CHAPTER 14 – SAFETY ANALYSIS

TABLE OF CONTENTS

SECTION	TITLE
14.0	SAFETY ANALYSIS
14.1	CORE AND COOLANT BOUNDARY PROTECTION ANALYSIS
14.1.1	ABNORMALITIES
14.1.2	ANALYSIS OF EFFECTS AND CONSEQUENCES
14.1.2.1	UNCOMPENSATED OPERATING REACTIVITY CHANGES
14.1.2.2	STARTUP ACCIDENT
14.1.2.3	ROD WITHDRAWAL ACCIDENT AT RATED POWER OPERATION
14.1.2.4	MODERATOR DILUTION ACCIDENT
14.1.2.5	COLD WATER ACCIDENT
14.1.2.6	LOSS OF COOLANT FLOW
14.1.2.7	STUCK-OUT, STUCK-IN, OR DROPPED CONTROL ROD ACCIDENT
14.1.2.8	LOSS OF ELECTRIC POWER
14.1.2.9	STEAM LINE BREAK
14.1.2.10	STEAM GENERATOR TUBE FAILURE
14.1.2.11	ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)
14.2	STANDBY SAFEGUARDS ANALYSIS
14.2.1	SITUATIONS ANALYZED AND CAUSES
14.2.2	ACCIDENT ANALYSES
14.2.2.1	FUEL HANDLING ACCIDENT
14.2.2.2	ROD EJECTION ACCIDENT
14.2.2.3	LARGE BREAK LOSS OF COOLANT ACCIDENT
14.2.2.3.1	IDENTIFICATION OF CAUSES
14.2.2.3.2	LARGE BREAK LOCA ANALYSIS
14.2.2.3.3	RESULTS - LARGE BREAK LOCA
14.2.2.3.4	ENVIRONMENTAL ANALYSIS OF LOSS OF COOLANT ACCIDENTS
14.2.2.3.5	POST ANALYSIS-OF-RECORD EVALUATIONS FOR LARGE BREAK LOCA
14.2.2.4	SMALL BREAK LOSS OF COOLANT ACCIDENT
14.2.2.4.1	IDENTIFICATION
14.2.2.4.2	SMALL BREAK LOCA ANALYSIS
14.2.2.4.3	RESULTS - SMALL BREAK LOCA
14.2.2.4.4	POST ANALYSIS-OF-RECORD EVALUATIONS FOR SMALL BREAK LOCA
14.2.2.5	MAXIMUM HYPOTHETICAL ACCIDENT
14.2.2.6	WASTE GAS TANK RUPTURE
14.2.2.7	LOSS OF FEEDWATER ACCIDENT
14.2.2.8	FUEL CASK DROP ACCIDENT
14.2.2.9	FEEDWATER LINE BREAK ACCIDENT
14.2.2.9.1	FEEDWATER LINE BREAK OUTSIDE CONTAINMENT
14.2.2.9.2	FEEDWATER LINE BREAK INSIDE CONTAINMENT
14.3	REFERENCES

LIST OF TABLES

TABLE	TITLE
14.0-1	EQUIPMENT AND RELATED SYSTEMS ASSUMED TO FUNCTION DURING ACCIDENT ANALYSIS
14.1-1	ABNORMALITIES AFFECTING CORE AND COOLANT BOUNDARY
14.1-2	UNCOMPENSATED REACTIVITY DISTURBANCES
14.1-3	STARTUP ACCIDENT PARAMETER
14.1-4	SUMMARY OF STARTUP ACCIDENT ANALYSIS
14.1-5	ROD WITHDRAWAL ACCIDENT PARAMETERS
14.1-6	SUMMARY OF ROD WITHDRAWAL ACCIDENT ANALYSIS
14.1-7	MODERATOR DILUTION ACCIDENT PARAMETERS
14.1-8	SUMMARY OF MODERATOR DILUTION ACCIDENT ANALYSIS
14.1-9	LOSS OF COOLANT FLOW ACCIDENT PARAMETERS
14.1-10	LOCKED ROTOR ACCIDENT PARAMETERS
14.1-11	SUMMARY OF LOSS OF COOLANT FLOW ACCIDENT ANALYSIS
14.1-11a	SUMMARY OF LOSS OF COOLANT FLOW ACCIDENT ANALYSIS – CURRENT CYCLE
14.1-12	NATURAL CIRCULATION CAPABILITY
14.1-13	DROPPED ROD ACCIDENT PARAMETERS
14.1-14	LOSS OF LOAD TRANSIENT PARAMETERS AND RESULTS
14.1-14a	LOSS OF ELECTRICAL LOAD PARAMETERS AND RESULTS
14.1-15	LOSS OF ALL A-C POWER EVENT (STATION BLACKOUT) RADIOLOGICAL ANALYSIS PARAMETERS AND RESULTS
14.1-15a	DELETED
14.1-16	STEAM LINE BREAK PARAMETERS
14.1-17	DELETED

14.1-18 RADIOLOGICAL CONSEQUENCES OF MAIN STEAM LINE BREAK ACCIDENT IN CONJUNCTION WITH ACCIDENT-INDUCED STEAM GENERATOR TUBE LEAK (REM)

14.1-19 DELETED

14.1-20 STEAM GENERATOR TUBE FAILURE PARAMETERS

14.1-21 SUMMARY OF STEAM GENERATOR TUBE FAILURE ANALYSIS

14.1-22 COLD WATER ACCIDENT PARAMETERS

14.1-23 DELETED

14.1-24 DELETED

14.2-1 SITUATIONS ANALYZED AND CAUSES

14.2-2 RADIOACTIVE RELEASE FOR THE FUEL HANDLING ACCIDENT

14.2-3 FUEL HANDLING ACCIDENT PARAMETERS AND RESULTS

14.2-4 FISSION PRODUCT INVENTORIES FOR THE CORE,THE AVERAGE ASSEMBLY, AND THE REACTOR COOLANT SYS

14.2-4a FISSION PRODUCT CORE INVENTORY FOR FHA IN REACTOR BUILDING AND LOCA

14.2-5 RADIOACTIVE RELEASE FOR THE POSTULATED FUEL HANDLING ACCIDENT AND DOSE RESULTS (IN THE REACTOR BUILDING)

14.2-6 ROD EJECTION ACCIDENT PARAMETERS (INITIAL CYCLE)

14.2-7 NOMINAL VALUES OF INPUT PARAMETERS FOR ROD EJECTION ACCIDENT ANALYSIS

14.2-8 COMPARISON OF SPACE-DEPENDENT AND POINT KINETICS RESULTS ON THE FUEL ENTHALPY (INITIAL CYCLE)

14.2-9 SUMMARY OF ROD EJECTION ACCIDENT ANALYSIS

14.2-10 REACTOR VESSEL PARAMETERS

14.2-11 ENVIRONMENTAL EFFECTS OF ROD EJECTION ACCIDENT

14.2-11 REACTOR VESSEL INTERNALS - DISSIMILAR METALS

14.2-13 ASSUMPTIONS AND RESULTS OF CONTROL ROD TEMPERATURE ANALYSIS

14.2-14 LOSS OF COOLANT ACCIDENT ANALYSIS

14.2-15	DELETED
14.2-16	SEQUENCE OF EVENTS OF STUCK OPEN PORV ACCIDENT
14.2-17	ACTIVITY RELEASE FROM MAXIMUM BREAK SIZE LOCA
14.2-18	IODINE ACTIVITY RELEASE FROM MHA
14.2-19	NOBLE GAS RELEASE FROM MHA
14.2-20	ENVIRONMENTAL DOSES RESULTING FROM MHA
14.2-21	WASTE GAS TANK INVENTORY
14.2-22	KEY INPUT PARAMETERS FOR LOFW ANALYSIS
14.2-23	DELETED
14.2-24	DELETED
14.2-25	NOBLE GAS AND IODINE GAP ACTIVITY (10 FUEL ASSEMBLIES) 120 DAY DECAY
14.2-26	SITE BOUNDARY DOSES
14.2-27	LOW PRESSURE INJECTION FLOW VS CORE FLOOD/LPI NOZZLE PRESSURE
14.2-28	HIGH PRESSURE INJECTION FLOW VERSUS HPI NOZZLE PRESSURE
14.2-29	SUMMARY OF ANALYSIS INPUT VALUES

LIST OF FIGURES

FIGURE	TITLE
14.1-1	STARTUP ACCIDENT FROM 10^{-9} RATED POWER USING A 1.5 PERCENT delta-k/k ROD GROUP;HIGH PRESSURE REACTOR TRIP IS ACTUATED
14.1-2	STARTUP ACCIDENT FROM 10^{-9} RATED POWER USING ALL RODS WITH A WORTH OF 10 PERCENT delta-k/k; HIGH FLUX REACTOR TRIP IS ACTUATED
14.1-3	PEAK THERMAL POWER VS ROD WITHDRAWAL RATE FOR A STARTUP ACCIDENT FROM 10^{-9} RATED POWER
14.1-4	PEAK NEUTRON POWER VERSUS ROD WITHDRAWAL RATE FOR A STARTUP ACCIDENT FROM 10^{-9} RATED POWER
14.1-5	PEAK THERMAL POWER VERSUS MODERATOR COEFFICIENT FOR A STARTUP ACCIDENT USING A 1.5 PERCENT delta-k/k ROD GROUP AT 1.09 X 10^{-4} (delta-k/k)/S FROM 10^{-9} RATED POWER
14.1-6	PEAK THERMAL POWER VERSUS MODERATOR COEFFICIENT FOR A STARTUP ACCIDENT USING A 1.5 PERCENT delta-k/k ROD GROUP AT 1.09 X 10^{-4}(delta-k/k)/S FROM 10^{-9} RATED POWER
14.1-7	PEAK THERMAL POWER VERSUS DOPPLER COEFFICIENT FOR A STARTUP ACCIDENT USING ALL RODS AT 7.25 X 10^{-4} (delta-k/k)/S FROM 10^{-9} RATED POWER
14.1-8	PEAK THERMAL POWER VERSUS MODERATOR COEFFICIENT FOR A STARTUP ACCIDENT USING ALL RODS AT 7.25 X 10^{-4} (delta-k/k)/S FROM 10^{-9} RATED POWER
14.1-9	ROD WITHDRAWAL ACCIDENT FROM RATED POWER USING A 1.5 PERCENT delta-k/k ROD GROUP AT 1.09 X 10^{-4} (delta-k/k)/S; HIGH FLUX REACTOR TRIP IS ACTUATED
14.1-10	PEAK PRESSURE VERSUS ROD WITHDRAWAL RATE FOR A ROD WITHDRAWAL ACCIDENT FROM RATED POWER
14.1-11	PEAK PRESSURE VERSUS TRIP DELAY TIME FOR A ROD WITHDRAWAL ACCIDENT FROM RATED POWER USING A1.5 PERCENT delta-k/k ROD GROUP
14.1-12	PEAK PRESSURE VERSUS DOPPLER COEFFICIENT FOR A ROD WITHDRAWAL ACCIDENT FROM RATED POWER USING A 1.5 PERCENT delta-k/k ROD GROUP

LIST OF FIGURES (cont'd)

FIGURE	TITLE
14.1-13	PEAK PRESSURE VERSUS MODERATOR COEFFICIENT FOR A ROD WITHDRAWAL ACCIDENT FROM RATED POWER USING A 1.5 PERCENT delta-k/k ROD GROUP
14.1-14	MAXIMUM NEUTRON AND THERMAL POWER FOR AN ALL-ROD WITHDRAWAL ACCIDENT FROM VARIOUS INITIAL POWER LEVELS
14.1-15	PEAK FUEL TEMPERATURE IN AVERAGE ROD AND HOT SPOT FOR AN ALL-ROD WITHDRAWAL ACCIDENT FROM VARIOUS INITIAL POWER LEVELS
14.1-16	PUMP STARTUP FROM 50 PERCENT POWER AND 50 PERCENT FLOW
14.1-17	PERCENT REACTOR COOLANT FLOW AS A FUNCTION OF TIME AFTER LOSS OF PUMP POWER
14.1-18	MINIMUM DNBR WHICH OCCURS DURING COASTDOWN FROM VARIOUS INITIAL POWER LEVELS
14.1-19	DNB RATIO VERSUS TIME FOR A LOCKED ROTOR ACCIDENT FROM 102 PERCENT OF RATED POWER
14.1-20	0.46 PERCENT delta-k/k ROD DROP FROM RATED POWER WITH AUTOMATIC RUNBACK TO 60 PERCENT DEMAND IN 12s
14.1-21	0.36 PERCENT delta-k/k ROD DROP FROM RATED POWER WITH AUTOMATIC RUNBACK TO 60 PERCENT DEMAND IN 12s
14.1-22A	DOUBLE-ENDED RUPTURE OF 24 IN STEAM LINE BETWEEN STEAM GENERATOR AND STEAM STOP VALVE (with Feedwater Isolation)
14.1-22B	DOUBLE-ENDED RUPTURE OF 24 IN STEAM LINE BETWEEN STEAM GENERATOR AND STEAM STOP VALVE (with Feedwater Isolation)
14.1-23	RCS HOT LEG PRESSURE (psia) RESPONSE FOR LOSS OF ALL AC POWER (STATION BLACKOUT)
14.1-24	DROPPED ROD W/ BAYONET FAILURE – RCS PRESSURE RESPONSE
14.1-25	DROPPED ROD W/ BAYONET FAILURE – REACTOR POWER RESPONSE
14.1-26	DROPPED ROD W/ BAYONET FAILURE – PRIMARY TEMPERATURE RESPONSE
14.2-1	PEAK NEUTRON POWER VARIATION WITH EJECTED CONTROL ROD WORTH

LIST OF FIGURES (cont'd)

FIGURE	TITLE
14.2-2	PEAK THERMAL POWER AS A FUNCTION OF EJECTED CONTROL ROD WORTH
14.2-3	PEAK ENTHALPY OF HOTTEST FUEL ROD VERSUS EJECTED CONTROL ROD WORTH
14.2-4	EFFECT ON PEAK NEUTRON POWER OF VARYING THE DOPPLER COEFFICIENT FOR AN EJECTED ROD WORTH OF 0.56 PERCENT delta-k/k AT 10^{-9} RATED POWER AND 0.46 PERCENT delta-k/k AT RATED POWER
14.2-5	EFFECT ON PEAK THERMAL POWER OF VARYING THE DOPPLER COEFFICIENT FOR AN EJECTED ROD WORTH OF 0.56 PERCENT delta-k/k AT 10^{-9} RATED POWER AND 0.46 PERCENT delta-k/k AT RATED POWER
14.2-6	EFFECT ON PEAK NEUTRON POWER OF VARYING THE MODERATOR COEFFICIENT FOR AN EJECTED ROD WORTH OF 0.56 PERCENT delta-k/k AT 10^{-3} RATED POWER AND 0.46 PERCENT delta-k/k AT RATED POWER
14.2-7	EFFECT ON PEAK THERMAL POWER OF VARYING THE MODERATOR COEFFICIENT FOR AN EJECTED ROD WORTH OF 0.56 PERCENT delta-k/k AT 10^{-3} RATED POWER AND 0.46 PERCENT delta-k/k AT RATED POWER
14.2-8	EFFECT ON PEAK THERMAL POWER OF VARYING THE TRIP DELAY TIME FOR AN EJECTED ROD WORTH OF 0.56 PERCENT delta-k/k AT 10^{-9} RATED POWER AND 0.46 PERCENT delta-k/k AT RATED POWER
14.2-9	PERCENT CORE EXPERIENCING DNB AS A FUNCTION OF EJECTED CONTROL ROD WORTH AT RATED POWER, BOL
14.2-10	LARGE BREAK ANALYSIS CODE INTERFACES
14.2-11	HOT SPOT CLAD TEMPERATURE VS TIME FOR A 36 IN ID, DOUBLE-ENDED, HOT LEG PIPE RUPTURE AND VARIABLE QUENCH COEFFICIENT
14.2-12	LBLOCA LIMIT CASE (BOL) – REACTOR VESSEL UPPER PLENUM PRESSURE
14.2-13	LBLOCA LIMIT CASE (BOL) – BREAK MASS FLOW RATE
14.2-14	LBLOCA LIMIT CASE (BOL – 2.506-FT) – HOT CHANNEL FUEL AND CLAD TEMPERATURE AT PEAK UNRUPTURED LOCATION
14.2-15	LBLOCA LIMIT CASE (BOL – 4.264-FT) – HOT CHANNEL FUEL AND CLAD TEMPERTURE AT PEAK UNRUPTURED LOCATION
14.2-16	LBLOCA LIMIT CASE (BOL – 6.021-FT) – HOT CHANNEL FUEL AND CLAD TEMPERATURE AT PEAK UNRUPTURED LOCATION

LIST OF FIGURES (cont'd)

FIGURE	TITLE
14.2-17	LBLOCA LIMIT CASE (BOL – 7.779-FT) – HOT CHANNEL FUEL AND CLAD TEMPERATURE AT PEAK UNRUPTURED LOCATION
14.2-18	LBLOCA LIMIT CASE (BOL – 9.536-FT) – HOT CHANNEL FUEL AND CLAD TEMPERATURE AT PEAK UNRUPTURED LOCATION
14.2-19	LBLOCA LIMIT CASE (BOL – 2.506-FT) – HOT CHANNEL FUEL AND CLAD TEMPERATURE AT PEAK RUPTURED LOCATION
14.2-20	LBLOCA LIMIT CASE (BOL – 4.264-FT) - HOT CHANNEL FUEL AND CLAD TEMPERATURE AT PEAK RUPTURED LOCATION
14.2-21	LBLOCA LIMIT CASE (BOL – 6.021-FT) – HOT CHANNEL FUEL AND CLAD TEMPERATURE AT PEAK RUPTURED LOCATION
14.2-22	LBLOCA LIMIT CASE (BOL – 7.779-FT) – HOT CHANNEL FUEL AND CLAD TEMPERATURE AT PEAK RUPTURED LOCATION
14.2-23	LBLOCA LIMIT CASE (BOL – 9.536-FT) – HOT CHANNEL FUEL AND CLAD TEMPERATURE AT PEAK RUPTURED LOCATION
14.2-24	LOCA LIMIT AXIAL POWER SHAPES
14.2-25	SBLOCA COMPARISON OF RCS PRESSURE (0.01 – 0.06 FT^2 BREAKS)
14.2-26	SBLOCA COMPARISON OF RCS PRESSURE (0.07 – 0.10 FT^2 BREAKS)
14.2-27	SBLOCA COMPARISON OF RCS PRESSURE (0.15 – 0.75 FT^2 BREAKS)
14.2-28	SBLOCA COMPARISON OF REACTOR VESSEL COLLAPSED LIQUID LEVEL (0.01 – 0.06 FT^2 BREAKS)
14.2-29	SBLOCA COMPARISON OF REACTOR VESSEL COLLAPSED LIQUID LEVEL (0.07 – 0.10 FT^2 BREAKS)
14.2-30	SBLOCA COMPARISON OF REACTOR VESSEL COLLAPSED LIQUID LEVEL (0.15 – 0.75 FT^2 BREAKS)
14.2-31	SBLOCA COMPARISON OF PEAK CLAD TEMPERATURES (0.01 – 0.06 FT^2 BREAKS)
14.2-32	SBLOCA COMPARISON OF PEAK CLAD TEMPERATURES (0.07 – 0.10 FT^2 BREAKS)
14.2-33	SBLOCA COMPARISON OF PEAK CLAD TEMPERATURES (0.15 – 0.75 FT^2 BREAKS)

LIST OF FIGURES (cont'd)

FIGURE	TITLE
14.2-34	CFT LINE BREAK SYSTEM PRESSURE
14.2-35	CFT LINE BREAK COLLAPSED LIQUID LEVELS
14.2-36	CFT LINE BREAK MIXTURE LEVELS
14.2-37	CFT LINE BREAK HOT CHANNEL CLAD TEMPERATURES
14.2-38	HPI LINE BREAK SYSTEM PRESSURE (OPERATOR ACTION)
14.2-39	HPI LINE BREAK COLLAPSED LIQUID LEVELS (OPERATOR ACTION)
14.2-40	HPI LINE BREAK MIXTURE LEVELS (OPERATOR ACTION)
14.2-41	HPI LINE BREAK HOT CHANNEL CLAD TEMPERATURES (OPERATOR ACTION)
14.2-42	HPI LINE BREAK SYSTEM PRESSURE (NO OPERATOR ACTION)
14.2-43	HPI LINE BREAK COLLAPSED LIQUID LEVELS (NO OPERATOR ACTION)
14.2-44	HPI LINE BREAK MIXTURE LEVELS (NO OPERATOR ACTION)
14.2-45	HPI LINE BREAK HOT CHANNEL CLAD TEMPERATURES (NO OPERATOR ACTION)
14.2-46	SBLOCA SPECTRUM – PCT VERSUS BREAK SIZE
14.2-47	CRAFT 2 NODING DIAGRAM FOR SMALL BREAK
14.2-48	BREAK SPECTRUM-AVERAGE SYSTEM VOID FRACTION WITH THE RC PUMPS OPERATIVE AND 2 HPI PUMPS
14.2-49	PRESSURE VS TIME – SMALL BREAKS WITH EMERGENCY FEEDWATER
14.2-50	PRESSURIZER LEVEL VS TIME – SMALL BREAKS WITH EMERGENCY FEEDWATER
14.2-51	PRESSURIZER LEVEL VS TIME FOR SMALL BREAK IN PRESSURIZER
14.2-52	0.01 FT^2 COLD LEG BREAK WITH NO EFW, 2 HPI'S & STUCK PORV AT 20 MIN-NODE 14 PRESSURE VS TIME
14.2-53	0.01 FT^2 COLD LEG BREAK WITH NO EFW, 2 HPI'S & STUCK PORV AT 20 MIN-PRESSURIZER LIQUID LEVEL

LIST OF FIGURES (cont'd)

FIGURE	TITLE
14.2-54	0.01 FT^2 COLD LEG BREAK WITH NO EFW, 2 HPI'S & STUCK PORV AT 20 MIN-UPPER PLENUM LIQUID LEVEL
14.2-55	0.01 FT^2 COLD LEG BREAK WITH NO EFW, 2 HPI'S & STUCK PORV AT 20 MIN-PORV LEAK FLOW
14.2-56	0.01 FT^2 COLD LEG BREAK WITH NO EFW, 2 HPI'S & STUCK PORV AT 20 MIN-PORV LEAK FLOW QUALITY
14.2-57	0.01 FT^2 COLD LEG BREAK WITH NO EFW, 2 HPI'S & STUCK PORV AT 20 MIN-COLD LEG BREAK FLOW
14.2-58	0.01 FT^2 CLOLD LEG BREAK WITH NO EFW, 2 HPI'S & STUCK PORV AT 20 MIN-COLD LEG BREAK LEAK FLOW QUALITY
14.2-59	SYSTEM PRESSURE VS TIME – SMALL BREAKS W/O EMERGENCY FEEDWATER
14.2-60	PRESSURIZER LEVEL VS TIME – CLASS 3 BREAKS W/O EMERGENCY FEEDWATER
14.2-61	INTEGRATED DIRECT DOSE FOLLOWING MHA WITH 3-1/2 FT REACTOR BULDING WALL THICKNESS
14.2-62	DELETED
14.2-63	BORON DILUTION AND PLATEOUT

APPENDICES

14 A	DESIGN REVIEW FOR CONSIDERATION OF EFFECTS OF PIPING SYSTEM BREAKS OUTSIDE CONTAINMENTS
14 B	IODINE REMOVAL CAPABILITIES OF THE TMI-1 REACTOR BUILDING SPRAY SYSTEM
14 C	EVALUATION OF ACCIDENT DOSE
14 D	DELETED
14 E	DELETED

Table 14.0-1
(Sheet 1 of 12)
Equipment And Related Systems Assumed To
Function During Accident Analysis

ACCIDENT	RPS	ICS	NNI	HSPS	TURBINE TRIP	ESAS	EFW	HPI
Uncompensated Operating Reactivity Changes (14.1.2.1)	YES	NO	NO	NO	NO	NO	NO	NO
Startup Accident (14.1.2.2)	YES	NO	NO	NO	NO	NO	NO	NO
Rod Withdrawal Accident At Rated Power Operation (14.1.2.3)	YES	NO	NO	NO	YES	NO	NO	NO
Moderator Dilution Accident (14.1.2.4)	YES	NO	YES	NO	NO	NO	NO	NO
Cold Water Accident (14.1.2.5)	NO	NO	NO	NO	NO	NO	NO	NO
Loss Of Coolant Flow (14.1.2.6)	YES	NO	NO	NO	NO	NO	NO	NO
Stuck-Out, Stuck-In, Or Dropped Control Rod Accident (14.1.2.7)	NO	YES	NO	NO	NO	NO	NO	NO
Loss Of Electric Power								
Elec. Load LOEL	YES	NO	YES	NO	YES	NO	NO	NO
All AC Power SBO	NO	NO	YES	YES	YES	NO	YES	NO
(14.1.2.8)								

Table 14.0-1
(Sheet 2 of 12)

Equipment And Related Systems Assumed To
Function During Accident Analysis

ACCIDENT	RPS	ICS	NNI	HSPS	TURBINE TRIP	ESAS	EFW	HPI
Steam Line Break (14.1.2.9)	YES	YES	NO	YES	YES	YES	YES	YES
Steam Generator Tube Failure (14.1.2.10)	YES	YES	YES	NO	YES	YES	NO	YES
Fuel Handling Accident (14.2.2.1)	NO	NO	NO	NO	NO	NO	NO	NO
Rod Ejection Accident (14.2.2.2)	YES	NO	NO	NO	NO	NO	NO	NO
Large Break LOCA (14.2.2.3)	YES	NO	NO	YES	YES	YES	YES	YES
Small Break LOCA (14.2.2.4)	YES	NO	NO	YES	YES	YES	YES	YES
Maximum Hypothetical Accident[3] (14.2.2.5)	YES	NO	NO	YES	YES	YES	YES	YES

3 Maximum Hypothetical Accident – Does not take credit for Auxiliary Building Ventilation System charcoal filters

Table 14.0-1
(Sheet 3 of 12)

Equipment And Related Systems Assumed To Function During Accident Analysis

ACCIDENT	RPS	ICS	NNI	HSPS	TURBINE TRIP	ESAS	EFW	HPI
Waste Gas Tank Rupture[4] (14.2.2.6)	NO	NO	NO	NO	NO	NO	NO	NO
Loss of Feedwater Accident								
W/stuck open PORV	YES	YES	YES	NO	YES	YES	YES	YES
Adequacy of 500gpm EFW[5]	YES	YES	NO	NO	YES	NO	YES	NO
ARTS Evaluation (14.2.2.7)	YES	YES	YES	NO	YES	NO	YES	NO
Fuel Cask Drop Accident[6] (14.2.2.8)	NO	NO	NO	NO	NO	NO	NO	NO
Feedwater Line Break Accident (14.2.2.9)	YES	NO	NO	YES	YES	YES	YES	YES

[4] Waste Gas Tank Rupture – Takes no credit for Auxiliary Building Ventilation System charcoal filters

[5] Loss of Feedwater Adequacy of 500 gpm EFW - NNI conditioned signals are assumed to reach the Control Room console, the ICS, and the plant computer

[6] Fuel Cask Drop Accident - Also takes credit for key operated interlock system to limit travel area of fuel handling crane

Table 14.0-1
(Sheet 4 of 12)
Equipment And Related Systems Assumed To
Function During Accident Analysis

ACCIDENT	LPI	MU	CFT	DHRS	PORV	PSV	PRESSURIZER SPRAY	PRESSURIZER HEATERS
Uncompensated Operating Reactivity Changes (14.1.2.1)	NO	NO	NO	NO	NO	NO	NO	NO
Startup Accident (14.1.2.2)	NO	YES	NO	NO	NO	NO	NO	NO
Rod Withdrawal Accident At Rated Power Operation (14.1.2.3)	NO	NO	NO	NO	NO	YES	NO	NO
Moderator Dilution Accident (14.1.2.4)	NO	YES	NO	NO	NO	NO	YES	NO
Cold Water Accident (14.1.2.5)	NO	NO	NO	NO	NO	NO	NO	NO
Loss Of Coolant Flow (14.1.2.6)	NO	YES	NO	NO	NO	NO	NO	NO
Stuck-Out, Stuck-In, Or Dropped Control Rod Accident (14.1.2.7)	NO	YES	NO	NO	NO	NO	NO	NO
Loss Of Electric Power (14.1.2.8)								
Elec. Load [LOEL]	NO	NO	NO	NO	NO	YES	YES	NO
All AC Power [SBO]	NO	NO	NO	NO	NO	YES	NO	NO

Table 14.0-1
(Sheet 5 of 12)

Equipment And Related Systems Assumed To
Function During Accident Analysis

ACCIDENT	LPI	MU	CFT	DHRS	PORV	PSV	PRESSURIZER SPRAY	PRESSURIZER HEATERS
Steam Line Break (14.1.2.9)	NO	NO	NO	NO	NO	NO	NO	NO
Steam Generator Tube Failure (14.1.2.10)	YES	YES	NO	YES	NO	NO	YES	NO
Fuel Handling Accident (14.2.2.1)	NO	NO	NO	NO	NO	NO	NO	NO
Rod Ejection Accident (14.2.2.2)	NO	NO	NO	NO	NO	YES	NO	NO
Large Break LOCA (14.2.2.3)	YES	NO	YES	NO	NO	NO	NO	NO
Small Break LOCA (14.2.2.4)	YES	NO	NO	NO	NO	NO	NO	NO
Maximum Hypothetical Accident[3] (14.2.2.5)	YES	NO	YES	NO	NO	NO	NO	NO

[3] Maximum Hypothetical Accident – Does not take credit for Auxiliary Building Ventilation System charcoal filters

Table 14.0-1
(Sheet 6 of 12)

<u>Equipment And Related Systems Assumed To Function During Accident Analysis</u>

ACCIDENT	LPI	MU	CFT	DHRS	PORV	PSV	PRESSURIZER SPRAY	PRESSURIZER HEATERS
Waste Gas Tank Rupture[4] (14.2.2.6)	NO	NO	NO	NO	NO	NO	NO	NO
Loss of Feedwater Accident								
W/stuck open PORV	NO	YES	NO	NO	YES	NO	NO	NO
Adequacy of 500gpm EFW[5]	NO	YES	NO	NO	NO	YES	NO	NO
ARTS Evaluation (14.2.2.7)	NO	YES	NO	NO	YES	NO	NO	NO
Fuel Cask Drop Accident[6] (14.2.2.8)	NO	NO	NO	NO	NO	NO	NO	NO
Feedwater Line Break Accident (14.2.2.9)	NO	NO	NO	NO	NO	YES	NO	NO

4 Waste Gas Tank Rupture - Takes no credit for Auxiliary Building Ventilation System charcoal filters

5 Loss of Feedwater Adequacy of 500 gpm EFW - NNI conditioned signals are assumed to reach the Control Room console, the ICS, and the plant computer

6 Fuel Cask Drop Accident - Also takes credit for key operated interlock system to limit travel area of fuel handling crane

Table 14.0-1
(Sheet 7 of 12)

Equipment And Related Systems Assumed To
Function During Accident Analysis

ACCIDENT	MFW	MSS	DHC	MSIV	FWIV	ADV	MSSV
Uncompensated Operating Reactivity Changes (14.1.2.1)	NO	NO	NO	NO	NO	NO	NO
Startup Accident (14.1.2.2)	YES	YES	NO	NO	NO	NO	NO
Rod Withdrawal Accident At Rated Power Operation (14.1.2.3)	NO	NO	NO	NO	NO	NO	NO
Moderator Dilution Accident (14.1.2.4)	NO	NO	NO	NO	NO	NO	NO
Cold Water Accident (14.1.2.5)	YES	YES	NO	NO	NO	NO	NO
Loss Of Coolant Flow (14.1.2.6)	YES	YES	NO	NO	NO	NO	NO
Stuck-Out, Stuck-In, Or Dropped Control Rod Accident (14.1.2.7)	NO	NO	NO	NO	NO	NO	NO
Loss Of Electric Power							
Elec. Load [LOEL]	NO	YES	NO	NO	NO	NO	YES
All AC Power [SBO]	NO	YES	NO	NO	NO	YES	YES
(14.1.2.8)							

Table 14.0-1
(Sheet 8 of 12)

Equipment And Related Systems Assumed To
Function During Accident Analysis

ACCIDENT	MFW	MSS	DHC	MSIV	FWIV	ADV	MSSV
Steam Line Break (14.1.2.9)	YES	YES	NO	NO	YES	YES	YES
Steam Generator Tube Failure (14.1.2.10)	YES	YES	YES	YES	YES	NO	YES
Fuel Handling Accident (14.2.2.1)	NO	NO	NO	NO	NO	NO	NO
Rod Ejection Accident (14.2.2.2)	NO	NO	NO	NO	NO	NO	NO
Large Break LOCA (14.2.2.3)	NO	NO	YES	YES	YES	NO	YES
Small Break LOCA (14.2.2.4)	NO	NO	YES	YES	YES	NO	YES
Maximum Hypothetical Accident[3] (14.2.2.5)	NO	NO	YES	YES	YES	NO	YES

[3] Maximum Hypothetical Accident – Does not take credit for Auxiliary Building Ventilation System charcoal filters

Table 14.0-1
(Sheet 9 of 12)

Equipment And Related Systems Assumed To
Function During Accident Analysis

ACCIDENT	MFW	MSS	DHC	MSIV	FWIV	ADV	MSSV
Waste Gas Tank Rupture[4] (14.2.2.6)	NO	NO	NO	NO	NO	NO	NO
Loss of Feedwater Accident							
W/stuck open PORV	NO	NO	NO	NO	NO	NO	NO
Adequacy of 500gpm EFW[5]	NO	NO	NO	NO	NO	NO	NO
ARTS Evaluation (14.2.2.7)	NO	NO	NO	NO	NO	NO	NO
Fuel Cask Drop Accident[6] (14.2.2.8)	NO	NO	NO	NO	NO	NO	NO
Feedwater Line Break Accident (14.2.2.9)	YES	NO	NO	NO	NO	NO	YES

4 Waste Gas Tank Rupture - Takes no credit for Auxiliary Building Ventilation System charcoal filters

5 Loss of Feedwater Adequacy of 500 gpm EFW - NNI conditioned signals are assumed to reach the Control Room console, the ICS, and the plant computer

6 Fuel Cask Drop Accident - Also takes credit for key operated interlock system to limit travel area of fuel handling crane

Table 14.0-1
(Sheet 10 of 12)

Equipment And Related Systems Assumed To
Function During Accident Analysis

ACCIDENT	TBV	TSV	BS	BWST	RBES	HSPS	OPERATOR ACTION
Uncompensated Operating Reactivity Changes (14.1.2.1)	NO	NO	NO	NO	NO	NO	NO
Startup Accident (14.1.2.2)	NO	NO	NO	NO	NO	NO	NO
Rod Withdrawal Accident At Rated Power Operation (14.1.2.3)	NO	NO	NO	NO	NO	NO	NO
Moderator Dilution Accident (14.1.2.4)	NO	NO	NO	NO	NO	NO	NO
Cold Water Accident (14.1.2.5)	NO	NO	NO	NO	NO	NO	NO
Loss Of Coolant Flow (14.1.2.6)	NO	NO	NO	NO	NO	NO	NO
Stuck-Out, Stuck-In, Or Dropped Control Rod Accident (14.1.2.7)	NO	NO	NO	NO	NO	NO	NO
Loss Of Electric Power							
Elec. Load [LOEL]	NO	YES	NO	NO	NO	NO	NO
All AC Power [SBO]	NO	YES	NO	NO	NO	NO	NO
(14.1.2.8)							

Table 14.0-1
(Sheet 11 of 12)

Equipment And Related Systems Assumed To Function During Accident Analysis

ACCIDENT	TBV	TSV	BS	BWST	RBES	HSPS	OPERATOR ACTION
Steam Line Break (14.1.2.9)	YES	YES	YES	YES	YES	YES	NO
Steam Generator Tube Failure (14.1.2.10)	YES	YES	NO	NO	NO	NO	YES
Fuel Handling Accident (14.2.2.1)	NO	NO	NO	NO	NO	NO	NO
Rod Ejection Accident (14.2.2.2)	NO	NO	NO	NO	NO	NO	NO
Large Break LOCA (14.2.2.3)	NO	NO	YES	YES	YES	NO	NO
Small Break LOCA (14.2.2.4)	NO	NO	YES	YES	YES	NO	YES
Maximum Hypothetical Accident[3] (14.2.2.5)	NO	NO	YES	YES	YES	NO	NO

[3] Maximum Hypothetical Accident – Does not take credit for Auxiliary Building Ventilation System charcoal filters

Table 14.0-1
(Sheet 12 of 12)

Equipment And Related Systems Assumed To
Function During Accident Analysis

ACCIDENT	TBV	TSV	BS	BWST	RBES	HSPS	OPERATOR ACTION
Waste Gas Tank Rupture[4] (14.2.2.6)	NO	NO	NO	NO	NO	NO	NO
Loss of Feedwater Accident							
w/stuck open PORV	YES	YES	NO	NO	NO	NO	NO
Adequacy of 500gpm EFW[5]	YES	YES	NO	NO	NO	NO	NO
ARTS Evaluation (14.2.2.7)	YES	YES	NO	NO	NO	NO	NO
Fuel Cask Drop Accident[6] (14.2.2.8)	NO	NO	NO	NO	NO	NO	NO
Feedwater Line Break Accident (14.2.2.9)	NO	YES	NO	YES	NO	NO	NO

4 Waste Gas Tank Rupture - Takes no credit for Auxiliary Building Ventilation System charcoal filters

5 Loss of Feedwater Adequacy of 500gpm EFW - NNI conditioned signals are assumed to reach the Control Room console, the ICS, and the plant computer

6 Fuel Cask Drop Accident - Also takes credit for key operated interlock system to limit travel area of fuel handling crane

14.0 SAFETY ANALYSIS

Since the performance of the original analyses in this chapter, additional analyses were performed to demonstrate plant safety with regard to response to postulated events. These analyses caused the Technical Specifications, operating procedures, and in some cases, the design to be changed in an effort to continue to confirm the results of this chapter.

At the end of Cycle 17, the original OTSGs were replaced. In support of Cycle 18 operation, evaluations and reanalyses of the UFSAR Chapter 14 events were performed to verify that the acceptance criteria for each event would be met with the replacement OTSGs. Either the analysis of record for the event was shown to be bounding for the replacement OTSGs or a new analysis produced results that met the applicable acceptance criteria. The revised analyses that are now part of the design basis analyses of record are MSLB for containment, LOFW, FWLB, and LOEL. The remaining Chapter 14 accident analyses continue to be bounding for the replacement OTSGs including the HELB outside containment analyses included in Appendix 14A.

It is noted that the LOFW, FWLB, SBLOCA and LBLOCA analysis results were directly affected by steam generator tube plugging. The replacement OTSG reanalyses modeled five percent OTSG tube plugging for the LOFW, FWLB, and SBLOCA events, while ten percent tube plugging was considered in the evaluation of the LBLOCA event. Consequently, if the actual TMI-1 replacement OTSG tube plugging were to exceed five percent or eventually 10 percent, the continued applicability of the reanalyses would need to be evaluated. The results of these evaluations as well as other safety analyses affecting the licensing basis are provided in References 124, 126, and 127.

14.1 CORE AND COOLANT BOUNDARY PROTECTION ANALYSIS

14.1.1 ABNORMALITIES

In previous Chapters of this report, both normal and abnormal operations of the various systems and components have been discussed. This Chapter summarizes and further explores abnormalities that either are inherently terminated or require the normal protection systems to operate to maintain integrity of the fuel and/or the Reactor Coolant System. Most of these abnormalities have been evaluated for a core power greater than or equal to 2535 MWt. Table 14.1-1 summarizes the potential abnormalities studied.

For TMI-1 Cycle 7 reload design, the rated power was upgraded from 2535 MWt to 2568 MWt. Each accident analysis in this chapter has been examined with respect to the power upgrade and corresponding accident parameter changes. The safety evaluation concluded that the power upgrade does not present any adverse safety impact and that the previously accepted design basis (safety analysis parameters) used in the FSAR bounds the power upgrade parameters (Reference 62).

The radiological source term data were reevaluated to reflect: (1) power upgrade to 2568 and (2) slight increase radioactivities due to increased fission yields of Pu-239. Furthermore, the Cycle 7 core inventory was deliberately increased by applying a conservatism factor of 1.10 for the purpose of enveloping future cycle variations. The updated source term data are presented in Table 14.2-4. Subsequent two-year cycle designs may result in some of the isotopic

activities in this table being exceeded. However, reload evaluations are performed each cycle to ensure that dose consequences for all affected accidents are not increased.

The radiological dose consequence based on the updated source term data are below the 10CFR50.67 dose acceptance criteria for the LOCA and FHA in the containment. For other accidents, the 10CFR100 dose acceptance criteria are used as incorporated in each subsection below.

For TMI-1 Cycle 17 reload design, the AREVA Mark-B-HTP fuel design was introduced. Each accident analysis in this chapter was evaluated relative to the effect the Mark-B-HTP fuel design could have on the sequence of events or the consequences of the events. DNB related events and LOCA were re-analyzed for the introduction of the Mark-B-HTP fuel design. The remaining non-LOCA events required no reload independent re-analyses. The evaluations and analyses provided a conclusion that the introduction of the Mark-B-HTP fuel design was acceptable relative to UFSAR described acceptance criteria and that the sequence of events was not affected. Consequently, the event descriptions in this chapter were not updated for the introduction of the Mark-B-HTP fuel design and the information is retained as representative of the accepted design basis or for historical purposes. Mark-B-HTP specific information due to the re-analysis of DNB related events, LOCA or Mark-B-HTP design specific data or correlations has been included where appropriate.

Additional documentation and references to specific calculational methods and criteria are identified in Reference 69.

14.1.2 ANALYSIS OF EFFECTS AND CONSEQUENCES

14.1.2.1 Uncompensated Operating Reactivity Changes

a. Identification of Cause

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

b. Analysis and Results

During normal operation, the automatic reactor control system senses any reactivity change in the reactor. A reactivity addition rate based on either predicted xenon buildup or fuel depletion was used to initiate the event. The analysis assumed a constant ramp insertion of the reactivity which continued throughout the analysis. The rate of reactivity change associated with the xenon buildup was based on using the maximum slope of the reactivity versus time curve for xenon buildup. Depending on the direction of the reactivity change, the reactor power increases or decreases. Correspondingly, the Reactor Coolant System average temperature increases or decreases, and the automatic reactor control system may act to restore reactor power to the power demand level and to reestablish this temperature at its set point. If manual corrective action is not taken or if the automatic control system malfunctions, the Reactor Coolant System average temperature changes to compensate for the reactivity disturbance. It is

assumed in the analysis that the secondary system follows the temperature changes in the Reactor Coolant System.

The acceptance criteria used in evaluating the event were: 1) the rate of reactivity addition was much less than the rate at which the operator could compensate for the addition, and 2) the rate of temperature change was much less than the rate at which the control system could compensate for the change. The analysis is driven by the moderator and fuel temperature changes which result from the reactivity increase. This difference is absolute and is not a function of power level.

Table 14.1-2 summarizes the reactivity changes and the corresponding change in the average moderator temperature. These reactivity changes are extremely slow and allow the operator to detect and compensate for the change. Operator actions to counter the reactivity changes would include boration and control rod motion.

14.1.2.2 Startup Accident

a. Identification of Cause

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 nuclear excursion. The uncontrolled reactivity addition, through rod withdrawal from zero power, is a startup accident. This excursion is terminated by the strong negative Doppler effect if no other protective action operates.

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

1) The control system is designed so that only one control rod group can be withdrawn at a time, except that there is a 25 percent overlap in travel between two regulating rod groups successively withdrawn. This overlap occurs at the minimum worth positions for each group because one group is at the end of travel and the other is at the beginning of travel.

2) Control rod withdrawal rate is limited.

3) A startup rate withdrawal stop and alarm are provided in the source range.

4) A startup withdrawal stop and alarm are provided in the intermediate range.

5) A high flux level and a high pressure trip are provided.

The criteria for the analysis of this accident is that the Reactor Protection System shall be designed to limit (1) the reactor thermal power to the design overpower condition (112 percent rated power) and (2) the Reactor Coolant System pressure so as not to exceed the ASME Code allowable pressure limit of 2750 psig (110 percent of Design Pressure).

b. Methods of Analysis

A B&W digital computer model 1 of the reactor core and coolant system was used to determine the characteristics of this accident. This model used full reactor coolant flow, but no heat transfer out of the system and no sprays in the pressurizer. The rated power Doppler coefficient was used, although the Doppler coefficient is much larger than this for the principal part of the transient. The rods were assumed to be moving along the steepest part of the rod-worth versus rod-travel curve, which results in the largest reactivity insertion rate calculated for the control rods as they were withdrawn from the core. The values of the principal parameters used in this analysis are listed in Table 14.1-3.

In addition, the criterion for minimum movable control rod worth is that a shutdown margin of 1 percent delta-k/k at the hot standby condition is required (see Item a of Section 3.2.3.1). The startup accident has been analyzed using the minimum tripped rod worth as part of the analysis.

The startup accident was analyzed from 1 percent delta-k/k subcritical at the hot, pressurized condition.

The results of the startup accident are not affected by the power uprate to 2568 MWt. The event is driven by the difference in heat generated in the core and the heat removed by the steam generators. Until a reactor trip is reached, the event evolution is the same, regardless of the power level. The high pressure trip setpoint is not affected by the core power level. Thus, for all reactivity insertion rates that result in a high pressure trip, the results are independent of the power upgrade. The high flux trip setpoint used in the analysis was 114 percent of 2535 MWt. This bounds the plant safety limit of 112 percent of 2568 MWt. Therefore, for reactivity insertion rates that result in a high flux trip, the existing analysis bounds TMI-1 at 2568 MWt.

Feedwater flow and temperature is assumed to be constant throughout the event. This assumption is conservative since the lack of primary to secondary heat transfer causes the primary system to retain more energy during the event.

c. Results of Analysis

Figure 14.1-1 shows the results of withdrawing the maximum worth control rod group at the maximum rod speed from 1 percent subcritical. This rod velocity and worth result in the maximum reactivity addition rate. The Doppler effect begins to slow the neutron power* rise, but the heat input to the reactor coolant increases the pressure past the trip point, and the transient is terminated by the high pressure trip.

Figure 14.1-2 shows the results of with drawing all control rod assemblies at the maximum speed from 1 percent subcritical. This results in a maximum possible reactivity addition rate. Although the calculated total rod worth (Table 14.1-3) is slightly higher, the sensitivity analyses on Figures 14.1-3 and 14.1-4 indicate that the difference

*Neutron power is defined as the total energy release from fission.

will have little effect on the results. The power rise is terminated by the negative Doppler effect. The high neutron flux trip takes effect after the peak power is reached and terminates the transient. The peak thermal heat flux is significantly less than the rated power heat flux.

A sensitivity analysis was performed on both of these startup accidents to determine the effect of varying several key parameters. Variation of the trip delay time from 0.1 to 0.7 seconds resulted in a change in peak thermal power of only a few percent. Figures 14.1-3 through 14.1-6 show typical results for the single rod group startup accident.

Figures 14.1-3 and 14.1-4 show the effect of varying the reactivity addition rate on the peak thermal power and peak neutron power. This reactivity rate was varied from more than an order of magnitude below the nominal single rod-group rate used for the analysis to a rate above that for simultaneous withdrawal of all rods. The slower rates will result in the pressure trip being actuated. Only the very fast rates actuate the high neutron flux level trip.

Figures 14.1-5 and 14.1-6 show the peak thermal power variation as a function of a wide range of Doppler and moderator coefficients for the single rod group. The peak thermal power varies a few percent from the nominal case for the moderator coefficient variation, and also by a few percent from the nominal for the range of Doppler coefficients. Figures 14.1-7 and 14.1-8 are the corresponding results for the withdrawal of all rods.

Table 14.1-4 summarizes the results for the postulated startup accidents.

It is concluded that the reactor is completely protected against any startup accident involving the withdrawal of any or all control rods, since in no case does the thermal power approach the design overpower condition, and the peak pressure never exceeds 2750 psig.

d. Re-analysis for 20% Average Steam Generator Tube Plugging

The startup accident was reanalyzed to justify a 20% average steam generator tube plugging (Reference 84). The RETRAN-02 Mod 5.2 computer code and a TMI plant model were used to perform this analysis. The startup accident is analyzed at hot zero power by modeling the control rod withdrawal as a reactivity insertion. For a slow reactivity insertion rate (RIR), the reactor trips on high RCS pressure. For a rapid RIR, the reactor trips on high flux before the high RCS pressure trip setpoint is reached. A variety of cases are run varying the RIR's to determine the RIR which causes the high pressure trip and high flux trip setpoints to be coincident. This RIR is the most limiting with regard to peak RCS pressure.

While it is not mechanistically possible to achieve this RIR, performing the analysis with this RIR results in an RCS pressure increase greater than withdrawing the maximum worth control rod at the maximum speed or withdrawing all the control rods at the maximum speed. In addition, the steam generators are conservatively assumed to remove reactor coolant pump heat only. The re-analysis of the startup accident with the assumption of 20% average steam generator tube plugging also included the following

(more conservative) changes from the original UFSAR analysis in initial conditions and assumptions (Reference 84):

- an initial power level of 2.568 Watts
- a pressurizer safety valve lift tolerance of +3 percent
- a pressurizer safety valve flowrate of 297,846 lbm/hr/valve
- a high flux trip setpoint of 112% rated power (2568 MWt)
- a high pressure trip setpoint of 2402 psia in the hot leg
- a high flux trip delay of 0.4 sec
- a high pressure trip delay of 0.6 sec
- a moderator temperature coefficient of +0.9 x 10^{-4} dk/k/F
- an initial RCS temperature of 525°F

The results of the re-analysis were reviewed and approved by the NRC (Reference 86) and show that the acceptance criteria for the event are met with considerable margin.

14.1.2.3 Rod Withdrawal Accident At Rated Power Operation

a. Identification of Cause

A rod withdrawal accident presupposes an operator error or equipment failure which results in accidental withdrawal of a control rod group while the reactor is at rated power. This uncontrolled withdrawal results in positive reactivity addition. As a result of this assumed accident, the power level increases, the reactor coolant and fuel rod temperatures increase, and, if the withdrawal is not terminated by the operator or protection system, core damage, or loss of the integrity of the primary system pressure boundary would eventually occur.

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

1) High reactor coolant outlet temperature alarms.

2) High reactor Coolant System pressure alarms.

3) High pressurizer level alarms.

4) High reactor coolant outlet temperature trip.

5) High reactor Coolant System pressure trip.

6) High power level (i.e., neutron flux level) trip.

The rod withdrawal accident analysis is performed with the criteria that the Reactor Protection System will limit: (1) the reactor thermal power to the design overpower condition and (2) the Reactor Coolant System pressure to 2750 psig.

The results of the rod withdrawal accident are not affected by the power uprate to 2568 MWt. The high pressure trip setpoint is not affected by the core power level. Thus, for all reactivity insertion rates that result in a high pressure trip, the results are independent of rated power level. The high flux trip setpoint used in the analysis was 114 percent of

2535 MWt. This bounds the plant safety limit of 112 percent of 2568 MWt. Therefore, for reactivity insertion rates that result in a high flux trip, the existing analysis bounds TMI-1 at 2568 MWt.

b. Methods of Analysis

A B&W digital computer code (Reference 1) was used to determine the characteristics of this accident. 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 were included. 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 the evaluation of this accident are specified in Table 14.1-5. The minimum control rod worth that satisfies the criterion for a shutdown margin of 1 percent delta-k/k at the hot standby condition is used throughout the analysis.

Assumptions were made to ensure the analysis of the Rod Withdrawal Accident is conservative yet realistic. These assumptions include the following:

1. Full reactor coolant flow
2. Inoperable pressurizer spray
3. Rated power doppler coefficient
4. Control Rod assemblies fully inserted prior to accident
5. Reactivity insertion rates generated from steepest section of integral rod worth versus position curve
6. Beginning-of-life conditions for fuel related parameters

c. Results of Analysis

Figure 14.1-9 shows the results of the nominal rod group withdrawal from rated power. 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 all quite small as shown in Table 14.1-6.

A sensitivity analysis of important parameters was performed around this nominal case, and the resultant Reactor Coolant System pressure responses are shown on Figures 14.1-10 through 14.1-13.

Figure 14.1-10 shows the pressure variation for a very wide range of rod withdrawal rates - more than an order of magnitude, smaller and greater than the nominal case. 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 rod withdrawal accidents. The high pressure trip is relied upon the slower transients (lower rod worths). In no case does the thermal power exceed the design overpower.

Figures 14.1-11 through 14.1-13 show the pressure response to variations in the trip delay time, Doppler coefficient, and moderator coefficient. In all cases, the neutron flux level trip is actuated.

An analysis has been performed extending the evaluation of the rod withdrawal accident for various fractional initial power levels up to rated power. This evaluation has been performed assuming simulated withdrawal of all control rods giving the maximum possible reactivity addition rate. This rate is a significantly higher rate than that used in the cases evaluated for withdrawal of a single group (Table 14.1-5). The results of this analysis are shown on Figures 14.1-14 and 14.1-15.

As seen on Figure 14.1-14, the peak thermal power occurs for the rated power case and is well below the design overpower. The peak neutron power for all cases slightly overshoots the assumed trip level. Figure 14.1-15 shows that the maximum fuel temperature reached in the average rod and the hot spot are well below melting. Even in the most severe case at rated power, the average fuel temperature increases only a few degrees. It is, therefore, readily concluded that no fuel damage would result from simultaneous withdrawal of all rods from any initial power level.

This analysis demonstrates that the high pressure trip and the high flux level trip adequately protect the reactor against any rod withdrawal accident from power operations. The acceptance criteria would also be met with considerable margin for 20% average generator tube plugging (Reference 86).

14.1.2.4 Moderator Dilution Accident

Boron, in the form of boric acid, in the reactor coolant helps control excess reactivity in the core. The moderator dilution accident addresses the concern that the boron could be diluted uncontrollably leading to a power escalation similar to that associated with a control rod withdrawal event. Uncontrolled dilution is analyzed at normal operating conditions and during refueling or maintenance conditions.

a. Identification of Cause

The values of system parameters used in the evaluation of this accident are listed in Table 14.1-7. 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. The dilution flow rate of 70 gpm is representative of the high side of normal flow rates during normal operation. The 100 gpm flow rate is a near maximum for the high pressure conditions seen at power.

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 high pressure injection pump operating with reduced Reactor Coolant System pressure, the resulting inflow rate could be much higher than the normal rate (Table 14.1-7). This constitutes the maximum dilution accident. A reactor trip would terminate unborated water addition to the makeup tank, and total flow into the coolant system would be terminated by high pressurizer level. The 500 gpm flow rate was chosen to bound all operating power conditions.

The criteria for reactor protection in this accident are:

1) The reactor thermal power will be limited to less than the design overpower.

2) The Reactor Coolant System pressure will be limited to less than the 2750 psig.

3) The reactor minimum shutdown margin of 1 percent delta-k/k subcritical will be maintained.

b. Analysis and Results

The reactor is assumed to be operating at rated power with a maximum initial boron concentration as listed in Table 14.1-7 in the Reactor Coolant System. The dilution water is uniformly distributed throughout the reactor coolant volume. Uniform distribution exists 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 the moderator coefficient and boron concentration values used are conservative. The effects of the three dilution rates discussed above on the reactor are as tabulated in Table 14.1-8.

While not credited in the analysis, the highest rate of dilution can be handled by the automatic control system, which inserts rods to maintain the power level and the Reactor Coolant System temperature.

If the reactor is under manual control with no control rod insertion, these reactivity additions will cause a high temperature or high pressure trip, which will cause the makeup valve to close, terminating the addition of deborated water to the makeup system. Peak pressures and thermal powers for this case are shown in Table 14.1-8 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 2750 psig. 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 only about 0.8 percent delta-k/k. This effect will only be seen when the initial boron concentration is high, which is at beginning of core life when sufficient rod worth is available to shut down the reactor by several percent even with the highest worth rod stuck out of the core. Thus, the minimum shutdown margin of 1 percent 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 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 rod 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 only the coolant in the reactor vessel is diluted), and the reactor minimum shutdown margin of 1 percent delta-k/k subcritical would be maintained.

In the cases analyzed at power, the reactor protection system provides a reactor trip with large margins to each of the acceptance criteria. During refueling, the analysis yields considerable margin in comparison to the acceptance criteria. The above analysis remains bounded for the increased power level of 2568 MWt. The results for the cases at power will remain well within the acceptance criteria with 20% average steam generator tube plugging. The moderator dilution accident in the refueling mode is not affected by steam generator tube plugging (Reference 86).

A review of the plant systems and procedures was performed to determine the potential pathways for a moderator dilution event. (Reference 106) The purpose of this review and evaluation is to determine if a possible moderator dilution event exists with a closed Reactor Coolant System (RCS) in a drained down state, which was not previously considered and would cause a loss of shutdown margin. Four general pathways have been identified, the Makeup and Purification System, Liquid Waste Disposal System, Reactor Building Spray System, and leakage through seals and/or packing.

The Reactor Building Spray System dilution path via the Decay Heat Removal System has been studied for a full RCS and the method for precluding this dilution event is considered acceptable by the NRC in Reference 12. This method is equally applicable during drained down conditions and therefore need not be addressed further in this context.

During drained down conditions, the systems which indirectly interface with the RCS through seals, motor coolers, or packings are either shutdown or create a potential for such small dilution rates as to be insignificant. Dilution events of this nature occur slowly enough so that operator action to terminate or compensate for deboration can be made based on periodic boron samples and/or the Control Room instrumentation.

The Makeup and Purification System provides one possible pathway for injection to the RCS of unborated water from the makeup tank. Under drained down conditions, or any cold shutdown condition, administrative measures are normally applied which prevent HPI injection of water from the makeup system. These include racking out the breakers for the makeup pumps, closing their associated makeup valves MU-V217, MU-V16A, MU-V16B, MU-V16C and MU-V16D, and opening their respective breakers. Also, injection of unborated water from the makeup tank would terminate with greater than the

minimum 1% delta-k/k shutdown margin remaining. In view of this, any likelihood of injection from the makeup tank during cold shutdown is very small and even if it would occur during drained down conditions, it would not result in a loss of the shutdown margin.

Since the Decay Heat Removal System is normally in operation when the RCS is in cold shutdown, this system provides another possible pathway by which unborated water from an RC Bleed Tank in the Liquid Waste Disposal System could be added to the RCS. However, this would also require an extensive valve lineup. Although highly unlikely, if inadvertent injection of unborated water from an RC Bleed Tank to the RCS were to occur, the consequences are even less severe than addition from the makeup tank resulting in an even larger shutdown margin remaining.

All calculations were performed using a conservative volume for the drained down RCS. Appropriate assumptions are listed in Table 14.1-8.

The conclusion of this review is that there are no dilution pathways with significant flow rates which could be established through a single failure or operator error during cold shutdown. Furthermore, the normally available shutdown margin would not be lost even in the unlikely event of multiple failures leading to the establishment of a significant dilution pathway.

c. Core Reload Evaluation

For a given flow rate of dilute, the reactivity insertion for a moderator dilution event at power is proportional to the initial boron concentration divided by the inverse boron worth, C_{cr}/IBW. As part of each reload core design process, the ratio of the beginning-of-cycle critical boron concentration to inverse boron worth, C_{cr}/IBW, is compared with the ratio of values used in the analysis of the moderator dilution accident at power (Table 14.1-7). The objective is to show that the core reload value for C_{cr}/IBW is less than the analysis value, ensuring the accident analysis remains bounding.

In addition to the reload evaluation at full power, the moderator dilution event during refueling is evaluated for each reload core. A calculation is performed to determine the boron concentration in the core after diluting one reactor vessel volume of liquid—initially at the refueling boron concentration—with one makeup tank volume of liquid at zero ppm. The resultant core boron concentration is compared with the critical boron concentration during refueling with all rods in minus the two maximum worth rods stuck out, to show that the core would remain subcritical following this event.

14.1.2.5 <u>Cold Water Accident</u>

a. Identification of Cause

This accident is not possible in this reactor because the Reactor Coolant System piping contains no check valves or other isolation valves, thus eliminating the basis for the cold water accident.

It is possible to reduce the average reactor coolant temperature in the core, however. When the reactor is operated with one or more pumps not running, and if these pumps

are then turned on, the increased flow rate will cause the average core temperature to decrease. If the moderator temperature coefficient is negative, reactivity will be added and a power rise will occur.

b. Protective Basis

Several different functions in the protective system are available to terminate any transient that might result from the starting of an idle pump. For the case where the power rise is slow, as is the normal situation for this accident, the high pressure trip will terminate the accident if the high pressure set point is reached. However, if the power rise is rapid, the power/flow comparator is set so that when the reactor power is greater than the flow by a given amount (the set point), the reactor will trip. The overpower trip will limit the power to the maximum design overpower in all cases.

In addition to these primary protective devices, the power/pump comparator acts to fix the initial conditions from which this transient can be started if fewer than four reactor coolant pumps are operating.

An additional protective function is provided in the form of an interlock in the reactor coolant pump electrical system wherein the operator is prevented from starting an idle pump if the reactor power is greater than a fixed power level.

Protection exists at four distinct levels for this accident:

1) The pump control interlock acts to prevent the accident from occurring.

2) The power/pump comparator fixes the initial conditions in such a way that a severe accident (several pumps starting while at high power) cannot occur.

3) The Reactor Protection System operates so that the results of this accident stay within design limits.

4) Pumps cannot be started above 30 percent power.

c. Method of Analysis

A detailed digital simulation of the plant (Reference 1) was used to evaluate the transient response to this accident. The model includes point kinetics, a multiregion fuel pin model, a pressurizer model, and a steam generator model.

To maximize the power response to the core temperature decrease, end-of-life core conditions were assumed in the analysis. At end-of-life the moderator temperature coefficient is most negative. Coupled with the reduction in core average temperature, the result was the largest power rise possible. To be consistent with end-of-life conditions, the end-of-life Doppler coefficient was used as well.

It was assumed that the plant was operating with two pumps (1 pump in each loop) at 50 percent of rated power when the remaining two pumps were started. Of the allowed operating conditions, the 1/1 pump status resulted in the greatest moderator temperature decrease for the cold water event. The mixed moderator average

temperature as a function of time was independently calculated for the pump startup and imposed as input to the reactor system model. It was found that the maximum temperature decrease for this case was proportional to the core temperature rise and occurred at the end of one loop time. The assumed value of the moderator coefficient corresponds to end-of-life conditions. Using the most negative moderator coefficient results in the greatest positive reactivity feedback due to the core average temperature decrease which occurs, thereby maximizing the peak thermal power. Conservative values for the moderator and Doppler feedback coefficients assumed were, -3.0×10^{-4} (delta-k/k)/F and -1.2×10^{-5} (delta-k/k)/F, respectively. The tripped rod worth used corresponds to the minimum worth available with the maximum-worth rod stuck out.

It should be noted that while the analyses presented in this section assumed an MTC of -3.0×10^{-4} delta-k/k/ºF, it has been determined that adequate margin is available with an MTC of -4.0×10^{-4} delta-k/k/ºF (Reference 65).

The values of the principal parameters used in this analysis are listed in Table 14.1-22. The power upgrade to 2568 MWt does not have a significant impact on the results of the cold water accident. Further, 20% average steam generator tube plugging will also not have a significant effect on the results (Reference 86).

d. Results of Analysis

The results of the calculation are shown on Figure 14.1-16. It is seen that a maximum power is reached several seconds after the pumps are started. The pressure very nearly reaches the 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 after the neutron power. These values are:

Maximum neutron power (at 7 sec) 72 percent

Maximum thermal power (at 8 sec) 61 percent

Maximum Pressure rise (at 12 sec) 200 psi

The high flux trip setpoint and the high pressure trip setpoint are not constrained by the cold water accident as the high flux trip and high pressure trip were not reached.

It is concluded that the pump control circuitry is adequate to protect the core in the event that an idle pump is started. Additionally, the automatic control system would serve to limit the imbalance between the reactor and steam generator and to reduce the pressure and power swings.

14.1.2.6 Loss Of Coolant Flow

a. Identification of Cause

A reduction in the reactor coolant flow rate occurs if one or more of the reactor coolant pumps fail. With the reactor at power, the result is an increase in reactor coolant system (RCS) temperature and a reduction in heat removal capability of the RCS. This

increase in RCS temperature and reduction in heat removal capability could result in departure from nucleate boiling (DNB) in the core. A pumping failure can occur from mechanical failures or from a loss of electrical power. With four independent pumps available, a mechanical failure in one pump will not affect operation of the others.

Each reactor coolant pump receives electrical power from one of the two electrically separate buses, as discussed in Chapter 8, one pump in each loop being connected to each bus. Loss of a unit auxiliary transformer to which one 6900 V bus is normally connected will initiate a rapid transfer to the second transformer source 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 is extremely unlikely and would occur only on a loss of all offsite power. In spite of the low probability of this event, the nuclear unit has been designed so that such a failure would not lead to core damage.

Three types of Loss of Coolant Flow accidents are analyzed; four pump coastdown, a single pump locked rotor, and singular pump coastdown. In addition, two steady state analyses were performed to demonstrate adequate DNB margin for an open reactor internals vent valve, Table 14.1-9. (Note: A core flow penalty for an open reactor internals vent valve is no longer required per Reference 88).

The four pump coastdown Loss of Coolant Flow event is the complete loss of forced flow in the RCS. Reactor protection for the four pump coastdown event is provided by the power/pump monitors trip function of the RPS. When the reactor is at full power, a trip condition results upon signal of a loss of power to two or more of the four reactor coolant pumps. After the pumps have stopped, natural circulation flows provide adequate heat transfer capability for core cooling and decay heat removal.

The single pump coastdown loss of coolant flow event is the complete loss of forced flow through a single pump with the affected pump impeller remaining free spinning. Reactor protection is provided by the flux-to-flow ratio that is included in the power/imbalance/flow RPS trip envelope. No credit for mitigative actions by the ICS to reduce power is taken. Once the reactor is tripped, the forced circulation flow is more than sufficient to remove the residual core decay heat and stored energy.

The locked rotor event occurs when one of the reactor coolant pump rotors seizes. When the rotor seizes, the pump in the affected cold leg no longer provides forced flow. The locked rotor event results in a more rapid reduction in flow than the four-pump coastdown event. Since the DNB analysis for the loss of flow events are controlled by the rate of reduction in flow, the locked rotor event is the most limiting. Reactor protection for this event is provided by the flux/flow trip that is part of the power-imbalance-flow envelope.

The criterion for reactor protection for the four and single pump loss of coolant flow accidents is that the minimum DNB ratio shall not be less than 1.3 for the W-3 correlation and B&W-2 correlation, 1.18 for the BWC correlation, and 1.132 for the BHTP correlation. The CHF correlation is fuel design specific and may change, but will be approved by the NRC prior to its application. For the locked rotor accident, the minimum DNB ratio shall not be less than 1.0.

The Statistical Core Design methodology (Reference 111) was applied to TMI Unit 1 core DNB analyses starting in Cycle 15. This methodology provides an increase in core thermal (DNB) margin by treating core state and bundle uncertainties statistically. The traditional method of treating uncertainties was to assume the worst level of each uncertainty simultaneously. Applying statistical techniques allows for a realistic assessment of core DNB protection. The Statistical Design Limit (SDL) developed for the BWC CHF correlation for B&W 177 FA plants provides 95 percent protection at a 95 percent confidence level against hot pin DNB. The corresponding core-wide protection on a pin-by-pin basis using real peaking distributions greater than 99.9 percent. The SDL is equivalent to the traditional DNBR limit of 1.18 (BWC), which only accounts for DNBR correlation uncertainty. A SDL was also developed for the BHTP correlation (Reference 118) using the LYNXT computer code. The BHTP SDL is equivalent to the traditional DNBR limit of 1.132 (BHTP), which only accounts for DNBR correlation uncertainty (Reference 119).

b. Methods of Analysis

The loss of coolant flow accident was analyzed by a combination of analog and digital computer programs. Analog simulation was used to determine the reactor flow rate following loss of pumping power. Reactor power, coolant flow, and inlet temperature are input data to the digital program which determines the core thermal characteristics during the flow coastdown.

The B&W digital computer KAPP model used to determine the neutron power following reactor trip includes six delayed neutron groups, control rod worth and rod insertion characteristics, and trip delay time. The Analog Model Pump used to determine flow coastdown characteristics includes description of flow-pressure drop relations in the reactor coolant loops. Pump flow characteristics are determined from manufacturers' zone maps. Flow-speed, flow-torque, and flow-head relationships are solved by affinity laws.

A transient, thermal hydraulic, B&W digital computer program RADAR (Reference 3) is used to compute channel DNBR continuously during the coastdown transient. System flow, neutron power, and core entering enthalpy are varied as a function of time. The representative hot channel flows and corresponding DNB ratios are obtained by using the average core pressure drop. The hot channel DNBR as a function of time is compared with the design DNBR at maximum overpower to determine the degree of heat transfer margin.

The four pump loss of coolant flow analyses have been carried out in the power range for coastdown from power levels between 100 percent and the design overpower condition. Conditions used in the analysis are specified in Table 14.1-9. To maximize the power response to the core temperature increase, beginning-of-life core conditions were assumed for all loss of coolant flow events, resulting in the maximum amount of energy transferred to the cladding.

The conditions of Table 14.1-10 were used in thermal calculations to determine the minimum value that the DNB ratio would reach during the loss of coolant flow accidents.

c. Results of Analysis

The results of the analyses show that the reactor can sustain a loss of coolant flow accident without damage to the fuel. The results of the four pump and locked rotor evaluations are presented on Figures 14.1-18 and 14.1-19, respectively. Figure 14.1-17 shows the percent reactor flow as a function of time after loss of power to all reactor coolant pumps. Figure 14.1-18 shows the minimum DNBR that occurs during coastdown from various initial power levels using the minimum tripped rod worth (assuming beginning of cycle reactivity coefficients and a one-percent delta-k/k subcritical margin at hot shutdown conditions). The degree of core protection during the four pump and single pump coastdown is indicated by comparing the minimum DNBR for the coastdown with the criterion value of 1.30 using either the W-3 or B&W-2 CHF correlation. Comparing the minimum DNBR for the coastdown with the criterion value of 1.0 indicates the degree of core protection during the locked rotor accident.

Under normal conditions, the maximum indicated reactor power level from which a loss of coolant flow accident could occur is 102 percent rated power. This power level provides an allowance of plus 2 percent rated power for heat balance error. Even with this error, Figure 14.1-18 and Table 14.1-11 show that an acceptable minimum DNBR exists for the four pump coastdown and locked rotor transients.

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 using conservative values for all resistance and form loss factors. No voids are assumed to exist in the core or reactor outlet piping. Table 14.1-12 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.

Also, as discussed in Appendix 1 of Reference 3, stable decay heat removal is accomplished by natural circulation.

The reactor is protected against reactor coolant pump failure(s) by the Reactor Protection System and the Integrated Control System. The Integrated Control System initiates a power reduction on pump failure to prevent reactor power from exceeding that permissible for the available flow. This action was not credited in the analysis. The reactor is tripped by the reactor protection system if insufficient reactor coolant flow exists for the power level.

Thermal calculations performed to determine the minimum value of DNBR that would be reached during the locked rotor accident are shown on Figure 14.1-19. 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 starts increasing as the transient is terminated by the flux flow trip. The determination of the flux flow trip set point that is necessary to meet the hot channel DNB ratio criteria is discussed in Section 3.2.3.2, Item e. The transient is terminated by the flux flow trip, and, as the figure shows, the DNBR increases in response to the decreasing power, going above 1.3, and continues to rise thereafter. At no time during the transient does the DNBR go below 1.0; therefore, no severe fuel rod or cladding temperature excursions are expected to occur as a result of this transient.

d. Re-analysis for 20% Average Steam Generator Tube Plugging

Note: The information in this section is historical and is preserved for historical purposes.

The loss of coolant flow accidents were re-analyzed to justify a 20% average steam generator tube plugging. The VIPRE-01 Mod 02 computer code and a TMI sub-channel model were used to perform the four-to-zero, four-to-three and locked rotor accidents. The re-analyses included the following initial conditions and assumptions (Reference 87):

- a RCS flow rate of 102% design flow
- a reference design peaking factor of 1.714
- a reference design axial power profile of 1.65 cosine
- a hot channel enthalpy rise factor of 1.0132
- a hot channel flow reduction factor of 0.97
- a flux flow trip delay time of 2.0 seconds
- a power/pump monitor trip delay time of 0.62 seconds
- reactor coolant pump coastdown as per Reference 87
- the use of the BWC correlation
- a DNB limit of 1.18 with the BWC correlation

The results of the re-analysis show that the acceptance criteria for the flow coastdown events with 20% average steam generator tube plugging are met with considerable margin. The re-analysis were reviewed and approved by the NRC (Reference 86).

In addition, as part of each reload core design the locked rotor, four-pump coastdown and one-pump coastdown events are evaluated with respect to DNB using the appropriate CHF correlation for the fuel designs being used. The evaluations account for cycle-specific power peaking factors, and ensure that the applicable acceptance criteria for these events are met.

e. Re-analysis Using Statistical Core Design Methodology

The loss of coolant flow accidents were re-analyzed using the Statistical Core Design methodology with replacement steam generators. The analysis considered a full core of Mark-B-HTP fuel. The LYNXT computer code and 112-channel model were used to perform the four-to-zero, four-to-three and locked rotor accidents. The re-analyses included the following initial conditions and assumptions (Reference 117):

- a RCS flow rate of 107% design flow (equivalent to 104.5% design flow plus the 2.5% flow uncertainty included in the SDL).

- analysis performed at a power level of 2611 MWt that is bounding and applicable for the current rated power level of 2568 MWt.

- a reference design radial-local peaking factor of 1.80.

- a reference design axial power profile of 1.65 cosine.

- hot channel factors included in the SDL.

- a flux/flow trip delay time of 2.0 seconds (includes 217 msec of engineering discretionary margin).

- a power/pump monitor trip delay time of 1.50 seconds (includes 217 msec of engineering discretionary margin).

- reactor coolant pump coastdown as determined using a TMI-specific RELAP5 model.

- the use of the BHTP correlation at the first HTP grid and above and the use of the BWC correlation below the first HTP grid.

  a DNB Thermal Design Limit (TDL) of 1.50 to retain additional margin to BHTP SDL to offset cycle-specific issues.

- The results of the re-analysis show that the TDL acceptance criteria for the flow coastdown events using the SCD methodology are met.

- In addition, as part of each reload core design the locked rotor, four-pump coastdown and one-pump coastdown events are evaluated with respect to DNB using the appropriate CHF correlation for the fuel designs being used. The evaluations account for cycle-specific power peaking factors, and ensure that the applicable acceptance criteria for these events are met.

- Results of the analysis are shown in Table 14.1-11a.

- The loss of coolant flow transients were re-analyzed to address legacy errors identified in IR 1493389, most notably that the initial power levels for the transients did not account for neutron errors. The re-analysis showed that the DNB acceptance criteria were met (Reference 139).

14.1.2.7 Stuck-Out, Stuck-In, Or Dropped Control Rod Accident (with and without ICS Action

a. Identification of Cause

A misaligned control rod is a misalignment of the entire control rod assembly. A stuck-out, stuck-in or dropped control rod are the three types of misalignment that can occur. Misalignment occurs on reactor trip if one rod fails to insert and remains stuck in the fully withdrawn position. This condition requires an evaluation to determine that

sufficient negative reactivity is available for tripping the reactor and maintaining a hot shutdown condition when considering the maximum worth stuck rod. The second type of rod misalignment occurs during withdrawal of control rods if one rod becomes stuck at some position as the other rods continue in motion. This condition will affect the power distribution in the core and could lead to excessive power peaking. The third type of rod misalignment occurs when a control rod drops into the core. The resulting transient causes a rapid reduction in power and temperature due to the rod drop. The decrease in temperature overcompensates for the worth of the control rod, and the moderator temperature coefficient causes temperature and power to rise again. The magnitude of the return to power, in consideration of the asymmetric power distribution, could lead to excessive power peaking.

In the event a control rod fails to insert for a reactor trip, there is sufficient negative reactivity for tripping the reactor and maintaining a hot shutdown condition. This requirement is met as a criteria of the core design and is verified for each new fuel cycle. All B&W reactor cores are required to be capable of maintaining 1 percent delta-k/k shutdown margin at hot shutdown conditions with the maximum worth control rod withdrawn from the core.

A dropped control rod is defined as the deviation of a control rod from the average group position by more than an indicated 5 inches (equivalent to a 9 inch absolute error). This definition then covers the action of a stuck-in control rod during withdrawal of the others and a dropped control rod. A stuck-in rod is less limiting due to the time required to raise the control rods. Raising a control rod completely out of the core from a fully inserted position requires approximately 6 minutes. If a rod becomes stuck, the operator is informed by several alarms and has time to take corrective action. Nevertheless, a dropped rod implies a stuck-in or dropped control rod assembly.

In the event that a control rod cannot be moved, localized power peaking and subcritical margin must be considered.

If a control rod is dropped while operating, a rapid decrease in neutron power would occur, accompanied by a decrease in core average coolant temperature. In addition, the power distribution might be distorted due to the new control rod pattern. In the presence of a distorted power distribution, the return to full power might lead to localized power densities and heat fluxes in excess of design limitations. The Control Rod Drive Control System (CRDCS) and Integrated Control System (ICS) does not attempt to pull control rods to correct the sudden decrease in power because the rod pull inhibit function is in effect when control rods are sensed to be out of alignment.

It has been postulated that secondary thermal aging and embrittlement of the CRDM leadscrew male coupling could lead to a complete failure of the bayonet during operation. If the bayonet fails, the control rod assembly would fall into the core without the leadscrew (position indicator) attached. As with the previous dropped rod scenario, a rapid decrease in core power will occur along with a decrease in core average coolant temperature and decrease in primary system pressure. However, if the bayonet coupling should completely fail, the position indicator on the leadscrew will remain in place. Because the control system believes that all the rods are in their normal position, the resulting decreasing core power will cause the ICS to pull rods to maintain the load demand. In this event, the control rod group is pulled to attempt to maintain the power

necessary to maintain the electric load demand and RCS average temperature. The core will either find a new equilibrium for this configuration, or trip upon reaching a reactor protection system setpoint.

b. Protective Basis

Adequate hot subcritical margin is provided by requiring a subcriticality of 1 percent delta-k/k with the control rod of greatest worth fully withdrawn from the core. The nuclear analysis reported in Section 3.2.2 demonstrates that this criterion can be satisfied.

This criterion has been analyzed in terms of the minimum tripped rod worth available in the loss of coolant flow, startup, rod withdrawal, and steam-line-failure accidents. In all cases, the available rod worth is sufficient to provide margins below any damage threshold.

For protective purposes, a dropped control rod is defined as the deviation of a control rod from its group reference position by more than a maximum of 9 inches. This definition then covers both the action of dropping a rod and sticking a rod while moving a group. The protective action taken is that all rod-out motion is inhibited and the steam generator load demand is run back to 60 percent of rated load at 0.5 percent/sec. The details of these actions are described in Sections 7.2.2 and 7.2.3.

If the dropped rod were due to a failed bayonet, the rod position deviation would not be sensed, and the rod pull inhibit function would not be in effect. In this event, the ICS would attempt to maintain core power sufficient for the electric load demand (ICS Load Following). The details of these actions are described in Sections 7.2.2 and 7.2.3.

The criteria for plant protection during these transients is that the system pressure remain below the 2750 psig, and that the DNBR limits for the dropped rod is greater than the design limit for the applicable approved CHF correlation throughout the core.

The criteria for plant protection during this transient is that the DNB ratio will not be less than 1.3 (W-3 or B&W-2 CHF correlation) and the system pressure will not exceed 2750 psig.

c. Method of Analysis

The transient response to a dropped control rod scenarios have been analyzed using a detailed B&W digital model. This program includes fuel pin, point kinetics, pressurizer, and loop models, including the steam generators.

<u>Dropped Rod with Leadscrew Attached (ICS/CRDCS rod pull inhibit)</u>

The reactor is assumed to be operating at rated power when the control rod is dropped. In order to achieve the most adverse response, the most negative values (end-of-life conditions) of the moderator and Doppler coefficients were used. In addition, the maximum rod worths expected to occur during full power operation were used. These rod worths are the maximum worths that correspond to operation at full power with Xenon and without Xenon. The accident occurring at end-of-life conditions results in

more severe conditions. It was assumed that the steam generator load demand was reduced linearly to 60 percent at 0.5 percent/second. The highest power levels resulting from the rod drop accident are well below the overpower trip condition of 112%. There is no RPS actuation involved for the rod drop accident. Therefore, the ICS runback rate does not significantly affect the peak power levels for this accident scenario. The parameters used in this analysis are shown in Table 14.1-13.

It should be noted that while the analyses presented in this section assumed an MTC of -3.0×10^{-4} delta-k/k/ºF, it has been determined that adequate margin is available with an MTC of -4.0×10^{-4} delta-k/k/ºF (Reference 65).

<u>Dropped Rod without Leadscrew (Bayonet Coupling Failure)</u>

A corollary to the normal dropped rod event is a failed bayonet dropped rod event that does not credit the ICS/CRDCS rod-pull inhibit function. This scenario is predicated by the bayonet coupling completely failing, and the control rod assembly falling into the core while the leadscrew (position indicator) remains in place. As a result, this event allows the ICS/CRDCS to withdraw the remaining control rods, which could potentially increase the severity of the event.

The dropped rod event from full power is characterized by a rapid decrease in core power with a subsequent decrease in fuel temperature and RCS temperature and pressure. The decrease in fuel and RCS temperature causes a positive reactivity insertion due to the negative feedback coefficients, which in turn, tends to increase core power. As core power increases, fuel and RCS temperatures increase which then inserts negative reactivity. The ICS responds to the overall decrease in core power by withdrawing rods. The net effect of the combination of dropped rod worth, reactivity feedback, and ICS controlled rod withdrawal is to increase core power above the initial condition.

Credit for the ICS/CRDCS rod withdrawal inhibit function is not taken in the analysis for this scenario, and therefore, the ICS will continue to insert positive reactivity until the rods are fully withdrawn and a new equilibrium condition is achieved, or the reactor trips on high power (neutron flux) or high RCS pressure.

Because of the ICS action after the bayonet failure dropped rod scenario, the peak reactor power occurs after, or during, the control rods being withdrawn. Several cases were analyzed, but only the transient case initiated from full power reached the high flux (112%) trip setpoint. The full power (102% of 2772 MWt) transient was analyzed using conservative EOC (MTC) conditions.

The parameters used in this analysis are shown in Table 14.1-13.

d. Results of Analysis

<u>Dropped Rod with Leadscrew Attached (ICS/CRDCS rod pull inhibit)</u>

The results of the analysis are presented on Figures 14.1-20 and 14.1-21. Figure 14.1-20 shows the response to a dropped rod with the maximum worth corresponding to the situation with no xenon. The neutron power decreases, causing a rapid decrease in both the core moderator temperature and fuel temperature. These temperature

decreases overcompensate for the worth of the control rod, and the power rises until the reduced steam generator demand begins to increase the inlet temperature and decrease the power.

The thermal power levels out briefly, but soon begins to decrease in response to the decreased steam generator demand. The pressurizer pressure varies above and below its initial value before stabilizing.

Figure 14.1-21 shows the results of the rod drop using the maximum worth with equilibrium xenon. The initial neutron power decrease is slightly less in this case, resulting in the thermal power leveling off at a slightly higher value than in the previous case. The pressurizer pressure peaks at a higher value due to this higher thermal power.

Several cases have been run for rod drops at beginning-of-life conditions. These transients yielded new power levels that are lower than the end-of-life conditions. Therefore, these are not included in this discussion because they represent less severe conditions.

The results are not significantly affected with the higher power level of 2568 MWt. Increasing the initial power level will result in greater peak thermal and core power; however, other B&W plants have been analyzed up to 2772 MWt. Large margins to the acceptance criteria will also be maintained with 20% average steam generator tube plugging (Reference 86).

Implementation of the Statistical Core Design Methodology (Reference 111) resulted in an increase to the maximum design radial-local peaking factor to 1.80. The impact of increased peaking due to dropped and misaligned control rod assemblies on core safety limits was evaluated in Reference 114. The evaluation concluded that margins to centerline fuel melt, cladding strain linear heat rate, and steady-state DNB limits are adequate for dropped rod worths up to 0.20%Δk/k for cores implementing SCD and designed with steady-state radial peaking factors up to 1.59. An updated DNB analysis (Reference 117) was performed for a full core of Mark-B-HTP fuel and higher flow rates available with replacement steam generators. The analysis developed maximum allowable peaking limits that are applied in reload licensing analyses to ensure dropped rod peaking criteria are met.

Dropped Rod without Leadscrew (Bayonet Coupling Failure)

Transient analyses were run at various initial power levels, and operating reactor coolant pump configurations. The case initiated from four-pump operation and 90% core power was bounded by the full power (102% of 2772 MWt) case.

The results from the full power, four RCS pump case is presented in Figures 14.1-24 through 14.1-26. The initial reactor behavior is that of the normal dropped rod event. The core power initially decreases due to the added negative reactivity of the failed control rod assembly. The ICS, not recognizing the asymmetric rod position, starts to pull the rod group to maintain electrical output and/or to maintain RCS average temperature.

The transient terminates on the RPS High Flux trip setpoint being reached (112% FP).

e. Conclusions

The pressure criteria and DNBR limits for the standard dropped rod analyses were both met. Reactor trip limits were not exceeded for this analysis. DNBR was greater than 1.3 (W-3 or B&W-2 CHF correlation) for all regions throughout the core. Adequate protection of any control rod malfunction is provided. Protection for the dropped rod accident is provided through a system which detects a dropped rod and inhibits out-motion of the control rods. The control system is designed to run back the steam generator load demand upon receiving the dropped rod signal from the rod drop detection circuitry. The reactor thermal power will assume a lower value that matches the load demand and will not result in exceeding any design limit.

Based upon the results of a single dropped rod w/o leadscrew (CRA bayonet coupling failure) analyses, it is concluded that the loss in peaking margin due to the maximum worth rod will not adversely impact the TMI-1 COLR limits or the ability to operate the core at rated thermal power.

As part of each reload core design, the maximum dropped rod worth is checked to ensure that the safety analysis results remain bounding.

14.1.2.8 Loss Of Electric Power

14.1.2.8.1 Description of Events Analyzed

This section summarizes the analytical basis for two events: (1) Loss of Electrical Load (LOEL) which is the limiting challenge to RCS pressure due to a secondary plant transient and (2) Loss of all AC power also known as a Station Blackout (SBO).

Loss of electric load (LOEL) events can be initiated by faults within the turbine generator, or more often are caused by circuit breaker operation anywhere in the power grid (for example, an opened switchyard circuit breaker). Thus, the loss of electric load may be caused by separation of the unit from the transmission system.

Section 14.1.2.8.2 describes the original Loss of electrical load analysis where a runback reduces power and the reactor remains critical. The bounding radiological consequences of this event are analyzed assuming initial conditions with 1 percent defective fuel and a 1 gpm primary to secondary tube leak. Due to changes in the RPS high pressure trip and PORV setpoints, a reactor trip would occur in such an event today. The loss of electrical load analysis in section 14.1.2.8.3 describes the limiting effect on RCS pressure. However, the original analysis remains a bounding analysis of the radiological consequences.

Section 14.1.2.8.3 describes analysis of the loss of electrical load event. A turbine trip without an anticipatory RPS actuation from full power was analyzed to simulate a load rejection where PLU actuation would close the turbine Control valves without causing a turbine trip and reactor trip.

Section 14.1.2.8.4 describes the analysis of a loss of all AC power event (Station Blackout). Emergency power systems are described in Section 8.2.3.
Analysis of the radiological consequences of a Station Blackout is described in Section 14.1.2.8.5.

The acceptance criteria for the LOEL and SBO events are:

1) Fuel damage will not occur (thermal power level remains less than 112% of rated power).

2) RCS pressure shall not exceed 110% of the design pressure (2750 psig).

3) Resultant doses must be within 10 CFR 100 limits.

14.1.2.8.2 Loss of Electrical Load with Unit Runback

After the TMI-2 accident, the RPS high pressure and PORV setpoints were changed (Reference 7). As a result of the setpoint changes, the unit cannot accommodate a loss-of-load from rated power without a reactor trip. The previous plant design allowed the unit to accept a loss of load without tripping the turbine or the reactor.

The radiological consequences of an LOEL event remain bounded by this event evaluation where the original design response to a loss of load which results in a runback (no reactor trip) is assumed. Initial unit operation with 1 percent defective fuel and a 1 gpm primary to secondary tube leak were assumed. The amount of steam relieved to the atmosphere and the radiological consequence is shown on Table 14.1-14.

The effect of the condenser in this event versus an event where the reactor trips, and the condenser is therefore unavailable was addressed. The steam relief accompanying a loss-of-load accident would not change the whole body dose. The whole body dose is primarily due to the release of Xe and Kr. Release of these gases is not increased by the steam relief because, even without relief, all of these gases are released to the atmosphere through the condenser vacuum pumps. The rate of release of iodine during relief would increase because the iodine is released in steam vented directly to the atmosphere rather than through the condenser and unit vent. However, the quantity released during this short time is small, as seen in Table 14.1-14; the total integrated thyroid dose from this release is also shown.

The description of this bounding historical event follows:

Under circumstances where the external system deteriorates, as indicated by system frequency deviation, the unit will automatically disconnect from the transmission system. When this occurs, a runback signal causes an automatic power reduction to 15 percent reactor power. To maximize the power response to the core coolant temperature increase, beginning-of-life, core conditions were assumed in the analysis. The unit is assumed to accommodate a loss-of-load condition without a reactor or turbine trip.

Following closure of the turbine governor valves and combined intermediate valves, steam pressure increases to the turbine bypass valve set point and may increase to the steam system safety valve set point. Steam is relieved to the condenser and to the atmosphere. Steam

venting to the atmosphere occurs for about three (3) minutes following loss-of-load from 100 percent initial power until the turbine bypass can handle all excess steam generated.

14.1.2.8.3 Loss of Electrical Load with Reactor Trip

Analysis of the loss of electrical load was performed. The loss of electrical load was conservatively modeled as a turbine trip without an anticipatory RPS actuation. The event analysis was performed using NRC-approved RELAP5/MOD2 systems analysis code (Reference 125). There is a summary of input parameters, analysis assumptions and results in Table 14.1-14a.

In the event analysis, an initial reactor power of 2827.44 MWt was used. The turbine stop valves were modeled to close in 0.05 seconds. Main Feedwater (MFW) flow was linearly decreased to zero over the first 8.9 seconds of the transient. The MSSV were modeled with lift setpoints at 103% and 3 percent valve accumulation. The effect of pressurizer sprays on the transient response was explicitly evaluated and determined to be insignificant. Neither control system response nor reactivity feedback were credited to reduce reactor power. No credit was taken for PORV operation, Emergency Feedwater (EFW) operation, or operation of the turbine bypass and atmospheric dump valves.

Peak RCS pressure was 2572 psia or less than 110% of RCS design pressure (2750 psig). Fuel damage will not occur since the thermal power remains at or below the initial reactor power level and forced flow and sub-cooling are maintained.

The evaluation of loss of electrical load in section 14.1.2.8.2 bounds the dose consequences of this event because the steam mass release through the MSSV and turbine bypass system for the runback scenario is significantly greater than for-the scenario where the reactor trips.

14.1.2.8.4 Loss of All Unit AC Power (Station Blackout)

The loss of all ac power (Station Blackout) transient is an event where all unit power except the station batteries are lost.

10 CFR 50.63 (Loss of All Alternating Current Power) was issued in 1988. This rule provides regulatory requirements for being able to withstand and recover from a Station Blackout of a plant specific duration. UFSAR Section 8.5 specifically addresses how TMI-1 complies with the 10 CFR 50.63 requirements.

This event has been analyzed assuming 20 percent OTSG tube plugging and revised EFW flow rates (reference 94). The event is initiated from rated power (2568 MWt plus 2% instrument error). To maximize the power response to the RCS temperature increase, beginning-of-life core conditions are assumed. The atmospheric dump valves are assumed unavailable. No credit is taken for PORV or Pressurizer spray operation. Key parameters are summarized in Table 14.1-15.

The sequence of events and the evaluation of consequences relative to the Station Blackout event are discussed below:

1) A loss of power results in gravity insertion of the control rods and trip of the turbine valves. Peak power is limited to the initial condition value of 102 percent of rated. This is well below the acceptance criteria limit of 112 percent.

2) Reactor coolant pumps, main feedwater and condensate booster pumps trip on loss of power.

3) After the turbine stop valves trip, excessive temperatures and pressures in the RCS are prevented by natural circulation with steam relief through the main steam line safety valves. Excess steam is relieved until the RCS temperature matches the pressure corresponding to the set point of the lowest set main steam safety valves (MSSVs). Thereafter, the lowest set MSSVs are used to remove decay heat.

4) Peak RCS pressure is limited by operation of the pressurizer safety valves. Maximum hot leg pressure is approximately 2575 psig, which is well below the 2750 psig acceptance criteria (Figures 14.1-23).

5) The RCS flow decays without fuel damage occurring. 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 coolant flow evaluation (Subsection 14.1.2.6).

6) The condensate storage tanks provide cooling water to the steam generators. The minimum condensate storage tank inventory is 150,000 gallons per tank. The inventory from both tanks provides sufficient water for decay heat removal without cooldown (assuming six years of irradiation at 2568 MWt) for a period in excess of 1 day. (Reference 82)

7) The turbine-driven emergency feed pump takes suction from the condensate storage tanks and is driven by steam from either or both steam generators. The emergency feedwater system provides feedwater for decay heat removal and is discussed in Section 10.6. The controls and auxiliary systems for the emergency feed pump operate on DC power from the battery backed DC bus for a minimum of two hours. An air bottle was added to the air supply for the EFW turbine throttle valve (MS-V-6) to assure a heat sink for a minimum of two hours, under loss of AC power operation.

In view of the above sequence, the loss of all unit AC power (Station Blackout) does not result in any fuel damage or excessive pressures on the RCS. There is no resultant radiological hazard to operating personnel or to the public from this accident as only secondary system steam is discharged to the atmosphere.

14.1.2.8.5 Radiological Consequences of SBO Event

The radiological consequences associated with the Station Blackout Event were evaluated at design RCS flow conditions using the design inputs specified in Reference 93. For this event, it is assumed that the plant has been operating with both 1 percent failed fuel and steam generator tube leakage of 1 gpm. The operation has been followed until the steam generator that does not have tube leakage can remove decay heat and the atmospheric dump valve associated with the leaking generator is closed. This results in the following sequence:

1) The atmospheric dump valves are assumed available for this case. It is assumed that the operator further opens the atmospheric dump valves 10 minutes after the loss of power to cooldown the plant at the maximum allowable rate.

2) Cooling down at the maximum available rate requires an additional 45 minutes to reach a temperature below the saturation temperature corresponding to the set point pressure for the steam safety valve having the lowest setting.

3) The steam generator with tube leakage is then completely isolated by closing its atmospheric dump valve, and the other steam generator is used to remove decay heat.

As in the loss of load transient evaluation, the whole body dose does not change due to steam relief. The total integrated thyroid dose is shown in Table 14.1-15.

14.1.2.9 Steam Line Break

a. Identification of Cause

The loss of secondary coolant due to a break of a steam line between the steam generator and the turbine causes a decrease in secondary system steam pressure. The turbine control valves will open in an attempt to maintain power generation. Increased steam flow through the turbine stop valves and the break lowers the average reactor coolant temperature.

Analyses have been performed to determine the effects and consequences of loss of secondary coolant due to a double-ended steam line rupture.

The criteria for unit protection and the release of fission products to the environment are as follows:

1) The core will remain intact for effective core cooling, assuming minimum tripped rod worth with a stuck rod (1 percent delta-k/k).

2) No steam generator tube break or separation from the tube sheet will occur due to a loss of secondary side pressure and the resultant temperature gradients.

3) Doses will be within 10CFR100 limits.

b. Analysis and Results

1) Accident Dynamics

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 largest inventory, at rated power, results in the greatest mass available for cooling.

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. This will cause control rod insertion in order to limit reactor power to 102 percent. A reactor trip occurs due to low reactor coolant pressure (1900 psig) or high neutron flux. Following reactor trip, the turbine trips, and the turbine stop valve and main feedwater control valves close. Low steam line pressure initiates automatic main feedwater isolation, which causes the steam generator associated with the rupture to blow dry. No credit is taken for operator action to terminate main feedwater flow.

Decay heat is removed by the unaffected steam generator with steam flow through the turbine bypass valve.

The analysis for the maximum break size at rated power shows results similar to those discussed above, but represents the worst condition for a steam line rupture accident. A controlled cooldown rate can be established by main feedwater isolation.

The reactor is assumed to be operating at rated power prior to the accident. Other parameters used in the analysis are shown in Table 14.1-16. Bounding end-of-life core conditions were conservatively assumed to maximize positive reactivity feedback due to decreases in the average core and coolant temperatures.

Since this transient is extremely asymmetric, the cooling of the faulted loop will preferentially cool the core half on the faulted loop side. A split core model is used in the analysis with a conservatively bounding mixing ratio between the faulted and unfaulted loops. The reactor trip is modeled to occur due to high neutron flux or low reactor coolant pressure. The high flux trip setpoint includes the nuclear instrumentation (NI) flux error. Transient NI flux errors are induced when a reduction in the coolant temperature entering the reactor vessel occurs. When the reactor vessel downcomer temperature decreases there is an increased attenuation of the neutron leakage flux. As a result, the NI channels will not accurately indicate the true core power level. Reload evaluations are performed each cycle to verify acceptable core response to overcooling conditions when the nuclear overpower trip string errors are effectively increased due to shielding of the out-of-core neutron detectors by the cooler reactor vessel downcomer fluid. The reload evaluations provide assurance that sufficient margin exists for both DNBR and kW/ft conditions even when the actual core power exceeds the design power limits.

The Moody choking model with a contraction coefficient of 0.6 is used to model the break. In addition, liquid carryover through the break is minimized. Main feedwater to the affected steam generator is shown on Figures 14.1-22A and 14.1-22B, and is based on ICS response, which would preferentially feed the affected steam generator, until the main feedwater control valve closes on a low steam generator pressure signal. In addition, the feedwater piping between the isolation valves and the affected steam generator is modeled, so that this additional mass of feedwater is available to cool the RCS. Emergency feedwater would be automatically initiated on a low OTSG level signal in the affected steam

generator. The EFW flow from two MDP's and TDP is assumed to be delivered to the affected OTSG for 10 minutes.

The high pressure injection (HPI) system would be actuated during the cooldown period following a large area steam line break. The system supplies borated water to the RCS to increase the shutdown margin. Boron addition to the reactor coolant, during the controlled cooling to atmospheric pressure, will prevent criticality at lower temperatures. While HPI flow was modeled in the analysis, taking credit for three pumps, no credit was taken for boron addition to the RCS.

These assumptions maximize the RCS cooldown, and are discussed in detail in Reference 95.

The following sequence of events then occurs following a steam line rupture:

a) Following reactor trip, the turbine stop valves close and the main feedwater control valves begin to close. Low steam line pressure (ICS or HSPS) initiates main feedwater isolation to both steam generators and allows the affected steam generator to blow dry.

b) The unaffected steam generator is isolated on the steam side by automatic closure of the turbine stop valves. Steam generator repressurization would cause the turbine bypass valve to open on high steam pressure. (The operator actuated steam line isolation valves are not considered since the steam generator blows dry before these valves can be closed.) Although the turbine bypass valve and atmospheric dump valve are not fully safety grade, they are modeled and operational in the analysis of record. This modeling conservatively provides additional RCS overcooling prior to the MSSVs lifting (Reference 95).

c) Continued Reactor Coolant System cooldown and decay heat removal are achieved by emergency feedwater flow to the unaffected generator with steam relief through the turbine bypass valve. ICS or HSPS provides control of minimum water level within the unaffected steam generator during decay heat removal.

2) Evaluation of Results

After a steam line rupture, the affected steam generator blows down resulting in a reactor trip. After the reactor trip, the unaffected steam generator is isolated on the steam side by automatic closure of the turbine stop valves.

Automatic feedwater isolation on low steam line pressure causes the affected steam generator to blow dry. Since this steam generator is dry, there is no means by which heat can be removed from the reactor coolant through this steam generator.

The unaffected steam generator (that has been isolated on the steam side) has the capability of removing core decay heat by venting steam through the turbine bypass valve or MSSVs.

Figures 14.1-22A and 14.1-22B show the response of the Reactor Coolant System for a double-ended steam line rupture. The affected steam generator blows down until a reactor trip occurs on high flux or low system pressure. After reactor trip, the control rod worth immediately makes the core subcritical. The reactor coolant temperature leaving the affected steam generator continues to decrease until it has blown dry. As the RCS fluid continues to cool, positive reactivity is introduced as a result of the negative moderator temperature coefficient. There is a temporary power increase due to the sub-critical multiplication effect of delayed neutrons, but the total reactivity remains negative and the system remains sub-critical. Since the unaffected steam generator turbine stop valves are closed, and the steam generator with the rupture is dry, the Reactor Coolant System temperature can only be lowered for long term cooling as a result of the steam flow from the isolated steam generator through the turbine bypass valve. Eventually, thermal equilibrium is reestablished; i.e., the heat removal rate (steam flow through the turbine bypass valve) is equal to the heat input (core decay heat). In addition to confirming that the post-trip subcritical multiplication did not result in a return to criticality, a DNBR analysis was performed to demonstrate that no fuel damage occurred for the pre-trip peak power condition (Reference 96).

The original OTSGs were replaced at the end of cycle 17. A specific MSLB core response analysis (Reference 130) was performed with the NRC-approved RELAPS/MOD2 systems analysis code. The post-trip power increase due to sub-critical multiplication was predicted to be less with the replacement OTSGs. In the replacement OTSG case, however, boron addition via HPI was credited 2 minutes into the event (after the initial approach to criticality) to suppress any additional challenges to criticality at lower temperatures. The SLB pre-trip DNBR was also evaluated for the replacement OTSG in Reference 130. It was concluded that the plant response with the replacement OTSG is bounded by the plant response for the original OTSG and the DNBR analysis in Reference 96 remains applicable.

The second reactor protection criterion for the MSLB event states that the steam generator tubes shall not fail as a result of the loss of secondary side pressure or the resultant temperature gradients. The analyses performed for the replacement OTSG unflawed and flawed tube scenarios considered the pressure differential load, axial load, and tensile loads resulting from the MSLB conditions predicted in Reference 130.

The analyses performed for the unflawed tube considered structural loadings based on normal, upset, emergency, and faulted transients, including the main steam line break accident. The results of these analyses (References 133, 135, and 136) show that the tube design meets the requirements of the ASME code for the design loadings identified in the TMI-1 Functional Specification.

Structural analyses that considered various transient loadings were also performed to determine allowable flaw configurations based on the structural integrity requirements outlined in current NEI 97-06 documents. The loads evaluated for the replacement OTSG included normal operating transients at 100

percent power as well as conservative accident (faulted) conditions for the MSLB and small break loss-of-coolant accidents (SBLOCAs) (Reference 134). The MSLB defines the limiting primary-secondary pressure differential load as well as the limiting bending loads. The limiting tube axial loads occur during the SBLOCA and bound those of the MSLB. Since the MSLB and SBLOCA tube loads do not result in tube failure (Reference 134), a flawed steam generator tube will not fail as a result of a steam line break as analyzed for limiting core response.

In addition to the evaluation of the tensile, compressive, and pressure loads on flawed and unflawed tubes, evaluations of the potential for tube failures caused by high cross flow velocities were also performed (Reference 135). These evaluations assess the effects of the high cross flow loading for stress on the steam generator tubes during a steam line break. These evaluations conclude that steam generator tube failures will not occur as a result of cross-flow steam velocities during a steam line break accident with the replacement OTSGs.

The analysis of pump trip in a large steam line break has been performed in Reference 8. The assessment of consequences of an imposed reactor coolant pump trip, upon initiation of the low reactor coolant pressure ESAS, was made there. In the event of a large steam line break (maximum overcooling) the blowdown may induce a steam bubble in the Reactor Coolant System. The maximum overcooling case has been analyzed by B&W, the Reactor Coolant Pump trip increased the amount of void formation in the hot leg "candy cane" of the pressurizer loop, however, natural circulation was not completely blocked. The steam bubble was collapsed and full natural circulation was restored. Core cooling was maintained throughout the transient and no void formation occurred in the core.

A more general concern exists with a large steam line break at EOL conditions and whether or not a return-to-power is experienced following the Reactor Coolant pump trip. If a return-to-criticality is experienced, natural circulation flow may not be sufficient to remove heat and to avoid core damage. B&W has analyzed and concluded that the subcritical return to power condition is bounded by the case where RCPs are not tripped. The tripped pump case results in a reduced overcooling effect.

The analysis of Intermediate Building environment following a postulated main steam break is discussed in Reference 9. The consideration of effects of piping system breaks outside containment is presented in Appendix 14A. The effects of breaks inside containment are discussed in Section d below. The primary system response is essentially the same for breaks inside or outside containment. Breaks inside containment pressurize the Containment Building, whereas breaks outside containment can result in higher dose rates.

c. Environmental Consequences

[Note that the consequences described in this “Environmental Consequences” Section are based on an analysis performed to evaluate Main Steam Line Break (MSLB) accident-induced steam generator tube leakage and the associated licensing basis radiological consequences. The MSLB analysis transient conditions used to evaluate accident-induced steam generator tube leakage and associated licensing basis

radiological consequences are not identical to the transient conditions that were used to evaluate the plant's reactor, Reactor Building, or Intermediate Building responses to a hypothetical MSLB as described in other portions of Section 14.1.2.9. (Reference NRC Safety Evaluation Report for Amendment No. 204, dated October 2, 1997). The environmental consequences from this accident are calculated by assuming that:

1) The unit has been operating with an RCS dose equivalent iodine activity of 0.35 μCi/g for the accident induced iodine spike analysis. This is the limit in Technical Specification 3.1.4. At the onset of the steam line break, the release rate of iodine from the fuel rods to the RCS is assumed to spike by a factor of 500.

2) No credit is taken for partitioning, demineralization, or radioactive decay after the break; thus, all iodine is assumed to be released directly to the atmosphere.

3) The steam line break occurs between the Reactor Building and a turbine stop valve.

4) Reactor coolant leakage into the steam generator continues until the Reactor Coolant System can be cooled down and the leakage terminated.

5) The total primary-to-secondary leakage assumption is integrated over the cooldown time to account for accident induced OTSG tube leakage.

The steam line break is assumed to result in the release of the activity contained in the steam generator inventory, the activity contained in feedwater, and the activity contained in reactor coolant leakage (see Table 14.1-18). Using these assumptions, the total integrated dose to the thyroid and the whole body dose have been calculated and are tabulated in Table 14.1-18. (Reference NRC Safety Evaluation Report for Amendment No. 204, dated October 2, 1997, 6710-97-3444).

An evaluation of the radiological consequences associated with the MSLB for the replacement of OTSGs shows that the dose consequences for the original OTSGs bound those of the replacement OTSGs (Reference 132).

d. Reactor Building Pressure

The effect of a main stream line break inside the Reactor Building has been evaluated using the GOTHIC computer code for the replacement OTSGs (Reference 98). The steam generator mass and energy release was generated with the RELAP/MOD2 computer code (Reference 130) using the assumptions shown in Table 14.1-16a. The peak containment pressure for the MSLB accident is 36 psig, which is less than the containment design pressure of 55 psig.

e. Conclusions

This analysis has shown that the reactor trips and remains subcritical. DNBR analysis demonstrated that no fuel damage occurs for the maximum pre-trip power condition. The maximum temperature differential that occurs in the steam generator does not produce excessive stresses, and steam generator integrity is maintained. The environmental doses are within acceptable limits.

14.1.2.10 Steam Generator Tube Failure

a. Steam Generator Tube Rupture

1) Identification of Accident

The environmental effects associated with steam generator tube leakage and subsequent release to the environment are evaluated in the preceding sections. An evaluation has also been performed for the complete severance of a steam generator tube. 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 directly to the atmosphere while the main steam line safety valves are open. The remainder of the noble gases and iodine would be released through the condenser air removal system. The unaffected steam generator was assumed to have no significant amount of leakage flow of reactor coolant to the secondary side of the steam generator prior to the rupture event.

Acceptance criteria for this event are as follows:

1) Radiological doses must be within the limits of 10CFR100.

2) Additional tube failures and loss of reactor coolant boundary integrity resulting from temperature gradients (thermally induced tube loading) shall not occur.

Flow rates through the failed tube from the RCS to the steam generator secondary were maximized for the double-ended severance of a tube by assuming that the tube offered no hydraulic resistance to the flow.

2) Analysis and Results

In analyzing the consequences of this failure, the following sequence of events is assumed to occur (input parameters are shown in Table 14.1-20 and a summary of results is given in Table 14.1-21):

a) A double-ended rupture of one steam generator tube occurs with unrestricted discharge from each end, at full rated power level with nominal operating conditions.

b) The initial leak rate exceeds the normal makeup to the Reactor Coolant System, and pressurizer level decreases causing the system pressure to decrease. No initial operator action is assumed, and a low Reactor Coolant System pressure trip will occur.

c) Following the reactor trip, the turbine stop valves will close. Steam line pressure will increase, opening the main steam safety valves (MSSVs). In less than two minutes, the turbine bypass system can handle all of the load rejection and the MSSVs close. This behavior is supported by the system transient analysis provided in Reference No. 110. Additional leakage through the MSSVs may occur due to variations in reseating pressure before the operator can further reduce the secondary system pressure to ensure that the MSSVs are fully closed. The doses

associated with these releases are insignificant, and assuming two minutes of full flow steam release in the dose calculations is appropriate.

d) Following reactor trip, the Reactor Coolant System pressure continues to decrease until high pressure injection is actuated. The capacity of the high pressure injection is sufficient to compensate for the leakage and maintains both pressure and volume control of the Reactor Coolant System. Thereafter, the reactor is assumed to be cooled down and depressurized at 100°F per hour, until the Reactor Coolant System temperature decreases below the saturation temperature corresponding to the minimum pressure setpoint on the main steam line safety valves.

e) After the Reactor Coolant System temperature decreases below the saturation temperature of the pressure set point of the main steam line safety valves, the operator isolates the affected steam generator, terminating the release. Cooldown continues with the unaffected steam generator until the temperature is reduced to 250°F. Thereafter, cooldown to ambient conditions is continued using the decay heat removal system.

Operator actions are assumed to initiate and control the cooldown of the primary system by use of the turbine bypass valves, isolate feedwater and steam lines on the affected steam generator, throttle the HPI flow for inventory control on the primary side, and to cool and depressurize the primary system to put the DHRS into operation.

Calculations of the radiological doses for the steam generator tube rupture accident take as inputs the mass released to the SG secondary and assumed to pass to the condenser, the concentrations of radionuclides in the reactor coolant for 1 percent defective fuel, and atmospheric dispersion factors. The calculations account for any direct releases to the atmosphere through the main steam safety valves. Credit for dilution of the concentrations in the reactor coolant has been taken to reflect the diluting effects of the HPI with no additional significant amount of fission product releases from the fuel following the rupture of the steam generator tube.

The first radioactivity release path is through the main steam safety valves. No partitioning of noble gases or iodines is assumed.

The second radioactivity release path during this accident is discharged through the turbine bypass to the condenser and then out the condenser vacuum pump exhaust. A gas-to-liquid partition factor of 100 is assumed for the iodine in the condenser, but noble gases are assumed to be released directly to the atmosphere.

The total dose to the body from all the Xenon and Krypton released is given in Table 14.1-21. The corresponding dose to the thyroid is also tabulated.

The atmospheric dilution is calculated using the dispersion factors developed in Section 2.5.

The above doses were calculated assuming isolation of the affected steam generator. The EOP for OTSG Tube Leakage does not isolate the affected OTSG and directs

cooldown with both steam generators to facilitate better risk management unless the actual or projected dose at the site boundary exceeds 0.5R whole body or 1.5R thyroid.

The original OTSGs were replaced at the end of Cycle 17. A specific steam generator tube rupture analysis was performed to determine if the SGTR analysis of record would remain bounding (Reference 124). The resulting maximum ruptured tube flow rate of 426 gpm is less than the 435 gpm documented in the current analysis of record. The conclusions of the SGTR analysis confirm that the existing analysis remains bounding for the TMI-1 plant with replacement OTSGs.

14.1.2.11 ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

a. Identification of Cause

The ATWS Rule (10CFR50.62, "Requirements for Reduction for Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants") requires improvements in the design and operation of commercial nuclear power facilities to reduce the likelihood of failure to shutdown the reactor following anticipated transients and to mitigate the consequences of an ATWS event.

Specifically, it required the Owners of B&W design plants to provide Diverse Scram Systems (DSS) and Mitigation Systems (ATWS Mitigation System Actuation Circuitry - AMSAC) to reduce the risk from ATWS.

In order to define a basis for the design of these systems, the functional and design requirements to comply with the ATWS rule for TMI-1 were assessed in coordination with the B&W Owners Group (Reference 67).

The limiting ATWS transient was determined by B&W to be the Loss of Main Feedwater (LOMFW). A TMI-1 specific RELAP5 model to predict the peak pressure resulting from a LOMFW ATWS event with the AMSAC functions of turbine trip and EFW actuation on low feedwater pump turbine control oil pressure was used. The model is benchmarked against B&W analysis and sensitivity studies demonstrate the effects of key parameters. Also, analyses were performed to establish the high pressure setpoint of 2500 psig for the DSS. The adequacy of the 2500 psig DSS setpoint was evaluated with 20% tube plugging, bounding kinetics parameters, and EFW flow based on the IST acceptance criteria using the RETRAN code (Reference 66), where it was demonstrated that the ATWS acceptance criteria are met with considerable margin.

TABLE 14.1-1
(Sheet 1 of 3)

ABNORMALITIES AFFECTING CORE AND COOLANT BOUNDARY

Event	Analysis Assumptions	Effect
Uncompensated operating reactivity changes	Automatic control system inoperative or unused.	Reduction in reactor system average temperature. Automatic reactor trip if uncompensated. No equipment damage or radiological hazard.
Startup accident	Uncontrolled single-group and all-group rod withdrawal from subcriticality with the reactor at zero power. Only high flux and high pressure trips were used to terminate the accident.	Power rise terminated by negative Doppler effect, control rod inhibit on short period, high Reactor Coolant System pressure or overpower. No equipment damage or radiological hazard.
Rod withdrawal accident at rated power operation	Uncontrolled single-group and all-group rod withdrawal with the reactor at rated power. Only high flux and high pressure trips were used to terminate the accident	Power rise terminated by overpower trip or high pressure trip. No equipment damage or radiological hazard.
Moderator dilution accident	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 temperature or pressure. During shutdown, a decrease in shutdown margin occurs, but criticality does not occur. No radiological hazard.

TABLE 14.1-1
(Sheet 2 of 3)

ABNORMALITIES AFFECTING CORE AND COOLANT BOUNDARY

Event	Analysis Assumptions	Effect
Loss of coolant flow	Reactor Coolant System flow decreases because of mechanical or electrical failure in one or more reactor coolant pumps. Reactor protection systems are the flux flow and power pump trips.	None. Core protected by reactor low-flow trip or loss-of-power trip. No radiological hazard.
Stuck-out, stuck-in, or dropped control rod	Asymmetric rod monitor operates to inhibit rod out-motion and run back of secondary load.	None. Subcriticality can be achieved if one rod is stuck out. If stuck in or dropped continue operation is permitted if effect on power peaking is not severe. No radiological hazard.
Loss of AC electric power (Station Blackout)	Both a blackout condition and a complete loss of all station AC power (Station Blackout) are considered. 1 percent defective fuel plus a 1-gpm steam generator tube leakage are assumed in radiological analysis.	Reactor trip, PSVs limit RCS peak pressure. See Table 14.1-15 for environmental effects.
Steam line break	Steam line break inside and outside containment are considered. Accident induced reactor coolant leakage into the steam generator is accounted for.	Reactor trips following a large rupture. See Table 14.1-18 for environmental effects.

TABLE 14.1-1
(Sheet 3 of 3)

ABNORMALITIES AFFECTING CORE AND COOLANT BOUNDARY

Event	Analysis Assumptions	Effect
Steam generator tube failures	Reactor coolant leakage into the steam generator continues for 34 minutes following reactor operation with 1 percent defective fuel. Isolation of the affected generator will be achieved when the Reactor Coolant System is cooled down and depressurized below the pressure set point on the main steam safety valves.	Reactor automatically trips if leakage exceeds normal makeup capacity to Reactor Coolant System. See Table 14.1-21 for environmental effects.

TABLE 14.1-2
(Sheet 1 of 1)

UNCOMPENSATED REACTIVITY DISTURBANCES

Cause	Maximum Reactivity Rate, (delta-k/k)/min	Rate of Average Temperature Change (Uncorrected),?F/min
Fuel depletion	-2.9×10^{-7}	-0.0006
Xenon buildup	-2.2×10^{-5}	-0.060
Xenon burnout	$+1.1 \times 10^{-3}$	+3.1

Note: The analysis is based on a Doppler coefficient of -1.17×10^{-5} (delta-k/k)/ºF and a moderator coefficient of 0.5×10^{-4} (delta-k/k)/ºF, Initial reactor power = 2535 MWt, Initial Average Reactor Coolant temperature = 579ºF.

TABLE 14.1-3
(Sheet 1 of 1)

STARTUP ACCIDENT PARAMETERS**
(Initial Cycle)

Maximum rod speed in/min.	30
Maximum number of CRAs	61
Maximum rod worth, all rods delta-k/k	10%***
Maximum reactivity addition rate, all 61 rods at maximum speed (delta-k/k)/sec	7.25×10^{-4}***
Nominal rod worth of single group when reactor is critical delta-k/k	1.5%
Nominal reactivity addition rate for single rod group (delta-k/k)/sec	1.09×10^{-4}
Doppler coefficient at rated power (delta-k/k)/°F	-1.17×10^{-5}
Moderator coefficient at rated power	Zero
Peak thermal power (original design overpower) permitted for assumed fuel damage*	114%
Trip Parameters	
Delay for high pressure trip sec	0.5
Delay for high-flux trip sec	0.3
Travel time to 2/3 insertion sec	1.4

* Original licensed power rating = 2535 MWt

** Refer to the re-analysis for 20% average steam generator tube plugging for the current values.

*** As per Reference 85, a maximum all rods worth of 12.9% dk/k was also analyzed. This corresponds to a maximum reactivity addition rate of 9.27×10^{-4} dk/k/sec.

TABLE 14.1-4
(Sheet 1 of 1)

SUMMARY OF STARTUP ACCIDENT ANALYSIS

1. Peak thermal power for withdrawal rates less than that corresponding to the withdrawal of all rods is always less than rated power.

2. The nominal single-group rod withdrawal causes a peak pressurizer pressure sufficiently high to actuate the pressurizer safety valves. These valves have sufficient capacity to handle the resultant coolant expansion.

TABLE 14.1-5
(Sheet 1 of 1)

ROD WITHDRAWAL ACCIDENT PARAMETERS
(Initial Cycle)

High pressure trip setpoint psig	2385
High flux trip setpoint (original design overpower value)*	114%
Trip delay time (high pressure trip) sec	0.5
Trip delay time (high flux trip) sec	0.3
CRA insertion time (2/3 insertion) sec	1.4
Doppler coefficient at full power (BOL) (delta-k/k)/ °F	-1.17×10^{-5}
Moderator coefficient at full power (BOL)	Zero
Maximum control rod speed in/min	30
Maximum rod worth, all rods (delta-k/k)	10.0%**
Nominal single control rod worth (delta-k/k)	1.5%
One rod group nominal reactivity addition rate(delta-k/k)/sec	1.09×10^{-4}
Maximum reactivity addition rate of all 61 control rods at maximum speed (delta-k/k)/sec	7.25×10^{-4}**
Minimum tripped rod worth used (delta-k/k)	2.36%
Initial RCS pressure psia	2200

* Original licensed power rating = 2535 MWt. The current high flux trip setpoint is 112% of 2568 MWt.

** As per Reference 85, a maximum all rods worth of 12.9% dk/k was also analyzed. This corresponds to a maximum reactivity addition rate of 9.27×10^{-4} dk/k/sec.

TABLE 14.1-6
(Sheet 1 of 1)

SUMMARY OF ROD WITHDRAWAL ACCIDENT ANALYSIS
(Initial Cycle)

Reactivity Rate, (delta-k/k)/sec	Peak Thermal* Power, % of Rated Power	Peak System Pressure Increase, psi
1.09×10^{-4}	109	118
7.25×10^{-4}	104	14

* The criterion for this transient is to stay at or below the maximum design overpower of 114 percent of reference core design power level (2535 MWt)

TABLE 14.1-7
(Sheet 1 of 2)

MODERATOR DILUTION ACCIDENT PARAMETERS
(Initial Cycle)

Flow Rates Considered

Dilution Flow Rate Condition	
Normal dilution gpm	70
Normal dilution with low Reactor Coolant System pressure gpm	100
Maximum considered gpm	500
Initial boron concentration in Reactor Coolant System ppm	1200
Boron reactivity worth ppm/1% (delta-k/k)	75
Moderator coefficient (delta-k/k)/ °F	0.5×10^{-4}
Doppler coefficient (delta-k/k)/ °F	-1.17×10^{-5}
Initial RCS Temperature, °F	554
Makeup Tank Volume, Ft^3	600

TABLE 14.1-7
(Sheet 2 of 2)

MODERATOR DILUTION ACCIDENT PARAMETERS
(Initial Cycle)

Flow Rates Considered

Reactor Protection System

High Pressure Trip Setpoint, psia	2400
Delay Time for High Pressure Trip, seconds	0.5

TABLE 14.1-8
(Sheet 1 of 1)

SUMMARY OF MODERATOR DILUTION ACCIDENT ANALYSIS
(Initial Cycle)

A. Dilution at Power

Condition	Dilution Water Flow, gpm	Reactivity Rate, (delta-k/k)/sec	Average Reactor Coolant System Temp. hange, °F/sec
Normal	70	$+2.2 \times 10^{-6}$	0.006
Normal (Maximum)	100	$+3.2 \times 10^{-6}$	0.009
Maximum considered	500	$+1.6 \times 10^{-5}$	0.044

1. Dilution to Trip

Dilution Water Flow, gpm	Peak Thermal Power, Percent of Rated*	Peak Pressure, psia	Time to Trip, sec
70	104.1	2420	111
500	107.3	2435	45

2. Dilution at Shutdown delta-k/k

Initial subcriticality margin delta-k/k	9.5%
Final subcriticality margin	4.5%

* Original licensed power rating = 2535 MWt

TABLE 14.1-9
(Sheet 1 of 2)

LOSS OF COOLANT FLOW ACCIDENT PARAMETERS
(Initial Cycle)

Initial Power and DNBR Conditions

Power level used in the analysis, % of rated power*	108
Nominal RCS flow rate, gpm	352,000
Initial core exit pressure, psia	2135
Steady-State DNBR based on 108% power	
At full flow	1.55
At 95% flow corresponding to an open internals vent valve**	1.40
Maximum indicated power, %	100
Maximum real power, %	102

* Original licensed rated power level = 2535 MWt
** Flow penalty for an open internals vent valve is no longer required

TABLE 14.1-9
(Sheet 2 of 2)

LOSS OF COOLANT FLOW ACCIDENT PARAMETERS
(Initial Cycle)

System Characteristics

The initial core inlet temperature for each power level is assumed to be 2°F greater than nominal in order to account for instrument error.

The initial system pressure is assumed to be 65 psi lower than nominal in order to account for instrument error.

The Power/Pump Monitors trip delay time is 620 msec.

The Flux Flow trip delay time is 650 msec.

The pump inertia is 70,000 lb-ft^2.

Flux Flow Trip Setpoint, % FP/ %Flow	1.1
Moderator Coefficient at Beginning of Life, (delta-k/k)/ °F	0.0
Doppler Coefficient at Beginning of Life, (delta-k/k)/ ºF	-1.27×10^{-5}

Note: The information on this page is historical and is preserved for historical purposes. See Section 14.1.2.6.e for current analysis parameters.

TABLE 14.1-10
(Sheet 1 of 1)

LOCKED ROTOR ACCIDENT PARAMETERS
(Initial Cycle)

Initial power, % rated power*	102
Initial flow, % thermal design flow	100**
Flux flow trip delay time, msec	650

System Characteristics

The initial core exit pressure and inlet temperature are 65 psi less than and 2ºF greater than nominal values, respectively, in order to account for instrument error and the limits of the ICS control band.

Maximum design conditions were assumed for the thermal conditions.

The flow decreases from 100 percent of its steady state value to 75 percent of the steady state value in approximately two seconds.

Film boiling is assumed to occur at DNBR = 1.0.

The reactor is tripped by the flux flow monitor.

* Original licensed rated power level = 2535 MWt
** The thermal design flow is 139.8×10^6 lbm/hr, or 106.5% of pump rated flow.

TABLE 14.1-11
(Sheet 1 of 1)

SUMMARY OF LOSS OF COOLANT FLOW ACCIDENT ANALYSIS
(Initial Cycle)

Minimum DNBR during coastdown for loss of all four pumps and the locked rotor case:

Situation	Criterion	Result	Correlation Used
Coastdown from 100% rated power* and 100% flow	1.3	1.82	w-3
Coastdown from 102% rated power* and 100% flow	1.3	1.75	w-3
Coastdown from 100% rated power* and 95% flow	1.3	1.68	w-3
Locked rotor at 102% rated power* and 100% flow	1.0	1.15	w-3
Coastdown from 102% rated power* and 106.5% flow using 930-msec trip time delay	1.3	1.50	(B&W-2)

* Original licensed rated power level = 2535 MWt

Note: The information on this page is historical and is preserved for historical purposes. See Section 14.1.2.6.e for current analysis description.

TABLE 14.1-11a
(Sheet 1 of 1)

SUMMARY OF LOSS OF COOLANT FLOW ACCIDENT ANALYSIS
(Current Cycle)

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

Situation	Criterion	Result	Correlation Used	Trip
4 pump coastdown from 104% power [a, b] and 100% flow [d]	1.50	1.68	BHTP	Pump power
1 pump coastdown from 106% power [a, c] and 100% flow [d]	1.50	1.77	BHTP	Flux/flow
Locked rotor at 106% power [a, c] and 100% flow [d]	1.50	1.62	BHTP	Flux/flow

[a] Analysis performed at a bounding power level of 2611 MWt

[b] Initial power includes a 2% power control deadband uncertainty and a 2% steady-state neutron power uncertainty. The transient neutron power uncertainty does not apply to the pump power monitor trip function. A 2% heat balance error is included in the statistical design limit.

[c] Initial power includes a 2% power control deadband uncertainty, a 2% steady-state neutron power uncertainty, and 2% transient neutron power uncertainty. A 2% heat balance error is included in the statistical design limit.

[d] 100% flow is defined as 107% of Cycle 1 design flow (1.07 * 352,000 gpm)

TABLE 14.1-12
(Sheet 1 of 1)

NATURAL CIRCULATION CAPABILITY
(Initial Cycle)

Time After Loss of Power (sec)	Decay Heat Core Power* (%)	Natural Circulation Core Flow Available (% Design Flow)	Flow Required for Decay Heat Removal, (% Design Flow)
3.6×10^{1}	5	4.1	2.3
2.2×10^{2}	3	3.3	1.2
1.2×10^{4}	1	1.8	0.36
1.3×10^{5}	0.5	1.2	0.20

* Original licensed rated power level = 2535 MWt

TABLE 14.1-13
(Sheet 1 of 1)

DROPPED ROD ACCIDENT PARAMETERS

Parameter	Dropped Road with Leadscrew	Dropped Rod without Leadscrew (bayonet failure)
Initial Reactor Power (% full power)	100	102% of 2772 MWt
Moderator coefficient (Δk/k)/ ºF	-3.0×10^{-4}	-4.0×10^{-4}
Doppler coefficient (Δk/k)/ ºF	-1.3×10^{-5}	-2.0×10^{-5}
Integral rod worth at full power, (%Δk/k)	n/a	0.54
Dropped control rod worth at full power		
with xenon, (%Δk/k)	0.36	0.22
without xenon, (%Δk/k)	0.46	(not analyzed)
Control rod drop time	2/3 insertion in 1.4 sec	
Maximum Reactivity Insertion Rate		
with xenon, ((Δk/k)/s)	-1.8×10^{-3}	-6.2×10^{-3} (NOTE 1)
without xenon, ((Δk/k)/s)	-2.3×10^{-3}	(not analyzed)
High flux trip setpoint, % full power	112	
High pressure trip setpoint, (psia) (NOTE 2)	2430	
Delay time for high flux trip, (sec)	0.4	
Delay time for high pressure trip, (sec)	0.5	
Original licensed rated power level = 2535 MWt		

TABLE 14.1-13 NOTES:

1. Averaged insertion rate of maximum worth rod - Integrated rod worth of 0.621 %Δk/k inserted 50.04 inches at a run speed of 30 inches per minute. This rod worth conservatively represents 115% of 0.54 %Δk/k.
2. Although a high pressure trip setpoint was modeled, all full power analyses terminated on high reactor power trip.

TABLE 14.1-14
(Sheet 1 of 1)

LOSS OF ELECTRICAL LOAD TRANSIENT PARAMETERS AND RESULTS

Initial RCS flow, gpm	352,000 (100% flow)
Initial Core Power, MWt	2568
Moderator Coefficient at BOL (delta-k/k)/^{o}F	+0.00
Doppler Coefficient at BOL	-1.17×10^{-5}
Steam relieved to the atmosphere lb_m	205,000
Atmospheric dispersion coefficient at exclusion distance sec/m^3 (Reference 91)	8.0×10^{-4}
Iodine released during relief (in Iodine-131 dose equivalent curies)	6.26×10^{-2}
Total integrated thyroid dose at exclusion distance rem (Reference 91)	2.6×10^{-2}

TABLE 14.1-14a
(Sheet 1 of 1)

LOSS OF ELECTRICAL LOAD PARAMETERS AND RESULTS

Initial Core Power (MWt) 102% of 2772 MWt	2827.44
Initial RCS Flow (gpm) minimum DNB flow	367,840
SG Tube Plugging (%)	0
Number of MSSVs	18
MSSV accumulation (%)	+3
MSSV drift (%)	+3
Number of PSVs	2
PSV capacity per valve, (lbm/hr)	297,846
PSV open setpoint (psig)	2500
PSV setpoint drift	3%
RCS High Pressure Reactor Trip (psia)	2400
Reactor Trip Delay Time (sec)	0.6
Moderator Coefficient at BOL (delta-k/k)/F	+0.0
Doppler Coefficient at BOL (delta-k/k)/F	-1.17×10^{-5}
Delayed neutron fraction at BOL	0.007
Maximum RCS pressure (psia)	2571.56

TABLE 14.1-15
(Sheet 1 of 1)

LOSS OF ALL AC POWER EVENT (STATION BLACKOUT)
RADIOLOGICAL ANALYSIS PARAMETERS AND RESULTS

Initial Core Power (MWt)	2620
Initial RCS Flow (Mlb/hr)	133.8 (102% of design)
OTSG Tube Plugging	20%
OTSG Tube Plugging Assymetry	25%/15%
OTSG Inventory (lbm/SG)	39,000
Terminate MFW flow (sec)	2.0
Total EFW Flow @ 1065 psia (gpm)	350
EFW Temperature (degF)	135
Moderator Coefficient at BOL (delta-k/k)/°F	+0.00
Doppler Coefficient at BOL	-1.17×10^{-5}
Steam relieved to atmosphere, lbm	203,900
Atmospheric dispersion coefficient at exclusion distance, sec/m^3 (Reference 91)	8.0×10^{-4}
Steam generator isolation time, min	55
Total integrated thyroid dose at exclusion distance, Rem (Reference 91)	4.9×10^{-1}

TABLE 14.1-15a
(Sheet 1 of 1)

Deleted

TABLE 14.1-16
(Sheet 1 of 1)

STEAM LINE BREAK MODELING ASSUMPTIONS

Initial "replacement" OTSG inventory (lbm)	63,500
Maximum pipe size assumed in analysis (in.) OD	24
Initial core power (MWt) 102% of 2568 MWt	2619.4
High flux trip setpoint, (% rated power)	105.1
High flux trip delay time (sec)	0.4
SG Tube Plugging (%)	0
Moderator Coefficient at EOL (delta-k/k)/F	-4.0×10^{-4}
Doppler Coefficient at EOL (delta-k/k)/F	-2.0×10^{-5}
Control rod movement time to 2/3 insertion during trip (sec)	1.4
Turbine stop valve closure time (sec)	0.5
Feedwater control valve closure time (sec)	10.0
Minimum tripped rod worth (%delta k/k)	3.86

TABLE 14.1-17

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TABLE 14.1-18
(Sheet 1 of 2)

Radiological Consequences of Main Steam Line Break Accident
In conjunction with
Accident-Induced Steam Generator Tube Leak
(rem)

	Thyroid	Whole Body
EAB	28	<1
LPZ	5.3	<1
Control Room	29	<1

Note: Assumptions used to calculate these consequences are listed on Table 14.1-18, Sheet 2 of 2. From the Standard Review Plan, the dose acceptance criteria for the EAB and LPZ are 2.5 rem whole body and 30 rem thyroid for the main steam line break with an accident initiated spike. The criteria for the control room are 5 rem whole body and 30 rem thyroid.

TABLE 14.1-18
(Sheet 2 of 2)

Radiological Consequences of Main Steam Line Break Accident In Conjunction with Accident-Induced Steam Generator Tube Leak
(rem)

Assumptions Used to Calculate Radiological Consequences Resulting from Main Steam Line Accident In conjunction with Accident-Induced Steam Generator Tube Leak

Parameter	Value
Reactor Power:	2620 MWt
Primary Coolant Iodine Concentration Limit:	0.35 μCi/gm DE I-131
Iodine Partition Factor in Steam Generator:	1.0
Primary Coolant Volume:	1.124E + 4 ft^3
Letdown Flow Rate:	45 gpm
Integrated Primary Coolant Released	
0 to 2 hours	3228 gallons
0 to 10 hours	9960 gallons
Average Leakage Rates (constant rate used)	
0 to 2 hours	26.9 gpm
2 to 10 hours	14.0 gpm
χ/Q Values (sec/m^3)[1]	
0 to 2 hour EAB	8.3E-4[2]
0 to 8 hour LPZ	1.1E-4
8 to 24 hours LPZ	6.7E-5
1 to 4 days LPZ	2.5E-5
4 to 30 days LPZ	6.0E-6
Control Room Pressure Boundary Volume:	1.26E + 5 ft
Control Room Air Recirculation Rate:	3.9E + 4 cfm
Control Room Unfiltered Inleakage Rate:	2.628E + 3 cfm
Emergency Filtration Filter Efficiencies:	90%
Control Room Makeup Air Inlet	3.0E + 3 cfm
Control Room χ/Q Values (sec/m^3)	
0 to 8 hours	7.45E-3
8 to 24 hours	4.15E-3
1 to 4 days	2.73E-3
4 to 30 days	1.44E-3

(1) NUREG-0107, Safety Evaluation Report for TMI Unit No. 2 (September 1976)
(2) NRC Fuel Handling Accident Safety Evaluation (December 1979)

TABLE 14.1-19

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TABLE 14.1-20
(Sheet 1 of 1)

STEAM GENERATOR TUBE FAILURE PARAMETERS

Initial Reactor Power, MWt	2568
Initial leak rate gpm	435
Normal makeup rate gpm	70
Low Pressure Trip setpoint, psig	1800
High pressure injection set point psig	1500
Assumed defective fuel	1%

TABLE 14.1-21
(Sheet 1 of 1)

SUMMARY OF STEAM GENERATOR TUBE FAILURE ANALYSIS

Low pressure trip occurs at min,	8
Time to isolation of affected OTSG, min	34
Reactor coolant leakage during depressurization, ft^3	1977
Activity Released to Atmosphere	
Noble gases, Curies	33,000
Iodine I-131 dose equivalent, Curies	21
Total Integrated Dose at Exclusion Distance	
Thyroid, REM	9
Whole body, REM	0.4

TABLE 14.1-22
(SHEET 1 of 1)

COLD WATER ACCIDENT PARAMETERS
(Initial Cycle)

Reactor Coolant System	
Initial Flow, gpm	176,000 (50% of 352,000)
Initial Core Power, MWt	1267.5 (50% of 2535)
Initial Pressure, psia	2200
Initial Temperature, ºF	554
Initial Pressurizer Level, inches	217
Moderator Coefficient at End-of-Life, (delta-k/k)/ºF	-3.00×10^{-4}
Doppler Coefficient at End-of-Life, (delta-k/k)/ºF	-1.20×10^{-5}
Reactor Protection System	
High Flux Trip Setpoint	112% of 2535 MWt
High Pressure Trip Setpoint, psia	2430
Delay Time for High Flux Trip, sec	0.3
Delay Time for High Pressure Trip, sec	0.5

TABLE 14.1-23
(SHEET 1 of 1)

DELETED

TABLE 14.1-24
(SHEET 1 of 1)

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14.2 STANDBY SAFEGUARDS ANALYSIS

14.2.1 SITUATIONS ANALYZED AND CAUSES

In this Section, accidents are analyzed in which one or more of the protective barriers are not effective and standby safeguards are required. All accidents evaluated are based on the rated core power level of greater than or equal to 2568 MWt. Table 14.2-1 summarizes the potential accidents studied.

14.2.2 ACCIDENT ANALYSES

14.2.2.1 Fuel Handling Accident

a. Identification of Accident

Spent fuel assemblies are handled entirely under water. Before refueling, the reactor coolant and the fuel transfer canal water above the reactor are increased in boron concentration so that, with all control rods removed, the k_{eff} of a core is not greater than 0.99. In the spent fuel storage pool, the fuel assemblies are stored under water in storage racks having an eversafe geometric array. 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.

b. Analysis and Results

1) Radiological Considerations in the Fuel Handling Building

The assumptions made for this analysis are shown in Table 14.2-3. The reactor is assumed to have been shut down for 72 hours, which is the minimum time allowed by Technical Specifications 3.8.10 for Reactor Coolant System cooldown, reactor head removal, and removal of the first fuel assembly. The activity released is based on 1.02 times the current licensed thermal power level of 2568 MWt for the entire core. A radial peaking factor of 1.7 is applied to an average fuel assembly. It is 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 entire content of the gap activities in the 56 rods damaged is assumed to be released to the spent fuel pool immediately after the accident. The radiation source term data and their bases are given in Table 14.2-2.

The gases released from the fuel assembly pass upward through the spent fuel storage pool water prior to reaching the Fuel Handling Building atmosphere. Normally, the spent fuel assembly rests within the spent fuel storage rack, where it is covered with a minimum of 23 feet of water. Although there is experimental evidence that a portion of the noble gases will remain in the water, no retention of noble gases is assumed. Per Regulatory Guide 1.183, 99.5 percent (or a DCF of 200) of the iodine released from the fuel assembly is assumed to remain in the water.

Atmospheric dilution is calculated using the 2 hour dispersion factor at the exclusion boundary of 8.0×10^{-4} sec/m^3, which is discussed in References 4, 89, and 90. The total effective dose equivalent (TEDE) at the exclusion area, the low population zone, and the Control Room can be seen in Table 14.2-3.

2) Fuel Handling Accident Occurring in the Reactor Building

An evaluation of the postulated Fuel Handling Accident Inside Containment (FHAIC) at TMI-1 has been performed using the Alternative Source Term (AST) methodology according to NRC Regulatory Guide 1.183. No credit was taken for Reactor Building isolation.

The analysis of the postulated fuel handling accident in the Reactor Building is based on the following:

a) The accident is assumed to happen after the reactor has been shut down for 72 hours. This is based on Technical Specification 3.8.10, which requires at least 72 hours between reactor shutdown and the removal of irradiated fuel. Radioactive decay of the core fission product inventory during this interval is taken into account.

b) All of the rods in one assembly (208) are assumed to rupture as a result of the accident.

c) The assembly damaged is assumed to be the highest powered assembly in the core region to be discharged. Table 14.2-4a gives the core inventory based on the conservative assumption that the entire core is irradiated at 102% of 2568 MWt and fuel burnup of 700 EFPDs. The dose consequences are based on a limiting steady-state radial peaking factor of 1.70.

d) All of the activity in the clad gap in the damaged rods is released to the refueling water. This activity is based on Regulatory Guide 1.183, Table 3, assumptions, i.e., 8% of the I-131, 5% other halogens, 10% Kr-85, 5% other noble gases, and 12% alkali metals in the rods at the time of the accident. However, since there have previously been fuel assemblies evaluated to exceed the Regulatory Guide 1.183 footnote 11 value of 6.3 kW/ft peak rod average power for burnups exceeding 54 GWD/MTU, the gap fractions cited above are doubled for conservatism. Particulate activity (i.e., alkali metals) is retained in the water.

e) The inventory of fission products in the reactor core and available for gap release from damaged fuel is based on the maximum power level of 2,619 MWt corresponding to current fuel enrichment and burnup, which is 1.02 times the current licensed rated thermal power of 2,568 MWt.

f) The refueling water decontamination factors are based on Regulatory Guide 1.183 assumptions, giving an effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is

retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. The depth of water above damaged fuel for a fuel handling accident in the containment is greater than 23 feet.

g) The radioactive material that escapes from the refueling water is released from the building to the environment without credit for passing through the charcoal filters in the purge exhaust. No retention by the refueling water of the noble gases is assumed.

The purge exhaust filter system at TMI-1 does not permit bypassing the filters during containment purging. That is, as presently installed, it is not possible to purge the containment without purging through both the HEPA and charcoal filters.

h) The iodine and noble gas activities from the damaged fuel assembly were assumed to be released instantaneously to the Reactor Building. The activity is then released to the outside atmosphere as a ground level release linearly over a two-hour period.

i) No credit is taken for holdup in the Reactor Building. However, Technical Specifications (T.S.) 3.8.6, 3.8.7, and 3.8.9, which provide assurance of automatic reactor Building isolation in the event of a fuel handling accident in the Reactor Building are as follows:

T.S. 3.8.6 During the handling of irradiated fuel in the Reactor Building, at least one door in each of the personnel and emergency air locks shall be capable of being closed.*
The equipment hatch cover shall be in place with a minimum of four bolts securing the cover to the sealing surfaces.

NOTE

The equipment hatch may be open if all the following conditions are met:

1. The Reactor Building Equipment Hatch Missile Shield Barrier is capable of being closed within 45 minutes.
2. A designated crew is available to close the Reactor Building Equipment Hatch Missile Shield Barrier, and
3. The Reactor Building Purge Exhaust System is in service.

* Administrative controls shall ensure that appropriate personnel are aware that air lock doors and/or penetrations are open, a specific individual(s) is designated and available to close the air lock doors and other penetrations as part of a required evacuation of containment. Any obstruction(s) (e.g., cable and hoses) that could prevent closure of an air lock door or other penetration will be capable of being quickly removed.

T.S. 3.8.7 During the handling of irradiated fuel in the Reactor Building, each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either: 1. Closed by an isolation valve, blind flange or manual, or 2. Be capable of being closed by an operable automatic containment purge and exhaust isolation valve, or 3. Be capable of being manually closed within 45 minutes.

T.S. 3.8.9 The Reactor Building purge system, including the radiation monitors which initiate purge isolation, shall be tested and verified to be operable no more than 1 week prior to refueling operations.

In addition, radiation monitors RM-G6 and RM-G7 monitor and alarm any excessive radiation in the vicinity of the refueling water surface. Also, Technical Specification 3.8.5, which requires that direct communications between the Control Room and the refueling personnel in the Reactor Building, is in effect whenever changes in core geometry are taking place. Therefore, by radiation monitoring and Technical Specification implementation assurance is provided that in the event of a fuel handling accident in the Reactor Building, the Control Room operators would have sufficient information to initiate isolation of the Reactor Building.

j) Atmospheric diffusion is calculated using a 0-2 hr dispersion factor at the exclusion boundary of 8.0×10^{-4} sec/m^3. This value is discussed in References 89 and 90.

The analysis of the radiological consequences of a FHAIC has been performed without taking credit for Reactor Building Purge Exhaust System (RBPES). The two-hour dose results at the exclusion boundary are given in Table 14.2-5.

OTSG tube plugging will have no impact on this accident.

14.2.2.2 <u>Rod Ejection Accident</u>

a. Identification of Accident

Reactivity excursions initiated by uncontrolled rod withdrawal (Section 14.1) were shown to be safely terminated without damage to the reactor core or Reactor Coolant System integrity. 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 assembly 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.

Since control rod assemblies are used to control load variations only 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 rod ejection accident is

inherently limited because the amount of reactivity available in the form of control rod worth is relatively small. The criterion for reactor protection in this assumed accident is that the reactor will be operated in such a manner that a control rod ejection accident will not further damage the Reactor Coolant System.

1) Accident Bases

Using an analytical method based on diffusion theory (Section 3.2.2.2.1) the worth of the most reactive control rod assembly in each rod group was determined for different control rod configurations.

The maximum rod worths and other important parameters used in the study are shown in Table 14.2-6.

The tripped rod worth used corresponds to the minimum worth available with the maximum worth rod stuck out at BOL and EOL.

The severity of the rod ejection accident is dependent upon the worth of the ejected rod and the reactor power level. The control rod group of greatest worth is the first of the entire rod pattern to be withdrawn. The maximum worth of a rod in this group can be as high as 2.5 percent delta-k/k but would only have this worth when the reactor was subcritical. The details of the control rod worth calculations and the methods of selecting the number of control rods in each group are presented in Section 3.2.2 and Item b. of Section 7.2.2.1.

When the reactor is subcritical, the boron concentration is maintained at a level which insures that the reactor is at least 1 percent subcritical with the control rod of greatest worth fully withdrawn from the core. Thus, a rod ejection will not cause a nuclear excursion when the reactor is subcritical and all the other rods are in the core.

As criticality is approached, the worth of the remaining rods decreases so that at criticality the maximum reactivity addition from a rod ejection would be 0.56 percent delta-k/k.

The rod worth continues to decrease as rated power is attained. Before equilibrium xenon is established, the total pattern worth remaining in the core at rated power is 2.8 percent delta-k/k, and the greatest single control rod worth is 0.46 percent delta-k/k. At equilibrium xenon the pattern worth is 1.8 percent delta-k/k and the maximum rod worth is 0.36 percent delta-k/k. A detailed analysis has been performed at worths up to 0.7 percent delta-k/k, however, to show the large margin that exists between the actual rod worths and those worths needed to approach any failure thresholds.

A rod must be fully inserted in the core to have the foregoing reactivity worth values. 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 control rod travel time to the top of the active region of the core is calculated to be 0.176 seconds. Since most of the reactivity

is added during the first 75 percent of this travel, only this distance is used in the analysis, resulting in an ejection time of 0.15 seconds for the analysis.

2) Fuel Rod Damage

The consequences of a rod ejection accident are largely dependent upon 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 which poses no threat to the integrity of the Reactor Coolant System. However, if the magnitude of the nuclear excursion is such that the fuel rod cladding does not remain intact, then fuel and clad may be dispersed into the coolant to such an extent as to cause a significant increase in the heat transfer rate.

Power excursions caused by reactivity disturbances of the order of magnitude occurring in rod 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 from 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.

The second failure mode occurs when significant fuel melting causes a rapid increase in internal fuel rod pressure* which, combined with clad loss of strength at higher temperatures, causes the fuel rod clad to rupture. 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 occurs when, as a result of a very large and rapid reactivity transient in which there is insufficient time for heat to be transferred from fuel to cladding, extensive fuel melting followed by vaporization occurs. Destructive internal pressures can be generated without increasing cladding temperatures significantly in this mode.

In evaluating the effects of the failure modes discussed above, two failure thresholds are considered. The first is associated with a gradual, and usually minor, cladding failure and may be approximately defined 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 the third failure mode described above.

A correlation of the results of different experiments conducted on Zircaloy 2 clad UO_2 fuel rods at TREAT 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_2 is about 260 cal/g, (Reference 22) and the heat of fusion is at least 78 cal/g (Reference 23). Thus, the 280 cal/g represents a condition where 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 measurements of pressure rise rates in the autoclave in the TREAT experiments (Reference 21). 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.

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[*] The increase in volume associated with the melting of UO_2 is 9.6 percent (Reference 20).

In computing the average enthalpy of the hottest fuel pellet during the excursion for the rated power 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. 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 (Reference 21) data for the meltdown experiments on Zircaloy 2 clad UO_2 fuel rods shows the threshold for the zirconium water reaction to be 210 to 220 cal/g energy input. A conservative threshold value of 200 cal/g is used in this study.

For the purpose of calculating the volume of the core which experiences prompt cladding rupture (the second failure threshold described in 14.2.2.2.a.2) in a given rod ejection accident, it is assumed that any DNB condition results in prompt cladding rupture for each rod where the DNB occurs. DNB in a rod ejection transient is assumed to occur whenever the peak thermal power of a given fuel rod exceeds the peak at steady state conditions which could result in a DNB, which in turn is assumed to occur for a DNBR of 1.919 using the BAW-2 CHF 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.

b. Method of Analysis

A B&W digital computer program has been used to analyze the rod ejection accident. This program agrees to within a few percent in all cases with CHIC-KIN (Reference 24). The B&W program is a point kinetics core model with a primary loop and pressurizer model. The core heat transfer calculation utilized a three-region fuel pin model. The 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 non-linear 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 25 segment curve of reactivity insertion during trip versus time, obtained by combining the

actual rod worth curve with a rod velocity curve. Nominal values for the various nuclear and physical parameters used as inputs are listed in Table 14.2-7.

As a check on the point kinetics calculation, the rod ejection accident was also analyzed for a limited number of cases in support of the technical specification rod worth using the exact, two-dimensional, space and time dependent TWIGL digital computer program (Reference 25). The point kinetics model assumes that the flux shape remains constant during a transient. This flux shape contains peaking factors which reflect unusual rod patterns such as the flux adjacent to a position where a high worth rod has been removed. Therefore, these points 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. But to have a transient where a rod is ejected from the core, one must start with a flux shape that is necessarily depressed in the region of the ejected rod. In fact, the higher the worth of the rod, the more severe becomes the depression. This flux depression also causes a fuel temperature depression. When the rod 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 rod, 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 rod ejection; the highest energy is reported in Table 14.2-8. 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 14.2-8, this result is uniformly true for all rod worths evaluated.

Thus, it can be seen that the space time dependent code gives a less conservative treatment of the accident analysis than does the point kinetics code.

For certain cases where the ejected rod has a low worth, or where at least one reactivity coefficient is very negative, or the initial power level is low, there is considerable pressure buildup in the Reactor Coolant System because of the increased heat being added to the coolant with no increase in heat demand. Many of these transients never reach the overpower trip point. For this class of possibilities, the high pressure trip must be relied on, and this is incorporated in the calculation.

c. Analysis and Results

1) Zero Power Level

The nominal BOL and EOL rod ejection analysis was performed at 10^{-3} rated power, and the results can be seen in Table 14.2-9. 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 rod worth were varied. Figure 14.2-1 shows the peak neutron power as a function of ejected rod worth from 0.2 to 0.8 percent delta-k/k. The curve shows two distinct parts corresponding to worths less than

the delayed neutron fraction (β) and values near to and greater than β. Figure 14.2-2 shows the corresponding results for the peak thermal power. It is seen that for rod 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.

In addition to the nominal cases discussed above, cases that represent an ejected rod worth of 1% Δk/k from hot zero power conditions and 0.65% Δk/k at rated power were performed. These points represent the fuel design limits. The fuel enthalpy does not exceed the acceptance criteria of 280 cal/g for these cases. For each new fuel reload, rod index limits are determined at various power levels and times-in-life to ensure that the ejected rod worth will not exceed the fuel design limits of 0.65% Δk/k at HFP and 1.0% Δk/k at HZP.

Figure 14.2-3 shows that the peak enthalpy in the fuel for the rod worths in the range being evaluated never exceeds 75 cal/g. Therefore, no threshold for damage is approached.

Figures 14.2-4 and 14.2-5 show the peak neutron and thermal power as a function of Doppler coefficient from -0.9 to -1.8×10^{-5} (delta-k/k)°F. It is seen that the variation is relatively small. Similar results are shown on Figures 14.2-6 and 14.2-7 for the variation of the moderator coefficient from –3.5 to $+1.5 \times 10^{-4}$ (delta-k/k)/°F. The slope of the curve for 10^{-3} rated power at BOL is the greatest slope for any of the four curves because this case relies on the pressure trip, which makes it a longer transient. It is also steeper because of the effect of the moderator coefficient, which is only noticeable in long transients due to the long time constant from fuel to coolant. Similarly, it is seen that the peak neutron power is higher for the EOL cases in both the Doppler and moderator studies, whereas the peak thermal powers are higher for the BOL cases. This again is because the EOL rod ejection cases are faster and the neutron power overshoots the trip point by a greater margin. It also trips more quickly, however, terminating the transient faster.

Figure 14.2-8 shows the effect of the trip delay time on the peak thermal power. It is seen that there is very little effect.

2) Rated Power

An analysis was performed for a BOL rod ejection at rated power with no Xenon present. The results of this analysis are shown in Table 14.2-9. A sensitivity study was made around this case and around the same rod worth at EOL. Figures 14.2-1 through 14.2-8 show these results.

Initial reactor power was assumed to be 2535 MWt. A higher initial power level would result in a higher peak neutron power. The event takes place so quickly that an increase in initial power would not significantly affect the results of the event and would not cause the acceptance criterion to be violated.

As seen on Figure 14.2-2, the peak thermal power shows relatively little change with increased rod worth. The peak neutron power on Figure 14.2-1 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 on Figure 14.2-3. The threshold for the zirconium-water reaction is not reached until values of BOL and EOL ejected rod worths are well above any which are considered feasible.

The results of varying the Doppler and moderator coefficients and trip delay time show very little effect on the peak neutron and thermal powers.

The results of the DNB calculation for BOL are shown on Figure 14.2-9. For the BOL analysis, ejection of the maximum rod worth of 0.65 percent delta-k/k at rated power results in 3.3 percent of the core volume in DNB. This corresponds to 17.5 percent of the rods.

d. Energy Required to Produce Further Reactor Coolant System Damage

(b) (5) [redacted]

[redacted]

[redacted]

[redacted]

e. Conclusions

The hypothetical rod ejection accident has been investigated in detail at two different initial reactor power levels: rated power 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 rod 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 calculated rod worths shown in Table 14.2-6 are about a factor of 3 less.

As a result of the postulated pressure housing failure associated with the accident (Item a. of this Subsection), reactor coolant is lost from the system. The rate of mass and energy input to the Reactor Building is considerably lower than that subsequently reported for the smallest rupture size considered in the loss of coolant analysis (Subsection 14.2.2.3). The maximum hole size resulting from a rod ejection is approximately 2.76 inches. This lower rate of energy input results in a much lower Reactor Building pressure than those obtained for any rupture sizes considered in the LOCA.

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. (Table 14.2-11 lists, by isotope, the gap activity expressed as a percent of total fuel pin activity). The fuel rods in DNB are calculated for the ejection of the maximum rod worth at BOL from rated power. Subsequently this gap activity and the activity in the reactor coolant from operation with 1 percent defective fuel is released.

The failure of the CRDM housing results in ejection of the control rod. It also results in a 2.76-inch Reactor Coolant System (RCS) rupture in the reactor vessel upper head. As the RCS pressure decreases, any pre-existing leakage to the steam generator would also decrease. When the pressure falls below the steam generator control pressure (of the turbine bypass or main steam safety valves), the leakage is terminated. In order to conservatively estimate the amount of reactor coolant that is released to the steam generators, a pre-existing leak rate of 1 gpm (total) was assumed. The effective leak area was based on the 1 gpm leak rate and the thermal-hydraulic conditions at rated power using the MOODY critical flow rate tables. This break area and the MOODY tables were used to calculate the time-dependent leak rate as the RCS pressure decreases due to the CRDM failure. It was conservatively assumed that the leak was not in critical flow until it was stopped and a discharge coefficient of 1.0 was modeled. Using these assumptions, it is calculated that less than 5 gallons of reactor coolant was released to the secondary system during the time that the RCS was depressurizing. The activity associated with this leakage is subsequently released to the condenser. A gas-to-liquid partition factor of 10^{-2} (equivalent to a decon factor of 100) is assumed for the iodine in the condenser (References 10 and 11), but the noble gases are assumed

to be released directly to the atmosphere. Therefore 99% of the total iodine released from the damaged rods into the condenser is retained by the condenser water.

All reactor coolant that is not released to the secondary system is released to the Reactor Building. Fifty percent of the iodine released to the Reactor Building is assumed to plate out. The activity released to the Reactor Building is shown in Table 14.2-11.

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 14.2-11 shows the resulting thyroid and whole body doses for a 2 hr exposure at the exclusion distance and for a 30 day exposure at the low population distance. In addition to the total thyroid and whole body doses, the table includes the dose contribution due to the activity released to the atmosphere via the secondary system and that released via Reactor Building leakage. Activity released due to normal operation within the Technical Specification limits was not considered in this accident analysis and is considered to be negligible. The doses resulting from the accident are well below the guideline values of 10CFR100.

The dose consequences reported in Table 14.2-11 were based on a 0.65 percent delta-k/k rod ejection accident with a nominal (zero) moderator temperature coefficient from hot full power (HFP) conditions. As shown on Figure 14.2-7, the rod ejection accident results are not strongly dependent upon the value of the moderator coefficient. Although the Technical Specifications require a negative moderator coefficient above 95 percent power, a moderator coefficient of up to $+0.9 \times 10^{-4}$ (delta-k/k)/°F is permitted at core power levels of 95 percent or less. Assuming that the fraction of the core that experiences DNB is proportional to changes in the peak thermal power prediction, the affect of the moderator can be approximated. Since the influence of the moderator coefficient on peak thermal power decreases as the initial power level increases, and that the initial fuel enthalpy decreases with decreasing initial core power level, it is concluded that the fraction of core experiencing DNB will be bounded by the HFP case, given the same ejected rod worth. If the fraction of the core that experiences DNB does not change, the offsite dose consequence will not change. Therefore, the offsite dose consequence at the exclusion distance reported in Table 14.2-11 is bounding.

f. Verification Analysis for Mark-B-HTP Fuel

The ejected rod accident DNB analysis was evaluated for a full core of Mark-B-HTP fuel using AREVA’s Statistical Core Design methodology and a Thermal Design Limit of 1.50. The evaluation (Reference 117) was performed in order to prove that the DNB pin census criterion from the analysis of record is met. A steady-state LYNXT case was run at 122.5%of rated core power (i.e., 2568 MWth) and produced an MDNBR of 1.6973, which gives approximately 19.7 DNBR points of thermal margin to the TDL of 1.50. The DNB pin census criterion for the ejected CRA is met; therefore, the ejected CRA analysis of record remains valid.

14.2.2.3 Large Break Loss of Coolant Accident

14.2.2.3.1 Identification of Causes

Loss of coolant accidents (LOCAs) are defined by 10CFR50.46 as hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, due to breaks in pipes of the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS. These accidents are non-mechanistically postulated; therefore, no method of reactor coolant pipe rupture is presented. They are classified as a limiting fault.

Break sizes in the RCS greater than 0.75 ft^2 are classified as large break LOCAs.

Simultaneously with the rupture of a reactor coolant pipe, all ac offsite power is assumed to have failed. This requires use of the onsite emergency electrical system.

Reactor trip, emergency core cooling systems (ECCSs), and containment and onsite emergency power supplies are provided to minimize the consequences of this accident.

Loss of Coolant Accident assumptions are listed in Table 14.2-14.

14.2.2.3.2 Large Break LOCA Analysis

a. Sequence of Events and System Operation

The sequence of events and system operation for a pipe break and its effects on the reactor and internals are discussed in Chapter 6 and in References 77 and 78. The evaluation of engineered safety features, other than ECCS, used to mitigate the consequences of this accident is discussed in Chapter 6.

b. Evaluation Model

Large break transients can be divided into three major phases: blowdown, refill, and reflood. The blowdown phase can be characterized as the decrease in the potential energy of the RCS to a level equal to that of its immediate surroundings. Core flow is variable and dependent on the nature, size, and location of the break. Departure from nucleate boiling (DNB) is generally calculated to occur very quickly; core cooling is by film boiling. Since film boiling amounts to only a small part of the steady state cooling, the cladding temperature increases by 600 to 1200°F by the end of the blowdown phase. During the very last seconds of blowdown, cooling is by convection of steam, and the cladding temperature is increasing.

Following blowdown, a short time is required for the ECCS to refill the bottom of the reactor vessel before final cooling can be established. During this period, core cooling is negligible, and the cladding experiences a near-adiabatic heatup. This phase is termed refill. When the ECCS water reaches the bottom of the core, the period of reflood commences. Core cooling by quenching is by steam generated below the rising water level and by water droplets entrained in the steam. The cladding temperature excursion is generally terminated before the core is covered by water because the steam water mixture is sufficient to remove the relatively low decay heat power being

generated at this time. The core is eventually covered by a mixture of steam and water, and the path to long term cooling is established through the use of pumped injection.

Figure 14.2-10 illustrates the interrelation of the computer codes used for large break analysis. The RELAP5/MOD2-B&W code calculates the evolution of system hydrodynamics and core power generation during blowdown. The REFLOD3B code is used to determine the length of the refill period and the flooding rates during reflood. The CONTEMPT code analyzes Reactor Building pressure. Finally, BEACH (a module in RELAP5/MOD2-B&W) is used with the output from RELAP5/MOD2-B&W and REFLOD3B to determine the peak cladding temperature response. The description of these computer codes can be found in References 32, and 79 through 81.

c. Core and System Performance

1) Emergency Core Cooling System

In the event of a LOCA, the emergency core cooling system (ECCS) provides protection to maintain core integrity. The high pressure injection (HPI), low pressure injection (LPI), and core flooding (CF) systems make up the ECCS. The internals vent valves within the reactor vessel are designed to promote core refill and reflooding following LOCA.

The emergency HPI system is actuated on RCS pressure less than 1600 psig. Backup HPI initiation signals not specifically credited in the analysis are (1) RCS Pressure less than 500 psig and (2) Reactor Building pressure greater than 4 psig. The 4 psig signal provides emergency core cooling in the event of a small break where the RCS pressure remains high. This two-pump system delivers water from the borated water storage tank (BWST) to the reactor vessel through the cold legs. The HPI System is cross-connected and has cavitating venturis installed. The cavitating venturis are designed to limit flow in each leg to less than 137.5 gpm when only one HPI pump is operating and the RCS is at atmospheric pressure.

The HPI system design, including the venturi design, ensures that the minimum required HPI flow is injected into the RCS. Table 14.2-28 shows HPI flow as a function of RCS pressure at the HPI nozzle for breaks in RCS piping and for the worst case HPI line break condition.

The LPI system is also initiated on an ESAS signal. The Low Pressure Injection flow rates (per loop) are greater than 2700 gpm at 109 psig Core Flood/LPI nozzle pressure for a LBLOCA. For additional flow rates as a function of core flood/LPI nozzle pressure refer to Table 14.2-27. The flow is deposited directly in the reactor vessel downcomer annulus through the CFT nozzles. Water supplied by the LPI system is contained in the BWST. When the BWST empties, suction is transferred to the Reactor Building sump manually and the recirculation mode is established. The LPI system has two separate and redundant injection paths. Each of the two flow paths is capable of delivering 100 percent of the design flow. Separate emergency power supplies provide power to the redundant flow paths in the event of loss of offsite power during an accident condition.

The self actuated core flooding system consists of two nitrogen-pressurized flooding tanks. Each tank is directly connected to a reactor vessel nozzle by a line containing two check valves and a normally open motor-operated stop valve. The system provides borated water if the RCS pressure falls below the nominal value of 600 psig. Each core flooding tank has a nominal liquid volume of 940 ft^3 borated water. For additional core flooding tank information refer to Table 14.2-14 (Sheet 2 of 2).

The analysis described in Reference 77, shows that the ECCS meets the requirements of 10CFR50.46.

A detailed description of the features of the model, including system noding, sources of heat, swelling, rupture, thermal properties, blowdown phenomena, single failure, and post-blowdown phenomena, is in Section 4.3 of Reference 78.

2) Examination of Core Component Structural Integrity

Many of the fuel rods may be expected to experience cladding perforation during the heatup portion of the transient in the LOCA as a result of fission-gas internal pressure and weakening of the clad as its temperature increases. The mechanical strength of M5 cladding begins to decrease when the temperature exceeds about 1400°F, causing the fuel rods with appreciable fission gas internal pressure to fail locally and to relieve the gas pressure when the temperature exceeds approximately 1500°F. Some local deformation of the rods will occur before perforation.

However, as analyses verify, cooling would still be effective since the fuel rods are submerged and cross-channel flow around the deformed area will continue to cool the rod. At worst, a local hot spot may occur, but 10CFR50.46 criteria would not be violated. It is calculated that a small number of fuel rods operating at peak power will experience a cladding temperature transient up to but not exceeding the criterion specified in Section 14.2.2.3.5. The major portion of the core will not experience as severe a transient. Heating of the fuel rod spacer grid requires heat flow from the clad to the structure; grid temperatures will lag the cladding temperature transient. As the fuel rod temperature rises, the fuel rods are expected to experience some bowing due to axial growth between supports. The spacer grids have substantial mechanical strength, even at the maximum expected temperatures, and will, therefore, retain sufficient strength to ensure spacing between fuel rods, to allow emergency coolant to reach the rods, and to keep the rods in their respective positions in the core by preventing gross change in core geometry.

The ability of the clad to maintain its strength and structural integrity during core reflooding has been confirmed by experimental work involving the rapid quenching of M5 tubing specimens. Test results confirm that the limiting criteria prescribed by 10 CFR 50.46, 2200°F PCT and 17 percent oxidation, are conservative for M5 cladding to prevent cladding embrittlement and fragmentation during a LOCA. The NRC staff has concluded that this design

criterion is acceptable for LOCA licensing analyses with M5 cladding up to the currently approved burnup levels.

Whenever two dissimilar metals are in contact with each other, there is a potential for the formation of a eutectic alloy which has a melting temperature lower than that of either pure metal. In the core and reactor vessel internals, dissimilar metals are present, with major elements being in the approximate proportions shown in Table 14.2-12. The binary phase diagram indicates that zirconium in the proportion 75 to 80 percent has a eutectic point with either iron, nickel, or chromium at temperatures of approximately 1715, 1760, and 2370°F, respectively. If one of these metals is in contact with Zircaloy and if the eutectic point is reached, then the material could theoretically melt even though the temperature is below the melting point of either metal alone.

One point of such dissimilar metal contact is between M5 clad fuel rods and the bottom-most spacer grid made of Inconel 718. However, the local power at the bottom of the heated length is insufficient to drive the contact point to eutectic formation.

Another area of dissimilar metal contact is that of a zirconium guide tube with the inconel or stainless steel cladding of the control rod. To determine whether the temperatures in the control rod following a LOCA could become high enough to approach either the temperature required for possible eutectic formation or the melting temperature of the Ag-In-Cd alloy, the thermal performance of a control rod assembly following a Large Break LOCA was examined with a bounding analysis. A conservative approach was taken in this analysis where power and PCT were artificially increased well above TMI actual values. Under these conditions, the fluid temperatures and time at elevated temperature were maximized to minimize the control rod heat removal during the transients. Despite the peaking and conservatisms used in this generic LBLOCA analysis, the control rod temperatures remained below the absorber melt temperature. Table 14.2-13 lists the assumptions made and the results of the control rod temperature study.

14.2.2.3.3 Results - Large Break LOCA

a. Large Break Spectrum

10CFR50, Appendix K requires that a spectrum of breaks be considered in determining the worst-case break size, configuration, and location. Results of analyses performed using the previous Evaluation Model (Reference 83) and the current Evaluation Model (Reference 78) determined that the typical worst break is a full-area double-ended guillotine break with a discharge coefficient (CD) of 1.0 located in the CLPD piping. This break location causes a significant reduction in the core flow and fuel pin heat removal during the first third of the blowdown period. The proximity of the break to the ECCS injection location also maximizes the potential for ECCS bypass during the later stages of blowdown. These two effects result in less fuel pellet stored energy removal and an increase in the reactor vessel lower plenum refill time. To confirm these results for all 177-LL B&W plants, a break spectrum analysis, which considered break size,

configuration, and location, was performed for the Oconee plants using the LOCA Evaluation Model. The results of these studies are summarized below.

Discharge Coefficient Analysis – The case with a CD of 1.0 resulted in the smallest positive hot spot core flow between one and eight seconds of the blowdown phase. The smaller flow reduced the fuel pin surface heat transfer. The liquid mass remaining in the lower plenum at the end of blowdown was also a minimum for this analysis, requiring a longer refill time during which the fuel pins heat up adiabatically. The calculated hot rod PCT of 1989 F was produced by the ruptured cladding segment. The calculated PCTs declined with decreasing discharge coefficient and switched to an unruptured segment, directly adjacent to the ruptured location. Further reductions in the discharge coefficient would result in additional surface heat transfer that would continue to reduce the calculated PCT. Therefore, no other calculations with smaller discharge coefficients were warranted. These results also confirmed that the transition break sizes discussed in the LBLOCA Evaluation Model did not need to be analyzed. The full-area, double-ended guillotine CLPD break with a discharge coefficient of 1.0 produced the most limiting results of the discharge coefficients studied. Since the results if this study can be applied to all 177-LL B&W plants, it is not necessary to demonstrate these results for the TMI-1 application analyses.

Break Type Analysis – Appendix K requires that instantaneous double-ended guillotine and longitudinal split break configurations be considered. The guillotine break is modeled as an instantaneous severance of the pipe, allowing separate discharges through the full pipe area from each side with no mixing of the flows from the two sides of the break allowed. The split break assumes discharge from the pipe through an area up to twice the cross-sectional pipe area. Mixing at the break location is allowed. The blowdown rates and system flow splits are somewhat different for the two break types, which can lead to differences in core flows and fuel pin heat removal.

Both breaks use discharge coefficients of 1.0. The split break produced higher core downflows during the later portion of blowdown, leading to better cooling and lower end-of-blowdown fuel pin and clad temperatures. The lower pin temperatures produce less boiling, decreasing the upper plenum pressure and increasing the core flooding rate. Consequently, the calculated PCT for the full-area split break with a discharge coefficient of one is lower than that produced by the guillotine break.

Split breaks performed with smaller discharge coefficients would increase the positive core flows during the first portion of blowdown. These higher flows would improve the cladding heat removal and cause additional reductions in the calculated PCTs. Therefore, CLPD split breaks will not produce core thermal-hydraulic conditions that can result in a PCT higher than that calculated for the guillotine break with a discharge coefficient of one. Since the results of this study can be applied to all 177-LL B&W plants, it is not necessary to demonstrate these results for the ANO-1/TMI-1 application analyses.

Break Location Analysis – There are three locations to consider for the large break LOCA: the hot leg piping, the cold leg pump suction piping, and the cold leg pump discharge piping. The hot leg break has been consistently shown to result in peak cladding temperatures far below those predicted for cold-leg breaks (Reference 78). The large positive core flow and no ECCS bypass combine to provide high fuel pin heat

removal for all hot leg breaks. Therefore, a hot leg LOCA analysis is not required to demonstrate that a hot leg break is not limiting for the 177-LL plants.

The pump suction break was analyzed to compare with the cold leg pump discharge break to determine the worst break location. The broken leg pump provided a significant resistance to flow trying to reach the break through the broken leg (Reactor Vessel side). The liquid was forced to reach the break via the hot legs, leading to positive core flows throughout blowdown and significantly increased hot pin heat removal. The lower pin temperatures allowed a higher core flooding rate and faster quench front advancement, and the amount of liquid remaining in the reactor vessel at End of Blowdown (EOB) led to a significantly shortened adiabatic heatup time. The PCT for the pump suction break was 160 F lower than that for the pump discharge break. Therefore, a break in the CLPD will produce more severe results. Since the results of this study can be applied to all 177-LL B&W plants, it is not necessary to demonstrate these results for the TMI-1 application analyses.

<u>Effect of Hot Leg LBLOCA on Steam Generator Tubes</u>

Due to the depressurization of the primary and secondary sides of steam generator from a LBLOCA, there are no significant pressure loadings on the steam generator tubes. There are also no large cross-flow loads which could produce bending loads in the tubes. Therefore, the only significant load in the tubes resulting from the LBLOCA event is the potential large axial loads caused by the large tube-to-shell differential temperatures which are created as the tubes rapidly cool in relation to the slow cooling shell.

An engineering analysis was performed to determine if loads on the replacement OTSG tubes from thermal expansion would produce material stress that remains within allowable ASME Code limits. The analysis also included stress and strain on the tube-to-tubesheet welds. The results of this analysis are provided in Reference 137. The conclusion of the analysis is that the stress and strain on the OTSG tubes and tube-to-tubesheet welds due to thermal effects of a hot leg LBLOCA are within allowable Code limits.

A plant specific analysis was performed for TMI-1 at a power level of 2772 MWt. The CLPD guillotine break with a C_D of 1.0 was analyzed with power peaks at various core elevations and at different core burnup values. Table 14.2-14 (Sheet 1 of 2) shows the peak cladding temperatures and linear heat rate limits resulting from that analysis. The LOCA linear heat rate limits for current fuel designs and operating cycles are documented in the cycle specific Core Operating Limits Report (COLR). The thermal-hydraulic response for the limiting break is shown on Figures 14.2-12 through 14.2-23. The axial power shapes used for the analysis are shown in Figure 14.2-24.

b. LOCA During Partial Power and Partial Loop Operation

Core power distribution analyses generally assume that the 100 percent full power LHR limits are preserved for all core power levels above 50 percent full power. Various LBLOCA analyses were performed with the 100 percent full power LOCA limits from another B&W designed plant to determine the maximum allowable MTC as a function of core power that result in a PCT that is below the PCT for the 100 percnet full power

case. The conclusions from the partial power analyses are applicable to TMI-1 because both plants are 177-fuel assembly, lowered loop plants. The resulting MTC as a function power curve is located in the TMI-1 COLR.

To allow an operating configuration with less than four reactor coolant pumps operating (partial loop), an analysis of the predicted consequences of a LOCA occurring during the proposed partial loop operation was evaluated. A LBLOCA analysis was performed with initial conditions based on three operating RCPs, with a core power set to 75 percent of full power (Reference 78) . Three CLPD guillotine break cases were performed with the locked rotor located in (1) the broken leg pump, (2) the intact leg pump of the broken loop, and (3) one of the intact loop pumps. The case with the locked rotor in the broken cold leg produced the most severe cladding temperatures at the end-of-blowdown. The EOB clad temperature is below that of the base case (CLPD) guillotine break from full power). Based on these results a case was performed with three operating RCPs at 80 percent full power with an MTC curve of +1.0 pcm/F. The inoperative pump was modeled in the broken cold leg. The results of the study showed that the calculated PCTs for the limiting three-pump case would be bounded by the 100% full power case. Therefore, the LOCA limits established for four-pump operation at full power are bounding for three-pump operation at 75 percent power.

c. Long Term Cooling

Criterion 5 of 10CFR50.46 states that a low core temperature must be maintained following the calculated successful initial operation and that decay heat must be removed for an extended period of time. This section describes the assumptions used to meet this condition.

1) Establishment of Long Term Cooling

The analysis of a LOCA is continued until the cladding temperatures at all locations in the core are decreasing and the fluid level in the core is rising. At this time the path to long term cooling is established. The capability of the ECCS to quench the reactor core will be demonstrated in reports on analyses of specific plant categories. Cooling for the long term is by circulation through the vessel maintained by a redundant pumped injection system.

The onset of long term cooling is defined as the time after a LOCA when operator action is required to ensure that ECC systems are properly aligned and minimum performance requirements are met. The onset of long term cooling, assuming a reasonable operator response time to evaluate the event, begins no earlier than 15 minutes after a LOCA. For large breaks, the onset of long term cooling occurs well after reflood and quenching, and for very small breaks, operator action may occur before the end of blowdown. The basis for the onset of long term cooling is related to the performance requirements of the ECC systems and reasonable operator response time.

Specific time periods during and following a LOCA can be identified as (1) blowdown, (2) refill, (3) reflood, and (4) long term cooling. The blowdown, refill, and reflood periods are short term (seconds and minutes) transient characteristics directed toward availability and rapid response to maintain the

cladding temperature and building pressures at or below prescribed design limits. No immediate operator action is required during these periods.

The period following reflood is of longer duration (minutes, hours, days); the cooling objectives are to remove decay heat and to provide gradual temperature reduction for the RCS and Reactor Building. Operator actions are required to verify and sustain long term cooling, and these actions may be initiated as soon as reasonable operator response can be assumed.

The duration of long term cooling is the period between its onset and the end of core cooling requirements by the ECC systems. The end of ECC cooling requirements occurs when the core is removed from the reactor vessel or when other permanent means are provided for core heat removal. The exact duration of long term cooling will vary depending on several factors, including the size of the break and the radiation release.

2) Boric Acid Concentration

During a hot leg large break LOCA (LBLOCA) coolant flow is in the forward direction through the core. LPI enters the reactor vessel and passes down the downcomer into the lower plenum, proceeds up through the core into the upper plenum, flows out of the vessel through the hot leg nozzle, and discharges out of the hot leg break. Therefore, boron acid does not concentrate in the core region. However, during a cold leg LBLOCA a majority of the LPI passes out of the downcomer through the cold leg nozzle and out of the cold leg break. The remainder of the LPI flow maintains a level in the reactor vessel downcomer, making up for boil-off from the core and a small amount of liquid carry-over through the reactor vessel internals vent valves (RVVV). Therefore, further analysis was required of the cold leg LBLOCA with respect to boron concentration.

The original evaluation to demonstrate compliance with 10CFR50.46 showed that following a large break LOCA (LBLOCA), there was a period during which a natural circulation loop would exist in the reactor vessel (Ref. 28). The flow path was downcomer-core-upper head-vent valves-downcomer. It was shown that circulation was adequate to prevent rapid increase in dissolved boron concentrations. During the natural circulation period, alternate flow paths could be established to maintain the solute boron concentration below their solubility limit.

The success of the previously evaluated natural circulation method in preventing the boron concentration from exceeding the solubility limit depends on the amount of liquid carry-over through the RVVV. The liquid carry-over mixes in the downcomer and some of the mixture flows out of the cold leg break. A review of the modeling techniques used in the original analysis indicated that the amount of liquid carry-over could be significantly less than originally predicted. A new analysis was performed, taking into account the physical geometry of the upper reactor vessel internals, to evaluate a previously uncredited recirculation path through the hot leg nozzle clearance gaps. No credit was taken for liquid carry-over through the RVVV for this analysis.

The reanalysis showed that the upper plenum mixture level rises causing liquid carry-over through the large flow holes in the upper plenum cylinder (Ref. 70). Some of the liquid flows back into the upper plenum through the three-inch holes in the core support shield opposite the hot leg nozzles. Liquid level rises in the hot leg, covering a large portion of the reactor vessel to hot leg nozzle interface region. A differential pressure is established between the outlet plenum and the downcomer forcing the RVVV to open partially and pass steam to the upper downcomer and out of the vessel through the cold leg break. During the LOCA the reactor vessel internals cooldown faster than the reactor vessel nozzle region. The gap between the reactor vessel hot leg nozzle and the plenum cylinder opens sufficiently to pass liquid from the outlet plenum into the upper downcomer. The liquid mixes in the downcomer and some of the mixture flows out of the cold leg break.

The reanalysis also showed that this passive method was acceptable at preventing the core boron concentration from exceeding the solubility limit with no credit for liquid carry-over through the RVVV. The active methods were reevaluated without taking credit for liquid carry-over through either the RVVV or the hot leg nozzle gap. Refer to Figure 14.2-63 for a simplified flow diagram. The results of that analysis indicated that the hot leg injection method through the auxiliary pressurizer spray line would not prevent boron concentration from exceeding the solubility limit following continuous operation at full power. This method does become effective once the core boil-off rate decreases to below the auxiliary spray flow rate, approximately 4.8 days. However, that condition does not occur before the boron concentration exceeds the solubility limit. The DHR drop line method, with either gravity flow to the RB Sump or suction of a LPI pump, adequately controls boron concentration as long as it is initiated within approximately eight hours. The DHR drop line valves are not single failure proof.

Further analysis of liquid carry-over through the RVVV (Ref. 73 & 74), without credit for hot leg nozzle gap flow, showed that the boron concentration would not exceed the solubility limit until approximately 24 hours. Initiating the drop line method within that time would be successful.

Approximately 10% of the calculated hot leg nozzle gap flow, is sufficient to prevent boric acid concentration from exceeding the solubility limit. Partial credit for this flow for up to approximately 4.8 days would allow initiation of the hot leg injection method. The hot leg nozzle gap flow method would be capable of preventing boron concentration from exceeding the solubility limit indefinitely at a fraction of the calculated flowrate. The referenced analyses assumed CFT and BWST concentration of 3000 ppmb. The initial core concentration was determined by calculation.

3) The method of preventing boron concentration from exceeding the solubility limit is to take partial credit for hot leg nozzle gap flow until the drop line method can be placed in service. If unable to establish the drop line method then the plant will initiate hot leg injection through the auxiliary spray line. If unable to establish either active method then the plant would rely on the passive means of hot leg

nozzle gap flow to prevent boron concentration from exceeding the solubility limit.

d. Mark-B 12 Fuel

The CLPD guillotine break with a CD of 1.0 has been analyzed for the Mark-B12 fuel type (Reference 116). The analysis serves as the basis for the allowable local power for the Mark-B12 fuel. A curve of allowable linear heat generation rate as a function of core elevation for times in life of fuel operation is provided in the COLR. The limiting PCT from the Reference 116 analysis is 1989°F.

e. Mark-B-HTP Fuel

The CLPD guillotine break with a CD of 1.0 has been analyzed at Beginning-of-Life, Middle-of-Life, and End-of-Life for the Mark-B-HTP fuel type (Reference 120) with 20% OTSG tube plugging. The analyses considered a mixed core of Mark-B-HTP and Mark-B12 fuel, using the hot pin methodology approved in Reference 121 to offset mixed core penalties. The analysis serves as the bases for the allowable local power for the Mark-B-HTP fuel. A curve of allowable linear heat generation rate as a function of core elevation for times in life of fuel operation is provided in the COLR. The limiting PCT from Reference 120 analysis is 1890°F.

14.2.2.3.4 Environmental Analysis of Loss of Coolant Accidents

a. Consequences of LOCA Radioactive Releases to the Environment

Safety injection is designed to prevent significant clad melting in the event of a loss of coolant accident. The analyses in the preceding sections have demonstrated that safety injection will prevent clad melting for loss of coolant accidents resulting from RCS ruptures which range in size from small leaks to the complete severance of the largest reactor coolant pipe. Without clad melting, only the radioactive material in the coolant at the time of the accident plus some gap activity is released to the Reactor Building.

The NRC-sponsored computer code RADTRAD was utilized to determine TEDE doses to control room operators, exclusion area boundary (EAB), and Low Population Zone (LPZ). Refer to Appendix 14C for additional discussion.

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 Reactor Building. Table 14.2-4 lists the activity in the coolant. In addition to the coolant activity, the activity associated with the gap of all fuel rods is also assumed to be released.

The noble gas activity released is shown in Table 14.2-17. All isotopes of iodine have been equated into dose equivalent curies of iodine-131. The dose equivalency factor is determined by considering the concentration and specific dose of each isotope present over the period of interest. The iodine dose to the thyroid, per curie, is obtained from the values given in Reference 100. The iodine activity released to the Reactor Building and available for leakage is given in Table 14.2-17.

The trisodium phosphate in the Reactor Building spray reduces the airborne iodine as described in Appendix 14B.

Specific parameters used and the calculated spray effectiveness are tabulated in Table 14B-3 of Appendix 14B.

Although the Reactor Building 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, since the Reactor Building will have returned to nearly atmospheric pressure, the rate is assumed to be reduced to one half the design leak rate and to remain at the reduced value for the duration of the accident.

The atmospheric dispersion characteristics of the site are described in Section 2.5. A breathing rate of 3.5×10^{-4} m^3/S is assumed for the 2 hour exposure. For the 24 hour exposure, a breathing rate of 3.5×10^{-4} m^3/sec is assumed for the first 8 hours, and a rate of 1.8×10^{-4} m^3/sec is assumed for the remaining 16 hours. For the 30 day exposure, a breathing rate of 2.3×10^{-4} m^3/sec is assumed.

The LOCA doses are bounded by the dose results of the MHA accident discussed in Appendix 14C. TEDE doses are maintained below 10CFR50.67 guidelines.

b. Effects of Reactor Building Purging

At times during the normal operation of the reactor it may be desirable to purge the Reactor Building while the reactor is operating. If a Large Break LOCA were to occur during purging operations, activity would be released to the environment. Assuming the worst rupture, essentially all of the reactor coolant will have been blowndown. (The activity in the Reactor Building is then taken to be the reactor coolant activity after operation with 1 percent failed fuel). For conservatism, the analysis assumes unrestricted flow through the purge line for a 1-minute closing time. No reduction in flow is assumed as the valve closes. The resulting maximum 2 hour TEDE dose at the exclusion area distance due to purge valve closing time is 0.026 rem. The additional TEDE dose that results from this release when added to the dose for a loss of coolant accident without purging is bounded by the MHA accident and is well below the 10CFR50.67 guidelines. The resulting contribution to the TEDE dose to the LPZ is 4.6×10^{-3} rem. The corresponding contribution to control room operator dose is 3.6×10^{-3} rem. Therefore, purging operations can be performed during reactor operation.

14.2.2.3.5 Post Analysis-of-Record Evaluations for Large Break LOCA

In addition to the analyses presented in Subsection 14.2.2.3.3, evaluations and reanalyses may be performed as needed to address emergent issues or to support plant changes. The issues or changes are evaluated, and the impact on the PCT is determined. The resultant increase or decrease in PCT is added to the analysis of record PCT (1989^{o}F for the limiting fuel type currently in use). These issues and their evaluations are reported to the NRC via the normal 10CFR50.46 reporting requirement. The latest 10CFR50.46 report is on file at the site.

The current peak clad temperature for a large break LOCA, including all penalties and benefits for evaluations/reanalyses performed since the analysis-of-record, 1954^{o}F.

The original OTSGs were replaced at the end of Cycle 17. The LBLOCA evaluation for the replacement OTSGs confirmed that the existing LBLOCA analyses for the original OTSGs remain bounding as long as the SG tube plugging levels in the replacement OTSGs do not exceed 10 percent.

The initial fuel conditions for TMI LOCA analyses are determined using the TACO3 and GDTACO fuel performance codes, which do not explicitly account for thermal conductivity degradation (TCD), but are adjusted to account for TCD effects. In 2014, AREVA determined that the previous TCD evaluation performed for the TACO3/GDTACO codes was non-conservative using the state-of-the-art COPERNIC code. The evaluation indicates that adequate LOCA peaking margins at the axial power imbalance operating limits specified in current COLRs are not supported without compensatory actions. A 2.0 kw/ft penalty to LOCA linear heat rates (LHR) is determined to offset the TACO3/GDTACO non-conservative TCD treatment [Ref. 140] until a new LBLOCA analysis is performed using updated TACO3/ GDTACO TCD adjustment factors. Exelon committed [Ref. 141] to perform a new LBLOCA analysis after the NRC approves the supplement to AREVA's LOCA Evaluation Model for OTSGs topical report, BAW-10192P-A.

14.2.2.4 Small Break Loss of Coolant Accident

14.2.2.4.1 Identification

Small break LOCAs are piping ruptures whose break areas range from 0.0007 ft^2 (3/8 inches diameter pipe) to as large as 0.75 ft^2 (12 inch diameter pipe).

The transient progression for SBLOCAs is summarized here to identify the key phenomena and controlling thermal-hydraulic behavior during each phase of the event.

A SBLOCA generally progresses through five phases: (1) subcooled depressurization (2) reactor coolant pump and loop flow coastdown and natural circulation, (3) loop draining, (4) boiling pot, and (5) refill and long-term cooling. The subcooled depressurization phase begins at the leak initiation. This phase is characterized by the period of time before the RCS begins to saturate and voids begin to form in the RV upper head and hot leg U-bends. During this period, the pressurizer will begin to empty, the RCS will depressurize to the low RCS pressure reactor trip setpoint, and the turbine will trip. With the assumption of a loss of offsite power coincident with reactor trip, the MFW pumps and RC pumps will trip and EFW will be initiated following a delay.

Following the RCP coastdown, the RCS flow tends to evolve to a natural circulation flow condition. The energy generated by the core is transferred by convection to the steam generators during the natural circulation flow phase. The continued loss of the RCS liquid inventory allows steam voids to form in the upper reactor vessel head and the upper hot leg U-bends. Natural circulation ends when the U-bend steam void displaces the hot leg mixture levels below the U-bend spillover elevation. Flow is usually interrupted first in the hot leg containing the pressurizer surge line connection, because of the additional flashing of the saturated pressurizer liquid that enters during the subcooled depressurization. Near the end of the flow phase, alternating periods of RCS repressurization can cause intermittent spillovers of hot-leg liquid into the steam generator primary region.

With the interruption of the RCS loop flow, the loop-draining phase begins. As the entire RCS approaches saturated conditions, the onset of subcooled and saturated nucleate boiling occurs in the core because of the high decay heat levels and the RCS depressurization. The flashing within the hot legs increases the size of the voids in the U-bends and eventually interrupts RCS flow and decreases the primary-to-secondary heat transfer. For the larger SBLOCAs, the RCS will continue to depressurize as the loops drain. For smaller breaks, however, the reduced heat transfer can interrupt the RCS depressurization. Also for these smaller breaks, the volumetric expansion of the RCS, due to continued steam formation, can exceed the volumetric discharge from the break, causing the RCS pressure to temporarily stabilize or increase.

In the reactor vessel, the steam void in the upper head displaces enough liquid to uncover the reactor vessel internals vent valves (RVVVs), creating a manometric imbalance between the core and the downcomer. The imbalance forces the RVVVs to open and pass steam into the reactor vessel downcomer. The downcomer steam volume grows until the cold leg nozzle is exposed to steam. As soon as the downcomer liquid level decreases below the cold leg nozzle spillunder elevation, a steam venting path develops from the core through the RVVVs to the cold leg break, enhancing the RCS depressurization.

During the loop draining phase, the steam voids that develop in the U-bends can become large enough that the primary liquid level is displaced into the steam generator tube region below the EFW nozzles. The improved primary-to-secondary heat transfer can then be restored, through condensation on the tubes wetted by the EFW. This heat transfer process within a once-through steam generator (OTSG) is referred to as boiler-condenser mode (BCM) cooling. When BCM cooling takes place near the location of the EFW nozzles, it is referred to as high-elevation BCM cooling. If high-elevation BCM cooling occurs, the RCS depressurization rate will be increased. Later in the loop draining phase, a different form of BCM cooling can occur if the RCS tube liquid level decreases below the secondary liquid level. This cooling process is referred to as pool BCM cooling, and will continue if (1) RCS condensation and ECCS injection do not cause the RCS liquid level to increase above the secondary level, (2) the secondary fluid temperature is maintained below the temperature of the steam on the primary side of the OTSG tubes, and (3) the secondary liquid level is high enough that the secondary OTSG thermal center remains several feet above the RCP spillover elevation. For the smaller breaks, the combination of leak flow (with upper-RV venting through the RVVVs), BCM cooling, and HPI cooling will cause the RCS pressure to again decrease.

Also during the loop draining phase, the reactor vessel outlet annulus mixture level will decrease to the hot leg nozzle spillunder elevation. If the top of the hot leg nozzles void, steam will flow up the hot leg riser section, and liquid from the hot leg risers will drain back into the vessel. This hot leg draining allows the mixture level in the outlet annulus to remain near the top of the hot leg nozzle until the hot leg level drops into the RV exit nozzle horizontal piping.

After the hot legs empty, another path for the direct venting of steam to the break can be opened if the loop seals in the RCP suction piping are cleared. The suction piping of the four RCPs contains a large total volume and the spillunder elevation for this piping is approximately 23 feet below the top of the core. For the larger SBLOCAs, the RCS depressurization can be rapid enough to cause significant flashing in the suction piping, causing the liquid level to decrease below the suction piping spillunder elevation. The loop seals will then be clear, creating another steam relief path, in addition to the path through the RVVVs.

When loop draining ends, the break site void fraction will be based on core steam plus broken loop HPI. At that point, the only RCS liquid available for core cooling is the liquid remaining in the reactor vessel and the ECCS flow from the intact loops. This portion of the transient is defined as the “boiling pot” phase. The increased void fraction at the break will further increase the RCS depressurization rate. The reactor vessel levels will continue to decrease, however, if the ECCS injection cannot match the reactor vessel liquid loss from flashing, decay heat, and passive metal heat.

The break flow allows the RCS depressurization to progress until either the CFT pressure is reached or the HPI flow rate exceeds the liquid loss rate, allowing the RCS to refill to the break elevation. Before either of these conditions occur, the mixture levels may descend into the core heated region resulting in a heat up of the fuel cladding in the uncovered portion of the core.

The clad temperature increases calculated for the upper elevations are conservative because the assumed power shape in the model places the peak power at the 9.536-ft core elevation. This power shape bounds the positive imbalance limits for core operation. During the period of partial core uncovering, the clad may swell and possibly rupture if the clad temperatures exceed 1300^{o}F. The potential for clad rupture is increased in the SBLOCA analytical model by assuming an initial internal pin pressure typical of the end of fuel life (EOL). If clad rupture is calculated, a sensitivity study is needed to show that the calculated PCT will bound the fuel pin conditions at any time-in-life condition.

An SBLOCA transient analysis is normally terminated at some point after the entire core is refilled and the cladding temperatures returned to within a few degrees of RCS saturation temperature. For the level to increase, core inflow (ECCS plus SG condensate) must exceed the liquid loss rate. Continued RCS depressurization permits higher ECCS injection rates that hastens core refill. The additional ECCS flow assures that the core can be kept covered. Once the core has been completely quenched, the analytical results are checked to ensure a path to long-term cooling is established. For long-term cooling to be assured, the HPI flow and/or LPI flow must match core boiling due to decay heat and wall metal heat plus flashing. When long-term cooling is assured, the LOCA analysis is terminated.

Loss of Coolant Accident assumptions are listed in Table 14.2-14. The original SBLOCA analysis took credit for operator action to balance the HPI flow between the four nozzles, which would increase the amount that reaches the reactor vessel. TMI-1 has since been equipped with cavitating venturis in each of the HPI lines to ensure that the minimum required HPI flow is injected into the RCS without the requirement for the operator to take action.

14.2.2.4.2 Small Break LOCA Analysis

a. Evaluation Model

The analysis method used for small break evaluation is described in References 77 and 78. The analysis uses the RELAP5/MOD2-B&W code to develop the history of the RCS hydrodynamics. Some sizes of small break result in partial core uncovering. RELAP5/MOD2-B&W properly calculates the inner vessel mixture height and core steam rate. Therefore, an additional computer code and code interface is not required. Reference 78 presents the conclusion that the simplified method is indeed conservative for demonstrating compliance with the criteria of 10CFR50.46.

b. Analysis Categories

The analyses provided herein can be categorized into the following general topics:

1) Cold Leg Pump Discharge (CLPD) Break Spectrum

2) CFT Line Break

3) HPI Line Break

4) Mark-B12 Fuel

5) EFW Temperature

6) Mark-B-HTP Fuel

7) Historical Analyses

Evaluation Model studies documented in Reference 77 demonstrated that the most limiting break location is in the bottom of the cold leg piping between the HPI injection nozzle and the reactor inlet vessel nozzle. Two special break locations are included: Core Flooding Tank line and High Pressure Injection line because they present unique challenges to the core cooling capacity of the ECCS because they result in reduced ECCS flow to the RCS.

c. Core Component Structural Integrity

An analysis was performed to confirm the structural integrity of both the zirconium guide tubes and the inconel or stainless steel cladding of the control rod if they were to come into contact with each other. To determine whether the temperatures in the control rod following a SBLOCA could become high enough to approach either the temperature required for possible eutectic formation or the melting temperature of the Ag-In-Cd alloy, the thermal performance of a control rod assembly following a SBLOCA was examined analytically. A conservative approach was taken in this analysis where power was artificially increased well above TMI actual values. Despite the higher power levels, control rod temperature did not exceed the melting temperature and the integrity of the control rod cladding was preserved. Table 14.2-13 lists the assumptions made and the results of the control rod temperature study.

14.2.2.4.3 Results - Small Break LOCA

a. Cold Leg Pump Discharge (CLPD) Breaks

A break spectrum analysis was performed with break sizes, in square feet, analyzed at the CLPD of 0.01, 0.03, 0.04, 0.05, 0.06, 0.07, 0.08, 0.09, 0.10, 0.15, 0.175, 0.2, 0.3, 0.05 and 0.75. Breaks smaller than 0.04 ft^2 will produce a slower RCS inventory loss rate and will be less challenging to the capacity of the ECCS to provide adequate core cooling. The smaller break sizes will also allow more time for mitigative operator actions, such as a manual depressurization of the secondary system, restoring an additional HPI pump to service, or initiating HPI/LPI piggyback operation. None of these potential beneficial operator actions are credited in these analyses.

1) Small SBLOCA

The smaller breaks, between 0.01 ft^2 and 0.08 ft^2, present the greatest challenge to the HPI system to replace lost liquid inventory before significant fuel or clad damage occurs. These break sizes are not large enough to rapidly depressurize the RCS to the CFT pressure (Figures 14.2-25 and 26). Consequently, the HPI flow reaching the reactor vessel must be sufficient to match the core decay heat. Therefore greater analytical emphasis was placed on those break sizes.

The RCS depressurization and voiding quickly interrupted loop flow. With primary-to secondary heat transfer interrupted, an RCS repressurization was predicted for all of the breaks in this category. For all the break sizes in this category, the RCS pressure response caused delays in ES actuation, and, consequently, in the start of HPI flow. Reverse (secondary-to-primary) heat transfer also had a detrimental effect on the RCS depressurization rate.

All of the breaks in this category, except the 0.01 ft^2 break, experienced partial core uncovering (Figures 14.2-28 and 29). The duration of core uncovering was a direct result of the sustained high RCS pressure, which limited the HPI flow and delayed the start of CFT injection. The limiting PCT for the entire spectrum analysis, 1444°F, was calculated for the 0.05 ft^2 break (Figure 14.2-46).

For the small SBLOCA cases analyzed, the RCS pressure did not depressurize sufficiently to allow LPI flow to enter the reactor vessel. In each case, at the time the analysis was ended, the core was completely recovered, the downcomer level was increasing, and the HPI flow was sufficient to absorb the decay heat and wall metal heat contributions. These conditions confirmed that the HPI flow, while inadequate to prevent partial core uncovering, was adequate to ensure long-term cooling.

2) Intermediate SBLOCA

Three break sizes were analyzed in the CLPD, with break areas of 0.09, 0.10, and 0.15 ft^2. These breaks caused the RCS to depressurize faster, and to enter the boiling pot mode sooner, than the small SBLOCA. The intermediate breaks continuously depressurized the RCS after the flow phase ended, although the depressurization rate slowed temporarily. The 0.09 and 0.10-ft^2 cases experienced a slight RCS repressurization. However, it did not significantly delay ESAS actuation. RCS depressurization rate increased after the steam venting path through the reactor vessel internals vent valves (RVVV) was established (Figures 14.2-26 and 27).

ESAS was actuated relatively early in the transient. The RCS depressurization rate also was not significantly affected when the RCS pressure decreased below secondary pressure and reverse (secondary-to-primary) heat transfer began. The intermediate breaks were able to discharge fluid mass, energy, and volume so efficiently that they cooled and depressurized both the primary and the secondary systems.

Even though the RCS depressurized fairly quickly, allowing the CFT injection to begin relatively early in the event, partial core uncovering did occur for the intermediate breaks (Figures 14.2-29 and 30). As a result, clad heatup occurred in the upper core regions.

When the analysis of the 0.09-, 0.10-, and 0.15-ft^2 breaks ended, the low-low pressure setpoint had been reached but the RCS pressure remained above the LPI shutoff head. Furthermore, the pressure was decreasing and sufficient liquid remained in the CFTs to sustain CFT injection until LPI flow could begin. Therefore, the analyses confirmed that, for intermediate breaks, the ECCS capacity was sufficient to recover the core and to provide adequate long-term cooling.

3) Large SBLOCA

Three large SBLOCAs, with break areas of 0.30, 0.50 and 0.75 ft^2, were analyzed. For these large SBLOCA sizes, the effects of the break dominated other factors (such as the timing and magnitude of EFW flow, BCM cooling, and reverse heat transfer) that could potentially affect the RCS depressurization rate.

For all three cases, the low-pressure reactor trip setpoint was reached within 1.5 seconds after break opening, and ESAS was actuated within the first 16 seconds. The transients progressed relatively quickly through the SBLOCA phases. In comparison to the smaller breaks analyzed, the HPI, CFT, and LPI delivery during the large SBLOCAs was enhanced by the lower transient RCS pressures (i.e., the actuation times were earlier and the flow rates were higher) (Figure 14.2-27). CFT injection started within 10 seconds after the onset of core uncovering. The rapid RCS depressurization rates caused by the large SBLOCAs produced significant core voiding. The improved ECCS performance significantly shortened the duration of core uncovering and produced lower peak clad temperatures (Figure 14.2-30).

The calculated PCTs for the 0.30, 0.50 and 0.75 ft^2 breaks were well below the limiting values calculate for the breaks in the small SBLOCA category (Figure 14.2-33). LPI flow was initiated in all three cases, but well after the HPI and CFT flow had limited the clad temperature increase and completely refilled the core. However, while the HPI and CFTs had demonstrated adequate short-term core cooling, the availability of the LPI ensured that adequate long-term cooling would continue after the CFTs emptied.

b. CFT Line Break

For the CFT line break, a more severe degradation of the ECCS capacity must be considered than for a CLPD break. During a CFT line break, the break location prevents one CFT and one LPI train from injecting into the reactor vessel. The other LPI train and one HPI train are assumed to be unavailable for core cooling due to a single failure. No credit is taken for injection from the normal makeup pump. Therefore, only one CFT and one train of HPI remain available for core cooling. However, since the HPI piping remains completely intact, all of the flow from the available HPI pump is able to enter the RCS. The break area was limited to 0.44 ft^2 by the cross-sectional area of the CFT line nozzle insert at the reactor vessel. The results of this case are shown on Figures 14.2-34 through 14.2-37.

The relatively large break size caused a rapid RCS depressurization. The timing of CFT actuation and the available HPI flow were sufficient to maintain core cooling throughout the event. The results of this case also demonstrated that one CFT and one

HPI pump provide adequate ECCS flow to mitigate the consequences of a CFT line break.

Framatome Technology's Preliminary Safety Concern (PSC) 2-00 brought into question the completeness of the previous CRAFT2 best-estimate RCP trip analyses. The concern was with a core flood tank (CFT) line break coupled with a more realistic 2-phase flow degradation assumption. However, this flow degradation assumption, referred to as M3, is not part of the approved ECCS evaluation model for SBLOCA's. With the approved SBLOCA model and assumptions, the 10CFR50.46 limits continue to be met for the scenario. Therefore, the best-estimate RCP trip analyses were reanalyzed with the NRC-approved RELAP5/MOD2-based evaluation model using the following revised assumptions from those in Table 14.2-14:

- Initial 100% power level is 2568 MWt.
- Reactor Coolant Pump performance follows the more realistic M3 prediction for two-phase flow degradation.

The results have been included in reference 108, which concludes that acceptable SBLOCA response requires manual Reactor Coolant Pump trip within 1 minute (with single-failure) or 10 minutes (Best-estimate with no failures) following LSCM.

For the best-estimate 10-minute RC pump trip, the analysis determined that the most limiting PCT was 1354^{o}F for the 0.08-ft^{2} CLPD break. With results well below the 10 CFR 50.46 acceptance criteria and bounded by the worst-case 1412^{o}F PCT for a Loss-of-Offsite Power scenario, these analyses reflected acceptable core cooling consequences. The availability of the two complete trains of ECCS coupled with the other less conservative assumptions have allowed the operators up to ten minutes to trip the RCPs following LSCM during a SBLOCA. This demonstration of adequate core cooling for the SBLOCA spectrum supports the previous conclusions from the best-estimate ten-minute RCP trip analyses performed with the CRAFT2 evaluation model.

Furthermore, for the most limiting single-failure scenario, which is the CFT line break, the analyses confirm that RC Pump trip after the RCS liquid fraction has dropped below 0.3, but before significant LPI flow is achieved will result in some core uncovering and heat up. Therefore, an RC pump trip immediately upon LSCM remains the best guidance and continues to be promulgated in the Generic Emergency Operating Guidance (GEOG) document (reference 109). Based on these results and proven operator performance, TMI-1's Abnormal Transient Procedures specify that the RC pumps are to be left running if not tripped within 1 minute of LSCM indication.

c. HPI Line Break

The HPI line break location was modeled in an HPI line just upstream of the HPI nozzle, and the thermal sleeve in the nozzle was assumed to be blown out coincident with the break opening. The break area is limited by the cross-sectional area of the pipe without the thermal sleeve (0.02463 ft^{2}). One of the four HPI flow paths to the RCS was completely interrupted by the break, and, because of a large differential pressure between the broken and intact lines, the broken HPI line was fed preferentially. Consequently, the HPI flow from the broken line was assumed to spill into the containment and was, therefore, not modeled.

For the CLPD breaks and the CFT line break, HPI was modeled to initiate automatically upon actuation of the low RCS pressure ESAS signal plus a delay. However, because of the slow RCS depressurization during the HPI line break, HPI was not available until ~1251 seconds. A second case was run that modeled operator action to manually start the available HPI pump at 10 minutes after the loss of core exit subcooling, providing HPI flow at ~675 seconds. Both cases assumed that the other HPI pump was unavailable for the entire transient, due to a single failure. No credit is taken for flow from the normal makeup pump.

The results of the HPI line break analysis with operator action to initiate HPI are presented in Figures 14.2-38 through 14.2-41. The results of the HPI line break analysis with no operator action to initiate HPI are presented in Figures 14.2-42 through 14.2-45. The timing of CFT actuation and the available HPI flow were sufficient to maintain core covering throughout the event. Therefore, the HPI system had sufficient capacity to provide adequate short-term and long-term core cooling for the HPI line break.

d. Mark-B12 Fuel

SBLOCA transient consequences are dominated by core power and emergency core cooling system flows. They are generally not sensitive to fuel design differences such as those found between the Mark-B9 fuel and the Mark-B12 fuel. A full spectrum of small break sizes has already been performed for TMI-1 based on the Mark-B9 fuel design. The change in fuel clad material does not invalidate the SBLOCA sensitivity studies that were previously performed for the TMI-1 plant (Reference 77). Nonetheless, a confirmatory analysis of the limiting break size from the Mark-B9 spectrum was performed using the Mark B-12 fuel parameters (Reference 116). The results show that the existing Mark-B9 results are conservative and appropriate for the Mark-B12 fuel, thus reanalysis of the full spectrum was unnecessary.

e. EFW Temperature

In the Reference 77 analysis, an EFW temperature of 120^{o}F is assumed. A value of 120^{o}F is a conservative estimate of the temperature for the liquid in the condensate storage tank. However, the liquid in the line between the tank and the pump can be at a temperature as high as 135^{o}F. An analysis is performed assuming 135^{o}F EFW temperature for the first 10 minutes, followed by a step change to 120^{o}F (Reference 116). The analysis was performed for the 0.04, 0.05, and 0.06 ft^2 CLPD breaks with the Mark- B9 fuel type. The results of the analysis show an increase in the PCT the 0.05 ft^2 break, and a decrease in PCT for the other two break sizes. The Mark-B9 fuel is no longer in operation (beginning with operating cycle 18). Further analysis of the Mark-B12 fuel is discussed in the next section below.

f. Mark-B-HTP Fuel

The Mark-B-HTP fuel design contains the same fuel rod design as Mark-B12 fuel and has hydraulically similar channels that result in nearly the same core uncovering and cladding heatup for the same initial stored energy. A full spectrum analysis to determine the limiting break size was performed using the Mark-B-HTP fuel parameters and

replacement OTSGs with 5 percent tube plugging (Reference 120). The higher initial EFW temperatures discussed above were considered in the analysis. The results of the analysis determined a limiting Mark-B-HTP PCT of 1444°F based on the 0.07 ft^2 break. This PCT is also bounding for the Mark-B12 fuel. The analysis also includes the effect (reduced EFW wetting) of the replacement OTSGs discussed in Subsection 14.2.2.4.4. The results of the analysis also concluded that the maximum break size for the Mark-B-HTP spectrum is the 0.50 ft^2 break size, and does not include the 0.75 ft^2 break that is part of the Mark-B9 /Mark-B12 spectrum.

g. Historical Analyses

1) Evaluation Model

The analysis method used for the small break evaluation in this section is basically that described in Reference 29, 39, and 40. The analysis uses the CRAFT2 code to develop the history of the RCS hydrodynamics. A schematic diagram of the model is shown on Figure 14.2-47 along with the node descriptions. Several of the breaks analyzed with the CRAFT code showed partial core uncovering. Thus, it was necessary to perform a FOAM analysis in order to more properly calculate the inner vessel mixture height and core steam rate. Reference 39 presents the conclusion that the simplified method is indeed conservative for demonstrating compliance with the criteria of 10CFR50.46.

2) Small Break LOCA Transient with the RC Pumps Operative

Small break analyses have been performed assuming a loss-of- offsite power (reactor coolant pump coastdown) coincident with reactor trip. These analyses support the conclusion that an early RC pump trip for a LOCA is a safe condition.

Additional evaluations have been performed to examine the primary system response during small breaks with the reactor coolant pumps operative. The breaks analyzed are assumed to be located in the cold leg piping between the reactor coolant pump discharge and the reactor vessel. Key assumptions which differ from those described in the Reference 40 are those concerning the equipment availability and phase separation. A detailed discussion is in Reference 35.

With the RC pumps operating during a small break, the steam and water will remain mixed during the transient.

This will result in liquid being discharged out of the break continuously. Thus, the fluid in the RCS can evolve to a high void fraction as shown on Figure 14.2-48. The maximum void fraction that the system evolves to, and the time it occurs, is dependent on the break size and location. Continued RC pump operation, even at high system void fractions, will provide sufficient core flow to keep cladding temperatures within a few degrees of the saturated fluid temperature.

Since the RCS can evolve to a high void fraction for certain small breaks with RC pumps on, a RC pump trip by any means (i.e., loss of offsite power, equipment failure, etc.) at a high void fraction during the small break transient may lead to inadequate core cooling. That is, if the RC pumps trip at a time period when the system void fraction is greater than approximately 70 percent, a core heatup will occur because the amount of water

left in the RCS is not sufficient to keep the core covered. The cladding temperature then increases until core cooling is reestablished by the ECC systems.

> [The following discussion on RCP tripping does not represent a Design Basis Accident. It is included in Chapter 14 to provide insight as to how the plant would respond to a SBLOCA and for ease of reference. It also demonstrates compliance with 10CFR50.46.]

Conservative analyses using Appendix K assumptions during the course of the SBLOCA indicated that an early trip of RC pumps is required to show conformance to 10CFR50.46 for a range of break sizes.

It is desirable to maintain forced circulation cooling and mixing through operation of RC pumps during non-LOCA events. This is particularly true for steam generator tube rupture where the RC pumps aid in the mitigation of the transient. The parameter that was chosen as indication of a LOCA and the trip of all four RC pumps is subcooling margin. A loss of subcooling will always occur for small breaks that have the potential to uncover the core and violate 10CFR50.46 criteria if the RC pumps are tripped under certain two phase conditions.

A best-estimate (as opposed to Appendix K assumptions) of RC pump trip following a SBLOCA was performed using a simplified six node CRAFT2 model. It was used to evaluate the RCS thermal hydraulic response to a SBLOCA. The FOAM2 and THETA codes were used to calculate steam rates, core mixture level and cladding temperatures during the most limiting transient (Reference 56).

In summary the results of the best-estimate analyses indicate:

1. Following a SBLOCA, if the RC pumps remain operative, core cooling is assured regardless of system void evaluation. However, the continuous RC pump operation in a highly voided system is not desirable for pump integrity reasons.

2. Prompt tripping of the RC pumps upon receipt of indication of loss of subcooling margin will maintain peak clad temperature well below the limits of 10CFR50.46.

3. Based on realistic assumptions, an RC pump trip at any time following a SBLOCA for break sizes 0.05 ft^2 and smaller will not result in peak clad temperature exceeding the 10CFR50.46 limit.

4. For breaks 0.2 ft^2 and smaller, more than 10 minutes time is available for manual RC pump trip following indication of a loss of subcooling margin.

5. Small breaks larger than 0.2 ft^2 cause a rapid depressurization of the RCS and early actuation of the LPI system and core flood tank system. A delayed RC pump trip for break sizes larger than 0.2 ft^2 will not result in peak clad temperature exceeding the 10CFR50.46 limit.

6. As a result of realistic analysis, at least 10 minutes is available for the manual trip of the RC pumps following a small break LOCA without exceeding 10CFR50.46 limits.

A second best-estimate study was performed to determine if a manual RC pump trip could occur at any time following the indication of a loss-of-subcooling margin without fear of exceeding the 10CFR50.46 temperature limits. Detailed analyses were performed for the 0.10 ft^2 and 0.15 ft^2 pump discharge breaks at multiple RC pump trip times. The 0.10 ft^2 break with a RC pump trip at 1300 seconds was identified as the worst break size/trip time combination. Using the THETA1-B code to determine the peak clad temperatures indicated, the limits of 10CFR50.46 were exceeded (Reference 57). Therefore, it could not be concluded that failure of RCPs at the worst time during a small break LOCA would meet the criteria of 10CFR50.46. The results of the analyses showed that there was sufficient time for the operator to manually trip the RCPs upon a loss of subcooling margin. Manually tripping RCPs immediately following a loss of subcooling margin is an acceptable approach to resolve NUREG 0737, Action Item II.K.3.5 (Reference 58) and has been implemented in the Abnormal Transient Procedures.

3) Small Break LOCA Transient with Emergency Feedwater

a) LOCA Large Enough to Depressurize Reactor Coolant System

Curves 1 and 2 of Figure 14.2-49 show the response of RCS pressure to breaks that are large enough, in combination with ECCS functioning, to depressurize the system to a stable low pressure. ECCS injection easily exceeds core boil-off and ensures core cooling. Curves 1 and 2 of Figure 14.2-50 show the pressurizer level transient. Rapidly falling pressure causes the hot legs to saturate quickly.

Cold leg temperature reaches saturation somewhat later as RC pumps coast down or the RCS depressurizes below the secondary side saturation pressure. Since these breaks are capable of depressurizing the RCS without the aid of the steam generators, they are essentially unaffected by the availability of emergency feedwater.

b) LOCA Which Stabilizes at Approximately Secondary Side Pressure

Curve 3 of Figure 14.2-49 shows the pressure transient for a break which is too small in combination with the operating HPI to depressurize the RCS. The steam generators are, therefore, necessary to remove a portion of core decay heat. The primary system pressure will initially stabilize near the secondary side pressure because, with both the primary and secondary systems saturated, the steam generator will become a heat source to the primary system as RCS pressure equalizes with or becomes lower than steam generator pressure. Curve 3 of Figure 14.2-50 shows pressurizer level behavior. The hot leg temperature quickly equalizes to the saturated temperature of the secondary side and controls primary system pressure at saturation. The cold leg temperature may remain slightly subcooled. If the HPI refills and repressurizes the RCS, the hot legs can become subcooled.

c) LOCA Which May Repressurize in a Saturated Condition

Curve 4 of Figure 14.2-49 shows the behavior of a small break that is too small, in combination with the HPI, to depressurize the primary system. Although steam generator feedwater is available, the loss of primary system coolant and the resultant RCS voiding will eventually lead to interruption of natural circulation. This is followed by gradual repressurization of the primary system. It is possible that the primary system could repressurize as high as the pressurizer safety valve set point before the pressure stabilizes. This is shown by the dashed line in Curve 4. Once enough inventory has been lost from the primary system to allow direct steam condensation in the regions of the steam generators contacting secondary side coolant, the primary system is forced to depressurize to the saturation pressure of the secondary side.

Since the cooling capabilities of the secondary side are needed to continue to remove decay heat, RCS pressure will not fall below that on the secondary side. HPI flow is sufficient to replace the inventory lost to boiling in the core, and condensation in the steam generators removes decay heat energy. The RCS is in a stable thermal condition, and it will remain there until the operator takes further action. The pressurizer level response is characterized by Curve 3 of Figure 14.2-50 during the depressurization and Curve 4 of Figure 14.2-50 during the temporary repressurization phase.

The dashed on Figure 14.2-50 line indicates the level behavior if pressure is forced up to the pressurizer safety valve set point. During this transient, hot leg temperature will rapidly approach saturation with the initial system depressurization, and it will remain saturated during the whole transient. Cold leg temperature will approach saturation as circulation is lost but may remain slightly subcooled during the repressurization phase of the transient. Later RCS depressurization could cause cold leg temperatures to reach saturation. Subsequent refilling of the primary system by the HPI might cause temporary interruption of steam condensation in the steam generator as the primary side level rises above the secondary side level. If the depressurization capability of the break and the HPI is insufficient to offset decay heat, the primary system will once more repressurize. This decreases HPI flow and increases loss through the break until enough RCS coolant is lost to once more allow direct steam condensation in the steam generator. This cyclic behavior will stop once the HPI and break can balance decay heat or the operator takes some action.

d) Small LOCA Which Stabilizes at Pressure Greater Than Secondary Pressure

Curve 5 of Figure 14.2-49 shows the behavior of the RCS pressure to a break for which high pressure injection is being supplied and exceeds the leak flow before the pressurizer has emptied. The primary system remains subcooled and natural circulation through the steam generator removes core decay heat. The pressurizer never empties and continues to control primary system pressure.

4) Small Breaks in Pressurizer

a) System Pressure Transient

The system pressure transient for a small break in the pressurizer will behave in a manner similar to that previously discussed. The initial depressurization, however, will be more rapid as the initial inventory loss is entirely in the form of steam.

The pressurizer level response for these accidents will initially behave like a very small break without auxiliary feedwater. The initial rise in pressurizer level shown on Figure 14.2-51 will occur due to the pressure reduction in the pressurizer and an insurge of coolant into the pressurizer from the RCS. Once the reactor trips, system contraction causes a decreasing level in the pressurizer.

Flashing will ultimately occur in the hot leg piping and cause an insurge into the pressurizer. This ultimately fills the pressurizer. For the remainder of the transient, the pressurizer will remain full. Toward the later stages of the transient, the pressurizer may contain a two phase mixture and the indicated level will show that the pressurizer is only partially full.

b) Stuck Open PORV

An analysis of a 0.01-ft^2 break in the cold leg pump discharge piping, without emergency feedwater to the SG, was performed wherein the PORV was actuated and assumed to stick open.

Small breaks in the primary system will not cause a repressurization to the PORV setpoint unless all feedwater is lost to the steam generators. In this situation, there exists a class of very small breaks, (less than 0.01 ft^2) wherein the system will repressurize to the PORV setpoint. An analysis is presented herein for a 0.01 ft^2 break, without feedwater to the steam generator, which results in a repressurization to approximately the PORV setpoint. At 20 minutes the PORV was actuated and was assumed to stick open.

Figures 14.2-52 through 14.2-58 show the system response during the transient and Table 14.2-16 presents a sequence of events for this accident. The resultant system pressure response of a 0.01 ft^2 cold leg break with no EFW is shown on Figure 14.2-52. This particular response is due to (1) the loss of the SG heat sink; (2) no automatic HPI actuation prior to the loss of the steam generator heat sink; and (3) the opening of the PORV and actuation of the HPI at 20 minutes. As seen on Figure 14.2-52, the pressure initially decreases following the break opening. During this depressurization period, the reactor trips, the pumps trip, the pressurizer empties, and the steam generator secondary inventory boils off. With the loss of the SG heat sink the primary system starts to repressurize before the ESAS signal is reached. Therefore, the HPI is not automatically actuated. The system repressurizes to 2450 psia by 30 minutes at which time the PORV was assumed to open. This is only 115 psi below the PORV set point which would have been reached approximately 2 minutes later.

However, the operator is instructed to manually open the PORV if the system repressurizes and the SG heat sink is lost. Thus, the opening at 20 minutes is not totally arbitrary. During the system repressurization the pressurizer level increases (Figure 14.2-53) and when the PORV is opened the pressurizer rapidly fills with two

phase mixture. At the time of the PORV opening, the two HPI pumps are manually actuated, and due to the addition of the cold makeup water and the additional leak path area, the RCS depressurizes.

The inner vessel mixture height is shown on Figure 14.2-54. As can be seen, operator action within 20 minutes to manually actuate the HPI prevents the core from becoming uncovered, and a minimum two phase mixture level of 4.5 feet above the top of the core is maintained. Long term cooling is established at 25 minutes as the HPI injection rate exceeds the core boil off rate. Thus, the acceptance criteria of 10CFR50.46 are satisfied.

While the analysis performed herein addressed the effect of operator action to manually actuate the HPI by 20 minutes, the effect of operator action to manually restore the emergency feedwater within 20 minutes can be qualitatively assessed. As has been shown in Section 6.2.1.3.5 of Reference 44, actuation of the emergency feedwater system at 20 minutes for a 0.01-ft^2 break results in a rapid system depressurization and the subsequent actuation of the HPI. For the case analyzed herein, the depressurization effect of the emergency feedwater would be faster than that shown in Reference 44 due to the effect of the loss of inventory through the PORV. Thus, the HPI would be actuated earlier and long term cooling would be established faster than that shown in Reference 44. Therefore, no core uncovering is expected if the operator only actuates the emergency feedwater system within 20 minutes and, contrary to the Small Break LOCA operators guidelines, does not manually actuate the HPI.

c) Pressurizer Safety Valve Stuck Open

The small break resulting from the failure of a pressurizer code safety valve needs no special analysis because of three main points: a) challenges to these valves are infrequent events, b) the valves are designed to relieve and reseat without leakage, and c) the leak that would result if one (or both) of these valves stuck open is bounded by already existing analyses.

Reducing the reactor overpressure trip setpoint to 2300 psig, further reduced the probability of challenging the code safety valves (reference NUREG-0737, II.K.3.2.and II.K.3.7). In fact, it was concluded from the analyses that neither the PORV (2450 psig) nor the code safety valves (2500 psig) are challenged by the bounding anticipated transients. The actual reactor protection high pressure trip setpoint was raised to 2355 psig in order to reduce the number of reactor trips. This higher setpoint was evaluated and meets the requirements of NUREG-0737, II.K.3.2 and II.K.3.7.

The design of the pressurizer safety valves is discussed in Chapter 4. Even in the unlikely event that the pressurizer code safety valves were opened to relieve RC pressure, and if one (or both) of those valves were to stick open, the resulting break would be bounded by existing LOCA analyses as confirmed in the following discussion. The general phenomena observed for the PORV stuck open cases would be observed for this break. However, since the area of the pressurizer safety valve is larger, the phenomena would occur earlier than that seen for the PORV breaks. Generally, the system behavior can be characterized by:

- A rapid system depressurization due to steam relief via the safety valve. This will result in reactor scram and ESAS actuation.

- The indicated pressurizer level will initially increase due to the break. After reactor scram, the pressurizer level decreases due to system contraction. Following saturation of the hot legs, there will be an insurge into the pressurizer resulting in the pressurizer going solid.

- After the pressurizer is filled by a two-phase mixture, a low quality mixture will be discharged through the valve. This will result in a large increase in the leak flow rate.

- Primary side pressure will ultimately be maintained at approximately 1200 psia.

- Long term cooling will be established at approximately 1000 seconds by one HPI train.

A stuck open pressurizer safety valve is bounded by the small break analyses performed at the cold leg RC pump discharge piping. The vent valves located in the reactor vessel cause the primary system to depressurize uniformly during a LOCA, maintaining equivalent mixture level in the cold and hot sides of the RCS. A break of a given size would, therefore, yield approximately the same leak flow rate in any part of the system. During a LOCA in the cold leg pump discharge piping, a fraction of the HPI flow, which injects at the pump discharge, is lost through the break. Thus, the core is cooled more effectively for breaks located in the hot leg piping and the pressurizer, since 100 percent of the HPI flow is delivered to the reactor vessel, and long term cooling is attained earlier. Therefore, the inventory loss from the system due to boiling is greater for breaks in the cold leg RC pump discharge piping and the results bound breaks located in other parts of the primary system.

5. Small Break LOCA Transients Without Emergency Feedwater

a) LOCAs Large Enough to Depressurize Reactor Coolant System

For Class 1 (Curve 1 of Figure 14.2-59), RC system pressure decreases smoothly throughout the transient. For the larger breaks in this class, CFT actuation and LPI injection will probably occur. For the smaller breaks of this class, only CFT actuation will occur. Emergency feedwater injection is not necessary for the short term stabilization of these breaks. The pressurizer level for this transient rapidly falls off scale.

b) LOCAs Which Reach a Semi-Stabilized State

For Class 2 (Curve 2 of Figure 14.2-59) breaks, the RC pressure will rapidly reach the low pressure ESAS trip signal (about two to three minutes). With the HPIs on, a slow system depressurization will be established coincident with the decrease in core decay heat. No CFT actuation is expected. Emergency feedwater is not necessary for the short term stabilization of these breaks. The pressurizer level for this transient rapidly falls off scale.

c) Small LOCAs Which do Not Actuate the ESAS

Note: This section describes the plant response to a small break LOCA that was performed without credit for emergency feedwater system actuation on either low OTSG level or on high (4 psig) containment pressure. This analysis was also performed before the RCP trip criteria was changed to loss of subcooling margin. With RCP's operating, the RCS will only begin to repressurizer when OTSG level is approaching the HSPS low level initiation setpoint.

Automatic ESAS actuation will not occur for Class 3 (Curve 3 of Figure 14.2-59) breaks. Once the SG secondary side inventory is boiled off, system repressurization will occur because the break is not capable of removing all the decay heat being generated in the core. System repressurization to the PORV or the pressurizer safety valves set points will occur for smaller breaks in this class. For the "zero" break case, repressurization to the PORV set point will occur in the first five minutes. Operator action is required within the first 20 minutes to ensure core coverage throughout the transient. This action can be either manual actuation of the emergency feedwater system or the HPI system.

The establishment of emergency feedwater will rapidly depressurize the RCS to the ESAS actuation pressure, and system pressure will stabilize at either the secondary side SG pressure or at a pressure where the HPI equals the leak rate. Upon receipt of the low pressure ESAS signal, the operator must trip the RC pumps. Further analyses demonstrated that tripping RC pumps based on subcooled margin is acceptable. Discussion on this treatment of RC pump trip following a SBLOCA is included in Section 14.2.2.4.3.d.2.

For the Class 3 breaks, pressurizer level response will be as shown on Figure 14.2-60. The minimum refill time for the pressurizer is that for the "zero" break and is shown on Figure 14.2-60. After initially drawing inventory from the pressurizer, the system will be repressurized, which will cause the pressurizer level to increase, possibly to full pressurizer level. Once the operator action to restore emergency feedwater has been taken, the system depressurization will result and cause an outsurge from the pressurizer. Complete loss of pressurizer level may result. For the smaller breaks in Class 3 which result in a system repressurization following the actuation of the HPI system, pressurizer level will increase and then stabilize.

Without emergency feedwater, both the hot and cold leg temperatures will saturate early in the transient and, for the Class 1 and 2 breaks, will remain saturated. For the Class 3 breaks, once emergency feedwater is established, the cold leg temperatures will rapidly decrease to approximately the saturation temperature corresponding to the SG secondary side pressure and will remain there throughout the remainder of the transient. Hot leg temperatures will remain saturated throughout the event.

14.2.2.4.4 Post Analysis-of-Record Evaluations for Small Break LOCA

In addition to the analyses presented in Subsection 14.2.2.4.3, evaluations and reanalyses may be performed as needed to address emergent issues or to support plant changes. The issues or changes are evaluated, and the impact on the PCT is determined. The resultant increase or decrease in PCT is added to the analysis of record PCT (1444°F for the limiting fuel type

currently in use). These issues and their evaluations are reported to the NRC via the normal 10CFR50.46 reporting requirement. The latest 10CFR50.46 report is on file at the site.

The current peak clad temperature for a small break LOCA, including all penalties and benefits for evaluations/reanalyses performed since the analysis-of-record, is 1669°F.

14.2.2.5 Maximum Hypothetical Accident

a. Identification of Accident

The analyses in the preceding sections have demonstrated that even in the event of a loss of coolant accident, no significant core melting will occur. However, to further assure that the operation of a nuclear power plant at the site does not present any undue hazard to the general public, an accident involving a gross release of fission products is evaluated. Fission products are assumed to be released from the core as stated in Regulatory Guide 1.183. Decay of fission products was assumed to occur while they were confined to the Reactor Building, but was not assumed to occur once they passed to the environment. Other parameters such as meteorological conditions, iodine inventory of the fuel, Reactor Building leak rate, etc., are the same as assumed for the loss of coolant accident in Subsection 14.2.2.3.4. The iodine activity and the noble gas activity released to the environment are shown in Tables 14.2-18 and 14.2-19. Accident analysis parameters and assumptions are tabulated in Table 14.2-20.

b. Analysis and Results of Environmental Analysis

As an upper limit on the consequences of this accident, it can be postulated that the leakage prevention systems are not completely effective in terminating all leakage. For this condition, leakage is assumed to continue at the design leak rate, and reduced after 24 hours to one half of its original value. The 2 hour thyroid dose at the exclusion distance (Appendix 14C) and the 30 day dose at the low population zone distance as summarized in Table 14.2-20 are less than the guideline values of 10CFR100.

The direct dose to the whole body following the accident is shown on Figure 14.2-61. No significant dose exists from this source at the exclusion distance. The dose to the whole body from the passing cloud has been calculated using the same meteorological conditions used for determining the thyroid dose. The whole body dose at the exclusion boundary (Appendix 14C) and low population zone distance as summarized in Table 14.2-20 are less than the guideline values of 10CFR100.

c. Iodine removal sensitivity analysis using various reactor building spray and fan cooler combinations.

In Appendix 14C the 2 hour exclusion area boundary TEDE doses were calculated for the following reactor building cooling system combinations:

1) One spray header pump and one air cooling unit fan operating

2) Two spray header pumps and two air cooling unit fans operating

The spray removal coefficients and decontamination factors developed in Appendix 14B were used in the sensitivity analysis. It is concluded that the analyzed combinations of reactor building cooling systems have an insignificant effect on the whole body dose and that the use of the second combination of systems will result in a larger TEDE dose.

d. Effects of Engineered Safeguards 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 safeguards external to the Reactor Building during the recirculation phase for long term core cooling. A detailed analysis of the potential leakage from these systems is presented in Section 6. With the exception of noble gases, all the fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the primary containment sump water at the time of release from the core. The total allowable ESF leakage from all components in the ESF recirculation systems is 15 gph. This ESF leakage is doubled and assumed to start at 29.44 minutes after onset of a LOCA. This corresponds to a 75 second building spray system startup time plus a 28.19 minute pump run time prior to BWST depletion (start of sump recirc). It should be noted that the 28.19-minute run time of the building spray pumps from the BWST is based on a pump flow rate of 1,250 gpm per pump. Assuming this while running the pumps at 800 gpm each is conservative since this causes ESF leakage to start sooner, thereby maximizing dose. A single spray pump operation in the sump recirc mode is conservative (i.e., yields higher dose). With the exception of iodine, all remaining fission products in the recirculating liquid are assumed to be retained in the liquid phase.

The conservative analysis assumes that ESF leakage into the Auxiliary Building occurs at a rate of 30 gph, twice the limit of Technical Specification 4.5.4. Once this leakage enters the building, some fraction of the iodine in the coolant goes airborne in the building and is ultimately released to the environment. The fraction used in the analysis was that agreed upon between Exelon and the NRC staff.

Radioactive decay is assumed for both airborne iodine and noble gas inventory from the occurrence of the MHA up to the beginning of recirculation when the sump water is circulated outside containment. Atmospheric dilution is calculated using the dispersion factors developed in Section 2.5. The leakage and resulting TEDE doses are shown in Table 14.2-20.

Another source of fission product leakage during the maximum hypothetical accident can occur from back-leakage to the BWST through system valves during the recirculation phase for long-term core cooling. The analysis assumes that leakage to the BWST begins at the onset of sump recirculation and starts out at a rate of 3 gpm, then decreases with time. Once leakage enters the BWST, it is assumed that 10% of the iodine in the coolant goes airborne in the tank. It is assumed that there are 300,000 gallons of empty space in the BWST when sump recirculation begins. The volumetric flow rate of air leaving the BWST is equal to the volumetric flow rate of leakage into the BWST. No credit is taken for iodine plateout in the BWST. Atmospheric dilution is calculated using the dispersion factors developed in Section 2.5. The leakage and resulting TEDE doses are shown in Table 14.2-20.

14.2.2.6 Waste Gas Tank Rupture

Rupture of a waste gas tank would result in the release of its radioactive contents to the Auxiliary Building environment.

A tank is assumed to contain the gaseous activity evolved from degassing all of the reactor coolant following operation with 1 percent defective fuel. The reactor coolant passes through purification demineralizers which remove 99 percent of the iodine, however, no credit is assumed for this iodine removal. The coolant is then degassed an additional 99 percent according to the liquid/gas partitioning for iodine. The resulting waste gas inventory is 1 percent of the iodine and all of the noble gas activities associated with one reactor coolant volume. All of this activity is assumed to be released to the Auxiliary Building and then to the environment as an instantaneous puff release. No radioactive decay is accounted for and no removal mechanisms for noble gases are assumed. The resulting leakage to the environment is 1 percent of the iodine and all of the noble gas activity associated with one reactor coolant volume. No credit for iodine removal by the Auxiliary Building ventilation charcoal filters is assumed in this analysis.

The gaseous activity in the tank is listed in Table 14.2-21. The total integrated doses at the exclusion area boundary are:

2 Hour Doses at Exclusion Area Boundary

Thyroid Dose	7.21 Rem
Whole Body Dose	2.11 Rem

These doses are well below the limits of the 10CFR100 guideline.

14.2.2.7 Loss Of Feedwater Accident

a. Identification

A loss of feedwater may result from abnormal closure of the feedwater isolation valves, control valve failure, or pump failure. The loss of feedwater flow results in a loss of heat sink, primary system heatup, increased pressurizer level and pressure, and reactor trip on high RCS pressure.

Acceptance Criteria

For the transient analyses, the acceptance criteria chosen were Prevention of Pressurizer Fill and, Prevention of Saturated Condition in the RC Hot Leg. The general criteria are as follows:

1) Core thermal power shall not exceed 112 percent of rated power.

2) RCS pressure shall not exceed 2750 psig.

3) Pressurizer does not become water solid during a loss of feedwater transient.

b. Analysis and Results

The events have been analyzed and are as follows:

1) Loss of main feedwater resulting in a stuck open PORV, no loss of offsite power, and one HPI train available for emergency core cooling.

2) The loss of feedwater analysis to establish acceptable EFW flow requirements.

3) The effects of feedwater line break outside the containment.

4) Evaluation of Anticipatory Reactor Trip (ART) for loss of feedwater events.

Results:

1) Loss of Main Feedwater Resulting in a Stuck Open PORV, No Loss of Offsite Power, and One HPI Train Available For Emergency Core Cooling.

Compliance with 10CFR50.46 for small break LOCA's is demonstrated in Section 14.2.2.4.2.

2) The Loss of Feedwater Analysis to establish acceptable EFW Flow Requirements.

The events which require emergency feedwater are:

a) Loss of coolant flow

b) Loss of offsite power (LOOP)

c) Main steam line break (MSLB)

d) Small break LOCAs (certain sizes)

e) Loss of feedwater (LOFW)

The most demanding event in terms of the need for heat removal via EFW is the LOFW without LOOP since this event requires the removal of reactor coolant pump heat as well as decay heat.

This Section of the FSAR evaluates that event.

The acceptance criteria for the loss of feedwater accident without anticipatory reactor trip are that:

1. RCS pressure will be limited to less than 110% of the RCS design pressure (2750 psig.)
2. Reactor thermal power shall not exceed 112% of rated power.

3. The pressurizer will not become water solid.

These acceptance criteria assure that the RCS pressure boundary and fuel integrity are not challenged. Failure of the RCS pressure boundary is prevented by limiting pressure/temperature conditions to stay within analyzed limits. The pressurizer safety valve is capable of performing its safety function while passing single phase liquid, depending on the thermodynamic state of the inlet fluid and the design inlet piping configuration. However, an acceptance criterion that prevents a water solid pressurizer precludes the need to address these issues. Limiting thermal power to 112% of rated power is a simple and conservative means of assuring that there is no fuel cladding failure.

Two separate loss of feedwater transients have been analyzed in this section. First, the event is analyzed to determine peak RCS pressure. No credit is taken for the PORV or pressurizer sprays. The second event determines a worse case pressurizer level. The PORV and spray perform as modeled in this analysis. Actuation of the PORV to control system pressure would increase the liquid insurge to the pressurizer by venting steam from the pressurizer at a lower pressure than the pressurizer safety valves. Pressurizer spray flow could worsen the pressurizer liquid level response during the event by condensing the pressurizer steam bubble. Consequently, the LOFW accident was analyzed with both the pressurizer spray and PORV to provide a conservative prediction of pressurizer liquid level.

Table 14.2-22 summarizes the key input parameters for the loss of feedwater transient.
For the both of the analyzed cases, the worst single failure is loss of one train of the Heat Sink Protection System (HSPS), which both prevents the automatic start of one EFW pump and the opening of one parallel injection valve to each OTSG. Consequently, the flow from two EFW pumps delivering flow through one valve per OTSG is used in this analysis. Once the EFW pumps have coasted up and the control valves open, the system begins delivering flow to the OTSGs. The HSPS attempts to maintain level at 25 inches. However, the EFW regulating valves remain full open for the duration of the transient, since OTSG level setpoint is not established up to the point of transient termination.

Shortly after reactor trip, only decay heat, RC pump and sensible heat from the RCS metal and fuel are added to the reactor coolant. Initially, the EFW flow is not sufficient to remove decay and pump heat, and the system stays pressurized with pressure relief through the pressurizer safety valves. Subsequently, the OTSG heat removal approaches the heat generated. Therefore, the reactor coolant continues to expand until the EFW heat removal equals the heat added to the RCS. Subsequent to this point in time, the reactor coolant will contract causing RCS pressure to decrease.

The peak RCS pressure is below the acceptance criteria of 2750 psig, and the pressurizer does not become water solid for either of the analyzed cases. Based on these results, the total EFW flow from 2 EFW pumps is considered acceptable for mitigating this event.

<u>Loss of Feedwater Accident With and Without PORV and Spray</u>

The rapid loss of flow to the OTSGs causes an immediate reduction in OTSG secondary tube region level and a consequent reduction in heat transfer. Reduction of primary-to-secondary heat transfer causes T_{hot} and T_{cold} in both loops to increase because OTSG heat transfer is inadequate to remove primary system heat. This temperature increase causes a reactor

coolant fluid expansion that compress the pressurizer steam space, causing the RCS pressure to increase.

The sequence of events for these cases are shown on Table 14.2-22, sheet 3.

The trip of the reactor and the resultant, momentary drop in average RCS temperature causes a shrink of RCS inventory and a decrease in RCS pressure. Reactor trip also initiates a turbine trip, so that steam flow out of the OTSGs is reduced, as is heat removal. Reactor power does not get higher than the initial value of 102% of 2568 Mwt, and so thermal power will not exceed this value. EFW is initiated on low OTSG level and conservatively assumed to start delivering flow at 43 seconds after the initiation signal with any combination of two EFW pumps (either 2 MDP's, or 1TDP and 1MDP). The EFW flow is modeled as a function of OTSG pressure. Initially, the EFW flow is not sufficient to remove decay and pump heat, and the system stays pressurized with pressure relief through the pressurizer PORV and/or Safety valves. After a delay, the OTSG heat transfer exceeds the heat generated and the RCS pressure and temperatures start to decrease.

The maximum RCS pressure (without PORV or spray) is shown on Table 14.2-22, sheet 3.

In the case allowing for pressurizer spray and PORV, the spray starts on high pressure and continues until a low pressure signal is reached. The PSV setpoint is reached quickly and the PORV cycles several times in the first few minutes of the event. The pressurizer level continues to increase due to spray condensing steam and transporting mass to the pressurizer from the cold leg. Moreover, PORV operation causes a fluid insurge to the pressurizer by reducing pressure in the steam space. Pressurizer level reaches a maximum slightly below the inlet of the PORV/PSV (see Table 14.2-22, sheet 3).

For both cases, the small main steam safeties control OTSG secondary pressure.

The results demonstrated that the acceptance criteria would be met with an EFW flow rate of 550 gpm (total delivered) to both steam generators at 1065 psia (Reference 128).

Main feed line break is a somewhat more abrupt case of LOFW. However, from a long term cooling standpoint, the heat removal requirements are identical to those occurring during an LOFW. Therefore, although the initial response to a feed line break may be more severe, the emergency feedwater sizing requirements based on LOFW considerations assure sufficient heat removal capacity to mitigate the line break accident. Therefore, the results of an LOFW and feed line break accident are essentially the same.

<u>Reanalyses for Replacement OTSGs</u>

The original OTSGs were replaced at the end of cycle 17. Specific loss of feedwater analyses without LOOP were performed with the NRC-approved RELAP5/MOD2-B&W systems analysis code to determine if the LOFW event with replacement OTSGs continues to meet the applicable acceptance criteria (Reference 128). The results described in Section 14.2.2.7, including the key parameters and sequence of events in Table 14.2-22, reflect the revised replacement OTSG analyses. The results of these analyses currently limit the amount of tube plugging in the replacement OTSGs to five percent.

1) The Effects of Feedwater Line Break Outside the Containment

This analysis is performed in Appendix 14A and in Section 14.2.2.9.

2) Evaluation of Anticipatory Reactor Trip (ART) For Loss of Feedwater Events

The primary purpose of ARTs is to reduce the probability of lifting the PORV for turbine trip/loss of main feedwater type events. The anticipatory reactor trip setpoints are described in Sections 7.1.2.2.c. 6) “Reactor Pressure Trip” and 7.1.2.2.c. 8) “Anticipatory Trip (Turbine Trip or FW Pump Trip)”, and Table 7.1-1.

14.2.2.8 Fuel Cask Drop Accident

a. Identification

A fuel cask drop accident is defined as the dropping of a fuel cask through the maximum drop height during transfer operations of a fuel cask onto a rail car. A fuel cask drop into the spent fuel pool is prevented by the Technical Specification requirement that the key operated travel interlock system for automatically limiting the travel area of the Fuel Handling Building crane shall be imposed whenever loads in excess of 15 tons are lifted and transported.

b. Analysis and Results

The analysis performed to determine the radiological consequences at the TMI site boundary for a gross release of activity from a fully loaded fuel cask is shown in Table 14.2-25 (Reference 64). The maximum height of a fuel cask during transfer operations at TMI-1 exceeds SRP 15.7.5 limiting criteria of 30 feet. Thus, the radiological consequences of a fuel cask drop are evaluated. It is assumed that the cask and its entire contents of ten fuel assemblies are sufficiently damaged to allow the escape of all the noble gases and iodine in the gap (See Table 14.2-25). The gap activity released from the fuel cask is based on a decay time of 120 days which is the minimum time before final assemblies can be loaded into a cask as required by Technical Specification. All 208 fuel pins in each of the 10 fuel assemblies are assumed ruptured.

The release of the noble gases and iodine is assumed to be directly to the atmosphere and to occur instantaneously. An atmospheric dispersion factor of $1.2x10^{-3}$ s/m^3 at the exclusion area distance (610m) for a 0-2 hour time period was assumed.

The resulting site boundary doses shown in Table 14.2-26 (Reference 64), even though calculated using this conservative approach, are still well within the limits specified in 10CFR100. Based on the results of this analysis, it is the applicant's opinion that no additional means need be provided to prevent dropping the fuel cask.

The Fuel Cask Drop Accident analysis is not affected by the operation of any reactor systems since the handling of irradiated fuel is not directly linked to reactor operation.

14.2.2.9 Feedwater Line Break Accident

A main feedwater line break accident (FWLB) is a piping rupture in the main feedwater system.

Feedwater is lost abruptly, as opposed to the flow coastdown that can occur for a LOFW. A FWLB break creates a loss of heat sink, a primary system heatup, increased pressurizer level and pressure, and a reactor trip on high RCS pressure.

The rapidly increasing RCS temperature and pressure of this event introduces the possibility for the RCS to fill water-solid. Since water is only slightly compressible, the water-solid condition has the potential to cause failure of the RCS pressure boundary or a fuel cladding failure. This situation would occur if volumetric relief out of the system is less than the volumetric expansion of the system caused by the rapid heatup. Reactor protection for these events is provided by the high RCS pressure trip function of the reactor protection system (RPS). Primary system pressure is initially controlled by the combined action of the high pressure trip and by volumetric relief through the pressurizer safety valve. Restoration of the secondary system heat sink occurs with emergency feedwater flow and steam relief through the main steam safety valves. When the RCS pressure reaches the high RCS pressure or the high Reactor Building setpoint, the reactor is tripped. Shortly after reactor trip, only decay and pump heat are added to the reactor coolant. Initially, the boiloff of EFW flow is less than the energy produced by decay, sensible, and RC pump heat. Therefore, the reactor coolant will continue to expand until the EFW heat removal matches heat input to the RCS. Subsequent to this point in time, RCS pressure will decrease and the reactor coolant will contract.

Two FWLB cases are analyzed to establish the most severe consequences: one postulated break is inside containment, and one postulated break is outside containment. The first case assumes a complete feedwater line break upstream of the main feedwater check valves closest to containment. Break locations can either be in the Intermediate or Turbine Buildings. A break outside containment causes an immediate loss of flow and also produces a harsh environment for the emergency feedwater system.

The break inside containment causes one OTSG to depressurize from the break and, in addition, creates a harsh environment inside containment, as well as creating a high reactor building pressure signal. This signal initiates reactor trip, and starts both EFW and high pressure injection. Although these signals are not necessarily modeled for the FWLB accident analysis, their effect is considered in determining the most bounding plant response.

The feedwater line break is a limiting fault transient, and the acceptance criteria are:

1. Reactor coolant system pressure shall not exceed code allowable limits of 2750 psig.
2. The reactor thermal power shall not exceed 112% of rated power.
3. No steam generator tube break or separation from the tube sheet will occur.

Table 14.2-29 (sheet 1) summarizes the analysis input values used. These values bound the TMI specific values.

Reanalyses for Replacement OTSGs

The original OTSGs were replaced at the end of cycle 17. Specific feedwater line break accident analyses were performed with the NRC-approved RELAP5/MOD2-B&W systems analysis code to determine if the FWLB event with replacement OTSGs continues to meet the applicable acceptance criteria (Reference 129). The results described in Sections 14.2.2.9.1 and 14.2.2.9.2, including the key parameters and sequence of events in Table 14.2-29, reflect

the revised replacement OTSG analyses. The results of these analyses currently limit the amount of tube plugging in the replacement OTSGs to five percent.

14.2.2.9.1 Feedwater Line Break Outside Containment

The sequence of events for this event is shown on Table 14.2-29 (sheet 2).

The feed line break outside containment is assumed to be upstream of the feedwater check valve. This break location could be such as to create a harsh steam environment in the intermediate or turbine buildings. Only two EFW pumps are assumed to be available resulting from a single active component failure. The combination assumed in this analysis results in the smallest EFW flow to the steam generators. The feedwater check valve prevents blowdown of either OTSG. Therefore, the event behaves like a rapid loss of feedwater.

A feedwater line break is assumed to occur while the reactor is at full power steady state operation resulting in a loss of feedwater to the SGs in 0.1 seconds. The feedwater piping upstream of the feedwater nozzles is not modeled. The rapid loss of flow to the OTSGs causes an immediate reduction in OTSG tube level. The loss of inventory in the tube region causes a reduction in heat transfer from the RCS causing T_h and T_c in both loops to increase.

Increasing RCS temperatures increase RCS liquid volume and, therefore, pressurizer level. Increasing level compresses the pressurizer steam bubble, increasing RCS pressure until the reactor trips on the high RCS pressure trip signal. The trip of the reactor and the resultant, momentary drop in average RCS temperature causes a shrink of RCS inventory and a decrease in RCS pressure. Reactor trip also initiates a turbine trip, so that steam flow out of the OTSGs is reduced, as is heat removal. Reactor power does not get higher than the initial value of 102% of 2568 Mw(t), and so thermal power will not exceed this value.

EFW is initiated when the OTSG low level setpoint (including instrument error) is reached. A 43 second delay is assumed after the low level setpoint is reached based on the turbine-driven pump (TDP) delivery time. Both OTSG's are pressurized, and MS-V-13A would be available to deliver steam to the TDP. This assumed delay is conservative since a motor-driven pump (MDP) would be ready to deliver flow in about 15 seconds after the initiating signal (5 second delay after initiation signal + 10 second coastup). Also as mentioned above, only two EFW pumps were credited in the analysis, and the flow delivered to each OTSG as listed in Table 14.2-29 (sheet 3). The flow is maintained constant for the duration of the transient, as no level is reached in the OTSGs up to the point of transient termination. OTSG pressure is controlled within code limits by the MSSVs.

Initially EFW flow is not sufficient to remove combined decay, sensible, and pump heat, and the system pressurizes with pressure relief through the pressurizer safety valves. Eventually the heat transfer to the secondary is sufficient to balance the decay and pump heat and the RCS temperature and pressure begin to decrease. The pressurizer level begins decreasing before the pressurizer becomes water solid. RCS pressure remains below code allowable pressure.

14.2.2.9.2 Feedwater Line Break Inside Containment

The sequence of events for this event is shown on Table 14.2-29 (sheet 2).

An instantaneous break inside containment is modeled at the OTSG just upstream of the feedwater nozzles. The break area is limited to the total area of the 32 feedwater nozzles, and is assumed to be on the ‘A’ OTSG. The generator depressurizes rapidly after the break, and there is no steam available to run the turbine-driven EFW pump until MS-V13B opens, providing motive power from the “B” OTSG. The depressurization of the upstream piping is assumed to cause immediate loss of feedwater flow to the “B” OTSG, with the feedwater check valve closing to prevent backflow out of the “B” OTSG. The large loss of mass from the “A” OTSG causes an immediate reduction in OTSG tube level in that generator. The loss of inventory in the tube region causes a reduction in heat transfer from the primary to secondary causing T_h and T_C to increase.

EFW is initiated on a low OTSG level signal. A high RB building pressure signal would occur prior to the low level signal but was conservatively ignored. However, the start of one MDP (the other MDP is assumed to fail as discussed below) is delayed based on the electrical loading sequence because of the high RB pressure ESAS signal. Because of the time delay associated with the opening of MS-V-13B, the worst single failure for this accident is the loss of 1 Heat Sink Protection System (HSPS) channel, which causes the failure of on motor-driven pump and one EFW parallel injection valve leading to each OTSG. Because the pressure is decreasing in the broken SG, all the flow from the MDP is assumed to go to this SG. For this analysis it is conservatively assumed that no EFW flow is available to either OTSG until the TDP starts delivering flow.

HPI would be initiated on the 4 psig building pressure signal. The accident model does not include a detailed containment model. Therefore, HPI flow is delayed by 20 seconds to conservatively account for both RB building pressure increase and the maximum stroke time of the HPI injection valves MU-V-16A/B/C/D. The HPI flow rate (see Table 14.2-29 (sheet 3)) is conservatively based on only one makeup pump, since a higher HPI flow will result in more core cooling.

The level decrease of the B OTSG is less rapid than the A OTSG. Increasing RCS temperatures increase the RCS liquid volume and pressurizer level. The RB high RPS pressure trip setpoint is reached before the RCS high pressure trip, but conservatively is not credited in this analysis. Reactor trip also initiates a turbine trip. Reactor power does not get higher than the initial value of 102% of 2568 Mw(t), and so thermal power will not exceed this value.

The TDP and MDP start delivering flow, with EFW flow as a function of pressure as shown on Table 14.2.29 (sheet 4). The “B” SG pressure is controlled by the lowest set main steam safety valve on each OTSG, while the “A” SG quickly depressurizes to near atmospheric conditions as a result of the FWLB.

Initially, the EFW flow is not sufficient to remove decay, sensible and RC pump heat, and the system stays pressurized with pressure relief through the pressurizer safety valves. The pressurizer goes water solid and stays solid, resulting in liquid relief through the safety valves. The pressurizer safety valves have sufficient capacity to limit the RCS pressure to below the code allowable pressure. While not credited in this analysis, the operator could trip two RCPs at 10 minutes, which would further decrease the heat load to the RCS and result in a faster cooldown of the RCS.

TABLE 14.2-1
(Sheet 1 of 1)

SITUATIONS ANALYZED AND CAUSES

Event	Analysis Assumptions	Effect
Fuel handling accidents	a. In the Fuel Handling Building the gap activity is released from the outer row of fuel rods in one assembly while in spent fuel storage pool. No retention of noble gases and only 99% retention of iodine are considered.	See Table 14.2-3 for environmental effects
	b. In the Reactor Building the gap activity is released from all of the rods in the highest powered assembly in the core.	The activity released is in Table 14.2-5
Rod ejection accident	All fuel rods which experience DNB are assumed to release their total gap activity to the reactor coolant (following operation with 1% defective fuel).	Some fuel clad failure. See T.14.2-11 for environmental effects
Loss of coolant accident	Reactor Coolant leakage through a spectrum of breaks within the RCS boundary. Environmental effects are based on the release of all the gap activity with the reactor operating with 1% failed fuel.	Clad temperature remains below 2200F. Analyses have been performed for large break LOCA and small break LOCA. HPI design includes cavitating venturis.
Maximum hypothetical accident	Release of noble gases, iodine, and solid fission products per Reg. Guide 1.183.	See Table 14.2-20 for environmental effects

TABLE 14.2-2
(Sheet 1 of 1)

RADIOACTIVE RELEASE FOR THE FUEL HANDLING ACCIDENT IN THE FUEL HANDLING BUILDING

Isoptope	Core Initial Inventory (curies)	Activity in Damaged Spent Fuel Assembly (curies)	Post-FHA Activity in FHB (curies/MWth)
KR-85*	1.05E+06	1.00E+04	2.064
KR-85M	2.33E+07	2.24E+05	22.99
KR-87	4.60E+07	4.41E+05	45.38
KR-88	6.48E+07	6.23E+05	64.00
I-131**	7.15E+07	6.87E+05	0.5646
I-132	1.03E+08	9.92E+05	0.5099
I-133	1.50E+08	1.44E+06	0.7380
I-134	1.66E+08	1.60E+06	0.8205
I-135	1.39E+08	1.34E+06	0.6882
Xe-131M	7.17E+05	6.89E+03	0.7079
Xe-133M	4.56E+06	4.38E+04	4.502
Xe-133	1.50E+08	1.44E+06	148.1
Xe-135	5.51E+07	5.29E+05	54.40
Xe-135M	2.85E+07	2.74E+05	28.14

Note:
* KR-85 activity is multiplied by a factor of 2 to account for additional fractional release.
** I-131 activity is multiplied by a factor of 1.6 to account for additional fractional release.

Number of damaged fuel rods = 56
Activity decay time allowed = 72 hours
99.5% of Iodines are retained in the pool (DF = 200)

TABLE 14.2-3
(Sheet 1 of 1)

FUEL HANDLING ACCIDENT PARAMETERS AND RESULTS
(IN THE FUEL HANDLING BUILDING)

Power level (MWt) 1.02 * 2568	2619
Damaged fuel rods in the assembly	56
Spent fuel pool water Decontamination factor (Iodine)	200
Atmospheric dilution factor sec/m^3	8.0×10^{-4}
Total activity release time (hour)	2

	FHA Occurring in Fuel Handling Building Without FHB Exhaust Filtration Post-FHA TEDE Dose (Rem)		
	Control Room	EAB	LPZ
Calculated Dose	0.669	1,21	0.211
Allowable Dose	5.000	6.30	6.30

TABLE 14.2-4
(Sheet 1 of 1)

FISSION PRODUCT INVENTORIES[1, 4]
FOR THE CORE, TOTAL GAP,
AND THE REACTOR COOLANT SYSTEM

Activity (Curies)

Isotope	Total Core[2] Inventory	Total GAP[2] Activities	RCS Activity[3] (ûCi/gr)
Kr-83m	1.02×10^7	1.24×10^4	0.534
Kr-85m	2.40×10^7	5.49×10^4	2.43
Kr-85	6.93×10^5	5.90×10^5	9.75
Kr-87	4.39×10^7	2.89×10^4	1.28
Kr-88	6.15×10^7	8.93×10^4	3.95
Xe-131m	4.87×10^5	7.10×10^4	2.68
Xe-133m	3.38×10^6	9.97×10^4	4.22
Xe-133	1.40×10^8	9.02×10^6	392.0
Xe-135m	3.63×10^7	1.34×10^4	.485
Xe-135	2.40×10^7	2.17×10^5	8.37
Xe-138	1.29×10^8	1.56×10^4	0.692
I-131	8.17×10^7	1.62×10^6	5.71
I-132	9.53×10^7	1.74×10^5	1.93
I-133	1.41×10^8	3.09×10^5	6.07
I-134	1.77×10^8	1.80×10^4	.757
I-135	1.40×10^8	9.75×10^4	3.08

(1) Based on Cycle 7 Bounding Analysis at 2568 MWth and at 460 EFPD. The source term is deliberately increased by applying a conservatism factor of 1.1. Subsequent two-year cycle designs may result in some of the isotopic activities in this table being exceeded. However, reload evaluations are performed each cycle to ensure that dose consequences for all affected accidents are not increased.
(2) Zero decay time
(3) Based on 1% failed fuel
(4) Not applicable to FHA in Reactor Building or LOCA (see Table 14.2-4a)

TABLE 14.2-4a
(Sheet 1 of 1)

FISSION PRODUCT CORE INVENTORY[1,2]
FOR FHA IN REACTOR BUILDING AND LOCA

Activity (Curies)

Isotope	Total Core Inventory	Isotope	Total Core Inventory
KR 83M	1.04E+07	XE131M	7.17E+05
BR 84	1.88E+07	TE132	1.02E+08
BR 85	2.30E+07	I132	1.03E+08
KR 85	1.05E+06	I133	1.50E+08
KR 85M	2.33E+07	XE133	1.50E+08
RB 86	1.64E+05	XE133M	4.56E+06
KR 87	4.60E+07	I134	1.66E+08
KR 88	6.48E+07	CS134	1.71E+07
RB 88	6.56E+07	I135	1.39E+08
SR 89	7.84E+07	XE135	5.51E+07
SR 90	8.45E+06	XE135M	2.85E+07
Y 90	8.72E+06	CS136	4.74E+06
SR 91	1.07E+08	CS137	1.15E+07
Y 91	9.59E+07	BA137M	1.09E+07
SR 92	1.12E+08	XE138	1.30E+08
Y 92	1.12E+08	CS138	1.42E+08
Y 93	1.25E+08	BA139	1.38E+08
ZR 95	1.24E+08	BA140	1.33E+08
NB 95	1.24E+08	LA140	1.35E+08
ZR 97	1.26E+08	LA141	1.26E+08
MO 99	1.36E+08	CE141	1.22E+08
TC 99M	1.19E+08	LA142	1.23E+08
RU103	1.09E+08	CE143	1.21E+08
RU105	7.27E+07	PR143	1.19E+08
RH105	6.86E+07	CE144	9.80E+07
RU106	4.11E+07	ND147	4.97E+07
SB127	7.48E+06	NP239	1.35E+09
TE127	7.41E+06	PU238	3.86E+05
TE127M	9.94E+05	PU239	3.01E+04
SB129	2.26E+07	PU240	3.24E+04
TE129	2.22E+07	PU241	1.34E+07
TE129M	3.33E+06	AM241	2.06E+04
I129	3.35E+00	CM242	4.85E+06
TE131M	1.02E+07	CM244	3.93E+05
I131	7.15E+07		

(1) Based on core operation at 102% of 2568 MWth and a 700 EFPD cycle length.
(2) Zero decay time

TABLE 14.2-5
(Sheet 1 of 1)

POSTULATED FUEL HANDLING
ACCIDENT DOSE RESULTS
(In the Reactor Building)

TWO HOUR DOSE RESULTS	DOSE (REM)
Control Room TEDE	2.52
EAB TEDE	4.49
LPZ TEDE	0.787

TABLE 14.2-6
(Sheet 1 of 1)

ROD EJECTION ACCIDENT PARAMETERS
(INITIAL CYCLE)

Worth of Ejected Rod	
Rated power, no xenon delta-k/k	0.46%(1)
Rated power, with xenon delta-k/k	0.36%(1)
Hot, zero power, critical delta-k/k	0.56%
Rod ejection time sec	0.150
Rated power level MWt	2535
Reactor Trip Delay Time	
High Flux Trip Setpoint, % FP	114(2)
High flux trip delay sec	0.3
High Pressure Trip Setpoint, psia	2430
High pressure trip delay sec	0.5
Trip time to 2/3 insertion sec	1.4

(1) An ejected rod worth of 0.65% delta-k/k at rated power has been shown to have acceptable dose results as reported in Table 14.2-11.

(2) Original licensed power rating = 2535 MWt. The current high flux trip setpoint is 112% of 2568 MWt.

TABLE 14.2-7
(Sheet 1 of 1)

NOMINAL VALUES OF INPUT PARAMETERS FOR ROD EJECTION ACCIDENT ANALYSIS

	BOL	EOL
Delayed neutron fraction, β	0.0071	0.0053
Neutron lifetime, microsec	24.8	23.0
Moderator coefficient, (delta-k/k)/°F	Zero	-3.0×10^{-4}
Doppler coefficient, (delta-k/k)/°F	-1.17×10^{-5}	-1.33×10^{-5}
Coolant inlet temperature,°F	554	554
Initial system pressure, psia	2200	2200
Total Nuclear Peaking Factor	3.24	2.92
Average fuel temperature of average pellet,°F	1540	1670
Average fuel temperature of hottest pellet,°F	2810	2685
Center line temperature of hottest pellet,°F	3759	3553

TABLE 14.2-8
(Sheet 1 of 1)

COMPARISON OF SPACE-DEPENDENT AND POINT KINETICS
RESULTS ON THE FUEL ENTHALPY
(INITIAL CYCLE)

Ejected Rod Worth, (% delta-k/k)	Peak-to-Average Values		Fuel Enthalpy (cal/g)	
	TWIGL	Point Kinetics	TWIGL	Point Kinetics
		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 14.2-9
(Sheet 1 of 1)

SUMMARY OF ROD EJECTION ACCIDENT ANALYSIS

Initial Power Level (% of rated power)	Ejected Rod Worth (% delta-k/k)	Peak Power (% of Rated Power) Neutron	Thermal
0.1 (BOL)	0.56	46	34
0.1 (EOL)	0.56	160	14
100.0 (BOL)	0.46[1]	274	121

[1] An ejected rod worth of 0.65% delta-k/k at 100 percent power has been shown to have acceptable dose results as reported in Table 14.2-11.

TABLE 14.2-10
(Sheet 1 of 1)

REACTOR VESSEL PARAMETERS

Vessel temperature,°F	600
Yield strength (0.2% offset), psi	55,000
Ultimate strength, psi	80,000
Ultimate strain (E_u), %	26
Strain energy (E_S) per unit volume up to a strain equal to 1/2 ultimate strain, in.-lb/in.^{3}	8,000
Strain energy (E_S) per unit volume up to ultimate strain, in.-lb./in.^{3}	17,000
Equivalent pressure vessel dimensions	
OD, in.	188.25
ID, in.	166.69
Thickness, in.	10.78

TABLE 14.2-11
(Sheet 1 of 2)

ENVIRONMENTAL EFFECTS OF ROD EJECTION ACCIDENT

Isotope	Fuel Pin, Gap/Total Activity, %	Radioactivity* Released To Reactor Building, Ci
Kr-83m	0.12	2.17×10^3
Kr-85m	0.23	9.61×10^3
Kr-85	85.14	1.03×10^5
Kr-87	0.07	5.06×10^3
Kr-88	0.15	1.56×10^4
Xe-131m	14.58	1.24×10^4
Xe-133m	2.95	1.74×10^4
Xe-133	6.44	1.58×10^6
Xe-135m	0.04	2.35×10^3
Xe-135	0.90	3.80×10^4
I-131	1.98	2.84×10^5
I-132	0.18	3.05×10^4
I-133	0.22	5.41×10^4
I-134	0.010	3.15×10^3
I-135	0.07	1.71×10^4

(*) Ci = (Total Core Inventory in Table 14.2-4)x(Fraction of DNB RODS = 0.175)x(Ratio of Fuel Pin Gap to Total Core Activity)

TABLE 14.2-11
(Sheet 2 of 2)

ENVIRONMENTAL EFFECTS OF ROD EJECTION ACCIDENT

	Total Dose (rem)
Two Hour Dose at Exclusion Distance	
Thyroid	5.2
Whole body	0.007
30-Day Dose at Low Population Distance	
Thyroid	9.72
Whole body	0.009

TABLE 14.2-12
(Sheet 1 of 1)

REACTOR VESSEL INTERNALS - DISSIMILAR METALS

Metals	Major Element Proportions
Type-304 Stainless Steel	19 percent chromium 10 percent nickel Remainder iron
Control Rod Poison Material	80 percent silver 15 percent indium 5 percent cadmium
Zircaloy-4	98 percent zirconium 1-3/4 percent tin
M5	98 percent zirconium 1 percent niobium
Inconel	53 percent nickel 19 percent chromium 2 percent molybdenum 5 percent Cb-Ta 1 percent titanium 0.5 percent aluminum Remainder iron

TABLE 14.2-13
(Sheet 1 of 2)

ASSUMPTIONS AND RESULTS OF CONTROL ROD TEMPERATURE ANALYSIS

Assumptions

1. Analyses using Mark-B 15x15 fuel assemblies, extended life control rod assemblies (ELCRAs) and non-ELCRAs, and a power level of 3026 MWt (LBLOCA and SBLOCA) or 2827 MWt (SBLOCA only) were conservatively analyzed.

2. An energy disposition for adjacent fuel pins at a Linear Heat Rate limit of 17.8 kW/ft or less was assumed for LBLOCA and a limit of 17.3 kW/ft or less was assumed for SBLOCA.

3. Temperatures for the LBLOCA analyses were conservatively pushed to achieve a peak clad temperature that approached or even slightly exceeded 2200°F (2230°F), the PCT acceptance criterion in 10 CFR 50.46.

4. Temperature for the SBLOCA analyses were conservatively pushed to achieve a peak clad temperature that slightly exceeded 1800°F. If SBLOCA limiting PCT begins to approach 1800°F, this analysis should be revisited.

5. Five axial peak locations along the active fuel length were analyzed for LBLOCA: 2.506-, 4.264-, 6.021-, 7.779-, and 9.536-ft.

6. The bounding axial peak location along the active fuel length was analyzed for SBLOCA: 10.811-ft.

7. A break at the Cold-Leg Pump Discharge (CLPD) was the break location for both the LBLOCA and SBLOCA, consistent with limiting Evaluation Model analyses.

8. Once-Through Steam Generators (OTSGs) with a maximum of 20% tube plugging were conservatively assumed for both LBLOCA and SBLOCA.

9. Eutectic temperature of stainless steel intersection with zircaloy (1715°F) was used as an acceptance criteria. The eutectic temperature of inconel intersection with zircaloy (1736°F) is bounded by this acceptance criteria.

Results

Using the assumptions above, the average temperature of the Ag-In-Cd goes up to 1435°F at 152 seconds. This temperature is 35°F less than the Ag-In-Cd melt temperature of 1470°F. Considering the melt temperature is not reached, the eutectic temperature is not reached (1715°F), so even if there is contact between the stainless steel or Inconel 625 control rod cladding sheath and the M5 guide tube, the control rod will remain intact.

TABLE 14.2-13
(Sheet 2 of 2)

ASSUMPTIONS AND RESULTS OF CONTROL ROD TEMPERATURE ANALYSIS

For SBLOCA, the Ag-In-Cd melting temperature was reached but not exceeded and the eutectic temperature was not reached. Even though the melting temperature was reached, it was determined to only be a few minutes and localized to the top of the core in which it would not affect the reactivity contribution of the control rod. Therefore, control rod integrity and reactivity contributions were preserved. The integrity of the control rod assemblies and surrounding guide tubes are maintained and continue to perform their function during and following a loss of coolant accident.

TABLE 14.2-14
(Sheet 1 of 2)
LOSS OF COOLANT ACCIDENT ANALYSIS
SUMMARY OF LOCA LIMITS ANALYSIS RESULTS

Elevation	BOL		MOL 40 GWd/mtU		EOL 62 GWd/mtU	
	LHR (kW/ft)	PCT (°F)	LHR (kW/ft)	PCT (°F)	LHR (kW/ft)	PCT (°F)
0.0 ft	*15.9*[4]	*< 1864.5*	*15.9*[4]	*< 1868.9*	*12.5*	*< 1653*[4]
2.506 ft	**16.8**	**1864.5**	**16.8**	**1868.9**	**12.5**	**1552.6**
4.269 ft	**16.8**	**1833.5**	**16.8**	**1822.9**	*12.5*	*< 1553*
6.021 ft	**17.0**	**1885.0**	**17.0**	**1847.9**	*12.5*	*< 1553*
7.779 ft	**17.0**	**1840.7**	**17.0**	**1847.9**	*12.5*	*< 1553*
9.536 ft	**16.8**	**1864.2**	**16.8**	**1837.1**	*12.5*	*< 1553*
12.0 ft	*15.9*[4]	*< 1864.2*	*15.9*[4]	*< 1837.1*	*12.5*	*< 1653*[4]

Notes:

1. The LHR limits represent the nuclear source power generated by the pin (i.e., all sources of useable energy caused by the fission process).
2. Analyses at BOL and MOL used a steady-state energy deposition factor (EDF) of 0.973 for initial core energy deposition and a transient EDF of 1.0. The analysis at EOL used a steady-state EDF of 0.987 for the initial core energy deposition and a transient EDF of 1.065.
3. Linear interpolation for LHR limits is allowed between core elevations and burnup intervals.
4. The LHR limits below 2.506 ft are reduced linearly to 0.95 - LHR2.506-ft at 0.0 ft. The LHR limits above 9.536 ft are reduced linearly to 0.95 - LHR9.536-ft at 12.0 ft. At EOL, a PCT increase of 100°F is applied to the adjacent elevation PCT (2.506-ft or 9.536-ft) for the 0.0 ft and the 12.0 ft instead of a LHR limit decrease.
5. LHRs are based on TACO3 LOCA initializations for fuel enrichments between 3.0 and 5.1 weight percent, with axial blankets of 2.5 to 3.0 weight percent, and pin prepressure of 245 psia.
6. All analyzed LHRs and PCTs are bold and the estimated LHR and PCT values are italicized.
7. See current COLR for any cycle-specific adjustments to LOCA LHR limits.

TABLE 14.2-14
(Sheet 2 of 2)

LOSS OF COOLANT ACCIDENT ANALYSIS
LOCA ANALYSIS ASSUMPTIONS

Initial Reactor Power	2827 MWt (102% of 2772 MWt)	
Decay Heat	120% ANS 5.1 1971 Standard With B&W Heavy Actinides	
Pressurizer Level	220 inches	
Core Bypass Flow	7.50%	
Reactor Trip (SBLOCA only)		
Low RCS Pressure Setpoint	1780 psig	
Low RCS Pressure Delay Time[1]	0.6 sec	
Time of Full Rod Insertion	2.3 sec	
ESAS Setpoint	HPI: 1480 psig LPI: 340 psig	
ECCS Delay Time	HPI: 35 sec LPI: Later of 35 sec after HPI Signal or 10 sec after LPI signal	
EFW (SBLOCA only)		
Flow	200 gpm	
Delay Time	120 sec	
Temperature	135^{o}F for the first 10 minutes, followed by a step change to 120^{o}F	
Level Setpoint	<LSCM+20 min: 50% OR >LSCM+20 min: 65% OR	
Core Flood Tank	(CLPD & HPI Line Break)	(CFT Line Break)
Pressure (psig)	565	635
Liquid Volume (ft^3)	985	895
Temperature (F)	140	140
BWST Temperature	120F	
Main Feedwater Pump Coastdown[2]	14 sec	
OTSG Tube Plugging	Broken Loop: 25% Intact Loop: 15%	
RCS Flow	LBLOCA: 133.9e6 lbm/hr SBLOCA: 134.4e6 lbm/hr[3]	

1 Includes instrument delay, trip module delay, breaker opening time delay, and CRDM unlatch time delay.

2 Loss of offsite power (LOOP) is assumed coincident with reactor scram. The LOOP is assumed to cause the reactor coolant pumps and main feedwater pumps to trip.

3 The SBLOCA was also evaluated at the minimum DNB flow of 104.5% design flow (including 2.5% uncertainty) and found to have no impact on PCT or transition break size [Ref. 142].

TABLE 14.2-15

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SUMMARY OF BREAK SPECTRUM

TABLE 14.2-16
(Sheet 1 of 1)

SEQUENCE OF EVENTS OF STUCK OPEN PORV ACCIDENT

Event	Time (sec)
1. 0.01-ft^2 cold leg break occurs	0.0
2. Reactor trip, loss of feedwater, and reactor coolant pump trip	54.5
3. Main feedwater coastdown ends	60.0
4. SG secondary boils dry	270.0
5. PORV opened	1200.0
6. HPI is manually initiated	1200.0
7. Long-term cooling established	1510.0

TABLE 14.2-17
(Sheet 1 of 1)

ACTIVITY RELEASE FROM MAXIMUM BREAK SIZE LOCA

Isotope	Activity(*) (Ci)
Noble Gas	
Kr-83m	1.24×10^4
Kr-85m	5.49×10^4
Kr-85	5.90×10^5
Kr-87	2.89×10^4
Kr-88	8.93×10^4
Xe-131m	7.10×10^4
Xe-133m	9.97×10^4
Xe-133	9.02×10^6
Xe-135m	1.34×10^4
Xe-135	2.17×10^5
I-131	8.10×10^5
I-132	8.70×10^4
I-133	1.55×10^5
I-134	9.00×10^4
I-135	4.88×10^4

* Activities in the Reactor Building atmosphere.

TABLE 14.2-18
(Sheet 1 of 1)

IODINE ACTIVITY RELEASE FROM MHA*

Isotopes	Activity (Ci)
I-131	4.4E+04
I-132	9.7E+02
I-133	8.8E+03
I-134	7.6E+02
I-135	3.5E+03

(*) Iodine activity released to the environment.
Basis of source term is given in Table 14.2-4.

TABLE 14.2-19
(Sheet 1 of 1)

NOBLE GAS RELEASE FROM MHA*

Isotope	Activity (Ci)
Kr-83m	1.1E+03
Kr-85m	6.3E+03
Kr-85	1.1E+04
Kr-87	3.3E+03
Kr-88	1.0E+04
Xe-131m	3.7E+03
Xe-133m	7.0E+03
Xe-133	1.7E+06
Xe-135m	5.6E+02
Xe-135	9.1E+04
Xe-138	1.8E+03

(*) Iodine activity released to the environment. Basis of source term is given in Table 14.2-4.

TABLE 14.2-20
(Sheet 1 of 1)

ENVIRONMENTAL DOSES RESULTING FROM MHA (Reference 103)

Post-LOCA Activity Release Path	Post-LOCA TEDE Dose (Rem) Receptor Location		
	Control Room	EAB	LPZ
Containment Leakage	6.6319E-01	2.0674E+01	5.8032E+00
ESF Leakage	4.0598E+00	2.6167E+00	1.9063E+00
BWST Leakage	3.6730E-02	3.0611E-02	4.4566E-02
Cont. Purge	3.5503E-03	2.6343E-02	4.6495E-03
External Cloud	0.0000E+00	0.0000E+00	0.0000E+00
Filter Shine	1.326E-02	0.0000E+00	0.0000E+00
Total	4.7765E+00	2.3347E+01	7.7587E+00

TABLE 14.2-21
(Sheet 1 of 1)

WASTE GAS TANK INVENTORY

Isotope	Activity (curies)
Kr-83m	1.21E2
Kr-85m	5.49E2
Kr-85	2.20E3
Kr-87	2.89E2
Kr-88	8.93E2
Xe-131m	6.06E2
Xe-133m	9.54E2
Xe-133	8.86E4
Xe-135m	1.10E2
Xe-135	1.89E3
Xe-138	1.56E2
I-131	1.30E1
I-132	4.40E0
I-133	1.39E1
I-134	1.70E0
I-135	7.00E0

TABLE 14.2-22
(Sheet 1 of 3)

KEY INPUT PARAMETERS FOR LOFW ANALYSIS

PARAMETER	ANALYSIS VALUE
Initial core power	2619.4 MWt (102% of 2568)
HFP BOC moderator temperature coefficient	0.0 pcm/^{0}F
HFP BOC doppler coefficient	-1.17 pcm/^{0}F
Control rod insertion time to 2/3 insertion	1.4 second
Minimum shutdown margin with maximum worth rod stuck	1% Δk/k
HFP delayed neutron fraction	0.007β_{eff}
Decay heat standard + B&W Heavy Actinides	ANS 1971
Number of pressurizer safety valves (PSV)	2
PSV capacity per valve	297,846 lbm/hr
PSV open setpoint	2500 psig
PSV setpoint drift	3 %
PORV flow rate at 2300 psig	106,451 lbm/hr
Pressurizer spray setpoint measured at hot leg pressure tap	2205 psig
High pressure trip measured at the hot leg pressure tap	2400 psia
High pressure trip delay time	0.6 sec
RCS average temperature	579°F
Initial RCS pressure, nominal	2170 psia

TABLE 14.2-22
(Sheet 2 of 3)

KEY INPUT PARAMETERS FOR LOFW ANALYSIS

PARAMETER	ANALYSIS VALUE
Initial pressurizer level	232 inches
Initial RCS flow rate, minimum DNB flow	367,840 gpm
Total EFW flow rate to both SG's (at 1065 psia)	550 gpm
EFW low SG level initiation startup range indicated level, including 10 in instrument error	0 inches
EFW time delay (after initiation signal)	43 sec
EFW temperature	135°F
Initial steam generator inventory per SG	~48,400 lbm
Initial SG level. Operate range	50%
Heat addition from each RC pumps	22.4MW (5.6MW per pump)
SG tube plugging	5%

TABLE 14.2-22
(Sheet 3 of 3)

Sequence of Events for LOFW Without ARTS Trip

Event	Time, Seconds PORV and Spray	Time, Seconds Without PORV or Spray
Loss of main feedwater initiated	0.0	0.0
Main feedwater flow reaches zero	8.9	8.9
Pressurizer spray on	11.24	N/A
RCS high pressure trip setpoint reached	20.65	20.13
Turbine trip	21.15	20.63
PORV lift (first)	21.99	N/A
PSV lift	24.32	23.22
Peak RCS pressure	2690.42 psia @ 24.67 sec	2704.75 psia @ 23.80 sec
SG low level setpoint reached	72.93	74.04
EFW flow initiated	115.93	117.04
PORV lift (last)	199.59	N/A
PSV lift (last)	No second lift	269.45
Peak RCS temperature reached	612.21°F @ 369.71 sec	611.72°F @ 483.49 sec
Pressurizer spray off	setpoint not reached	N/A
Peak pressurizer level reached	437.6 in @ 673.06 sec	390.5 in @ 747.73 sec
End of transient	800.0	800.0

TABLE 14.2-23
(Sheet 1 of 1)

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TABLE 14.2-24
(Sheet 1 of 1)

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TABLE 14.2-25
(Sheet 1 of 1)

NOBLE GAS AND IODINE GAP ACTIVITY
(10 FUEL ASSEMBLIES)
120-DAY DECAY

Isotope	Activity Ci
Kr-85	8.11×10^4
Xe-131m	8.64
Xe-133	0.173
I-131	5.48

TABLE 14.2-26
(Sheet 1 of 1)

Fuel Cask Drop Accident

SITE BOUNDARY DOSES*

	Whole Body (Rem)	Thyroid (Rem)
Kr-85	5.11×10^{-2}	---
Xe-131m	8.55×10^{-6}	---
Xe-133	1.88×10^{-6}	---
I-131	---	3.38
	5.11×10^{-2}	3.38

* Site Meteorological Dispersion Factor (X/Q) = 1.2×10^{-3} sec/m^3 (Reference 55).

TABLE 14.2-27
(Sheet 1 of 1)

LOW PRESSURE INJECTION FLOW VERSUS CORE FLOOD/LPI NOZZLE PRESSURE

Large Break LOCA[1]

Pressure At Centerline Core Flood/LPI Nozzle (PSIG)	Low Pressure Injection Flow (per loop) (GPM)
0	3150
109	2700
145	1830
160	1350
169	900
178	0

Small Break LOCA[2]

Pressure At Centerline Core Flood/LPI Nozzle (PSIG)	Low Pressure Injection Flow (per loop) (GPM)
0.0	3150
98	2700
133	1830
148	1350
157	900
163.9	0

1 LPI flows for LBLOCA are based on a full BWST.
2 LPI flows for SBLOCA are based on a half-full BWST.

TABLE 14.2-28
(Sheet 1 of 1)

HIGH PRESSURE INJECTION FLOW VERSUS
HPI NOZZLE PRESSURE

LOCA on RCS Piping[1]

Pressure At HPI Nozzle (PSIG)	Total Flow[2] (GPM)
0	438.0
600	431.0
1200	380.0
1500	346.0
1600	335.0
1800	310.0
2400	190.0

[1] Breaks other than Cold Leg Pump Discharge (CLPD) break and HPI Line break.
[2] Total Flow is evenly distributed between all four HPI lines.

CLPD Break

Pressure At HPI Nozzle (PSIG)	Flow to Intact Legs[1] (GPM)
0	306.6
600	301.7
1200	266.0
1500	242.2
1600	234.5
1800	217.0
2400	133.0

[1] No HPI flow is injected into the broken Cold Leg.

HPI Line Break

Pressure At HPI Nozzle (PSIG)	Flow to Intact Legs[1] (GPM)
0	308.96
600	308.96
1200	230
1500	182
1600	165
1800	130

[1] No HPI flow is injected into the broken Cold Leg.

TABLE 14.2-29
(Sheet 1 of 4)

FEEDWATER LINE BREAK SUMMARY OF ANALYSIS INPUT VALUES

PARAMETER		ANALYSIS VALUE
HFP BOC Doppler Coefficient, pcm/^oF		-1.17
HFP Delayed Neutron Fraction, βeff		0.007
HFP BOC MTCoefficient, pcm/^oF	(note that Tech Specs limit operation above 95% power to a non- negative MTC)	0.0
Initial Core Power, Mw(t)	102% of 2568	2619.4
PSV Capacity, lbm/hr/valve	at 2575 psig	297,846
PSV Setpoint Drift, %		3
Decay Heat model		ANS 1971 + B&W Heavy Actinides
Decay Heat multiplier		1.0
SG Tube Plugging		5%
High RCS Pressure Trip, psia		2400 outside containment 2446 inside containment
High RCS Pressure Trip Delay Time, sec		0.6
High RB Pressure Trip, psia	Not Credited	N/A
RCS Average Temperature, °F		579
Initial RCS Pressure psia		2170
RCS Flow Rate, gpm	minimum DBN flow	367,840
EFW Flow Rate		sheet 4
EFW Temperature, °F		135
HPI Delay time, sec		0
HPI Flow Rate		sheet 3
Feedwater nozzle area 32 nozzles		0.6 ft^2

Table 14.2-29
(Sheet 2 of 4)

Feedwater Line Break Sequence of Events

Event	Time, seconds	
	outside containment	inside containment
Main feedwater line break	0.0	0.0
Main feedwater flow reaches zero	0.0	0.0
RCS high pressure trip setpoint reached	16.29	10.88
Turbine trip	16.39	10.98
PSV lift (first)	18.8	12.5
Peak RCS pressure	2705.1 psia @ 19.3 sec	2755.15 psia @ 13.5 sec
SG low level setpoint reached	69.28	15.28
HPI initiated	N/A	10.88
PSV lift (last)	~ 258	~ 565
EFW flow initiated	114.0	152.0
Peak pressurizer level	600.0	**~ 100.0**
Peak RCS temperature	611.92^{o}F @ 478 sec	609.53^{o}F @ 140 sec
End of transient	600.0	600.0

TABLE 14.2-29
(Sheet 3 of 4)

HPI FLOW RATE ASSUMPTIONS FOR FWLB ANALYSIS

Cold Leg Pressure (psig)	Outside Containment 1 HPI (gpm)	Outside Containment 3 HPI (gpm)
0	535.4	954
536	535.4	
600	530	
1300	459	
1395		
1500	435	
1600	420	
1800	395	
1860		954
2000	363	890
2200	330	800
2400	290	690
2600	240	550
2800	154	346

TABLE 14.2-29
(Sheet 4 of 4)

EMERGENCY FEEDWATER Assumed Flows

PUMP COMBINATION	DBA ANALYZED	OTSG PRESSURE A psia	OTSG PRESSURE B psia	NET FLOW TO OTSG A gpm	NET FLOW TO OTSG B gpm	TOTAL DELIVERED FLOW, gpm
1 MDP & 1TDP or 2 MDPs to both OTSGs	FWLB (Outside Containment)	1065 1090	1065 1090	275 250	275 250	550 500
1 MDP & 1 TDP or 2 MDPs to both OTSGs	FWLB (Inside Containment)	15 15	1065 1090	450 445	120 85	570 530

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