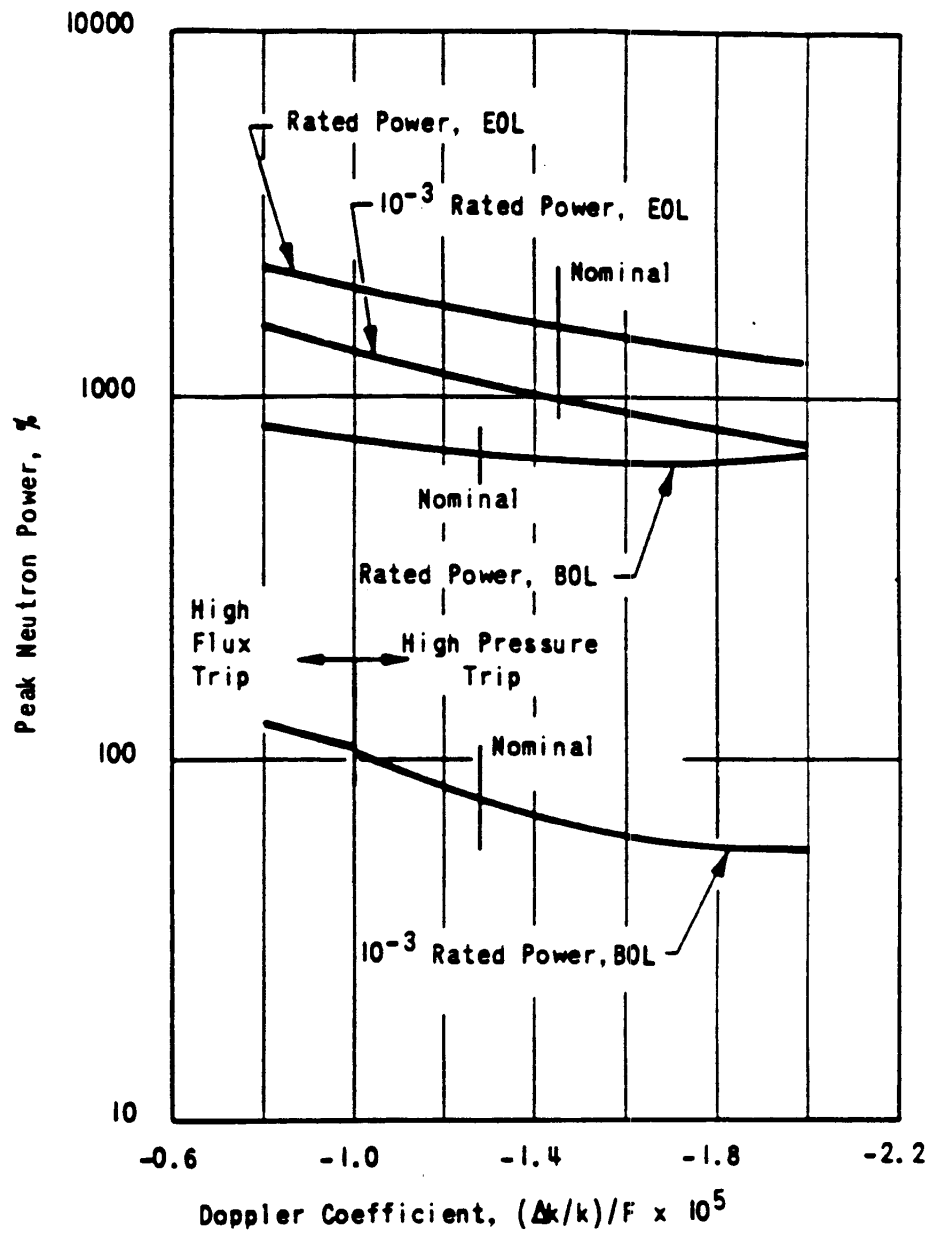
DAVIS-BESSE NUCLEAR POWER STATION  
 PEAK THERMAL POWER AS A FUNCTION OF EJECTED  
 CRA WORTH  
 FIGURE 15.4.3-2

REVISION 0  
 JULY 1982



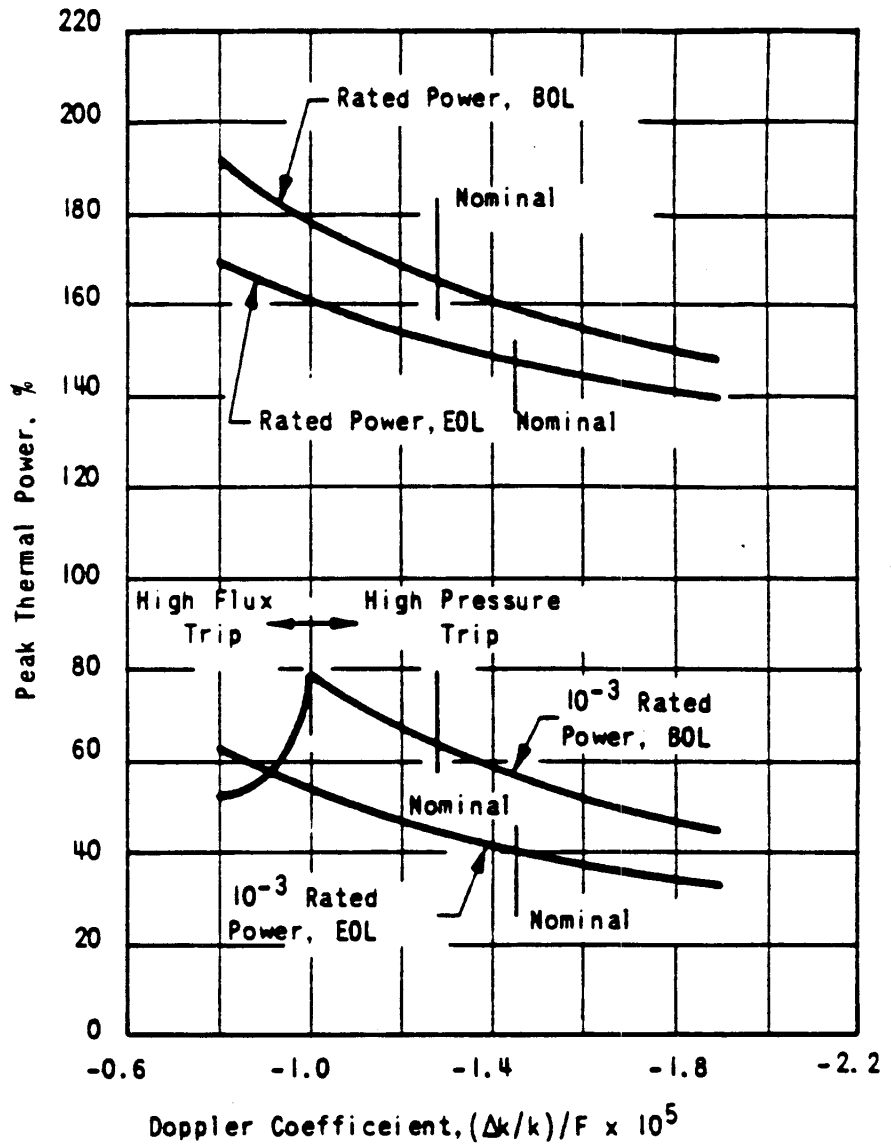
DAVIS-BESSE NUCLEAR POWER STATION  
 PEAK ENTHALPY OF HOTTEST FUEL ROD AS A  
 FUNCTION OF EJECTED CRA WORTH  
 FIGURE 15.4.3-3

REVISION 0  
 JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION  
 PEAK NEUTRON POWER AS A FUNCTION OF DOPPLER  
 COEFFICIENT FOR AN EJECTED CRA WORTH  
 OF 0.65%  $\Delta k/k$  AT BOTH  $10^{-3}$  RATED  
 POWER AND RATED POWER  
 FIGURE 15.4.3-4

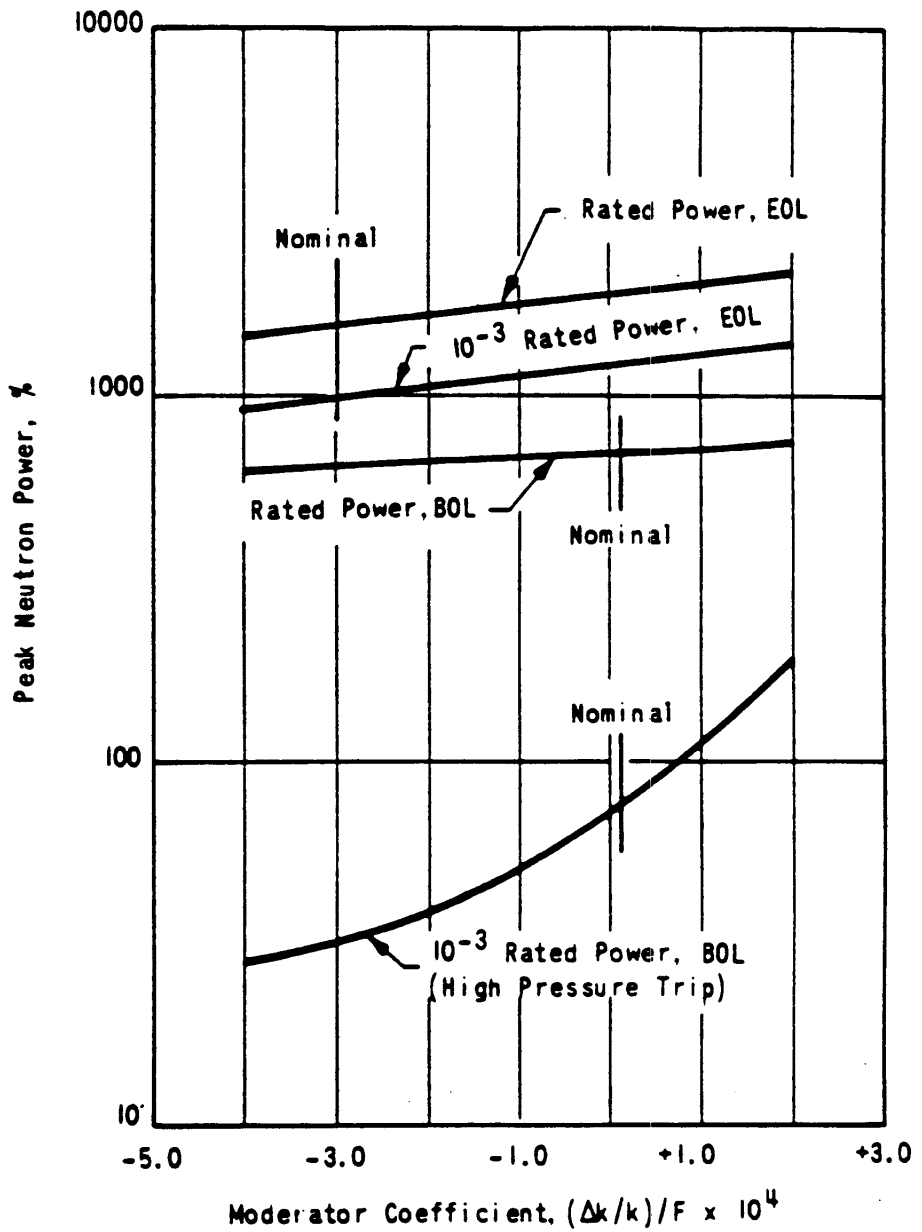
REVISION 0  
 JULY 1982



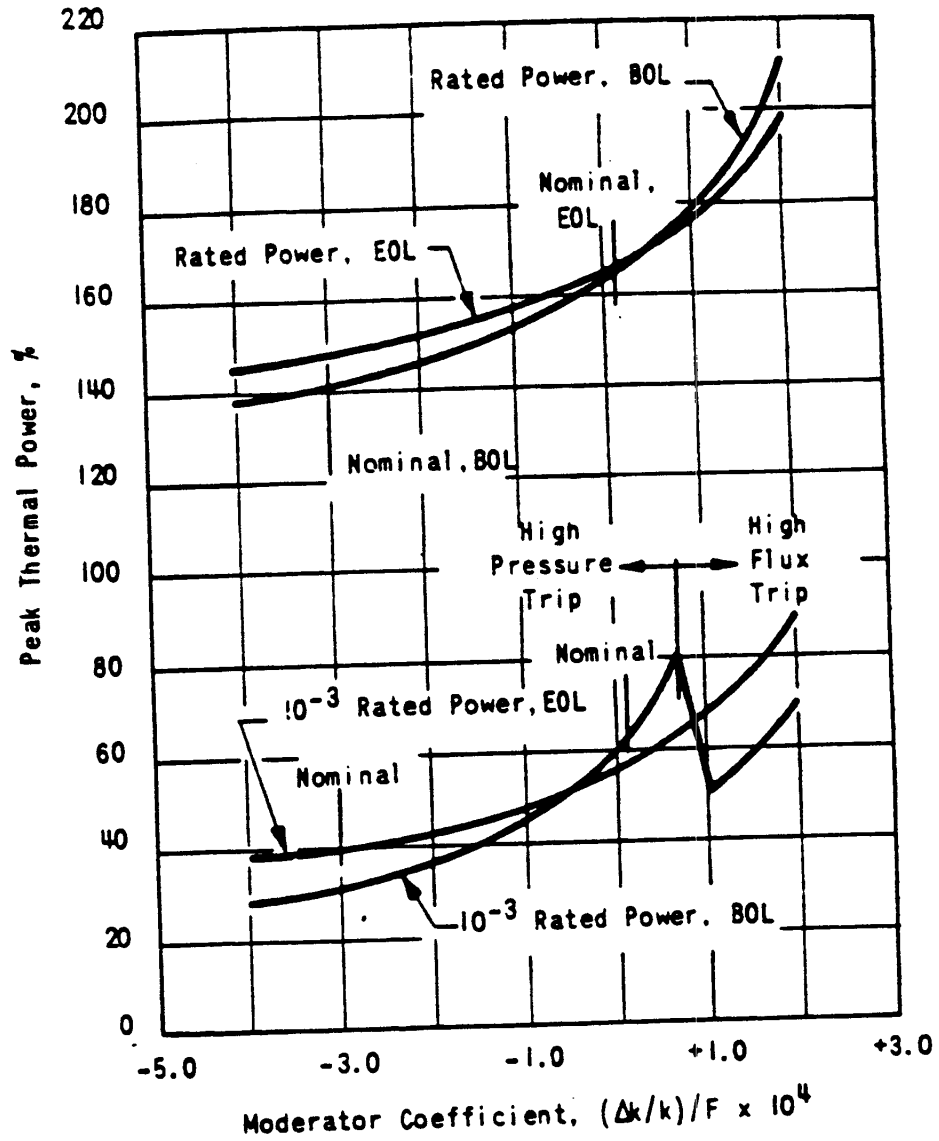
DAVIS-BESSE NUCLEAR POWER STATION  
 PEAK THERMAL POWER AS A FUNCTION OF DOPPLER  
 COEFFICIENT FOR AN EJECTED CRA WORTH  
 OF 0.65%  $\Delta k/k$  AT BOTH  $10^{-3}$   
 RATED POWER AND RATED POWER  
 FIGURE 15.4.3-5

REVISION 0  
 JULY 1982

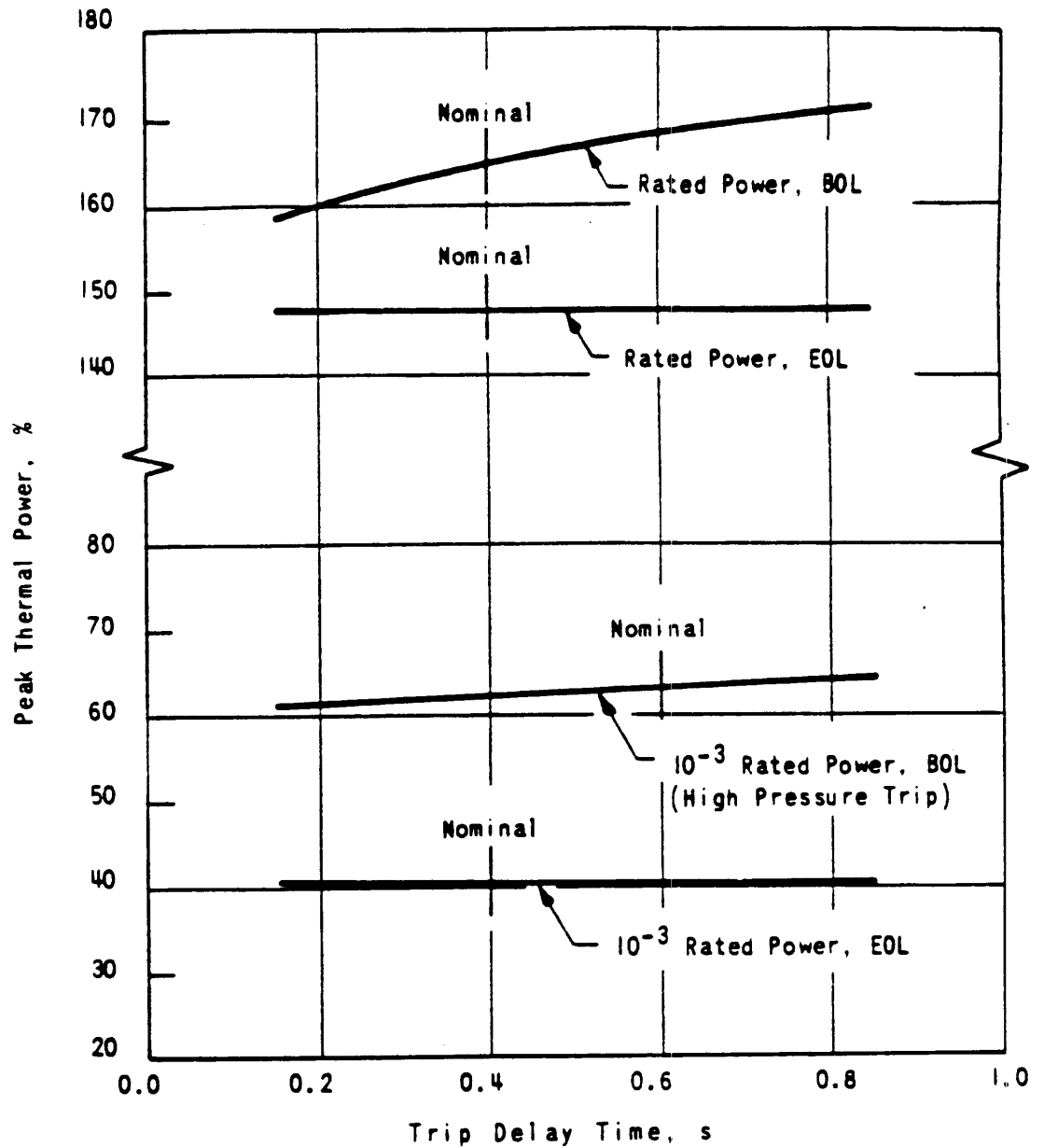




DAVIS-BESSE NUCLEAR POWER STATION  
 PEAK NEUTRON POWER AS A FUNCTION OF MODERATOR  
 COEFFICIENT FOR AN EJECTED CRA WORTH  
 OF 0.65%  $\Delta k/k$  AT BOTH  $10^{-3}$   
 RATED POWER AND RATED POWER  
 FIGURE 15.4.3-6

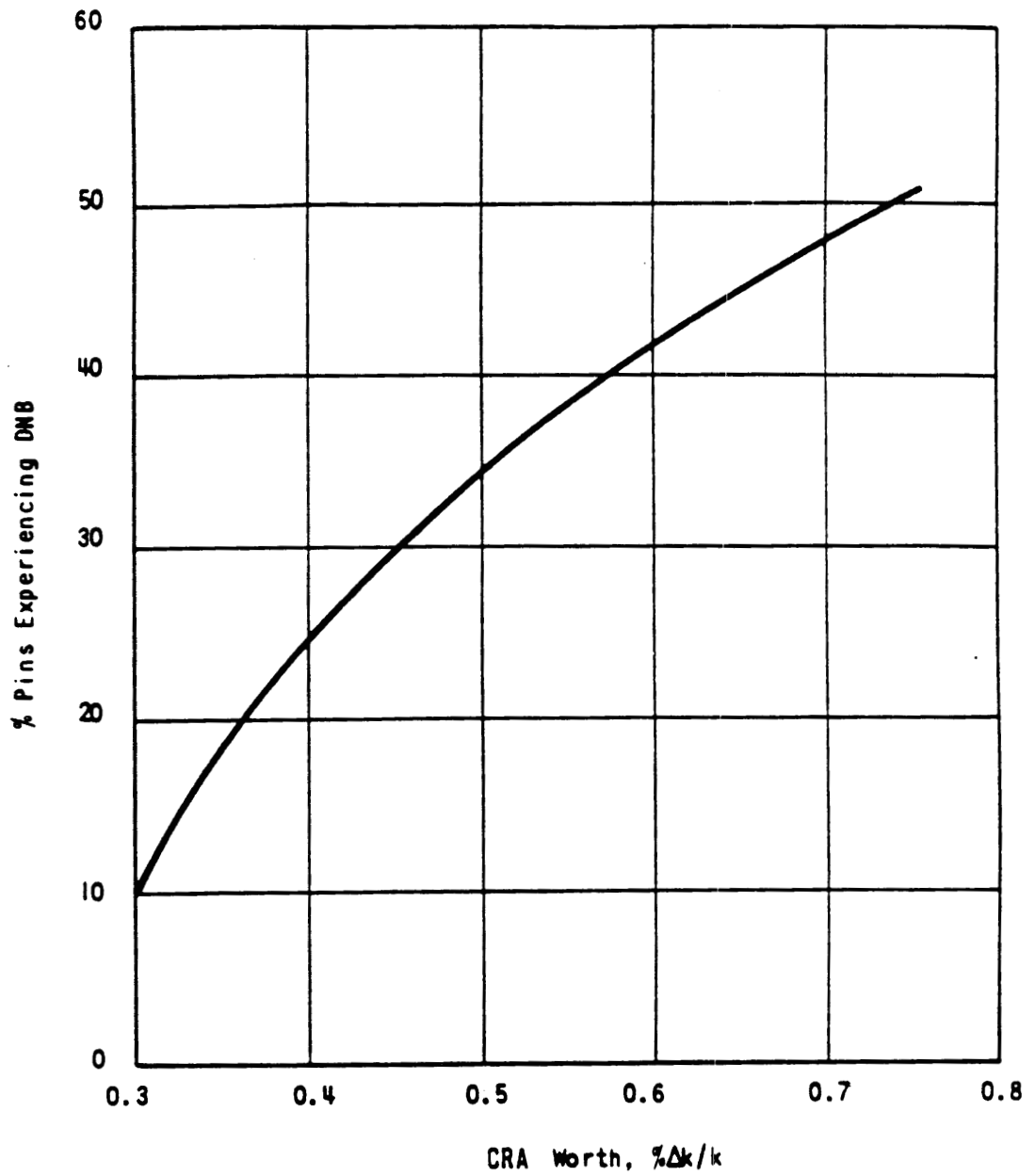


DAVIS-BESSE NUCLEAR POWER STATION  
 PEAK THERMAL POWER AS A FUNCTION OF MODERATOR  
 COEFFICIENT FOR AN EJECTED CRA WORTH  
 OF 0.65%  $\Delta k/k$  AT BOTH  $10^{-3}$   
 RATED POWER AND RATED POWER  
 FIGURE 15.4.3-7



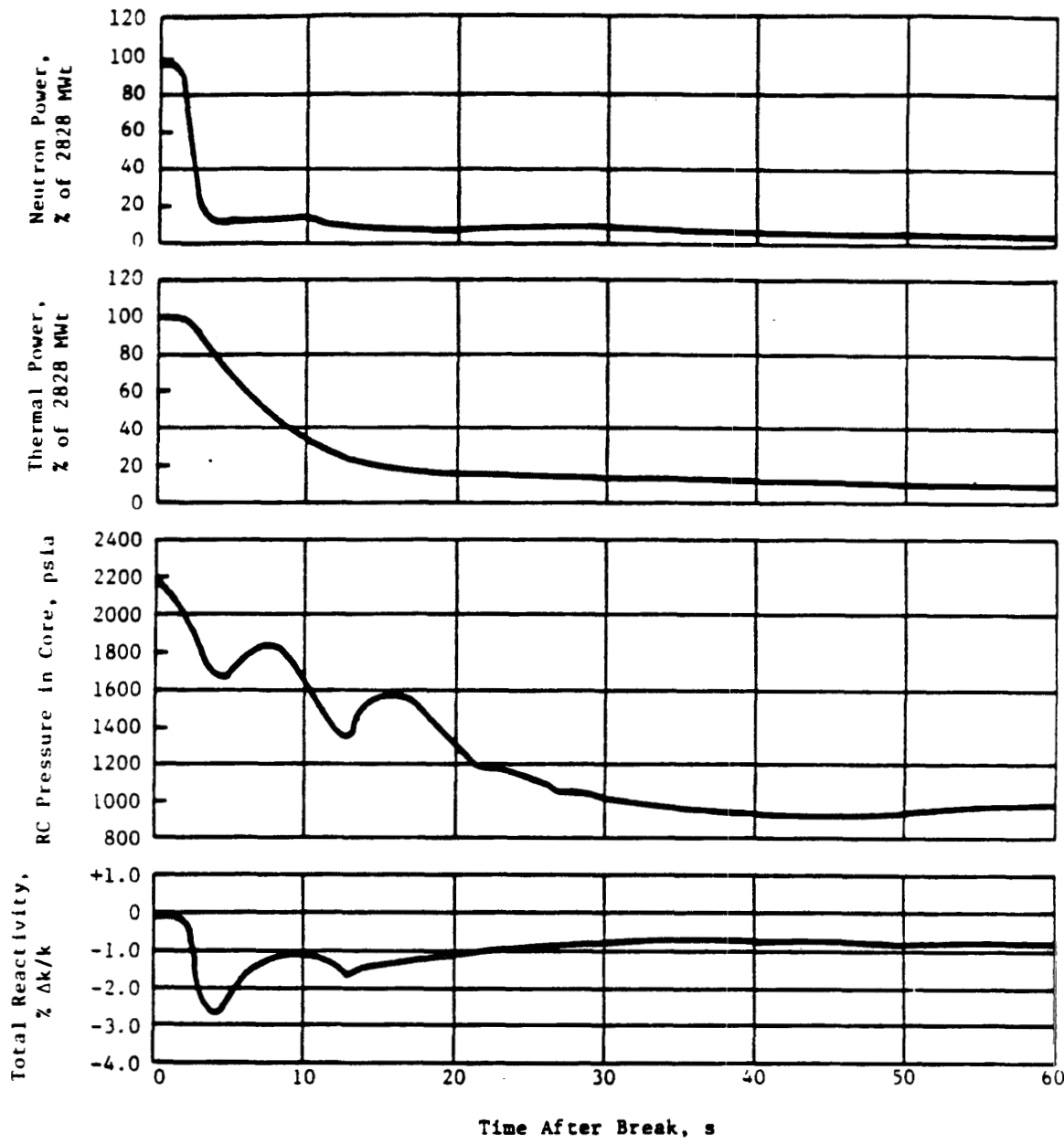
DAVIS-BESSE NUCLEAR POWER STATION  
 PEAK THERMAL POWER AS A FUNCTION OF TRIP  
 DELAY TIME FOR AN EJECTED CRA WORTH  
 OF 0.65%  $\Delta k/k$  AT BOTH  $10^{-3}$   
 RATED POWER AND RATED POWER  
 FIGURE 15.4.3-8

REVISION 0  
 JULY 1982



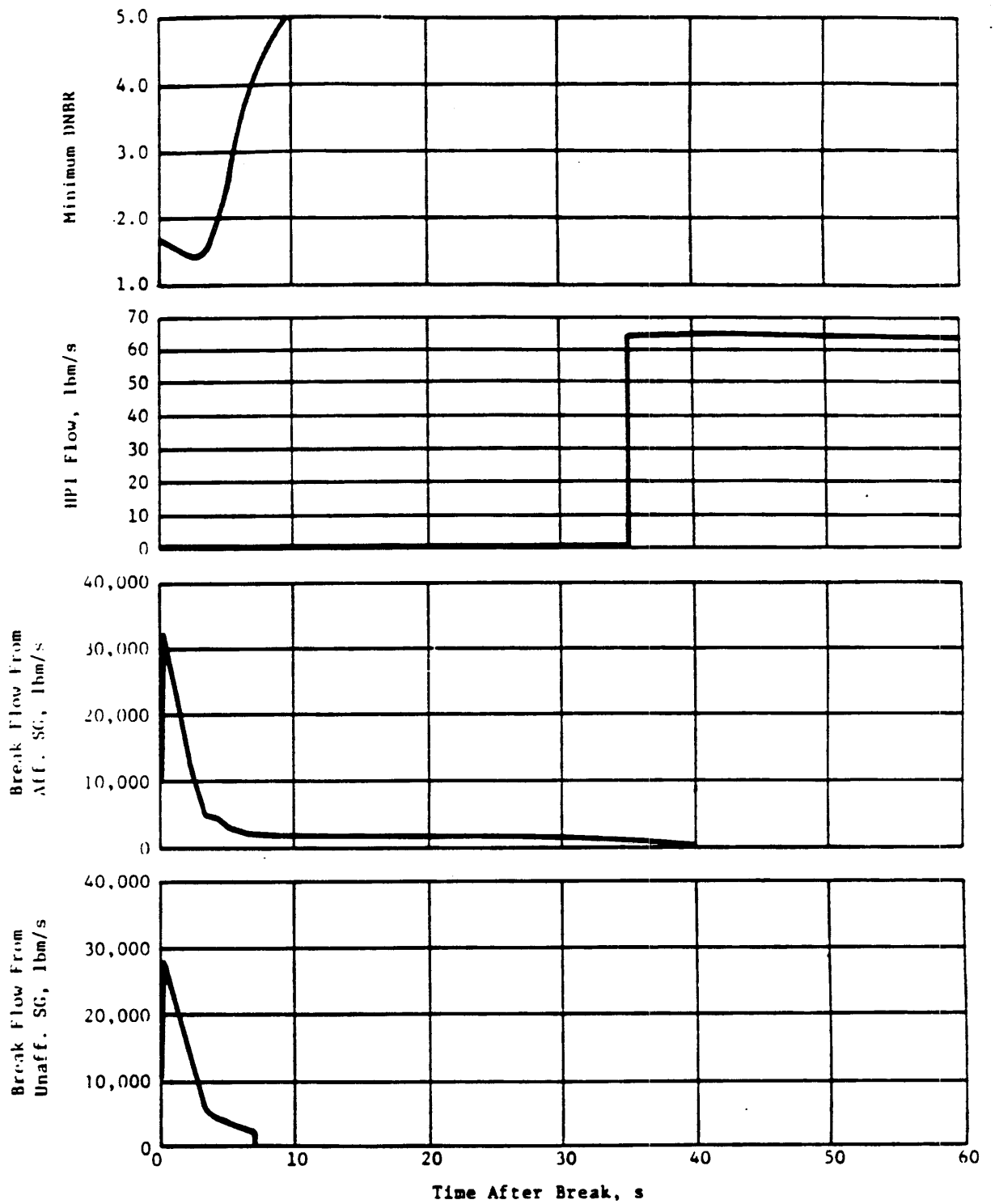
DAVIS-BESSE NUCLEAR POWER STATION  
PERCENT PINS EXPERIENCING DNB AS A FUNCTION  
OF EJECTED CRA WORTH AT  
RATED POWER, BOL  
FIGURE 15.4.3-9

REVISION 0  
JULY 1982



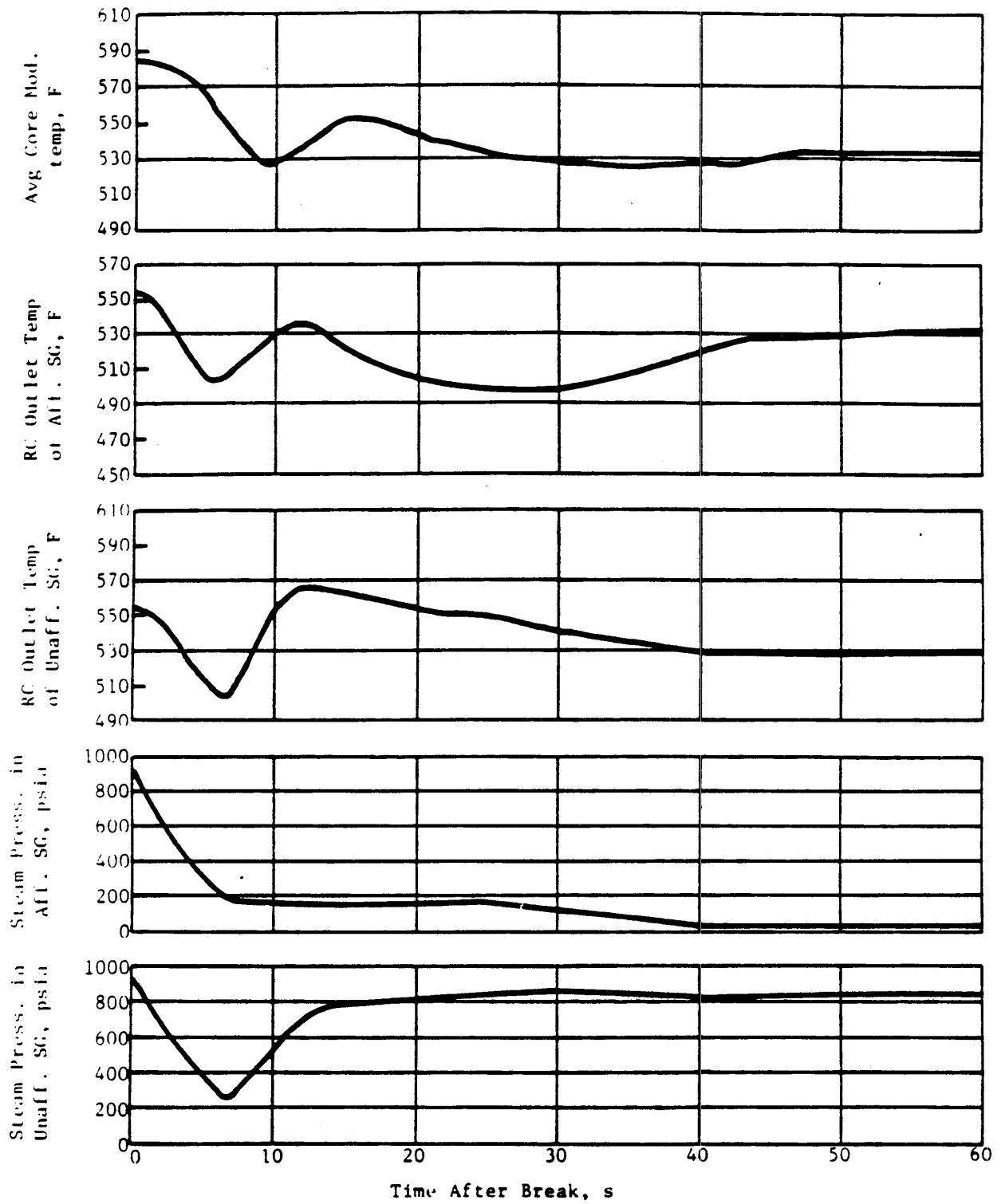
DAVIS-BESSE NUCLEAR POWER STATION  
 DOUBLE-ENDED RUPTURE OF 36-INCH STEAM  
 LINE BETWEEN STEAM GENERATOR  
 AND MAIN STEAM ISOLATION VALVE  
 FIGURE 15.4.4-1

REVISION 0  
 JULY 1982



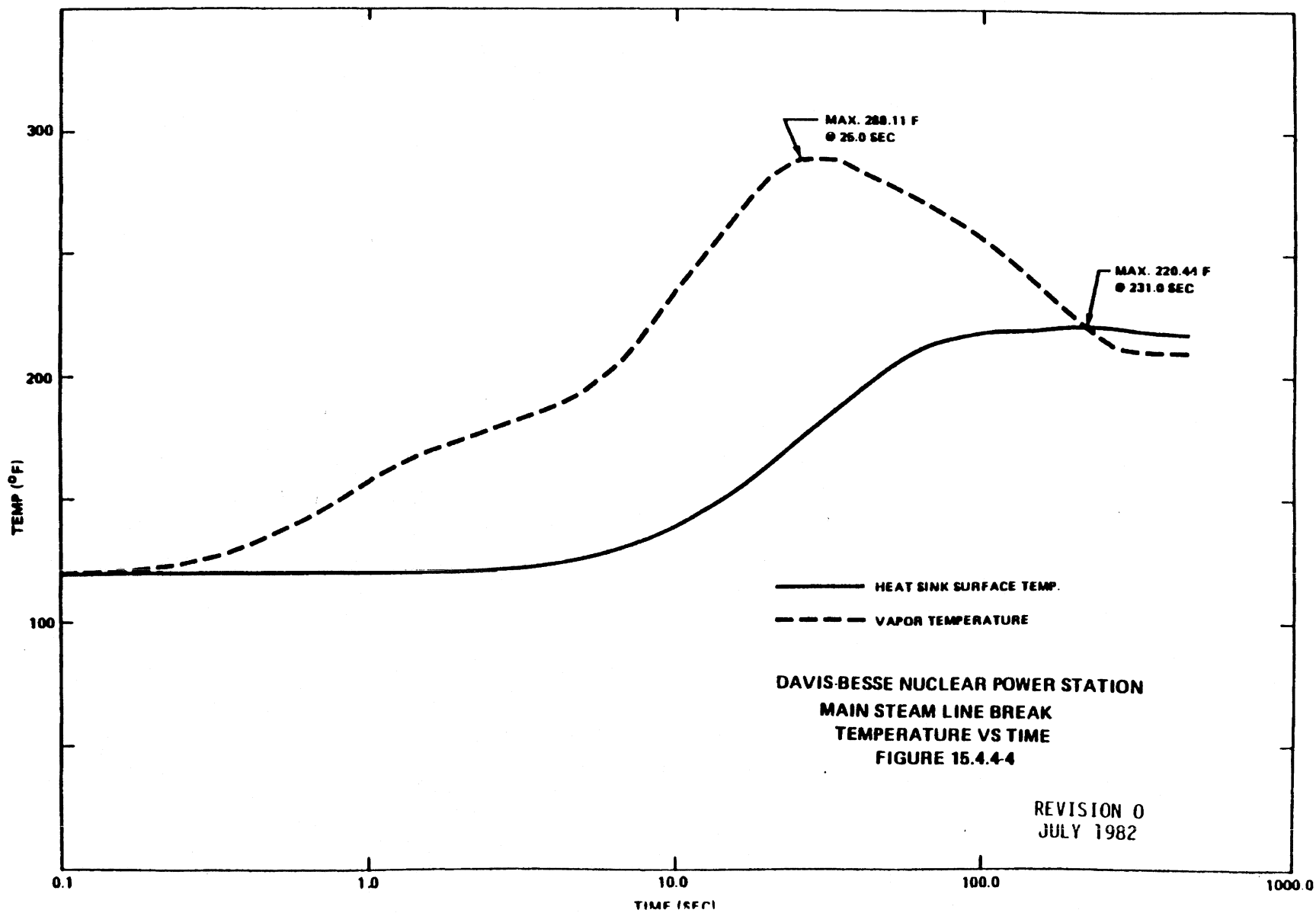
DAVIS-BESSE NUCLEAR POWER STATION  
 DOUBLE-ENDED RUPTURE OF 36-INCH STEAM  
 LINE BETWEEN STEAM GENERATOR  
 AND MAIN STEAM ISOLATION VALVE  
 FIGURE 15.4.4-2

REVISION 0  
 JULY 1982

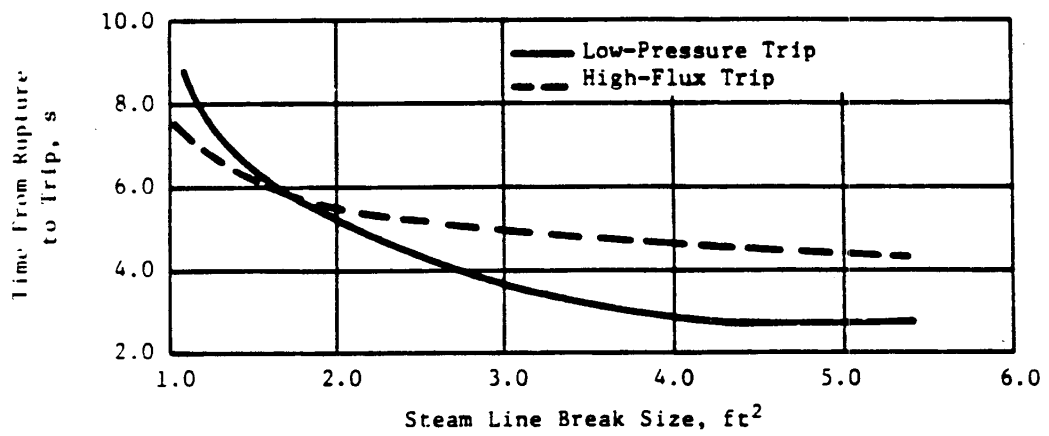


DAVIS-BESSE NUCLEAR POWER STATION  
 DOUBLE-ENDED RUPTURE OF 36-INCH STEAM  
 LINE BETWEEN STEAM GENERATOR  
 AND MAIN STEAM ISOLATION VALVE  
 FIGURE 15.4.4-3

REVISION 0  
 JULY 1982



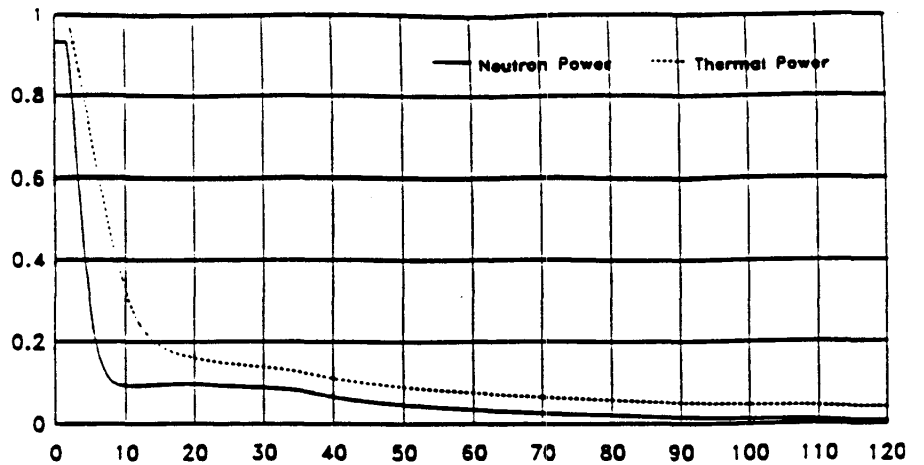




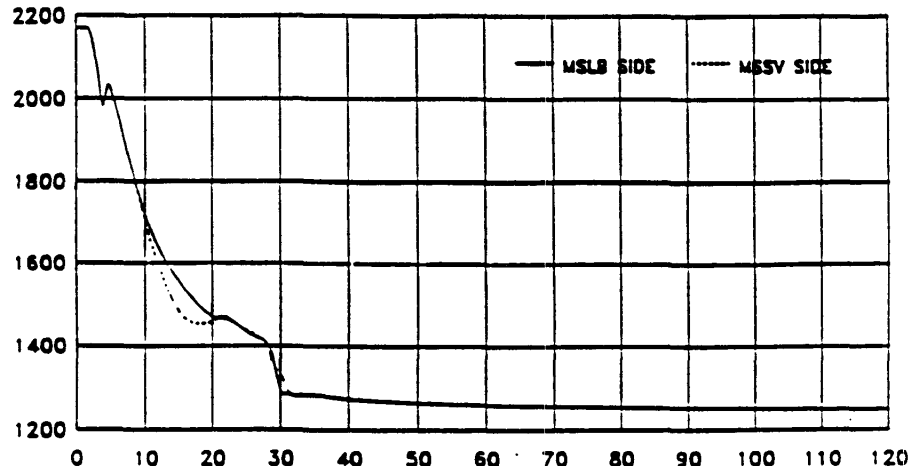
DAVIS-BESSE NUCLEAR POWER STATION  
 TIME FROM RUPTURE TO TRIP  
 (INCLUDING DELAYS) VERSUS  
 STEAM LINE BREAK SIZE

FIGURE 15.4.4-5

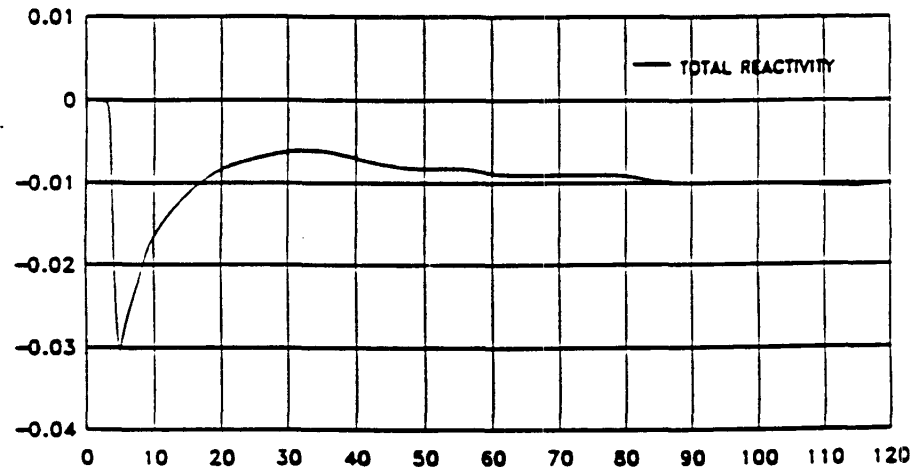
Reactor Power,  
Fraction of 2828 MWT



RCS Pressure (Hot Leg),  
psia



Total Reactivity,  
 $\Delta k/k$



Note: \* MSLB occurs  
at 2 seconds

Transient Time, sec\*

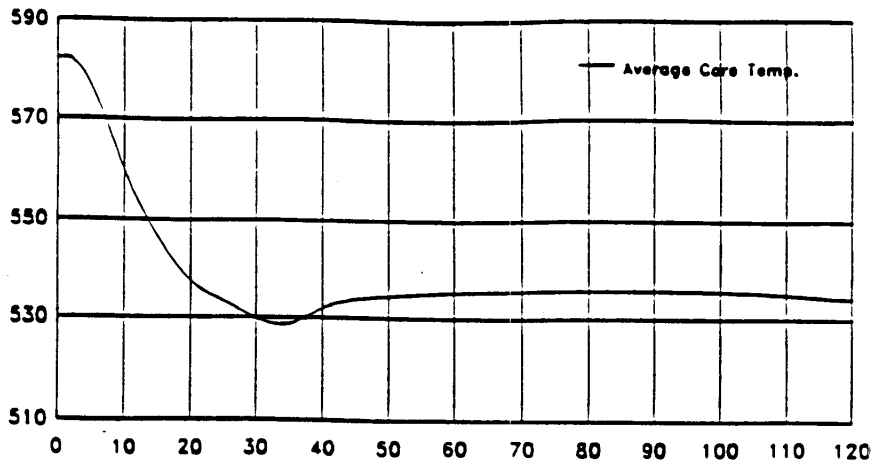
DAVIS-BESSE NUCLEAR POWER STATION

Double-Ended Rupture of 36-inch Steam Line  
Between Steam Generator and Main Steam  
Isolation Valve with Safety Valve Stuck  
Open at Rupture on Unaffected Steam  
Generator

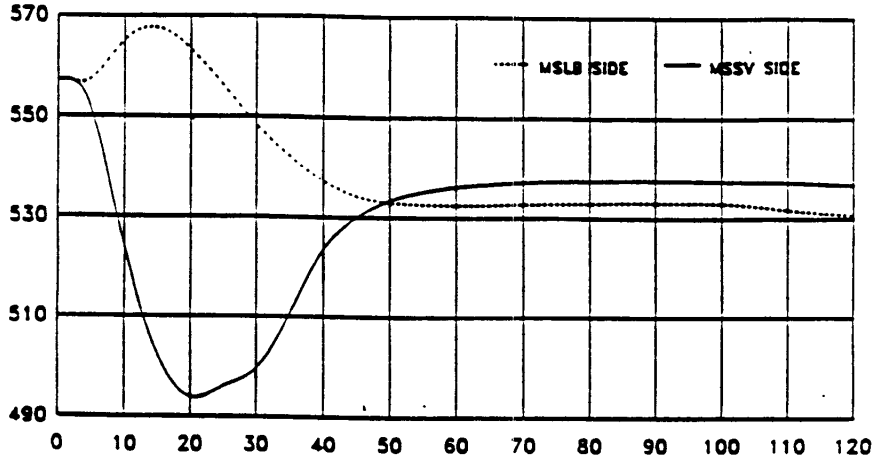
Figure 15.4.4-6

REVISION 14  
JULY 1991

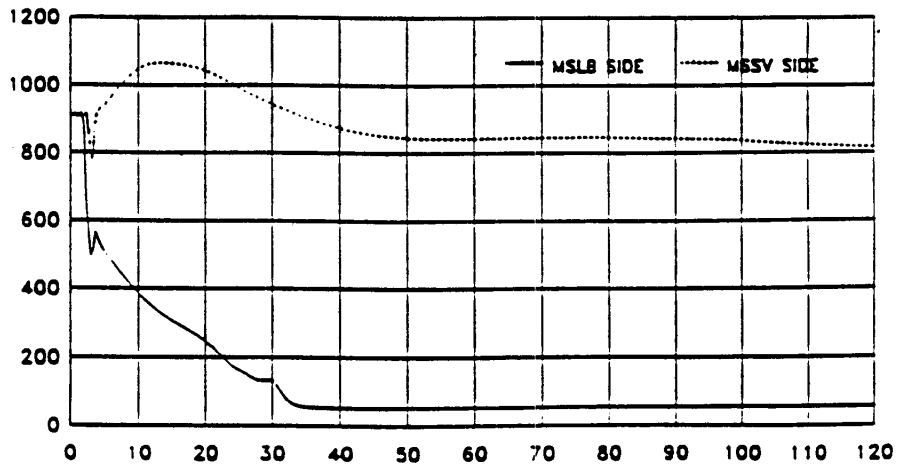
Average Core Moderator  
Temperature, °F



RCS Cold Leg Temperature,  
°F



Steam Generator  
Secondary Pressures,  
psia



Note: \* MSLB occurs  
at 2 seconds

Transient Time, sec\*

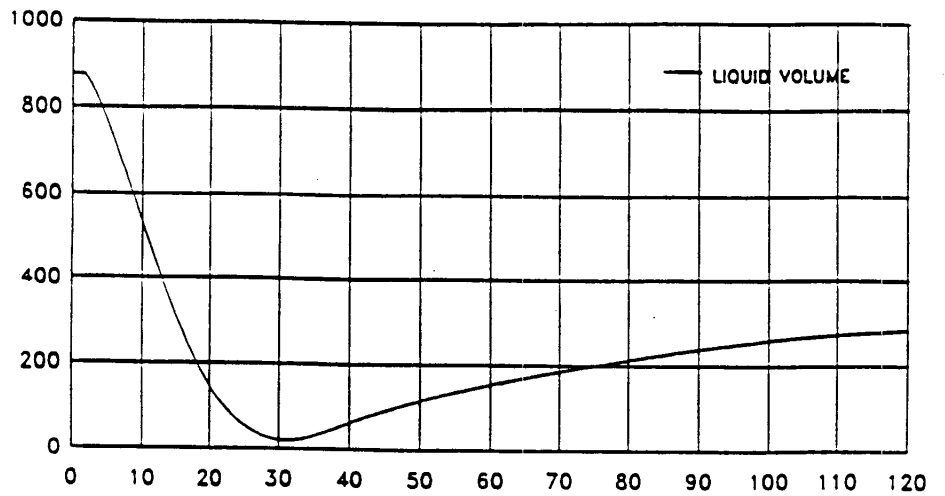
DAVIS-BESSE NUCLEAR POWER STATION

Double-Ended Rupture of 36-inch Steam Line  
Between Steam Generator and Main Steam  
Isolation Valve with Safety Valve Stuck  
Open at Rupture on Unaffected Steam  
Generator

Figure 15.4.4-7

REVISION 14  
JULY 1991

Pressurizer Liquid  
Volume, ft<sup>3</sup>



Note: \* MSLB occurs  
at 2 seconds

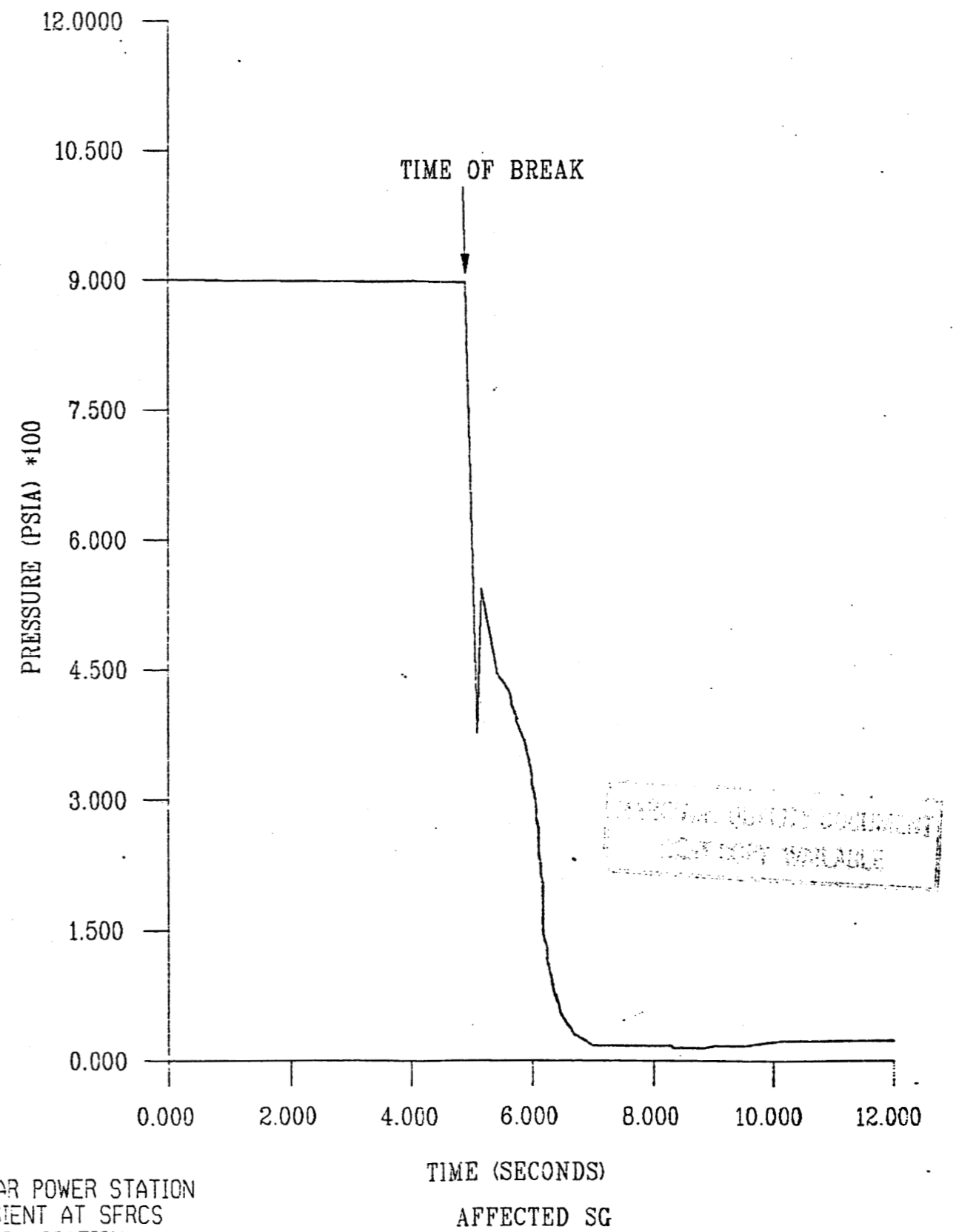
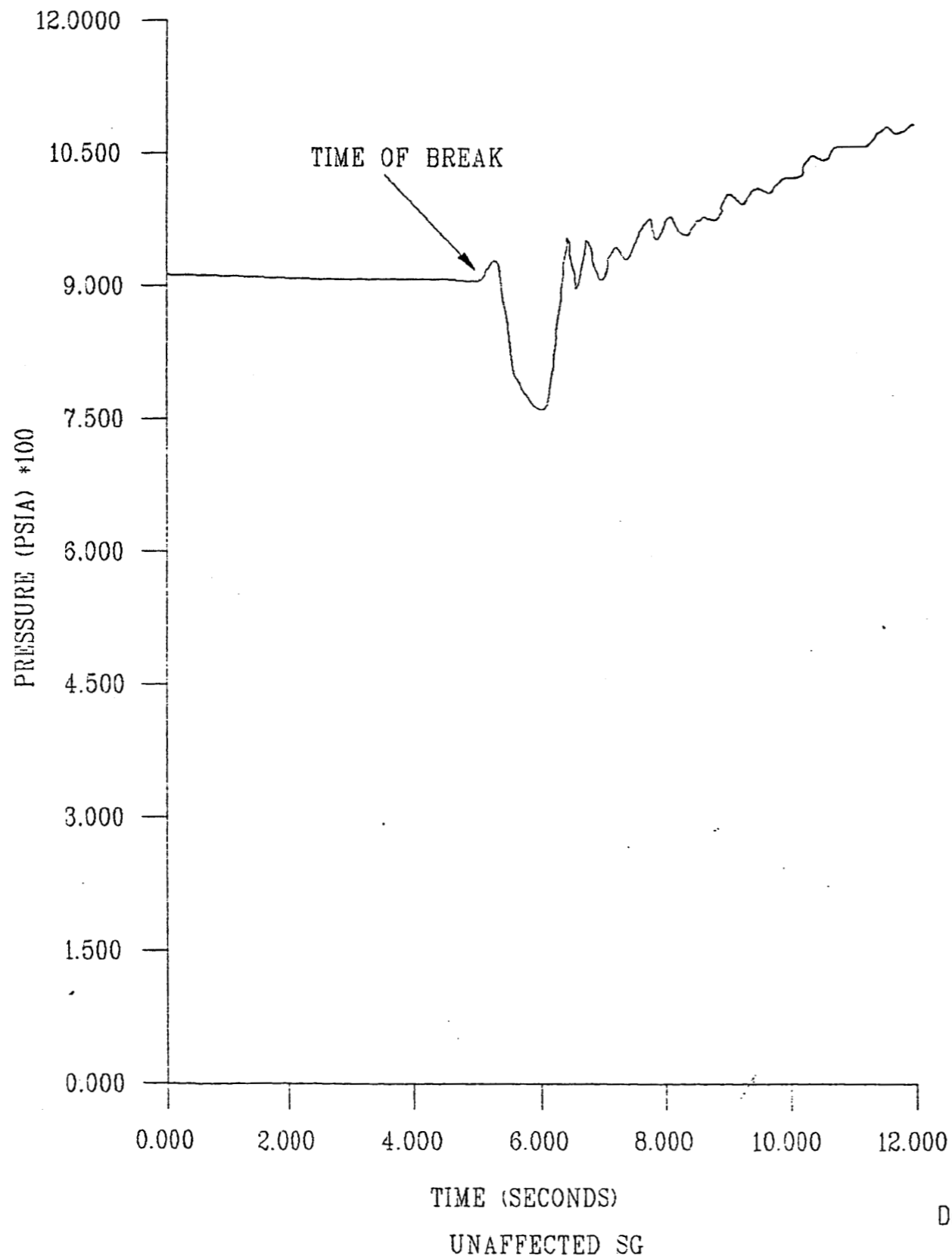
Transient Time, sec\*

DAVIS-BESSE NUCLEAR POWER STATION

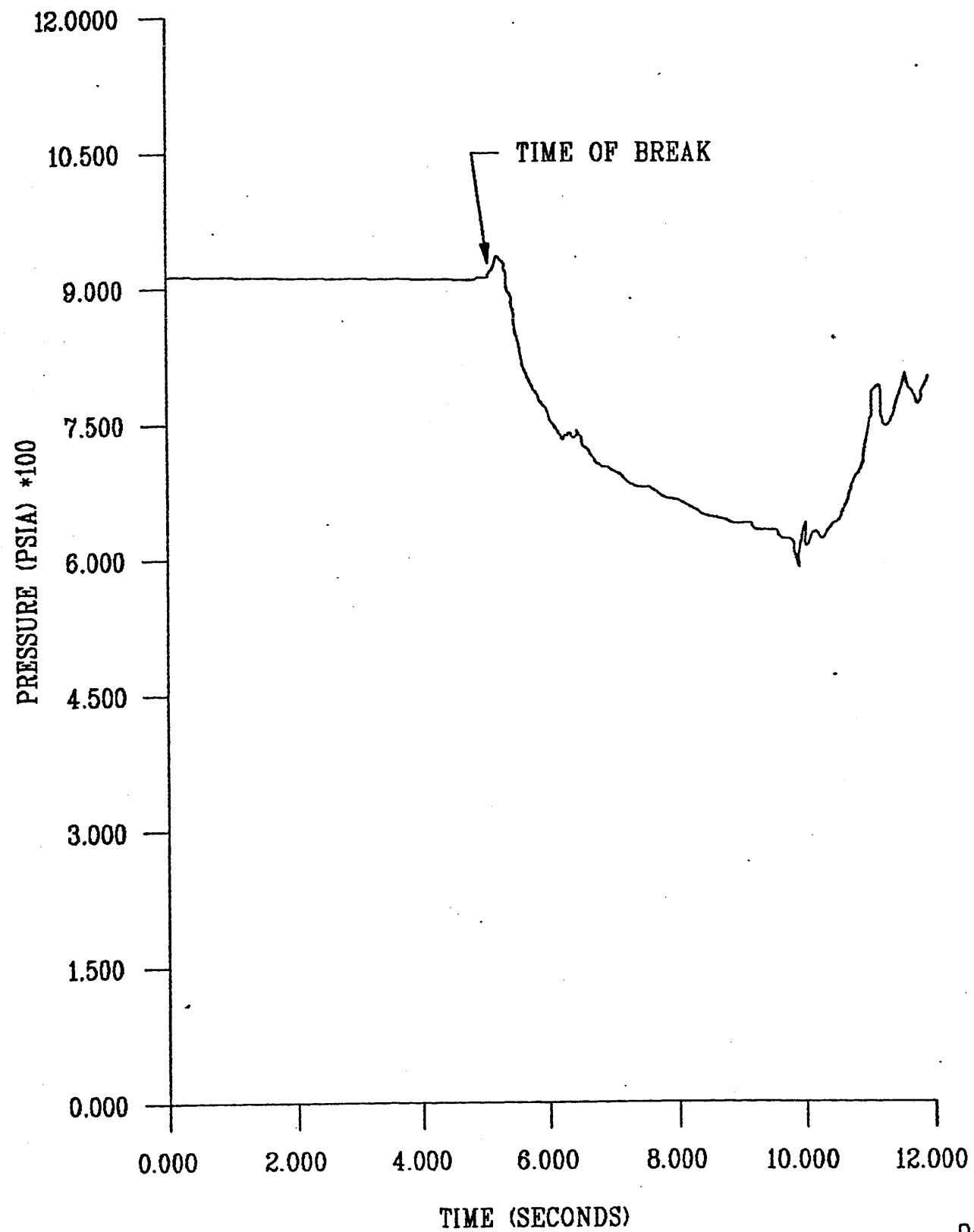
Double-Ended Rupture of 36-inch Steam Line  
Between Steam Generator and Main Steam  
Isolation Valve with Safety Valve Stuck  
Open at Rupture on Unaffected Steam  
Generator

Figure 15.4.4-8

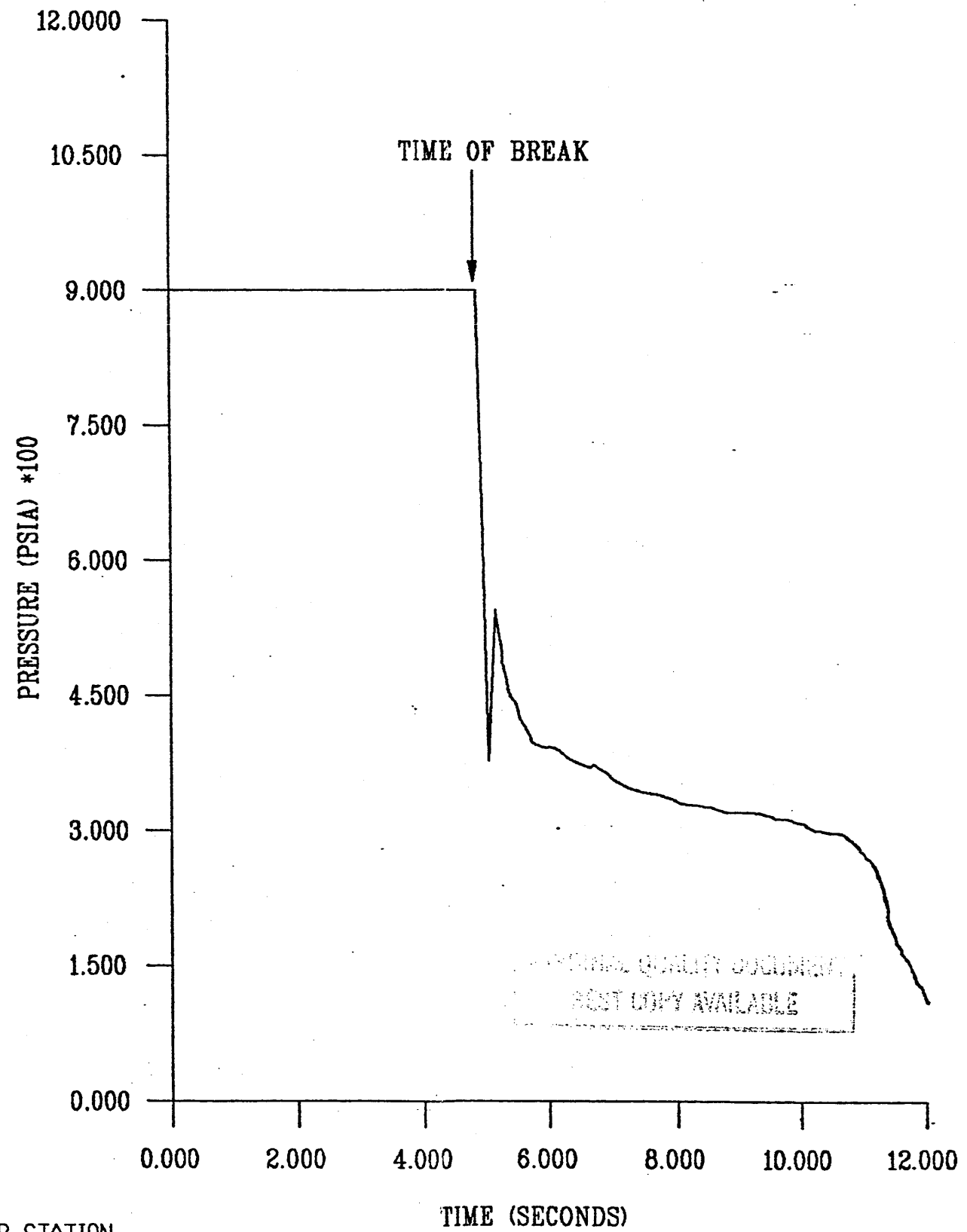
REVISION 14  
JULY 1991



DAVIS-BESSE NUCLEAR POWER STATION  
 PRESSURE TRANSIENT AT SFRCS  
 PRESSURE TAP LOCATION  
 FIGURE 15.4.4-9



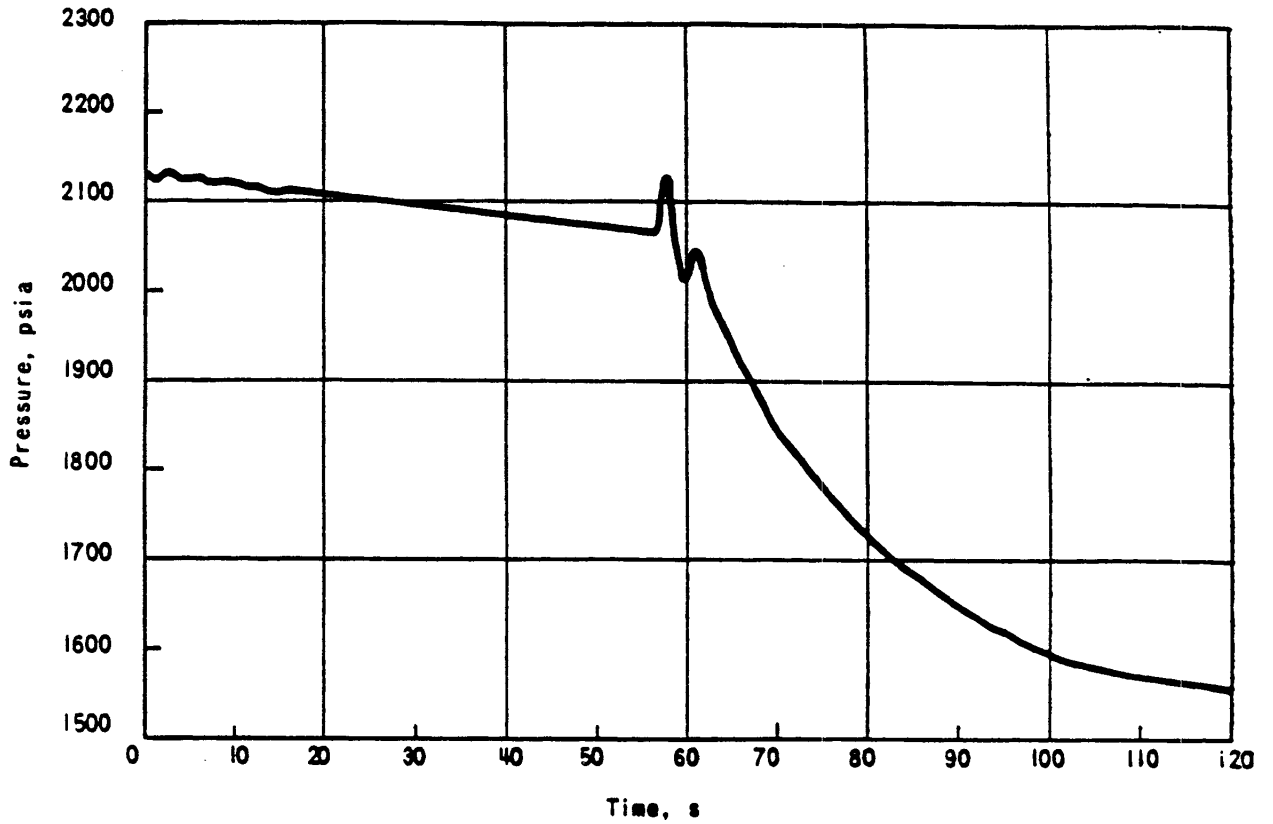
UNAFFECTED SG



AFFECTED SG

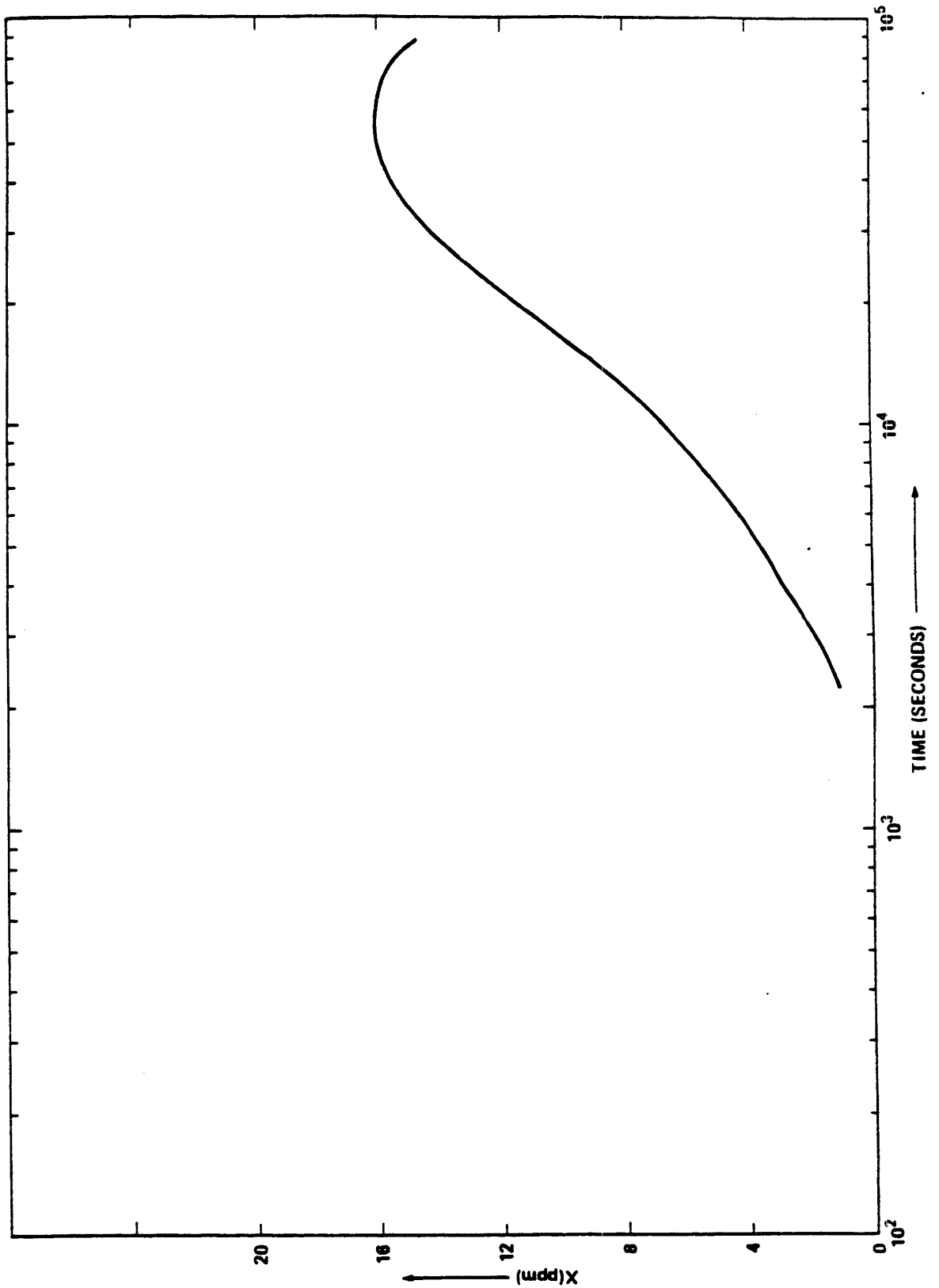
DAVIS-BESSE NUCLEAR POWER STATION  
 PRESSURE TRANSIENT AT SFRCS  
 PRESSURE TAP LOCATION  
 FIGURE 15.4.4-10

Revision 5  
 July, 1987



DAVIS-BESSE NUCLEAR POWER STATION  
REACTOR COOLANT PRESSURE AS A FUNCTION  
OF TIME FOR THE COMPLETE SEVERANCE  
OF A LETDOWN LINE  
FIGURE 15.4.5-1

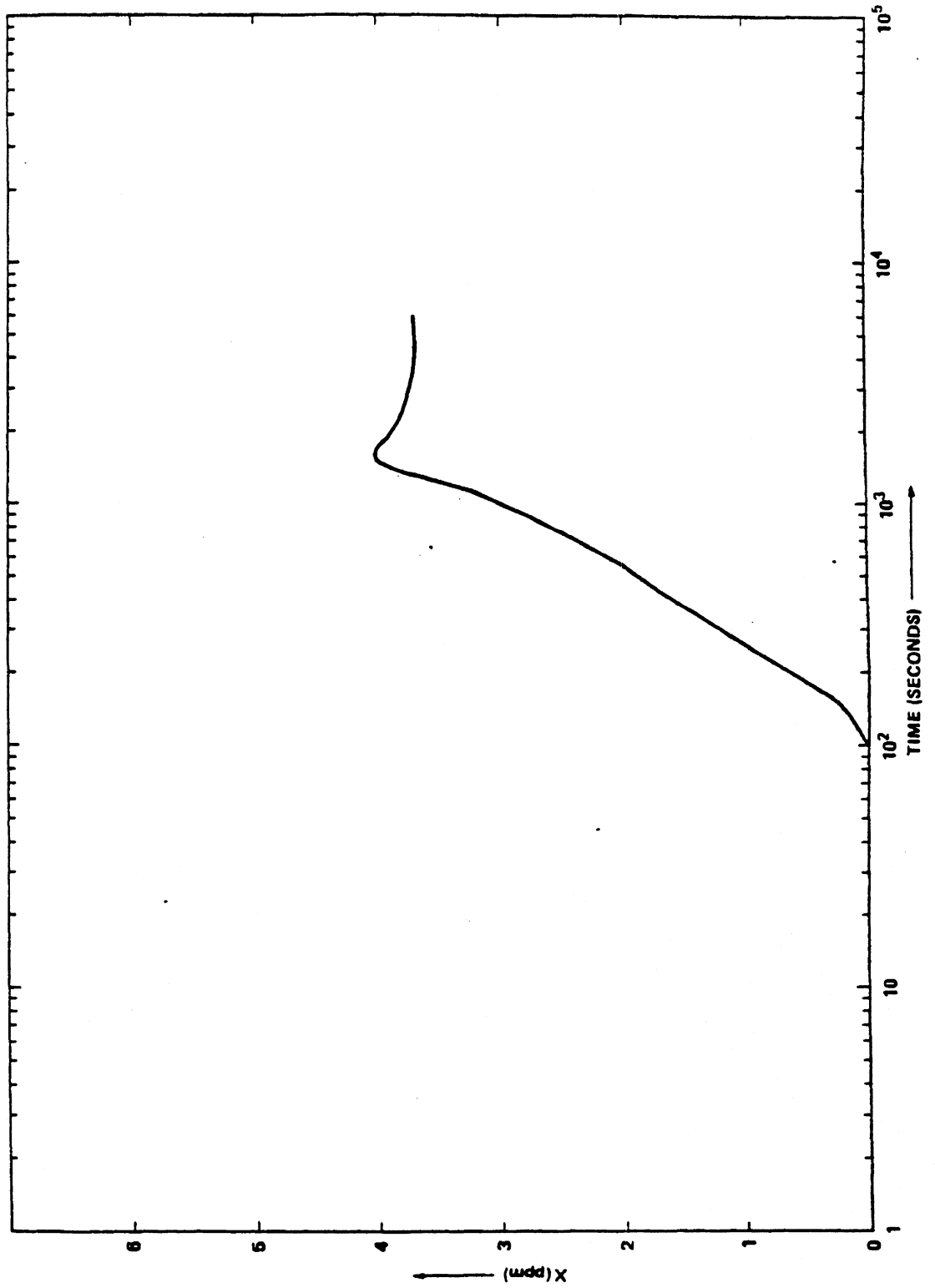
REVISION 0  
JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION  
CHLORINE CONCENTRATION  
AFTER TANK CAR RUPTURE  
FIGURE 15.4.8-1

REVISION 0  
JULY 1982





DAVIS-BESSE NUCLEAR POWER STATION  
 CHLORINE CONCENTRATION AFTER  
 A PIPE LINE BREAK  
 FIGURE 15.4.8-2

REVISION 0  
 JULY 1982

## Davis-Besse Unit 1 Updated Final Safety Analysis Report

### 15.5 REFERENCES

1. L.C. Watson, A.R. Bancroft, and C.W. Hoelke, "Iodine Containment by Dousing in NPD-11," AEC-1130.
2. M.A. Styrikovich, et al., "Transfer of Iodine from Aqueous Solutions to Saturated Vapor," Soviet Journal of Atomic Energy 17, July 1964.
3. H.R. Diffey, et al., "Iodine Cleanup in a Steam Suppression System," International Symposium on Fission Product Release and Transport Under Accident Conditions, Oak Ridge, Tennessee, CONF-65047, Vol 2, pp 776-804 (1965).
4. A.J. Barthoux, et al., "Diffusion of Active Iodine Through Water, With the Iodine Being Liberated in CO<sub>2</sub> Bubbles at High Temperature," AEC-TR-6144, June 1962.
5. A.E.J. Eggleton, "A Theoretical Examination of Iodine-Water Partition Coefficients," AERE-R-4887, February 1967.
6. B. Weidenbaun and S. Naymark, "Potentialities of Molten UO<sub>2</sub> as a Reactor Fuel," Transactions of the ANS, 7, No. 2, pp 527-528 (1964).
7. R.C. Liimatainen and F.J. Testa, "Studies in TREAT of Zircaloy-2 Clad, UO<sub>2</sub> -Core Simulated Fuel Elements," Argonne National Laboratory Chemical<sup>2</sup> Engineering Division Semi-Annual Report, ANL-7225, January-June 1966.
8. J.F. Whited, GE-NMPO, direct communication of experimental data to be published.
9. L.N. Grossman, "High-Temperature Thermal Analysis of Ceramic Systems," Paper presented at the American Ceramic Society 68th Annual Meeting, May 1966.
10. J.A. Redfield, "CHIC-KIN - A Fortran Program for Intermediate and Fast Transients in a Water-Moderated Reactor, WAPD-TM-479, January 1965.
11. A.F. Hery and A.V. Vota, "WIGL2 - a Program for the Solution of the One-Dimensional, Two-Group, Space-Time Diffusion Equations Accounting for Temperature, Xenon, and Control Feedback," WAPD-TM-532, October 1965.
12. W.R. Wise, Jr., and J.F. Proctor, "Explosion Containment Laws for Nuclear Reactor Vessels," NOLTR-63-140, August 1965.
13. W.R. Wise, Jr., An Investigation of Strain Energy Absorption Potential as the Criterion for Determining Optimum Reactor Vessel Containment Design, NAVORD Report 5748, June 1958.
14. D.H. Slade, Ed., Meteorology and Atomic Energy, USAEC, 1968.
15. USNRC, Regulatory Guide 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," Revision 1, December 2001.
16. A.E. Howerton, "Estimating Area Affected by a Chlorine Release," The Chlorine Institute, Inc., New York, 1967.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

17. USNRC, Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," 1975.
18. J. W. Pegram, C. W. Tally, and J. A. Weimer, "Justification for Raising Setpoint for Reactor Trip on High Pressure," BAW-1890, Babcock & Wilcox, September 1985.
19. C. L. Ritchey, "DB-1 Startup Event Reanalysis," B&W Document 32-1167957-00, Babcock & Wilcox, July 17, 1987.
20. C. L. Ritchey, "DB-1 Startup Event Reanalysis Summary," B&W Document 86-1167960-02, Babcock & Wilcox, August 31, 1987.
21. D. J. Skulina, "MSLB Analysis with Failed Open MSSV" B&W Document 32-1178648-01, Babcock & Wilcox, August 10, 1990.
22. R. L. Harne and J. H. Jones, Thermal-Hydraulic Crossflow Applications, BAW-1829, Rev 0, Babcock & Wilcox, Lynchburg, VA, April, 1984.
23. BAW-10179P-A, Latest rev. per App.4-B "Safety Criteria and Methodology For Acceptable Cycle Reload Analysis," (See App.4-B).
24. DELETED
25. DELETED
26. DELETED
27. DELETED
28. DELETED
29. DELETED
30. S. A. Skidmore, "DB-1 FHA Thyroid Doses", B&W Document 51-1164134-00, Babcock & Wilcox, dated May 2, 1986.
31. J. R. Worsham III, "Safety Analysis to Change MTC" at Davis-Besse, B&W Document 51-1201479-00, Babcock & Wilcox, dated December 21, 1990.
32. B&W Calculation 32-1171668-00, "DB-1 Letdown Line Break", dated May, 1988.
33. TED Calculation C-NSA-065.01-010, Rev. 0.
34. Docket No. 50-346, Amendment No. 149 to Facility Operating License No. NPF-3 (TAC No. 69569), June 4, 1990.
35. Docket No. 50-346, Amendment No. 160 to Facility Operating License No. NPF-3 (TAC No. 80347), August 23, 1991.
36. TED Calculation C-NSA-60.00-001, Rev. 1.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

37. TED Calculation C-NSA-059.01-010, Impact of 1900 PSIG RPS Trip on Containment Response to MSLB.
38. M. A. Rutherford, "DB FHA Extd BUP Calculation," B&W Document 32-1224894-00, B&W Nuclear Technologies, dated September 24, 1993.
39. Log No. 3077 (EXT-89-07488), Evaluation of the Davis-Besse Nuclear Power Station Compliance with 10 CFR 50.62 Requirements for Reduction of Risk from Anticipated Transients without Scram (ATWS) (TAC 59086)," September 29, 1989.
40. B&W Calculation 32-1224901-01, "DB-1 Rod Ejection Analysis," dated August, 1994.
41. TED Calculation C-NSA-63.01-005, Line Breaks in Mode 3, Rev. 0.
42. TED Calculation C-NSA-63.01-007, MSLB In Mode 3 with the MFWP, Rev. 0.
43. Docket No. 50-346, Amendment 202 to Facility Operating License No. 20 NPF-3 (TAC No. M92533), November 17, 1995.
44. TED Calculation C-NSA-028.01-002, Control Room Radiation Dose due to Fuel Handling Accident Inside Containment, Rev. 2.
45. B&W Calculation 32-1159090-01, Davis-Besse Unit 1 SFRCS Accident Analysis with Licensing Assumptions, March 2, 1987.
46. B&W Calculation 32-1171148-00, Davis-Besse Loss of Feedwater 220 inch Pressurizer Setpoint, March 14, 1988.
47. BWNT Document 51-1245290-00 "D-B Cy10 EOC T<sub>ave</sub> Reduct Man," 2/15/96.
48. Docket No. 50-346, Serial Number 2180, "Supplemental Information for License Amendment 181," October 5, 1993.
49. TED calculation C-NRE-062.02-072, "Two Year Cycle Source Term Analysis, Rev. 1."
50. DELETED
51. TED Calculation, C-NSA-064.02-036, "DB-1 LOCA Summary Report."
52. DELETED
53. FTI Document 51-5006137-01, "DB Cy13 PSC 15-74 Statept.," 2/2/00.
54. Bechtel Calculation 61.01, FSAR Chapter 15 Accident Analysis.
55. Bechtel Calculation 60.16 Accident Analysis.
56. BAW 10193PA, RELAP5/MOD2-B&W for Safety Analysis of B&W Designed Pressurized Water Reactors, Rev 0, January 2000.
57. C-NSA-060.00-014 Revision 0, Loss of Feedwater (LOFW) Analysis (Framatome ANP document 86-5027672-00, "DB LOFW Analysis at 2827.44 MWT," September 26, 03).

## Davis-Besse Unit 1 Updated Final Safety Analysis Report

58. Calculation, C-NSA-028.01-005, Revision 3, "Control Room Radiation Doses Following a Maximum Hypothetical Accident."
59. Calculation, C-NSA-049.01-007, Revision 0, "DB1 Post-LOCA pH Analysis."
60. Calculation, C-NSA-028.01-007, Revision 0, "Control Room Doses due to ECCS Leakage to the BWST and Auxiliary Building."
61. Calculation, C-NSA-028.01-008, Revision 0, "Dose in the Control Room due to EVS and CREVS filter activity."
62. C-NSA-060.00-016, Revision 0, "Reanalysis of Startup Event" (Areva Document 86-5027875-00, "DB-1 Startup Event Analysis Results," June 24, 2003).
63. Areva NP Document 51-9004090-005, "Davis-Besse MUR Summary Report."
64. Areva NP Document, 51-9209222-001, "DB-2 ROTSG Disposition of Events."
65. Areva NP Document, 51-9211669-003, "Disposition of ROTSG for DB-1 LOCA Analysis."
66. Calculation, C-NSA-081.05-001, "DB Caldon Dropped Rod Analysis."
67. Areva NP Document 86-5007079-00, "Davis-Besse SG Overpressure Protection."

APPENDIX 15A

RADIATION SOURCES

APPENDIX 15A

RADIATION SOURCES

TABLE OF CONTENTS

<u>Section</u>	<u>Title</u>	<u>Page</u>
15A.1.0	GENERAL	15A-1
15A.2.0	ACTIVITY IN CORE	15A-1
15A.3.0	ACTIVITY IN HIGHEST POWER FUEL ASSEMBLY	15A-1
15A.4.0	FISSION PRODUCT ACTIVITY IN REACTOR COOLANT	15A-2
15A.5.0	FISSION PRODUCT ACTIVITY IN SECONDARY COOLANT	15A-2
15A.6.0	DELETED	
15A.7.0	SOURCE TERM FOR EXTENDED CYCLE EVALUATION	15A-7

Davis-Besse Unit 1 Updated Final Safety Analysis Report

LIST OF TABLES

<u>Table</u>	<u>Title</u>	<u>Page</u>
15A-1	Escape Rate Coefficients	15A-4
15A-2	Total Core Fission Product Inventory in Fuel and Fuel Rod Gaps	15A-4
15A-3	Fission Product Inventory in Fuel and Fuel Rod Gap of Highest Power Fuel Assembly	15A-5
15A-4	Maximum Fission Product Activity in Reactor Coolant	15A-6
15A-5	Activities Expected in Secondary Coolant	15A-6
15A-6	Comparison of Core Fission Product Inventory for a 24 Month Fuel Cycle with Source Terms from Table 15A-2 and TID 14844	15A-8



APPENDIX 15A

RADIATION SOURCES

15A.1.0 GENERAL

This appendix lists the quantities of fission products in the fuel, fuel rod gap, reactor coolant and secondary system that are used to evaluate the environmental consequences of the Chapter 15 accidents. The calculational methods discussed in Section 11.1 were used, but more conservative assumptions have been used to calculate these accident sources. This results in higher fission product activities and conservative dose values.

See Section 15A.7.0, Source Term for Extended Cycle Evaluation.

15A.2.0 ACTIVITY IN CORE

The fission product activity in the fuel and the fuel rod gap is calculated by the method discussed in Subsection 11.1.1, using the following assumptions:

- a. Full power operation at 2772 MWt for a 433 day initial cycle and two 277 day equilibrium cycles.
- b. Operation with no defective fuel rods so the maximum core fission product inventory is calculated. The calculation was performed with the escape rate coefficients shown in Table 15A-1.

The resulting fuel and fuel rod gap fission product inventories are given in Table 15A-2.

See 15A.1.0 referenced re-analyses for the assumptions used to determine the activity in core.

15A.3.0 ACTIVITY IN HIGHEST POWER FUEL ASSEMBLY

The fission product activity in the fuel and gap of the highest power fuel assembly is calculated by the same method as was used for the core activity (Section 15A.2), except that:

- a. The activity inventory is based on the highest power fuel assembly removed from the core at the end of any core cycle.
- b. The assembly contains no defective fuel rods so that the maximum fission product inventory is calculated.

The resulting fuel and gap fission product inventories in the highest power fuel assembly are given in Table 15A-3.

See 15A.1.0 referenced re-analyses for the calculated activity in the highest power fuel assembly.

#### 15A.4.0 FISSION PRODUCT ACTIVITY IN REACTOR COOLANT

The fission product activity in the Reactor Coolant System is calculated by the method discussed in Subsection 11.1.2.1, using the following assumptions:

- a. Full power operation at 2772 MWt for a 433 day initial cycle and two 277 day equilibrium cycles.
- b. One percent of the fuel rods in the core develop cladding defects at the beginning of the third operating cycle. All of the defects are assumed to occur in the high burnup fuel rods which have been irradiated for the 433 day initial cycle and one 277 day equilibrium cycle. The defective fuel rods are assumed to leak continuously during the entire 277 day irradiation cycle.
- c. Base loaded operation.
- d. Continuous reactor coolant purification through the purification demineralizer at an average flow rate of one Reactor Coolant System volume per day with a zero removal efficiency for krypton, xenon, molybdenum, cesium, yttrium, tellurium and tritium, and a 99% removal efficiency for all other nuclides.
- e. All isotopes, except krypton, xenon and tritium are removed with a 99.9% efficiency during the processing of the reactor coolant bleed. Krypton and xenon are removed with a 99.9% efficiency during the first 267 days of an equilibrium cycle, but after that time, when the deborating demineralizers are used for bleed stream processing, the removal of krypton and xenon ceases.

The maximum reactor coolant activities are given in Table 15A-4.

See 15A.1.0 referenced re-analyses for the fission product activity in secondary coolant.

#### 15A.5.0 FISSION PRODUCT ACTIVITY IN SECONDARY COOLANT

The equilibrium iodine activity in the secondary coolant is given by the ratio of production terms to loss terms.

Production is from a 1 gpm steam generator leak. Losses are:

- a. 90% removal in the 67% of the flow which sees the demineralizers.
- b. Secondary leakage of 10 gpm.
- c. A DF of  $10^4$  in the steam jet air ejector.
- d. Decay.

## Davis-Besse Unit 1 Updated Final Safety Analysis Report

The flow rate of the secondary system is 70 systems/day.

The noble gas activity will be negligible due to gas removal in the steam generator and the condenser. Table 15A-5 lists the activities expected in the secondary coolant.

See 15A.1.0 referenced re-analyses for the fission product activity in reactor coolant.

15A.6.0 DELETED

TABLE 15A-1<sup>(1)</sup>

Escape Rate Coefficients

Element	Escape rate coefficient, sec <sup>-1</sup>
Krypton	1.0 x 10 <sup>-7</sup>
Xenon	1.0 x 10 <sup>-7</sup>
Iodine	2.0 x 10 <sup>-8</sup>

<sup>(1)</sup>See 15A.1.0 referenced re-analyses for the escape rate coefficients used.

TABLE 15A-2<sup>(1)</sup>

Total Core Fission Product Inventory in Fuel and Fuel Rod Gaps

<u>Isotope</u>	Activity at end of third cycle	
	Fuel activity, Ci	Fuel rod gap activity, Ci
Kr-83m	8.26(+6)	1.06(+4)
Kr-85m	2.46(+7)	5.58(+4)
Kr-85	5.32(+5)	4.33(+5)
Kr-87	4.54(+7)	3.06(+4)
Kr-88	6.85(+7)	9.84(+4)
Xe-131m	5.9(+5)	8.72(+4)
Xe-133m	3.45(+6)	1.03(+5)
Xe-133	1.43(+8)	9.21(+6)
Xe-135m	3.53(+7)	3.40(+4)
Xe-135	1.76(+7)	1.86(+5)
Xe-138	1.27(+8)	1.87(+4)
I-131	7.46(+7)	1.46(+6)
I-132	1.09(+8)	2.01(+5)
I-133	1.44(+8)	3.09(+5)
I-134	1.83(+8)	1.92(+4)
I-135	1.4(+8)	9.73(+4)

<sup>(1)</sup>See 15A.1.0 referenced re-analyses for the total fission product inventory in fuel and fuel rod gap

TABLE 15A-3<sup>(1)</sup>

Fission Product Inventory in Fuel and  
Fuel Rod Gap of Highest Power Fuel Assembly

<u>Isotope</u>	<u>Fuel activity, Ci</u>	<u>Fuel rod gap activity, Ci</u>
Kr-83m	6.15(+4)	7.56(+1)
Kr-85m	1.76(+5)	3.99(+2)
Kr-85	5.14(+3)	4.59(+3)
Kr-87	3.23(+5)	2.19(+2)
Kr-88	4.89(+5)	7.03(+2)
Xe-131m	4.22(+3)	6.23(+2)
Xe-133m	2.46(+4)	7.38(+2)
Xe-133	1.02(+6)	6.58(+4)
Xe-135m	2.52(+5)	2.42(+2)
Xe-135	1.26(+5)	1.33(+3)
Xe-138	9.07(+5)	1.34(+2)
I-131	5.33(+5)	1.05(+4)
I-132	7.79(+5)	1.44(+3)
I-133	1.03(+6)	2.21(+3)
I-134	1.31(+6)	1.37(+2)
I-135	1.0(+6)	6.95(+2)

<sup>(1)</sup>See 15A.1.0 referenced re-analyses for the fission product inventory of highest power fuel assembly.

TABLE 15A-4<sup>(1)</sup>

Maximum Fission Product Activity in Reactor Coolant<sup>1</sup>

<u>Isotope</u>	<u>Activity, <math>\mu</math> Ci/cc</u>
Kr-83m	3.12(-1)
Kr-85m	1.65
Kr-85	9.30
Kr-87	9.04(-1)
Kr-88	2.90
Xe-131m	2.32
Xe-133m	3.03
Xe-133	2.63(+2)
Xe-135m	1.00
Xe-135	5.50
Xe-138	5.52(-1)
I-131	3.48
I-132	5.20
I-133	4.11
I-134	5.39(-1)
I-135	2.05

<sup>1</sup> Coolant density = 44.5 lbm/ft<sup>3</sup>

<sup>(1)</sup>See 15A.1.0 referenced re-analysis for the maximum fission product activity in reactor coolant.

TABLE 15A-5<sup>(1)</sup>

Activities Expected in Secondary Coolant

<u>Isotope</u>	<u>Activity, <math>\mu</math> Ci/gm</u>
I-131	3.44(-4)
I-132	4.39(-4)
I-133	3.99(-4)
I-134	3.66(-5)
I-135	1.92(-4)

<sup>(1)</sup>See 15A.1.0 referenced re-analysis for the activities expected in secondary coolant.

#### 15.A.7.0 Source Term for Extended Cycle Evaluation

The evaluation below discusses the impact, from a source term standpoint (i.e., for dose consequences), of extending the fuel cycle to 24 months.

Among the accidents analyzed in Chapter 15, only the control rod ejection accident (15.4.3), the Loss of Coolant Accident (15.4.6), and the fuel handling accident (15.4.7) utilize the activity in the fuel and/or in the fuel rod gap in the calculation of off site doses. For the other accidents, the dose consequences were evaluated based on the fission product activity in the reactor coolant. Since the coolant activity used in the accident analysis is significantly higher than the Technical Specifications limits, the USAR dose analysis is bounding for accidents that do not result in a release from the fuel rods. The fuel handling accident has been evaluated in USAR sections 15.4.7.2.5.1 and 15.4.7.3.4.1 for a maximum burnup of 60,000 megawatt days per metric ton. This burnup limit is not changed by the 24 month fuel cycle. The remaining two accidents are evaluated below.

The total fission product inventory in the core (i.e., in the fuel matrix as well as in the fuel rod gap) was analyzed in Reference 49. Also, implicit in the analysis is the core's operation for 690 Effective Full Power Days (EFPDs) at a power level of 2827 MWt (i.e., 102% of 2772 MWt). Based on the expected fuel batch size and 60,000 megawatt days per metric ton (MWD/MTU) burnup limit, fuel enrichments of approximately 4.5 to 5 percent are needed to support full power operation for two years. However, in order to maximize the calculated iodine inventory, the 24 month fuel cycle source term was analyzed by assuming a fuel enrichment of 4% by weight.

The core inventory for a 277 day equilibrium cycle was given in USAR Table 15A-2. However, when Davis-Besse was licensed this was not used in the evaluation of dose consequences due to a postulated Maximum Hypothetical Accident (MHA) in USAR section 15.4.6. Instead, the core inventory used in the evaluation of the MHA was based on TID-14844. TID-14844 is the historical document that describes a source term and methodology for calculation of offsite dose consequences. A comparison between the source terms from the original USAR Table 15A-2, those for the 24 month cycle and TID-14844 (used in the analysis of the MHA) is in Table 15A-6. This table shows that, although there are minor variations in the calculated core inventories, the source term for the 24 month cycle is not significantly different than the one given in the original USAR Table 15A-2.

The original USAR source term was not used in the MHA analyses and assumptions used in the rod ejection accident for the original USAR analysis (with respect to availability of the condenser) were not as conservative as the NRC's independent analysis in the Safety Evaluation Report (SER) for Davis-Besse. Therefore, additional evaluations using the 24 month cycle source term for these two accidents were performed as described in sections 15.4.6.4 and 15.4.3.2.7.

TABLE 15A-6

Comparison of Core Fission Product Inventory for a 24 Month Fuel Cycle with Source Terms from Table 15A-2 and TID 14844\*

Total Activity in the Fuel and Fuel Rod Gaps, Ci

<u>Isotope</u>	<u>From Table 15A-2</u>	<u>24 Month Fuel Cycle</u>	<u>TID 14844</u>
Kr-83m	8.27E6	9.24E6	1.17E7
Kr-85m	2.47E7	1.93E7	3.67E7
Kr-85	9.65E5	9.32E5	1.16E6
Kr-87	4.54E7	3.69E7	6.60E7
Kr-88	6.86E7	5.18E7	9.05E7
Xe-131m	6.77E5	8.59E5	7.34E5
Xe-133m	3.55E6	4.68E6	3.91E6
Xe-133	1.52E8	1.51E8	1.59E8
Xe-135m	3.53E7	3.08E7	4.40E7
Xe-135	1.78E7	3.31E7	1.52E8
Xe-138	1.27E8	1.28E8	1.35E8
I-131	7.61E7	7.71E7	7.08E7
I-132	1.09E8	1.12E8	1.08E8
I-133	1.44E8	1.56E8	1.59E8
I-134	1.83E8	1.71E8	1.86E8
I-135	1.40E8	1.46E8	1.44E8

\*Note: These values are calculated values and do not reflect limits.