15.0 ACCIDENT AND TRANSIENT ANALYSIS
TABLE OF CONTENTS

15.0 ACCIDENT AND TRANSIENT ANALYSIS
TABLE OF CONTENTS

Page

15.0 ACCIDENT AND TRANSIENT ANALYSIS
TABLE OF CONTENTS

Page

15.0 ACCIDENT AND TRANSIENT ANALYSIS
TABLE OF CONTENTS

Page

15.0 ACCIDENT AND TRANSIENT ANALYSIS
TABLE OF CONTENTS

15.0 ACCIDENT AND TRANSIENT ANALYSIS
TABLE OF CONTENTS

Page

15.0 ACCIDENT AND TRANSIENT ANALYSIS
TABLE OF CONTENTS

15.0 ACCIDENT AND TRANSIENT ANALYSIS
LIST OF TABLES

15.0 ACCIDENT AND TRANSIENT ANALYSIS
LIST OF FIGURES

15.0 ACCIDENT AND TRANSIENT ANALYSIS

The evaluation of the safety of a nuclear power plant includes analyses of the plant's response to postulated disturbances in process variables and to postulated malfunctions or failures of equipment. These safety analyses provide a significant contribution to the design and operation of components and systems from the standpoint of public health and safety.

In previous chapters, the important structures, systems, and components are discussed. In this chapter, the effects of anticipated process disturbances and postulated component failures are examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such failures and situations (or to identify the limitations of expected performance).

The following Equipment Out-of-Service (EOOS) conditions are analyzed or evaluated for impact on thermal limits and the results are reported for the applicable transient analyses: feedwater temperature reduction, TBV out of service, one SRV out of service, single loop operation, TCV slow closure, PLU out of service, pressure regulator out of service, one TCV stuck closed, one TSV stuck closed, and one MSIV out of service (at 75% power). Additional details related to the combination of EOOS options with ARTS and MELLLA operation can be found in cycle specific reload documentation and Reference 3. The EOOS conditions have been analyzed for the impact on fuel thermal limits and the design basis of the fuel, but the transient analysis does not evaluate the effect of the EOOS condition on the design basis of the system. Any plant modifications, procedure revisions, or other permanent changes to the facility involving the above EOOS should consider the effect of the proposed changes on the design basis of the affected systems.

15.0.1 Frequency Classification

The effects of various postulated anticipated operational occurrences and incident events are investigated for a variety of plant conditions. With the exception of Anticipated Transients Without Scram (ATWS), transients and accidents are categorized into the following three groups according to frequency of occurrence:

A. Incidents of moderate frequency - incidents that may occur with a frequency greater than once in 20 years for a particular plant. These events are referred to as anticipated (expected) operational occurrences.

B. Infrequent incidents - incidents that may occur during the life of the particular plant (spanning once in 20 years to once in 100 years). These events are referred to as abnormal (unexpected) operational occurrences. For conservatism, infrequent events can be analyzed as if they were moderate frequency events.

C. Limiting faults - incidents that are not expected to occur but are postulated because their consequences may result in the release of significant amounts of radioactive material. These events are referred to as design basis (postulated) accidents.

Treatment of ATWS events is discussed in Section 15.8.

15.0.2 Transients and Accidents Analyzed

AREVA ATRIUM 10XM methods and fuel are only applicable to Unit 3.

Events analyzed in this chapter are categorized as either transients (anticipated and abnormal operational occurrences) or accidents, depending on the frequency classifications described in Section 15.0.1. Transients and accidents have different acceptance criteria for their analyses. Listings of transients and accidents, a summary of analysis methods, and a description of specific transients which are reanalyzed for each fuel cycle are provided in the following subsections.

The core-wide AOOs were analyzed to support the Extended Power Uprate (EPU) conditions (includes the MELLLA domain) and the incorporation of the APRM Rod Block Monitor Tech Spec (ARTS) power and flow limits Improvement Program. This included re-evaluating a broad set of the most limiting transient events at the EPU conditions. The ARTS power and flow dependent limits for equipment out of service (OOS) and other flexibility options are addressed in Reference 3. The basis for selection of the transient events for re-analysis is documented in "NEDC-32424P-A, Licensing Topical Report Generic Guidelines for General Electric BWR Extended Power Uprate, February 1999, Appendix E" where it stipulates, "Analysis will be performed for the limiting transient events. This includes all events that establish core thermal operating limits and the events that show bounding conformance to the other transient protection criteria (e.g., ASME Overpressure Limits)". The transient events which are reanalyzed with power uprate conditions from 2527 MWt to 2957 MWt core thermal power are documented in the Dresden EPU PUSAR Report section 9.1. The parameters used for the transient analysis are provided in Table 15.0-1. The limiting transient results representative of GE transient analysis for EPU are provided in Table 15.0-2.

The existing licensing bases of DNPS were reviewed to determine the potentially limiting analyses that must be done on a cycle-specific basis or on a one-time basis to support the introduction of SVEA-96 Optima2 fuel. The basis for selection of the limiting events is discussed in the Westinghouse reload licensing methodology document for Dresden (Reference 4). A summary of the results of the events that are re-analyzed is documented in the cycle-specific reload licensing report (RLR). The parameters used for the transient analysis and the limiting transient results are documented in the OPL-W.

The existing licensing bases of DNPS were reviewed to determine the potentially limiting analyses that must be done on a cycle-specific basis or on a one-time basis to support the introduction of ATRIUM 10XM fuel. The basis for selection of the limiting events is discussed each unit's cycle-specific Reload Safety Analysis report (RSAR). Additionally, a summary of the results of the events that are re-analyzed are also documented in the cycle-specific RSAR. The parameters used for the transient analysis and the limiting transient results are documented in the plant parameters document (PPD).

15.0.2.1 Transients

Transients which occur as a consequence of a single equipment failure or malfunction or single operator error are evaluated in the sections listed below:

Analysis		Section
A.	Decrease in feedwater temperature (loss of feedwater heating)	15.1.1
B.	Increase in feedwater flow (feedwater controller malfunction) - maximum flow	15.1.2
C.	Increase in steam flow	15.1.3
D.	Steam pressure regulator malfunction	15.2.1
E.	Generator load rejection without bypass	15.2.2.1
F.	Generator load rejection with bypass system (loss of electrical load)	15.2.2.2
G.	Turbine trip without bypass	5.2.2.2.2 and 15.2.3.1
H.	Turbine trip with partial bypass – maximum power	15.2.3.2
I.	Inadvertent closure of main steam line isolation valves	15.2.4
J.	Loss of main condenser vacuum	15.2.5
K.	Loss of offsite ac power	15.2.6
L.	Loss of normal feedwater flow (feedwater controller malfunction) - zero flow	15.2.7
M.	Single and multiple recirculation pump trips	15.3.1
N.	Recirculation flow controller failure (malfunction) - decreasing flow	15.3.2
O.	Recirculation pump shaft break	15.3.5
P.	Jet pump malfunction	15.3.6
Q.	Uncontrolled control rod assembly withdrawal - subcritical or startup condition	15.4.1
R.	Rod withdrawal error	15.4.2
S.	Control rod maloperation	15.4.3
T.	Startup of idle recirculation loop at incorrect temperature (cold recirculation loop)	15.4.4
U.	Recirculation flow controller failure (malfunction) - increasing flow	15.4.5
U.1	Thermal Hydraulic Instability	15.4.11
V.	Inadvertent actuation of high pressure coolant injection during power operation	15.5.1
W.	Inadvertent opening of a safety valve, relief valve, or safety relief valve	15.6.1
X.	Radioactive gas waste system leak or failure	15.7.1
Y.	Postulated liquid releases due to liquid tank failure	15.7.2
Z.	Loss of auxiliary power	8.3.1
AA.	Power bus loss of voltage	8.3.1
BB.	Instrument air failure	9.3.1.2

CC.	Failure of one diesel generator to start	8.3.1.5
DD.	Loss of Stator Cooling	15.2.8

In addition to the above transients, the following events have been analyzed as transients although they are not anticipated operational occurrences when considered without scram:

A.	Closure of main steam isolation valves without scram	15.8.1
B.	Loss of normal ac power without scram	15.8.2
C.	Loss of normal feedwater flow without scram	15.8.3
D.	Turbine-Generator trip without scram	15.8.4
E.	Loss of condenser vacuum without scram	15.8.5

15.0.2.2 Design Basis Accidents

In order to evaluate the ability of the plant safety features to protect the public, a number of accidents are analyzed herein. These accidents are of very low probability; they are considered in order to include the far end of the spectrum of challenges to the safeguards and the containment system. The accidents evaluated are discussed in the following sections:

Analysis		Section
A.	Control rod drop	15.4.10
B.	Loss of coolant	15.6.2 and 15.6.5
C.	Main steam line break	15.6.4
D.	Recirculation pump shaft seizure	15.3.3
E.	Recirculation pump shaft seizure while in single loop operation	15.3.4
F.	Fuel assembly drop during refueling	15.7.3
G.	Mislocated fuel assembly	15.4.7*
H.	Misoriented fuel assembly	15.4.8*

*The mislocated and misoriented fuel assembly are characterized as infrequent events (infrequent incidents) in AREVA methodology (Reference 6).

15.0.2.3 Method of Analysis

Sections 15.1 through 15.8 provide analyses for each transient and accident given above, from the initiating event to the propagation of the event including effects on other systems. Generally, for each transient or accident analysis there are subsections which delineate the cause identification, frequency classification, sequence of events and system operation, core and system performance, barrier performance, and radiological consequences.

15.0.2.4 Transients Reanalyzed for each Fuel Cycle

Some of the transients listed above in Section 15.0.2.1, are reanalyzed for each fuel cycle to account for the characteristics specific to the fuel type and configuration for that cycle. The results of these transient analyses are used to set reactor thermal limits for that cycle in order to prevent fuel damage or reactor coolant pressure boundary (RCPB) overpressurization.

The remaining transients are not reanalyzed for each fuel cycle since they have been found to be always bounded by (i.e., less severe than) those transients that are reanalyzed.

The transients currently reanalyzed for each cycle are as follows:

A. Loss of feedwater heating (15.1.1)

B. Feedwater controller failure (15.1.2)

C. Generator load rejection without bypass (15.2.2.1)

D. Turbine trip without bypass (15.2.2.2.2 and 15.2.3.1)

E. Rod withdrawal error event (15.4.2)

F. Recirculation loop flow controller failure with increasing flow (15.4.5)

G. Thermal hydraulic Instability (15.4.11)

H. Main steam line isolation valve closure without direct scram or credit for relief valves (ASME overpressure event) (15.2.4)

I. Inadvertent actuation of high pressure coolant injection during power operation (15.5.1)

J. Loss of Stator Cooling (15.2.8)

Transients A through G and I through J are reanalyzed for each cycle to determine thermal margins. ASME overpressure events (Transient H) are reanalyzed to confirm the maximum pressure is within 110% of the reactor coolant system design pressure (Section 15.2.4.2). The most limiting transients for determining thermal limits to prevent fuel damage are usually feedwater controller failure, generator load rejection without bypass, turbine trip without bypass, or rod withdrawal error. More detailed descriptions of these transients are given in the identified sections above.

See Reference 4 for details on the Westinghouse reload method. See References 5 and 6 for details on the application of AREVA methodology to reload licensing.

The results of cycle specific transient analyses have indicated that operation must be maintained within a range of pressure and feedwater temperature, determined by the inputs to the transient analyses.

15.0.2.5 Radiological Reassessments of Design Basis Accidents

Alternative Source Terms are utilized for the evaluation of the onsite and offsite dose consequences of the Design Basis Accidents of Loss of Coolant Accident (LOCA), Control Rod Drop Accident (CRDA), Fuel Handling Accident (FHA), and Main Steam Line Break (MSLB). The power Level used in the radiological assessment of design basis accidents is 102% of the extended power uprate; i.e., 3016 MWt.

15.0.2.6 References

1. "Licensing Topical Report Generic Guidelines for General Electric BWR Extended Power Uprate," NEDC-32424P-A Appendix E, February 1999.

2. "Safety Analysis Report for Dresden 2 & 3 Extended Power Uprate," NEDO-32962P Revision 2, August 2001.

3. "Dresden 2 and 3 Quad Cities 1 and 2 Equipment Out-Of-Service and Legacy Fuel Transient Analysis," GE-NE-J11-03912-00-01-R3, Revision 3, September 2005.

4. "Westinghouse BWR Reload Licensing Methodology Basis for Exelon Generation Company Dresden Nuclear Power Station Units 2 and 3," WCAP-16588-P Revision 3, August 2006.

5. XN-NF-80-19(P)(A) Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, June 1986. (Unit 3 only)

6. ANP-3338P, Revision 1, "Applicability of AREVA BWR Methods to the Dresden and Quad Cities Reactors Operating at Extended Power Uprate," AREVA, August 2015. (Unit 3 only)

Table 15.0-1
(Historical)
Parameters Used For Transient Analysis

Parameter	Units	Base UFSAR	Extended Power Uprate by GE
Rated Thermal Power	MWt	2527	2957
Analysis Power	% Rated	100	100*
Dome Pressure	psig	1005	1005
Turbine Pressure	psig	950	926
Rated Steam and FW Flow	Mlbm/hr	9.81	11.71
Analysis Steam and FW Flow	% Rated	9.81	11.71
Rated Core Flow	Mlbm/hr	98	98
Rated Power Core Flow Range	% Rated	-	95 – 108
Analysis Core Flow**	% Rated	100	108
Analysis Feedwater Temperature	°F	340	356
Number of Safety Plus Relief Valves		13	13***
SRV set pressure (nominal/analysis values) (psig)		1135/1146.4	No change
SRV set pressure (nominal/analysis values) (psig)		2 @ 1240/1252.4	No change
		2 @ 1250/1262.5	No change
		4 @ 1260/1272.6	No change
SRV relief function (nominal/analysis values) (psig)		1125/1135	No change
RV setpoints (nominal/analysis values) (psig)		2 @ 1101/1115	No change
		2 @ 1124/1135	No change
MCPR Safety Limit		N/A	1.09

* GEMINI analysis at 100%
** All analysis at maximum core flow unless explicitly noted otherwise.
***The lowest pressure setpoint valve is assumed to be out of service for the MCPR transient analysis.

Table 15.0-2
(Historical)
GE EPU Transient Analysis Results

Event	Peak Neutron Flux (% of Rated EPU)	Peak Heat Flux (% of Rated EPU)	ΔCPR	MCPR Operating Limit	
				Option A	Option B
Load Rejection with Bypass Failure	557	131	0.26	1.58	1.41
Turbine Trip with Bypass Failure	555	131	0.26	1.57	1.40
Feedwater Controller Failure Max Demand	572	143	0.31	1.63	1.46
Loss of Feedwater Heating(1)	-	-	0.21	1.33	
Inadvertent HPCI	Bounded by Loss of Feedwater Heating				
Rod Withdrawal Error(1)	-	-	0.30	1.39	
Slow Recirculation Increase(1)	-	-	-	MCPR(F)	
Fast Recirculation Increase	96	82	0.17	Bounded by ARTS K(P) and MCPR(F)	
Load Rejection With Bypass	505	128	0.23	Bounded by Load Rejection with Bypass Failure	
MSIV Closure All Values	100	100	0.00	Bounded by Load Rejection with Bypass Failure	
MSIV Closure 1 Valve	131	108	0.09		

(1) Neutron and heat fluxes are not reported for these (slow) transients.

15.1 INCREASE IN HEAT REMOVAL BY THE REACTOR COOLANT SYSTEM

AREVA ATRIUM 10XM methods and fuel are only applicable to Unit 3.

Events described in this section that result in decreased feedwater temperature may also result in a core thermal hydraulic instability transients.

This section covers transients which involve an unplanned increase in heat removal from the reactor. Excessive heat removal, i.e., heat removal at a rate in excess of the heat generation rate in the core, causes a decrease in moderator temperature which increases core reactivity and can lead to an increase in power level and a decrease in shutdown margin. The power level increase, if sufficient, would be terminated by a reactor scram. An unplanned power level increase, however, has the potential to cause fuel damage (defined in Section 4.2.1.1) or excessive reactor coolant system pressure.

The following design basis transients are covered in this section:

A. Feedwater system malfunctions that result in a decrease in final feedwater temperature;

B. Feedwater system malfunctions that result in an increase in feedwater flow; and

C. Steam pressure regulator malfunctions that result in an increase in steam flow.

The events described in this section may not be reanalyzed for the current fuel cycle since they may continue to be bounded by analyses for previous fuel cycles. These events, including the associated assumptions and conclusions, continue to be part of the plant licensing basis. The conclusions of these analyses are still valid; however, specific details contained in the descriptions and associated figures should be used only to understand the analysis and its conclusions. These specific details should not be used as sources of current fuel cycle design information.

The limiting events (in terms of minimum critical power ratio (MCPR)) evaluated to determine the impact of EPU, are the Loss of Feedwater Heating (LFWH) and Feedwater Controller Failure (FWCF). For Westinghouse reloads. the inadvertent HPCI event would also decrease the core coolant temperature similar to LFWH, but it was shown to be bounded by the LFWH event. For AREVA reloads, the inadvertent HPCI event is not necessarily bounded by the LFWH event and is analyzed on a cycle-specific basis. For Westinghouse analysis results for these events, refer to the cycle-specific reload licensing report (RLR). For AREVA analysis, results for these events, refer to the cycle-specific reload safety analysis report (RSAR).

15.1.1 Decrease in Feedwater Temperature

A decrease in feedwater temperature due to loss of feedwater heating would result in a core power increase due to the increase in core inlet subcooling and the reactivity effects of the corresponding increase in moderator density.

15.1.1.1 Identification of Causes and Frequency Classification

Feedwater heating can be lost in at least two ways: if the steam extraction line to the heater is closed, or if the feedwater is bypassed around the heater.

The first case would produce a gradual cooling of the feedwater. In the second case, the feedwater would bypass the heater, and the reduction of heating would occur during the stroke time of the bypass valve (about 1 minute, similar to the heater time constant). In either case the reactor vessel would receive feedwater that is cooler than normal. The maximum number of feedwater heaters which can be tripped or bypassed by a single event represents the most severe transient for analysis considerations. The loss of feedwater heating would cause an increase in

core inlet subcooling. Due to the decrease in coolant void fraction and the effect of the negative void reactivity coefficient, the result is a gradual initial increase in reactor power.

The loss of feedwater heating event is classified as a moderate frequency event.

15.1.1.2 Sequence of Events and System Operation

The following plant operating conditions and assumptions form the principal basis for which reactor behavior is analyzed during the loss of feedwater heating transient:

A. The plant is operating at steady-state equilibrium.

B. The plant is operating in the manual recirculation flow control mode. There would be compensation for the reactor power increase by modulation of core flow, and the event generally would be less severe, if operation were in the automatic recirculation flow control mode.

For this event power would increase at a very moderate rate. If power exceeded the APRM Rod Block Setpoint (see Figure 4.4-1 for a typical schematic illustration of the power-flow map), the operator would be expected to insert control rods to return power and flow to their normal range. If this were not done the neutron flux could exceed the scram setpoint; however, a scram is not assumed in the calculations.

Westinghouse analysis assumes an instantaneous 145°F decrease in feedwater temperature.

For AREVA reload cores, two scenarios are evaluated – a fast and a slow event. The fast event is a 80˚F decrease in feedwater temperature over a time period of 60 seconds. The slow event supports a larger 145˚F decrease in feedwater temperature over a time period greater than 80 seconds (Reference 10).

15.1.1.3 Core and System Performance

The SEP analysis for the loss of feedwater heating transient showed the event would result in a mild transient in which the fuel surface heat flux would increase to a maximum value below that corresponding to steady-state operation at the scram setpoint. The increase in power would be partially offset by the beneficial effect of the increased core inlet subcooling on the critical power ratio (CPR).

The analyses considered that the loss of feedwater heating transient could result in both a gradual decrease in core inlet water temperature, which would be consistent with the closure of a steam extraction line to a feedwater heater, and a relatively rapid decrease in temperature, which would be consistent with bypassing feedwater around the heater.

For cycle specific calculations, the event is analyzed using the three-dimensional steady state simulator code POLCA7 for Westinghouse methods.

The fast 80˚F decrease in feedwater temperature is analyzed using the COTRANSA2/XCOBRA/XCOBRA-T code package. The longer term event supporting a 145˚F decrease in feedwater temperature is evaluated using the Reference 9 methodology.

The gradual power change allows the fuel thermal response to maintain pace with the increase in neutron flux. The loss of feedwater heating is a relatively slow transient and can be modelled by analyzing equilibrium conditions in the initial and final state points.

The basic assumptions applied for this analysis include the following:

A. The reactor is in steady-state equilibrium before and after the event.

B. The reactor is initially at 100% rated core thermal power. For AREVA reload cores, an off-rated statepoint is typically included corresponding to 60% rated core flow.

C. The xenon concentration does not change during the event.

D. The total core flow is constant during the event. This assumption conservatively implies that the plant is operating in the manual flow control mode. The analysis is performed at 95.3%, 100% and 108% of rated core flow. For AREVA reload cores, three core flows may also be analyzed at rated power in addition to the off-rated statepoint that is typically at 60% rated flow (described above).

E. The criticality eigenvalue is constant throughout the event.

F. Although the neutron flux levels due to the event may be sufficiently high to cause an average power range monitor (APRM) reactor protection system (RPS) trip, reactor scram is not allowed in the calculations. For AREVA reload cores, a high pressure scram may be credited if the steam flow exceeds the maximum combined flow limiter capacity.

Refer to the applicable reload licensing documents for the cycle-specific calculation results.

A typical loss of feedwater heating transient is shown in Figures 15.1-1 through 15.1-3.

The Loss of Feedwater Heating (LFWH) event was evaluated under EPU conditions. The analysis assumes a feedwater temperature reduction of 145^0F. A LHGRFACp multiplier was determined with and without credit for a high flux scram in Reference 4. Because the LFWH event is a cycle specific analysis, an applicable LHGRFACp reduction multiplier (if required) would need to be determined on a cycle specific basis and would be included in the Core Operating Limits Report (COLR), which is part of the Dresden Technical Requirements Manual or applicable cycle specific reload documents.

15.1.1.4 Barrier Performance

Since the maximum drop in CPR is typically about 0.20 or less and a typical operating limit MCPR is 1.46, the MCPR will remain above the Technical Specification Safety limit and the fuel cladding integrity safety limit is not violated. These results are re-evaluated for the current operating cycle. A cycle specific ΔCPR for the loss of feedwater heaters analysis can be found in the applicable cycle specific reload documents.

Since this transient does not appreciably increase reactor temperature or pressure, there is no possibility of overstressing the reactor coolant pressure boundary (RCPB).

15.1.1.5 <u>Radiological Consequences</u>

Since the fuel cladding integrity safety limit would not be violated, a radiological consequence analysis has not been performed.

15.1.2 <u>Increase in Feedwater Flow</u>

See the introduction to Section 15.1 for information regarding the use of details from this analysis description which may not be applicable to the current fuel cycle.

15.1.2.1 Identification of Causes and Frequency Classification

The increase in feedwater flow event is postulated on the basis of a single failure of a control device, specifically one which can directly cause an increase in coolant inventory by increasing the feedwater flow. The most severe applicable event is a feedwater controller failure during maximum flow demand. The feedwater controller is forced to its upper limit at the beginning of the event.

An increase in feedwater flow is classified as a moderate frequency event.

15.1.2.2 Sequence of Events and System Operation

The operating conditions and assumptions considered in this analysis are as follows:

A. Feedwater controller fails during maximum flow demand;

B. Maximum feedwater pump runout occurs;

C. The reactor is operating in the manual recirculation flow control mode, which provides for the most severe transient;

D. One relief valve is out-of-service; and

E. The final feedwater temperature is reduced by up to 120°F below rated feedwater temperature by out-of-service feedwater heaters.

A feedwater controller failure under these circumstances would produce the following sequence of events:

A. The reactor vessel receives an excess of feedwater flow;

B. This excess flow results in an increase in core subcooling (which results in a rise in core power) and an increase in the reactor vessel water level; and

C. The rise in the reactor vessel water level eventually leads to a high water level turbine trip and a feedwater pump trip. The reactor scram is initiated by the closure of the turbine stop valves.

Following the turbine and reactor trips, the plant would be in a stable condition. To recover, the operator must correct the feedwater controller malfunction and initiate a normal return to power. If an extended plant shutdown is required, steam bypass to the main condenser can be used to remove decay heat.

Turbine/Feed water pump trip logic provides signals to automatically trip turbine and FW pumps including the motor breakers. Two out of two trip logic receives input from four medium range level transmitters and four dedicated slave units. Two trip systems are comprised of two channels each. This arrangement ensures that a single channel failure will not result in a spurious trip; or a failure in one channel will not inhibit a valid trip from the other trip system. Analytical value for the trip set point is chosen at 201 inches above the TAF (TAF is 360 inches above vessel zero) for Westinghouse analysis and 203 inches for AREVA analyses (see cycle specific vendor input document). This elevation is well below the main steam nozzle to prevent water from entering the steam piping.

Additionally, procedures address the potential for RPV overfill events that may occur via the condensate system during low-pressure operation following the turbine and reactor trips. Procedure steps require an operator to trip all condensate pumps and close the isolation valves for the feedwater regulating valves if water level exceeds the high-level trip setpoint and continues to rise uncontrollably.

15.1.2.3 Core and System Performance

This transient is reanalyzed each fuel cycle as described in Section 15.0.2.4. A typical excess feedwater flow transient due to a maximum feedwater demand by the feedwater controller is shown in Figures 15.1-4 through 15.1-6.

Failure of the feedwater control system is postulated to lead to a maximum increase of feedwater flow into the vessel. As the excessive feedwater flow subcools the recirculating water returning to the reactor core, the core power rises and attains a new equilibrium if no other action is taken. The water level in the vessel rises until the sensed level exceeds the high-level trip point. At that point, the turbine trips, closing the stop valves to protect against spillover of subcooled water to the turbine. The compression wave created, though mitigated by bypass flow, increases the core pressure leading to a power excursion. The power increase is terminated by scram and by pressure relief from the bypass valves opening. The change in CPR calculated for this transient represents a bounding result, as shown in Figure 15.1-6.

The turbine bypass system performance assumptions for analyses using AREVA or Westinghouse methods can be found in the cycle-specific transient analysis input parameters document (i.e., PPD or OPL-W).

A feedwater controller failure at partial power gives a larger steam/feedwater flow mismatch. However, failure at rated power can be more severe in terms of maximum reactor pressure and minimum critical power ratio (MCPR). A feedwater controller failure event at rated power is similar to the turbine trip event at rated power with turbine bypass operable. However, for the feedwater controller failure event, the turbine trip signal occurs when the reactor is above rated power. Hence, this event can be limiting with respect to MCPR and is evaluated in reload analyses. The typical maximum drop in CPR (ΔCPR) has been calculated to be 0.38. Specific MCPR results for the Feedwater Flow Controller Failure Analysis can be found in the applicable cycle specific reload documents.

This event, also known as Feedwater Controller Failure – Maximum Demand (FWCF), was evaluated for EPU conditions in References 4 and 5. The FWCF event under EPU conditions is consistent with analysis at 2527 MWt rated power. The EPU analysis showed fuel thermal margin results are within acceptable limits.

The FWCF event is also analyzed assuming turbine bypass valves are not available to provide a basis for this possible mode of operation. The thermal limits and limitations associated with implementation of this EOOS option are provided in the Core Operating Limits Report.

For AREVA and Westinghouse analysis, this event is also considered for a cycle specific ASME overpressure analysis where the turbine bypass valves are assumed to be out of service and there is no direct scram off turbine stop valve closure.

This event is re-evaluated by AREVA and Westinghouse on a cycle-specific basis to show that fuel thermal margin results are within acceptable limits.

15.1.2.4 Barrier Performance

The fuel-specific operating limit MCPR is determined for each reload core. The operating limit MCPR is established to preclude violation of the fuel cladding integrity safety limit and ensure the resultant ΔCPR for events like the feedwater controller failure are within the thermal margin set by the operating limit MCPR. The reactor coolant pressure boundary integrity would be maintained since the maximum vessel pressure resulting from the feedwater controller failure event was evaluated to be less than 1150 psig, well below the 1375 psig maximum vessel pressure limit (typical value: cycle specific results can be found in applicable cycle-specific reload documents).

15.1.2.5 Radiological Consequences

Since the fuel cladding integrity safety limit would not be violated, a radiological consequence analysis has not been performed.

15.1.3 Increase in Steam Flow

See the introduction to Section 15.1 for information regarding the use of details from this analysis description which may not be applicable to current fuel cycle.

15.1.3.1 Identification of Causes and Frequency Classification

The event postulated is the failure of the turbine pressure regulator in the valve-open direction.

An increase in steam flow is classified as a moderate frequency event.

15.1.3.2 Sequence of Events and System Operation

The Main Steam Line Pressure setpoint has been changed from the value specified in reference 4. Reference 6 describes the change to Main Steam Line Pressure setpoint and demonstrates that the change does not affect conclusion of reference 4.

It is postulated that the pressure regulator malfunction would occur at reactor rated thermal power. If the pressure regulator fails in a wide open direction, the maximum control valve plus bypass valve demand would be limited by the control system. The vessel pressure, due to the excess steam flow to the turbine, would drop 100 psi in the first 10 seconds. Core flux is decreased significantly as the pressure drop increases the moderator void fraction of the core. When the steam line pressure decreases by about 100 psi, closure of the main steam isolation valves (MSIVs) is initiated by a Group 1 isolation signal, which occurs on reactor pressure less than 785 psig. In order to prevent unnecessary MSIV closure from spurious main steam line low-pressure signals, the logic actuation is delayed by a time delay relay in the circuit. The total channel response time, from the time the main steam line pressure reaches the analytical limit of ≤ 785 psig until power is removed from the MSIV solenoid is 500 milliseconds. The scram occurs when the MSIVs reach the analytical limit of ≤ 10% closed. The depressurization is stopped as soon as the isolation becomes effective. After the reactor is shut down and isolated, the pressure would rise slowly. The isolation condenser can dissipate the decay heat for long-term shutdown. A typical pressure regulator failure is shown in Figures 15.1-7 and 15.1-8 (based on Group I isolation at 850 psig main steam line pressure).

The above analysis depends on the Group 1 isolation signal (with consequent MSIV closure and scram) occurring approximately 10 seconds into the event. If the pressure regulator failure event were to occur when the initial RPV water level was just below the high-level alarm point, the resultant level swell would reach the alarm point in approximately 5 seconds. This alarm would trip the feedwater pumps and main turbine, causing a reactor scram. The bypass valves would open fully and depressurize the RPV.

The Dresden scram procedure directs the operator to turn the mode switch to the shutdown position immediately following a scram signal. If the operator were to shift the mode switch within 4 to 5 seconds after the high-level trip occurred, the Group 1 signal would be bypassed. The scram procedure further directs the operator to the normal shutdown procedure, which maintains vessel cooldown rate below the Technical Specification limit of 100°F per hour.

If the operator does not enter the shutdown procedure after turning the mode switch and does not manually close the MSIVs, then the vessel will blow down through the bypass valves and the water level will drop to the low-low level isolation analytical limit of -59 inches. The MSIVs would then automatically close and the high pressure coolant injection (HPCI) system would inject water into the vessel until the water level reached the high-level HPCI shutoff point. The introduction of relatively cold water from the condensate storage tank will cause cooldown of the

vessel. This cooldown will not exceed the design basis limit of 240°F and will cease following HPCI trip on low pressure. This cooldown would be bounded by inadvertent HPCI injection at power as described in Section 15.5.

Water from the condensate system will not reach the vessel, since vessel pressure remains above the condensate pump shutoff head (approximately 350 psig). In the unlikely event that the pressure falls to a point where appreciable flow from the condensate system to the vessel could occur, the feedwater regulating valves will control the flow to prevent overfilling the vessel.

Vessel level swell during the depressurization could be postulated to a point where water enters the HPCI steamline and thus renders HPCI unavailable as a consequence of the event. In this case, after closure of the MSIVs the isolation condenser (IC) would normally provide long term core cooling as discussed above. In the unlikely event that both HPCI and IC are not available during this event, ADS and low pressure ECCS (Core Spray and LPCI) would provide the necessary long term core cooling. Analysis of this event has been bounded by the analysis in Reference 7. The NRC approved the use of ADS and low pressure ECCS for long term cooling following a transient (loss of offsite power) and subsequent failure of both HPCI and IC during the Systematic Evaluation Program reviews (Reference 8).

15.1.3.3 Core and System Performance

This transient is not analyzed for reload cores since the fuel-specific operating limit MCPR is determined for each reload core based on bounding events for the cycle. The typical transient response to the failure of turbine pressure regulator in the open position is shown in Figures 15.1-7 and 15.1-8.

15.1.3.4 Barrier Performance

The increase in steam flow event is not limiting with respect to peak system pressure or MCPR. Therefore, the fuel cladding integrity safety limit would not be violated, and the RCPB integrity would be maintained.

15.1.3.5 Radiological Consequences

Since the fuel cladding integrity safety limit would not be violated, a radiological consequence analysis has not been performed.

15.1.4 References

1. Deleted

2. Deleted

3. Deleted

4. “Dresden and Quad Cities Extended Power Uprate, Task T0900: Transient Analysis,” GE-NE-A22-00103-10-01 Revision 0, October 2000.

5. “Dresden and Quad Cities Evaluation of Extended Final Feedwater Temperature Reduction” GE-NE-A13-00487-00-01P, Revision 1, August 2002.

6. GENE-0000-0010-4202-01P Rev. 0, “Engineering Evaluation of Impact on Transient and Safety Analyses of Reducing the Low Pressure Isolation Setpoint Analytical Limit to 785 psig Dresden Units 2 & 3 and Quad Cities Units 1 & 2.

7. "Dresden 2&3, Quad Cities 1&2, HPCI Steamline Water Carryover Evaluation," GENE-0000-0026-2389-00-P, Revision 0, February 2004

8. Letter from USNRC, Dresden 2 – SEP Topics V-10.B, RHR Reliability; V-11.B, RHR Interlock Requirements; and V11-3, Systems Required for Safe Shutdown (Safe Shutdown Systems Report), dated April 24, 1981.

9. ANF-1358(P)(A), Revision 3, “The Loss of Feedwater Heating Transient in Boiling Water Reactors” Framatome ANP, September 2005. (Unit 3 only)

10. ANP-3516P, Revision 0, “Dresden Unit 3 Cycle 25 Reload Safety Analysis,” AREVA, September 2016. (Unit 3 only)

15.2 DECREASE IN HEAT REMOVAL BY THE REACTOR COOLANT SYSTEM

AREVA ATRIUM 10XM methods and fuel are only applicable to Unit 3.

Some events described in this section have not been reanalyzed for the current fuel cycle because these events continue to be bounded by other events which are analyzed for the current fuel cycle. Although not reanalyzed, these events, including the associated assumptions and conclusions, continue to be part of the plant's licensing basis. The conclusions of these analyses are still valid; however, specific details contained in the descriptions and associated figures should be used only to understand the analysis and its conclusions. These specific details should not be used as sources of current fuel cycle design information.

For operation under EPU conditions, the events resulting in a decrease in heat removal by the reactor coolant system were analyzed under Reference 1. The events in this category are primarily represented in the EPU analysis guidelines by the turbine trip and load rejection transient events with the assumed failure of the turbine steam bypass function. The feedwater controller failure (maximum demand) event also includes some aspects of this area, since it involves a turbine trip (from high water level). Other pressurization events analyzed in Reference 1 include the MSIV closure with direct scram, load rejection with bypass, and a single MSIV closure. The loss of condenser vacuum is another type of turbine trip with bypass and is bounded by analyses without bypass operation. The Loss of Offsite AC Power and Loss of Normal Feedwater are similar events. These events result in initial power decreases and are not limiting with respect to thermal limits. For fuel thermal limits, these events are bound by the load rejection event.

For operation with SVEA-96 Optima2 fuel, the events resulting in a decrease in heat removal by the reactor coolant system were evaluated in Reference 5. The limiting events in this category are the load rejection without bypass and the turbine trip without bypass. Since these two events are nearly identical at full power operating conditions, an analysis is performed that bounds both events. The other events in this category (pressure regulator malfunction, load rejection with bypass, turbine trip with bypass, inadvertent closure of MSIVs with direct scram, loss of condenser vacuum, loss of offsite power, loss of normal feedwater flow) are more benign as the resulting power increases are not as severe.

For AREVA reloads, the events resulting in a decrease in heat removal by the reactor coolant system were evaluated in Reference 7. This evaluation determined that the potentially limiting events in this category are the load rejection without bypass (LRNB), the turbine trip without bypass (TTNB), and loss of stator cooling (LOSC). The other events in this category (Pressure regulator malfunction, load rejection with bypass, turbine trip with by bypass, inadvertent closure of MSIVs with direct scram, loss of condenser vacuum, loss of offsite power, and loss of normal feedwater flow) are benign and/or the consequences are bound by the LRNB and/or TTNB. For follow-on reloads, only the potentially limiting events will be evaluated on a cycle-specific basis.

15.2.1 Steam Pressure Regulator Malfunction

15.2.1.1 Identification of Causes and Frequency Classification

For the steam pressure regulator malfunction, the turbine pressure regulator is assumed to fail low (i.e., zero output). This event is classified as a moderate frequency event.

15.2.1.2 Sequence of Events and System Operation

If one of the three processors in the pressure controller failed low, the pressure controller would maintain control of the turbine valves with no change in pressure. If either one or two of the three pressure transmitters providing input to the pressure controller failed low, the turbine control valves would adjust to the pressure sensed by the functioning transmitter and a small change in pressure would occur. If a second processor or a third transmitter failed low, the turbine control valves will close resulting in an increase in reactor pressure.

15.2.1.3 Core and System Performance

If one of the three pressure transmitters and two of the three processors in the pressure controller remained functional, the transient would be similar to a pressure setpoint increase as shown in Section 4.3.2.3.4.4.

If a second processor or the third transmitter failed low, the turbine control valve will close resulting in an increase in reactor pressure leading to a reactor scram on high flux.

15.2.1.4 Barrier Performance

This transient is not analyzed for reload cores since the fuel-specific operating limit minimum critical power ratio (MCPR) is determined for each reload core based on bounding events for the cycle. The operating limit MCPR is established to preclude violation of the fuel cladding integrity safety limit. The steam pressure regulator malfunction is not considered as one of the limiting events for the fuel cycle.

15.2.1.5 Radiological Consequences

Since the fuel cladding integrity safety limit would not be violated, a radiological consequence analysis was not performed.

15.2.2 Load Rejection (Generator Trip)

A loss of generator load would cause the turbine-generator to speed up. The turbine speed governor would react by closing the turbine control valves. The reduction of steam flow would cause the reactor vessel pressure to rise, and the pressure regulator would open the turbine bypass valves in an attempt to maintain reactor pressure constant. If the load reduction were sudden and of a greater magnitude than the bypass valve capacity, the reactor pressure would rise. To prevent fuel damage and the lifting of reactor safety valves, a sudden generator load rejection will lead to a scram by the fast closure of the turbine control valves as discussed in Section 7.2.2.

The loss of generator load has been evaluated both with operation of the turbine steam bypass and with the failure of the turbine steam bypass, Section 15.2.2.2 and 15.2.2.1, respectively.

15.2.2.1 Load Rejection Without Bypass

The generator load rejection without bypass has been identified as one of the most limiting core-wide anticipated operational transient events relative to thermal limits to preclude violation of fuel cladding integrity. This transient is reanalyzed for reload cores.

This event, Load Rejection without Bypass (LRNBP), is identified as one of the limiting AOOs for Dresden licensing analyses. Therefore, this event was analyzed for EPU conditions (2957 MWt rated power) to determine operating limits and to verify safety margins. The transient results of Reference 1 show that the EPU ΔCPR for this event is about 0.02 higher than for the 2527 MWt rated power. This event is reanalyzed on a cycle specific basis. This event is also considered for a cycle-specific ASME overpressurization analysis where direct scram on turbine control valve fast closure is assumed unavailable. See the applicable cycle-specific reload documents.

15.2.2.1.1 Identification of Causes and Frequency Classification

The following plant operating conditions and assumptions form the principal bases for which reactor behavior is analyzed during a load rejection:

A. The reactor and turbine generator are initially operating at full power when the load rejections occurs;

B. All of the plant control systems continue normal operation;

C. Auxiliary power is continuously supplied at rated frequency;

D. The reactor is operating in the manual flow control mode when load rejection occurs (although the results do not differ significantly for operation in the automatic flow control mode);

E. The turbine bypass valve system fails in the closed position;

F. One relief valve is out-of-service; and

G. A final feedwater temperature reduction is not assumed each cycle for load rejection without bypass analyses. Unit 3 Cycle 15 analyses were performed with a feedwater temperature reduction of 100°F. These analyses resulted in a ΔCPR 0.04 lower than the ΔCPR with normal feedwater temperature.

This event is classified as a moderate frequency event.

The LRNBP event is also analyzed or evaluated for three EOOS options: the power/load unbalance system out of service option, TCV slow closure option, and pressure regulator out of service. The thermal limits associated with implementation of each EOOS option are provided in the Core Operating Limits Report.

15.2.2.1.2 Sequence of Events and System Operation

A complete loss of the generator load would produce the following sequence of events:

A. The power/load imbalance actuation steps the load reference signal to zero and closes the turbine control valves at the earliest possible time. The turbine accelerates at a maximum rate until the valves start to close. The power load imbalance generates a fast closure signal and the turbine control valves close in 0.150 seconds for the full valve stroke.

B. Reactor scram initiates upon sensing control valve fast closure signal.

C. If the pressure rises to the pressure relief setpoint, some or all of the relief valves would open, discharging steam to the suppression pool.

In parallel, the generator protective relaying will result in a generator lockout and turbine trip. Hence, a load rejection occurring, while the power/load actuation is being tested will not result in a more severe event than that analyzed in a cycle specific basis for the generator load rejection without bypass event.

Above 50% power, a scram is initiated by sensing the turbine generator load imbalance and sending an electrical signal to the fast acting solenoid. This results in control valve closure and reactor scram. It has been identified that at power levels between 38.5% and 50% rated thermal power, the Power Load Unbalance (PLU) device may not actuate and the turbine control system will initiate turbine control valve closure at normal speed, which would not generate a scram (Reference 2). This would occur if the PLU was calibrated to actuate at power levels between 38.5% and 50% rated thermal power.

15.2.2.1.3 Core and System Performance

Fast closure of the turbine control valves would be initiated whenever electrical grid disturbances result in significant loss of load on the generator. The turbine control valves would close as rapidly as possible to prevent overspeed of the turbine generator rotor. The closing would cause a sudden reduction of steam flow which would result in a reactor coolant system pressure increase. The reactor would be scrammed by the fast closure of the turbine control valves.

A typical transient response to the load rejection without bypass is shown in Figures 15.2-1, 15.2-2, 15.2-3. A typical calculated ΔCPR is 0.31. The cycle-specific MCPR for the generator load reject without bypass event can be found in the applicable cycle-specific reload documents.

For the situation that turbine control valve fast closures do not occur between 38.5% and 50% rated thermal power, a GE analysis accounting for how the plant actually behaves has been performed. This analysis credits the generator protection logic, which would initiate a turbine trip within 0.625 seconds of load rejection resulting in a turbine stop valve position scram. The analysis concludes that the equipment-in-service thermal limits, as confirmed in Reference 4, are bounding for this event (Reference 3). Similarly, AREVA and Westinghouse analyses assume TCV fast closure not occurring between 38.5% and 50% rated thermal power. See details in the applicable cycle-specific reload documents.

15.2.2.1.4 Barrier Performance

The fuel-specific operating limit MCPR is determined for each reload core. The operating limit MCPR is established to preclude violation of the fuel cladding integrity safety limit. The resultant ΔCPR for. the load rejection without bypass transient would be within the thermal margin set by the operating limit MCPR with a maximum ΔCPR of 0.35 (typical ΔCPR value from the limiting AOO such as the feedwater controller failure analysis in Section 15.1.2). The reactor coolant pressure boundary (RCPB) integrity would be maintained since the maximum vessel pressure resulting from the load rejection without bypass event was evaluated to be 1294 psig (typical value: Cycle specific results can be found in applicable cycle-specific reload documents), well below the 1375 psig maximum vessel pressure limit.

15.2.2.1.5 Radiological Consequences

Since the fuel cladding integrity safety limit would not be violated, a radiological consequence analysis was not performed.

15.2.2.2 Load Rejection with Bypass

15.2.2.2.1 Identification of Cause and Frequency Classification

The cause and frequency classification of the load rejection with bypass transient are the same as that for load rejection without bypass discussed in Section 15.2.2.1.

15.2.2.2.2 Sequence of Events and System Operation

A loss of generator load would cause a load rejection trip of the turbine generator. The control valves would fully close in about 0.15 seconds. The bypass system would be actuated simultaneously. The automatic load rejection scram signal is bypassed when the first stage turbine pressure is less than that corresponding to 38.5% rated core thermal power. Because steam flow is assumed to exceed the capacity of the bypass system, 33.5% of turbine design steam flow, an anticipatory load rejection scram would occur. Scram is initiated by pressure switches on the control valve solenoids which indicate fast control valve closure.

15.2.2.2.3 Core and System Performance

The pressure rise due to the valve closure would cause voids in the moderator to collapse and result in a spike in neutron flux before the scram shuts down the reactor. The pressure rise also would result in an increase in coolant saturation temperature and a momentary decrease in nucleate boiling and heat transfer from the fuel cladding.

15.2.2.2.4 Barrier Performance

This transient is not analyzed for reload cores since the load reject without bypass event is a more severe pressurization transient. The fuel-specific operating limit MCPR is determined for each reload core based on bounding events for the cycle. The operating limit MCPR is established to preclude violation of the fuel cladding integrity safety limit.

The Load Rejection With bypass (LRBP) event was analyzed for EPU conditions. This event assumes the fast closure of all turbine control valves (TCVs) without the failure of bypass valves. The EPU results of Reference 1 indicate that the event is bounded by the Load Rejection Without Bypass (LRNBP) event.

15.2.2.2.5 Radiological Consequences

Since the fuel cladding integrity safety limit would not be violated, a radiological consequence analysis was not performed.

15.2.3 Turbine Trip (Stop Valve Closure)

The turbine trip analyses without bypass and with bypass, presented in Sections 15.2.3.1 and 15.2.3.2, respectively, assume a reactor scram due to turbine trip (stop valve closure).

The analysis of a turbine trip coincident with failure of the turbine bypass system which was used to establish the design basis for the required capacity of the electromatic relief valves is discussed in Section 5.2.2.2.2. This event is also considered for a cycle-specific ASME overpressurization analysis where direct scram on turbine stop valve position is assumed unavailable. See the applicable cycle-specific reload documents.

15.2.3.1 Turbine Trip Without Bypass

15.2.3.1.1 Identification of Causes and Frequency Classification

A variety of turbine or nuclear system malfunctions will initiate a turbine stop valve closure and turbine trip. Some examples are moisture separator high levels, loss of control fluid pressure, low condenser vacuum, and reactor high water level. A turbine stop valve closure would cause a sudden reduction in steam flow which would result in a nuclear system pressure increase and the shutdown of the reactor.

This event is classified as a moderate frequency event.

15.2.3.1.2 Sequence of Events and System Operation

The plant operating conditions and assumptions are identical to those of the generator load rejection.

The sequence of events for a turbine trip would be similar to that for a generator load rejection. Stop valve closure occurs over a period of 0.10 second. Position switches at the stop valves sense the stop valve closure and initiate reactor scram signal before the stop valves are less than 90% open (Analytical Limit: 10% closed from full open).

As in the case with the load rejection scram, this scram signal would be bypassed if first stage turbine pressure were less than that corresponding to 38.5% of rated core thermal power. If the pressure were to rise to the pressure relief setpoints at equal to or less than 1112 and 1135 psig respectively, the relief valves would open and discharge steam to the suppression pool. Please refer to Table 5.2-1 for settings and relief valve identification.

15.2.3.1.3 Core and System Performance

The turbine stop valves would close as rapidly as possible. The closing would cause a sudden reduction of steam flow which would result in a nuclear system pressure increase. The reactor would be scrammed by the closure of the turbine stop valves.

15.2.3.1.4 Barrier Performance

The maximum drop in CPR (ΔCPR) calculated (typical value of 0.30) is adequate for protection of all fuel types against boiling transition. Since a typical rated conditions operating limit MCPR is 1.46 (typical value for OLMCPR, the cycle specific OLMCPR can be found in the Core Operating Limits Report or applicable cycle specific reload documents), the MCPR will remain above the Technical Specification Safety Limit and the fuel cladding integrity safety limit is not violated. The reactor coolant pressure boundary (RCPB) integrity would be maintained since the maximum vessel pressure resulting from the turbine trip without bypass event was evaluated to be 1279 psig (typical value), well below the 1375 psig maximum vessel pressure limit.

The Turbine Trip Without Bypass (TTNBP) event was analyzed at EPU conditions and the results presented in Reference 1 show that the event is similar to the Load Rejection Without Bypass (LRNBP) event. See Reference 1 for further guidance. See the cycle-specific reload licensing report (RLR) for details on the Westinghouse transient analysis results. See the cycle-specific reload safety analysis report (RSAR) for details on the AREVA Transient analysis results.

15.2.3.1.5 Radiological Consequences

Since the fuel cladding integrity safety limit would not be violated, a radiological consequence analysis was not performed.

15.2.3.2 Turbine Trip With Bypass

15.2.3.2.1 Identification of Causes and Frequency Classification

A turbine stop valve closure can be initiated by a variety of turbine or reactor system malfunctions (see Section 15.2.3.1.1).

This event is classified as a moderate frequency event.

15.2.3.2.2 Sequence of Events and System Operation

The sudden closure of the stop valves would cause a rapid pressurization of the steam line and reactor vessel with resultant void collapse and power increase. The reactor would scram from position switches mounted on the stop valves (turbine trip scram).

Closure of the stop valves would immediately initiate bypass valve opening via action of the electrohydraulic control (EHC) system.

15.2.3.2.3 Core and System Performance

If the steam flow exceeded the capacity of the bypass system, the sudden closure of the stop valves would cause a rapid pressurization of the steam line from the vessel resulting in a void collapse and neutron flux spike similar to that described for the load rejection. The relief valves would handle steam flow in excess of the bypass valve capacity.

15.2.3.2.4 Barrier Performance

This transient is not analyzed for reload cores since the turbine trip without bypass event is a more severe pressurization transient. The fuel-specific operating limit MCPR is determined for each reload core based on bounding events for the cycle. The operating limit MCPR is calculated to preclude violation of the fuel cladding integrity safety limit.

For plant operation under EPU conditions, the Turbine Trip With Bypass (TTBP) event was re-analyzed and the results presented in Reference 1 show that the event is milder than the Turbine Trip Without Bypass (TTNBP) event, since the availability of the bypass valves greatly reduced vessel pressurization rate and hence the associated power increase. Based on the non-limiting results from the similar Load Rejection With Bypass (LRBP) event, the TTBP event is also a non-limiting event. See the cycle-specific reload licensing report (RLR) for details on the Westinghouse transient analysis results. See the cycle-specific reload safety analysis report (RSAR) for details on the AREVA Transient analysis results.

15.2.3.2.5 Radiological Consequences

Since the fuel cladding integrity safety limit would not be violated, a radiological consequence analysis was not performed.

15.2.4 Inadvertent Closure of Main Steam Isolation Valves

The inadvertent closure of the main steam isolation valves (MSIVs) with direct scram and without direct scram is discussed in Sections 15.2.4.1 and 15.2.4.2, respectively.

15.2.4.1 Inadvertent MSIV Closure with Direct Scram

15.2.4.1.1 Identification of Causes and Frequency Classification

Various steam line and nuclear system malfunctions, or operator actions, can initiate MSIV closure. Some examples are low steam line pressure, high steam line flow, high steam line radiation, low water level, and manual action.

This event is classified as a moderate frequency event.

15.2.4.1.2 Sequence of Events and System Operation

A MSIV closure was assumed to isolate the steam lines at the containment boundary within 3 seconds. A scram would be initiated when the valves reach the 10% closed position.

15.2.4.1.3 Core and System Performance

A typical transient response to inadvertent closure of these valves from 2527 MWt is shown in Figures 15.2-4 and 15.2-5. No significant neutron flux or surface heat flux peaks would be encountered since the first 10% of the valve stroke (i.e., when the scram is initiated) does not reduce valve flow area. The relief valves would open to remove excess stored heat; safety valves, other than the Target Rock safety relief valve, would not actuate since the pressure would peak at 1144 psig well below the safety valve lowest setpoint of 1240 psig. The isolation condenser could be actuated to handle long-term decay heat removal.

For Core and system performance in EPU conditions, see Reference 1.

15.2.4.1.4 Barrier Performance

The inadvertent MSIV closure with direct scram event is not reanalyzed for each reload cycle since this event is bounded by the Load Rejection Without Bypass (Section 15.2.2.1) event due to the faster closure of the Turbine Control Valves (0.150 seconds versus 3 seconds).

The MSIV closure with direct scram (MSIVD) event was analyzed at EPU conditions under Reference 1. Results contained in Reference 1 show that the MSIVD event is not limiting event and that the transient response is very mild compared to the MSIV flux scram and Load Rejection Without Bypass (LRNBP) events. See Reference 5 for details on the Westinghouse reload method. AREVA concludes the MSIVD event is non-limiting in Reference 7.

15.2.4.1.5 Radiological Consequences

Since the fuel cladding integrity safety limit would not be violated, a radiological consequence analysis was not performed.

15.2.4.2 Inadvertent MSIV Closure Without Direct Scram

Closure of the MSIVs without direct scram and without credit for the relief valves is analyzed to assure compliance with the ASME Pressure Vessel Code, Section III.

The MSIV closure without direct scram event was analyzed under EPU conditions in Reference 1, and detailed results are contained in Reference 1.

15.2.4.2.1 Identification of Causes and Frequency Classification

Various steam line and nuclear system malfunctions, or operator actions, can initiate MSIV closure. Some examples are low steam line pressure, high steam line flow, high steam line radiation, low water level, and manual action.

The Summer 1968 Addenda to the 1968 Edition of Section III to the ASME code revised the conditions to be considered when performing pressure vessel stress analyses. Loads were to be considered from four categories of conditions:

1. Normal;

2. Upset;

3. Emergency; and

4. Faulted.

The Addenda defines an upset condition as any deviation from normal operating conditions caused by any single error, malfunction or a transient which does not result in a forced outage. These events are anticipated to occur frequently enough that design should include the capability to withstand the upsets without operational impairment. Emergency conditions are stated as having "...a low probability of occurrence..." and require shutdown for correction but cause no gross damage to the system. Additionally, faulted conditions are "...those combinations of conditions associated with extremely low probability postulated events..." which may impair the integrity and operability of the nuclear system to the point where public safety is involved.

As described in the Summer 1968 Addenda of Section III, the following pressure limits are applied to the operating limit category:

1. Under upset conditions, the code requires that reactor pressures are not to exceed 110% of design pressure (1.1 x1250 = 1375 psig).

2. For emergency conditions, it allows up to 120% of design pressure (1.2 x 1250 = 1500 psig).

3. For faulted conditions, it allows up to 150% of design pressure (1.5 x 1250 = 1875 psig).

For conservatism, the ASME overpressure analysis may assume the safety function of the Target Rock valve to be inoperable, although the Dresden Technical Specifications require all 9 safety valves to be operable. See the applicable reload analysis documents for the assumptions used by AREVA and Westinghouse.

In addition to assuring that the lower vessel pressure is below the conservatively applied ASME code limit of 1375 psig, the peak pressure calculated in the vessel steam dome is verified to be less than the Technical Specification Safety Limit for Reactor Coolant system Pressure (1345 psig as measured by the steam space sensor).

15.2.4.2.2 Sequence of Events and System Operation

Though the closure rate of the MSIVs is substantially slower than that of the turbine stop or control valves, the compressibility of the fluid in the steam lines would provide significant damping of the compression wave associated with the

turbine trip events to the point that the slower MSIV closure without direct scram results in nearly as severe a compression wave. Once the containment was isolated, the subsequent core power production would need to be contained within a smaller system volume than that associated with the turbine trip events. Comparative analyses have demonstrated that the containment isolation event under these conservative assumptions could result in a higher overpressure than either the turbine trip or the generator load rejection without bypass.

15.2.4.2.3 Core and System Performance

Due to valve characteristics and steam compressibility, the vessel pressure response would not be noted until about 3 seconds after the beginning of the valve stroke. Since credit is not taken for the MSIV closure scram in this analysis, effective power shutdown would be delayed until after 5 seconds following initiation of the MSIV stroke. Assuming a delay of the scram until the high flux trip setpoint is reached results in a more severe transient. The power operated relief valves (including the relief mode of the Target Rock valve) are assumed to fail, preventing that mechanism from assisting in the shutdown. The recirculation pump trip was assumed to occur at 1250 psig for Westinghouse analysis and 1200 psig for AREVA analysis. AREVA analyzes this event with and without the recirculation pump trip (Reference 7).

A typical transient response is shown in Figures 15.2-6, 15.2-7, and 15.2-8. Pressure in the steam lines for this typical transient was calculated to peak at 1307 psig at approximately 7.2 seconds. The maximum vessel pressure was calculated to be 1329 psig in the lower plenum occurring at 6.6 seconds. These values for peak pressures are for a typical MSIV Closure without Direct Scram and do not apply to a particular cycle. For the cycle specific results for the ASME overpressurization analysis, see the applicable cycle-specific reload documents.

15.2.4.2.4 Barrier Performance

The ASME overpressure event (as described in 15.2.4.2) is analyzed for every reload cycle to assure that this low probability, multiple failure event will not result in peak reactor pressure greater than that associated with the most conservative classification in ASME Section III (upset conditions).

15.2.4.2.5 Radiological Consequences

Since the reactor pressure safety limit would not be violated, a radiological consequence analysis was not performed.

15.2.5 Loss of Condenser Vacuum

15.2.5.1 Identification of Causes and Frequency Classification

The main condenser vacuum is assumed to be suddenly lost while the unit is operating at rated conditions.

This event is classified as a moderate frequency event.

15.2.5.2 Sequence of Events and System Operation

The following would occur due to the loss of the condenser vacuum:

A. Turbine low vacuum alarm

B. Condenser low vacuum alarm

C. Scram

D. Turbine stop valve closure

E. Turbine bypass valve closure

The worst case would occur if the loss of vacuum were instantaneous. In this event, the transient would become identical to the turbine trip with bypass failure included in Section 5.2.2.2 for sizing the primary system relief valves. The relief valves would preclude safety valve operation.

15.2.5.3 Core and System Performance

The majority of the stored heat would be removed by the relief valves and the isolation condenser would handle the remaining decay heat. Slower losses of condenser vacuum would produce less severe transients because the scram would precede the stop valve closure and some bypass flow would be permitted to remove stored heat.

15.2.5.4 Barrier Performance

This transient is not analyzed for reload cores since it is not a limiting transient as discussed above. The fuel-specific operating limit MCPR is determined for each reload core based on bounding events for the cycle. The operating limit MCPR is calculated to preclude violation of the fuel cladding integrity safety limit.

For operation under EPU conditions, the loss of condenser vacuum event was analyzed in Reference 1. The results contained in Reference 1 show that the loss of condenser vacuum event is another type of turbine trip with bypass and is bounded by analyses without bypass operation. See Reference 5 for details on the Westinghouse reload method. AREVA concludes the loss of condenser vacuum event is non-limiting in Reference 7.

15.2.5.5 Radiological Consequences

Since the fuel cladding integrity safety limit would not be violated, a radiological consequence analysis was not performed.

15.2.6 Loss of Offsite AC Power

The onsite power systems provide power to vital loads in the event of a loss of auxiliary power from offsite sources. The loss of offsite ac power is addressed in Section 8.3.

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Identification of Causes and Frequency Classification

A loss of feedwater transient is assumed to occur due to a feedwater controller malfunction demanding closure of the feedwater control valves.

This event is classified as a moderate frequency event.

15.2.7.2 Sequence of Events and System Operation

With an initial power level of 2527 MWt, the feedwater control valves are assumed to close at their maximum rate. The unit response to simultaneous tripping of all feedwater pumps would be very similar to the transient analyzed. The reactor water level would decrease rapidly due to the mismatch between the steam flow out of the vessel and the shut off feedwater flow. Low water level scram would occur after about 7.4 seconds. The recirculation loop Adjustable Speed Drives (ASD) for Unit 2 or Unit 3 would run down to 30 % speed demand when the feedwater flow drops below 20 % and a 15 second time delay expires. This interlock is used to protect the recirculation drive pumps from steady-state net positive suction head (NPSH) problems and the jet pumps from inefficiency due to cavitation.

For ASDs, the reactor scram recovery procedure directs the operator to verify runback of both pumps, and if needed, the operator also has the option to run back or to trip the recirculation pumps.

15.2.7.3 Core and System Performance

For operation at 2527 MWt rated power, transient response to this event is shown in Figures 15.2-9 and 15.2-10.

For operation at EPU conditions transient response, refer to figures in Reference 1.

The decrease in moderator subcooling would slightly decrease the neutron flux until scram occurs and completely shuts down the reactor. Vessel steam flow would closely follow the decay of fuel surface heat flux.

Analysis of the transient was discontinued at 16 seconds since the model was not programmed to handle the situation when core inlet subcooling becomes negative. Subsequent events would be complete drive motor trip, main steam isolation valve closure, and high pressure coolant injection (HPCI) initiation, all occurring when the water level drops to the low-low level setpoint. The time when this would occur, estimating from the established rate of level decrease, is about 33.5 seconds. Pressure would rise following the isolation and would eventually actuate the isolation condenser to handle the long-term shutdown heat removal.

Water inventory loss from 16 seconds until 36.5 seconds (the time the isolation valves would be closed) was conservatively estimated to be less than 550 cubic feet of saturated water. (At 16 seconds, vessel steam flow was 45 % of rated. For extreme conservatism, this rate was considered to exist until 36.5 seconds.)

Accounting for the above conservative inventory loss after 16 seconds and assuming the recirculation pumps would trip, an estimate of the final water level was made. All steam existing as carry-under and as voids in the core, upper plenum, standpipes and separators at 16 seconds was allowed to condense. The volume of water discharged to the scram discharge volume was assumed to be removed from the vessel. Even neglecting the inventory makeup from the HPCI system, the calculations showed that greater than 5 feet of water would remain above the core.

15.2.7.4 Barrier Performance

No thermal limits would be violated since the transient would be less severe than the turbine or generator trips. The fuel-specific operating limit MCPR is determined for each reload core based on bounding events for the cycle. The operating limit MCPR is calculated to preclude violation of the fuel cladding integrity safety limit.

For operation under EPU conditions, the Loss of Normal Feedwater event was analyzed under Reference 1. The results for the Loss of Normal Feedwater event contained in Reference 1 shows that power decreases at the initiation of the event and therefore, the event is not limiting with respect to thermal limits. For fuel thermal limits, this event is bound by the load rejection event.

See Reference 5 for details on the Westinghouse method. AREVA concludes the Loss of Normal Feedwater event is non-limiting in Reference 7.

15.2.7.5 Radiological Consequences

Since the fuel cladding integrity safety limit would not be violated, a radiological consequence analysis was not performed.

15.2.8 Loss of Stator Cooling

To protect the generator from over heating from the consequences of a loss of stator cooling, protection logic initiates an automatic runback of the turbine. The Loss of Stator Cooling (LOSC) event is characterized by a very slow turbine control valve (TCV) closure. The slow valve closure does not scram the reactor directly. Subsequently, the steam flow capacity of the turbine is reduced. If the reactor power exceeds the turbine plus bypass capacity, an increase in reactor pressure will occur. Without sufficient turbine plus bypass capacity, the event is terminated by an automatic reactor protection system (RPS) actuation on high reactor pressure or high neutron flux (high APRM scram). The LOSC event was analyzed by Westinghouse (Reference 6) and found to be bounded by other limiting events (e.g. LRNBP and FWCF). AREVA analyzed the LOSC event (Reference 7) and found it to be close to limiting at some off-rated powers and analyzes the LOSC event on a cycle-specific basis.

15.2.8.1 Identification of Causes and Frequency Classification

A loss of stator cooling signal can be generated in a number of ways, including low stator water flow, low stator pressure, or a high stator water temperature signal. This event is classified as a moderate frequency event.

15.2.8.2 Sequence of Events and System Operation

A loss of stator cooling would produce the following sequence of events:

A. The TCVs start to close slowly when a loss of stator cooling condition is sensed. The rated power analysis assumes the load set begins to runback from its initial setting of 105% to approximately 24% over a period of 157 seconds.
B. The bypass valve(s) opens as soon as the load set drops below actual load.
C. It is assumed there is a 20 second delay in feedwater temperature response.
D. Reactor pressure begins to increase when the steam production is greater than the combined capacity of the TCV and turbine bypass valves.
E. Reactor scram initiates upon sensing high pressure or high flux.

15.2.8.3 Core and System Performance

The turbine runback and very slow closure of the TCVs would be initiated whenever there is a loss of stator cooling. The runback of the TCVs would cause a relatively slow increase in reactor pressure, once the combined capacity of the TCV and turbine bypass is less than the steam production in the reactor vessel. The increase in coolant pressure causes a subsequent increase in reactor power. The reactor would scram on high pressure or possibly high flux.

15.2.8.4 Barrier Performance

As described in Reference 6, the loss of stator cooling is not considered to be one of the limiting events for the fuel cycle. This LOSC transient is checked but typically not analyzed for reload cores, since the fuel-specific operating limit minimum critical power ratio (MCPR) is determined for each reload core based on events that are more limiting than the LOSC event. For AREVA reloads, the LOSC event is considered a potentially limiting event at certain power levels and analyzed on a cycle-specific basis (Reference 7). The operating limit MCPR is established to preclude violation of the fuel cladding integrity safety limit. Cycle-specific results can be found in applicable cycle-specific reload documents for AREVA reloads.

15.2.8.5 Radiological Consequences

Since the fuel cladding integrity safety limit would not be violated, a radiological consequence analysis was not performed.

15.2.9 References

1. “Dresden and Quad Cities Extended Power Uprate Task T0900: Transient Analysis,” GE-NE-A22-00103-10-01 Revision 0, October 2000.

2. “SC04-15, “Turbine Control System Impact in Transient Analyses,” 10 CFR Part 21 Communication, October 31, 2004.

3. “Dresden Units 2 and 3 and Quad Cities Units 1 and 2 Offrated Analyses Below the PLU Power Level,” GE-NE-0000-0040-2860-R0, General Electric Company, July 2005.

4. “Dresden 2 and 3 Quad Cities 1 and 2 Equipment Out-Of-Service and Legacy Fuel Transient Analysis,” GE-NE-J11-03912-00-01-R3, Revision 3, September 2005.

5. “Westinghouse BWR Reload Licensing Methodology Basis for Exelon Generation Company Dresden Nuclear Power Station Units 2 and 3,” WCAP-16588 Revision 3, August 2006.

6. "Loss of Stator Cooling Evaluation," NF-BEX-12-139, Revision 1, January 2014.

7. ANP-3516P, Revision 0, “Dresden Unit 3 Cycle 25 Reload Safety Analysis”, AREVA, September 2016. (Unit 3 only)

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOWRATE

This section describes events which cause a decrease in reactor coolant system flowrates. The recirculation flow control system is described in Section 7.7.3.1.

The events described in this section have not been reanalyzed for the current fuel cycle because they continue to be bounded by other events which are analyzed for the current fuel cycle. Although not reanalyzed, these events, including the associated assumptions and conclusions, continue to be part of the plant's licensing basis. The conclusions of these analyses are still valid; however, specific details contained in the descriptions and associated figures should be used only to understand the analysis and its conclusions. These specific details should not be used as sources of current fuel cycle design information.

The NRC evaluated the loss of forced reactor coolant flow including trip of pump motor and flow controller malfunctions as a part of the Systematic Evaluation Program (SEP) for Dresden Unit 2. The NRC concluded that the Dresden Unit 2 design, with regard to transients that are expected to occur during plant life and result in a loss or decrease in forced reactor coolant flow, is acceptable and meets the relevant requirements of General Design Criteria (GDC) 10, 15, and 26. This conclusion was based on the following:

A. The specified acceptable fuel design limits would not be exceeded for these events since analysis showed that the thermal margin limits were satisfied;

B. The reactor coolant pressure boundary limits would not be exceeded for these events since analysis showed that the calculated maximum pressure of the reactor coolant and main steam systems would not exceed 110% of the design pressure; and

C. The reactivity control system would provide adequate control of reactivity during these events while including appropriate margin for stuck rods because the specific acceptable fuel design limits would not be exceeded.

The AOO events in this category are not limiting for any GE BWR. These events are not reevaluated for reloads and are not required to be in Reference 4 because they are not limiting, even after 20% power uprate. The decrease in core flow causes a decrease in reactor power and thermal limits are not challenged. However, the SLO pump seizure accident was analyzed under Reference 5 for the introduction of GE14 fuel and EPU conditions. Reference 6 evaluates which of the licensing basis events require evaluation as a result of the transition to SVEA-96 Optima2 fuel. Reference 6 concludes that the licensing basis events in this category are not limiting and do not require re-evaluation. Reference 7 identifies which of the licensing basis events require evaluation as a result of the transition to ATRIUM 10XM fuel. Reference 7 concludes that the licensing basis events in this category are not limiting and do not require re-evaluation except for the pump seizure while in SLO.

AREVA ATRIUM 10XM methods and fuel are only applicable to Unit 3.

15.3.1 Single and Multiple Recirculation Pump Trips

The transient responses of the plant have been analyzed for the trip of one recirculation pump or both recirculation pumps due to the trip of Unit 2 or Unit 3 ASDs while operating at full power. No reactor scram is assumed during these transients. However, a simultaneous trip of both drive motors implies a loss of auxiliary power, which would subsequently result in reactor scram.

Extensive tests and analyses were conducted during the original design of the reactor coolant system to evaluate the performance characteristics of the jet pumps and the recirculation system, particularly with respect to pump design requirements and the effect of the pumping system on hydraulic and nuclear stability. These analyses included the evaluations discussed in Section 5.4.1.3 and also included the evaluation of recirculation pump malfunctions discussed in the following subsections.

If one of the recirculation pumps were to fail, half of the jet pumps would coast down. Depending upon the speed of the active pump, the flow through jet pumps in the inactive loop may be forward, reverse, or stagnant. In this case, however, flow would reverse through the 10 idle jet pump diffusers. Recirculation flow and core flow would decay to a value lower than rated.

The other 10 jet pumps would continue to function. Since the core pressure drop would be reduced at the lower flow due to the core flow reduction in the active loop, the active jet pump flow ratio, i.e., inducted flow to driving flow, would increase. The driving flow would remain essentially constant since the loop hydraulic characteristics would not change.

Calculations for typical BWRs (see APED-5460[1]) show that the 10 active jet pumps would provide nearly 150% of their normally rated flow at the lower core pressure drop. Therefore, the total flow injected by the jet pumps would be 75% of rated flow. About 22% of rated flow bypasses the core through the idle diffusers resulting in a net of about 53% of rated flow going through the core. This lower-than-normal core flowrate would result in more core coolant void formation. Core power would drop to and stabilize at about 70% of rated. This one-pump trip transient would be less severe than the complete loss of pumping power transient which itself does not violate the minimum critical power ratio (MCPR) criteria. A gradual power decrease would be the only result. After a pump trip, power could be raised only by a flow increase or by rod motion. In either case there would be no violation of the MCPR. Protection against excessive power generation at a given flow is provided by the rod block interlocks. Loss of a driving pump, therefore, would not result in a serious flow loss or unstable operation of the remaining pumps.

Subsequent to the original design evaluations, a series of tests performed at the GE Moss Landing Test Facility verified previous performance predictions. Throughout these tests, basic performance data were collected under conditions duplicating, in all important respects, the temperatures, pressures, and flowrates expected to be encountered by the recirculation system. Concurrent with performing the tests at Moss Landing, the loop in which the tests were being performed was analytically modeled. Agreement between the analytical model results and the actual test results was quite good. The recirculation system analyses obtained are presented in Sections 15.3.2, 15.4.4, and 15.4.5, and in the following subsections.

15.3.1.1 Deleted.

15.3.1.2 Trip of One M-G Set Drive Motor

The results of a transient due to the trip of one drive motor would be less severe than the trip of both drive motors or the stall of one pump. Therefore, the thermal margins during this transient would be greater than either of those cases.

Flow would increase through the active loop jet pump diffusers and would finally provide about 75% of the original total jet pump diffuser flow. This flow would be split in the lower plenum with about 60% going through the core and the remainder providing reverse flow through the jet pump diffusers of the tripped loop. A small amount of forward flow would still be induced in the tripped drive loop due to the static pressure difference between the downcomer and the jet pump throat. The unit's response to the trip of one drive motor is shown on Figures 15.3-3 and 15.3-4.

This transient is not analyzed for reload cores since the fuel-specific MCPR LCO is determined for each reload core based on bounding events for the cycle. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit.

15.3.1.3 Trip of One Recirculation Pump Motor

For the ASDs this is the same as a single trip in Section 15.3.1.2.

15.3.1.4 Trip of Two Recirculation Pump Motors

The trip of two recirculation pump motors will be the same for the Unit 2 ASDs and the Unit 3 ASDs. Loss of power to both recirculation pump motors causes a transient similar to the trip of both M-G set drive motors, with a faster flow coastdown because the pump and motor would be decoupled from the M-G set inertia. The thermal margins would be greater than for the recirculation pump shaft seizure transient, which is the most limiting of the reactor coolant flow decrease transients.

This transient is not analyzed for reload cores since the fuel-specific MCPR LCO is determined for each reload core based on bounding events for the cycle. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit. The trip of two recirculation pump motors as it pertains to Anticipated Transients Without Scram is discussed in Section 15.8.

15.3.2 Recirculation Flow Controller Malfunctions

The equipment associated with the variable speed recirculation pump motors is designed with the basic objective that in spite of any failure, the operating pump speed should be maintained. However, the potential for failure in either direction (full speed or pump tripped) does exist. These failures have been analyzed and are discussed here and in Section 15.4.5.

For Unit 2 and Unit 3 ASDs, a failure in a ASD speed signal would cause the ASD output to either go to zero frequency, thus shutting off the motor voltage the same as any other trip, or would attempt to increase frequency until the programmed limit is reached. Additionally, if the programmed limit function failed and the motor continued to accelerate, the over frequency protective relays would trip the ASD feed and shut down the motor. The failure results in increase in pump speed, which has been analyzed at 2.5% rated rpm/sec for maximum run-up rate for normal operation and 42% rated rpm/sec for maximum run-up rate for faulted conditions (single failure).

The recirculation flow and thermal power decay resulting from the output going to zero frequency would be the same as analyzed in Section 15.3.1 for a trip of the recirculation pump ASD drive.

15.3.2.1 Recirculation Flow Controller Failure with Decreasing Flow

For Unit 2 and Unit 3, the ASD control ramp for reducing speed is also limited to approximately 2-1/2% per second, so this transient is not analyzed for reload cores since the fuel-specific MCPR LCO is determined for each reload core based on bounding events for the cycle. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit.

15.3.3 Recirculation Pump Shaft Seizure

Recirculation pump shaft seizure is assumed to occur as a consequence of an unspecified instantaneous stoppage of one recirculation pump shaft while the reactor is operating at full power.

The sudden decrease in core coolant flow while the reactor is at power would result in a degradation of core heat transfer. The reactor power would decrease in response to the reduced circulation flow and increased void formation. A reactor trip would not occur.

The transient response of the unit to seizure of one recirculation pump from 2527 MWt is shown in Figures 15.3-5 and 15.3-6. This case (assuming a 2-second shaft decay) represents the most rapid decrease of flow in a single drive loop. Jet pump diffuser flow on this loop reverses about 1.7 seconds after the seizure. This steady-state flow pattern is similar to the one pump trip; however, it is reached more quickly. The MCHFR for this transient was analyzed using the standard thermal evaluation power distribution giving the resulting minimum value of 1.20 near 2.14 seconds after the seizure.

The NRC evaluated the reactor coolant recirculation pump rotor seizure event for the Dresden Station as a part of the SEP. The NRC concluded that the consequences of such an event would meet the requirements set forth in the GDC 27, 28, and 31 regarding control rod insertability and core coolability. This conclusion was based upon the following:

A. There would be no fuel damage as a result of a postulated reactor coolant recirculation pump rotor seizure accident since the MCHFR would remain above the allowable limit; and

B. The requirements of GDC 31 with respect to integrity of the primary system boundary to withstand the postulated accident would be met.

MCPR is typically used in recent accident analyses rather than MCHFR. Since MCPR and MCHFR are not interchangeable, the MCHFR terminology in this analysis has not been revised. Westinghouse and AREVA BWR reload licensing analysis processes do not require re-analysis of this event (References 6 and 7, respectively).

15.3.4 Recirculation Pump Shaft Seizure While in Single Loop Operation

Seizure of the active recirculation pump shaft while the plant is operating with the other recirculation pump out of service is a postulated accident. During single loop operation, water flows through the jet pump diffusers in the inactive loop in the reverse direction. As a result of this seizure, inactive jet pump flow will change from negative flow to positive flow. Plant response to this accident is shown in Figures 15.3-7 through 15.3-9. Thermal hydraulic analysis has shown that less than 10% of the rods in the core would experience boiling transition during the accident. Vessel pressure would not increase and RCPB integrity would not be jeopardized. Offsite dose from any postulated fuel failures would not exceed 10% of 10 CFR 100 limits.

AREVA analyzes the SLO pump seizure event on a cycle-specific basis as an AOO. The results of this event are potentially limiting at off-rated conditions in single-loop operation. See the cycle specific Reload Safety Analysis report (RSAR) for details on the AREVA analysis results.

15.3.5 Recirculation Pump Shaft Break

The results of analyses for other BWR plants indicate that the single reactor coolant recirculation pump rotor seizure is more limiting than the pump shaft break event. This is because it produces a greater initial power to flow mismatch and more of a decrease in the MCPR. The recirculation pump rotor seizure is discussed in Section 15.3.3.

15.3.6 Jet Pump Malfunction

The effects of a malfunction of a single jet pump have been analyzed. If one of the jet pump nozzles is assumed to be plugged while the plant is operating at full power, two effects would be observed: flow would reverse through the blocked jet pump diffuser, bypassing some core flow, and the remaining 19 jet pumps would operate at a slightly higher flow ratio due to the altered hydraulic characteristics. The net effect would be a reduction in core flow to approximately 96% of rated (power approximately 97% of rated).

If malfunctions are assumed at the inlet of the diffuser, it should be noted that as long as the injection nozzle is functioning at least the nozzle flow will be injected into the bottom plenum of the vessel. The nozzle flow is about a third of the rated jet pump flow. The malfunction of one jet pump is bounded by the malfunction of one recirculation pump, since this would cause the loss of half the jet pumps.

This transient is not analyzed for reload cores since the fuel-specific MCPR LCO is determined for each reload core based on bounding events for the cycle. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit.

15.3.7 References

1. "Design and Performance of GE BWR Jet Pumps," General Electric Company, September 1968, APED-5460.

2. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," General Electric Company, January 1977, NEDO 10958-A.

3. Deleted

4. “Safety Analysis Report for Dresden 2 & 3 Extended Power Uprate,” NEDO-32962 P Revision 2, August 2001.

5. “Dresden and Quad Cities Extended Power Uprate, Task T0900: Transient Analysis,” GE-NE-A22-00103-10-01 Revision 0, October 2000.

6. “Westinghouse BWR Reload Licensing Methodology Basis for Exelon Generation Company Dresden Nuclear Power Station Units 2 and 3,” WCAP-16588-P Revision 3, August 2006.

7. ANP-3561P, Revision 0, “Dresden Unit 3 Cycle 25 Reload Safety Analysis”, AREVA, September 2016. (Unit 3 only)

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Some events described in this section have not been reanalyzed for the current fuel cycle because these events continue to be bounded by generic analyses or analyses for previous fuel cycles. Although not reanalyzed, these events, including the associated assumptions and conclusions, continue to be part of the plant's licensing basis. The conclusions of these analyses are still valid; however, specific details contained in the descriptions and associated figures should be used only to understand the analysis and its conclusions. These specific details should not be used as sources of current fuel cycle design information.

The events in this category that were analyzed for EPU conditions under Reference 24 are the Rod Withdrawal Error (RWE), Mislocated Fuel Assembly and the Recirculation Loop Flow Controller Failure.

AREVA ATRIUM 10XM methods and fuel are only applicable to Unit 3.

Reference 31 provides the basis for the events in this category that are analyzed to support operation with SVEA-96 Optima2 fuel. Reference 49 provides the basis for the events in this category that are analyzed to support operation with ATRIUM 10XM fuel.

Reference 43 evaluates replacing the M-G set with an ASD.

15.4.1 Uncontrolled Control Rod Assembly Withdrawal - Subcritical or Startup Condition

This transient was analyzed for the most severe initial condition, a reactor core just subcritical and the IRM subsystem not yet on-scale. The full withdrawal of the worst-case control rod was evaluated. The power at the peak was demonstrated to be within thermal limits. A detailed description of this evaluation may be found in section 7.6.1.4.3.

15.4.1.1 Control Rod Removal Error During Refueling

15.4.1.1.1 Identification of Causes and Frequency Classification

This event considers an arbitrary full withdrawal of the most reactive control rod during refueling. Such an event is categorized by the frequency of a limiting fault. The probability of the initiating causes alone is considered low enough to warrant its being categorized as an infrequent incident because there is not a practical set of circumstances which can result in an inadvertent RWE while in the REFUEL mode.

15.4.1.1.2 Sequence of Events and Systems Operation

The refueling interlocks prevent any condition that could lead to a control rod withdrawal error during refueling, thus an inadvertent criticality is precluded.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the core. This requirement is backed up by refueling interlocks on rod withdrawal and on movement of the refueling platform. When the mode switch is in the "REFUEL" position, the interlocks prevent the platform from being moved over the core if a control rod is withdrawn and fuel is on the hoist. Likewise, if the refueling platform is over the core and fuel is on the hoist, control rod motion is blocked by the interlocks.

When the mode switch is in the REFUEL position, only one control rod can be withdrawn. A second rod cannot be selected (select block), which thereby prevents the withdrawal of more than one rod at a time. Because the core is designed to meet shutdown requirements with the highest worth rod withdrawn, the core remains subcritical even with one rod withdrawn.

In addition, the design of the control rod, incorporating the velocity limiter, does not physically permit the upward removal of the control rod without the simultaneous or prior removal of the four adjacent fuel bundles. This precludes any hazardous condition.

No operator actions are required to preclude this event because the plant design prevents its occurrence. Even if the operator somehow withdraws one rod, the electrical interlocks prevent withdrawal of the second rod.

15.4.1.1.3 Core and System Performance

Subsection 4.3.2 contains the shutdown margin analysis.

No mathematical models were involved in this event. The need for input parameters or initial conditions were not required as there are no results to report. Consideration of uncertainties is not appropriate.

The probability of inadvertent criticality during refueling is precluded, hence the core and system performances were not analyzed. However, it is well known that withdrawal of the highest worth control rod during refueling results in a positive reactivity insertion but not enough to cause criticality. This is verified experimentally by performing shutdown margin tests during the startup series.

15.4.1.1.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since there is not a postulated set of circumstances for which this event could occur.

15.4.1.1.5 Radiological Consequences

An evaluation of the radiological consequences was not made for this event since no radioactive material is released from the fuel.

15.4.2 Rod Withdrawal Error - At Power

See the introduction to Section 15.4 for information regarding use of details from this analysis description which may not be applicable to the current fuel cycle.

15.4.2.1 Identification of Causes and Frequency Classification

The rod withdrawal error (RWE) transient evaluation assumes the reactor is operating at a power level above 25% of rated power at the time the control rod withdrawal error occurs; that the reactor operator has followed procedures; and, up to the point of the withdrawal error, the reactor is in a normal mode of operation (i.e., the control rod pattern, flow setpoints, etc., are all within normal operating limits). For these conditions, it is assumed that the withdrawal error occurs with the maximum worth control rod. Therefore, the maximum positive reactivity insertion would occur.

While operating in the power range in a normal mode of operation, the reactor operator is assumed to make a procedural error by withdrawing the maximum worth control rod to its fully withdrawn position. Due to the positive reactivity insertion, the core average power would increase. More importantly, the local power in the vicinity of the withdrawn control rod would increase and could cause cladding damage either by overheating, which may accompany the occurrence of boiling transition, or by exceeding the 1% plastic strain limit imposed on the cladding.

The rod withdrawal error is considered a moderate frequency event.

15.4.2.2 Sequence of Events and System Operation

The following list depicts the sequence of events for this transient.

A. Event begins, operator selects the maximum worth control rod, acknowledges any alarms, and withdraws the rod at the maximum rod speed, at 0 seconds;

B. Core average power and local power increase causes local power range monitor alarm, at ≤5 seconds; and

C. Event ends - rod block by rod block monitor (RBM), at ≤30 seconds (Cycle-specific analysis may not credit the RBM in which case the rod withdrawal error event terminates when the rod is its full out position).

The worst-case situation is established for the most reactive reactor state and assumes that no xenon is present. The absence of xenon ensures that the maximum amount of reactivity excess must be controlled with the movable control rods.

During a normal startup, sufficient time would be available to achieve some xenon and samarium buildup, and after some short period of operation, samarium would always be present. This assumption makes it possible to obtain a worst-case situation in which the maximum worth control rod is fully inserted and the remaining control rod pattern is selected in such a way as to achieve design thermal limits in the fuel bundles near to the inserted maximum worth control rod which is to be withdrawn. It should be noted that this control rod configuration would be highly abnormal and could be achieved only by deliberate operator action or by numerous operator errors during rod pattern manipulation prior to the selection and complete withdrawal of the maximum worth rod.

15.4.2.3 Core and System Performance

The cycle-specific analysis considers the continuous withdrawal of the maximum worth control rod at its maximum drive speed from the reactor, which is assumed to be operating at rated power with a control rod pattern which results in the core being placed on thermal design limits, i.e., MCPR at the Technical Specification limiting values. A worst case condition is analyzed to ensure that the results obtained are conservative.

Results for this worst case condition for the reload are given in the cycle-specific licensing documents. The analysis results from a typical reload licensing package indicate that the ΔCPR calculated for the RWE is 0.3 (for an unblocked rod withdrawal). The calculated peak linear heat generation rate (LHGR) during the RWE is compared to the LHGR limit associated with 1% plastic strain in the cladding to ensure that fuel damage is not expected during the event. The maximum control rod drive withdrawal speed is 5.14 inches/second when the Operating Limit MCPR established in the Core Operating Limits Report (COLR) is set greater than or equal to the value corresponding to a RWE – at Power analysis for an “unblocked” condition (Reference 27).

The RWE was analyzed under EPU conditions in Reference 24. This event met the mechanical overpower (MOP) limits criteria.

The RWE was analyzed to support the introduction of SVEA-96 Optima2 fuel. For Westinghouse reload cores, the RWE event is analyzed for potentially limiting cycle reactivity conditions, including the most reactive statepoint during the cycle. All control rods with a potential for being limiting are evaluated. The analysis is performed to select control rod patterns that would approach the thermal limits (e.g., MCPR and LHGR limits) in the region of the core where the rod is being erroneously withdrawn. The control rod patterns analyzed need not be consistent with normal control rod patterns. As a result, the control rod patterns analyzed are highly unlikely to occur, are very conservative, and establish a limiting analysis. The results of the analysis, which are documented in cycle-specific reload licensing reports, show that the MFLPD and maximum steam flow limit criteria are met. In the event that the maximum combined steam flow limit criteria cannot be met, power restrictions will be established based on the complement of bypass valves in service.

Similarly, the RWE was analyzed to support the introduction of the ATRIUM 10XM fuel design. For AREVA reloads, the RWE is analyzed as a potentially limiting event on a cycle-specific basis for potentially limiting reactivity conditions including the most reactive statepoint in the cycle. All control rods with a potential for being limiting are evaluated. The control rod patterns are chosen to put the core on the CPR limit in the region surrounding the error rod. Consequently, the error rod patterns are not typical of actual operation but are conservative and establish a limiting basis for the analysis. The results of the analysis are provided in the cycle specific Reload Safety Analysis Report (RSAR) and may include power dependent LHGR multipliers to ensure that the Fuel Design Limit (FDL) for each fuel design is protected. In the event that the maximum combined flow limit criteria cannot be met, the analysis will include the impact of pressurization up to the scram pressure limit.

15.4.2.4 Barrier Performance

The fuel-specific operating limit MCPR is determined for each reload core based on bounding events for the cycle. The operating limit MCPR is established to preclude violation of the fuel cladding integrity safety limit. The cycle specific value for the ΔCPR due to the RWE event is included in the applicable cycle-specific reload documents.

For AREVA reload cores, power dependent LHGR multipliers are established as required to ensure that the FDL for each fuel design is protected. These LHGR multipliers are included in the cycle specific RSAR.

15.4.2.5 Radiological Consequences

Since the fuel cladding integrity safety limit would not be violated, a radiological consequence analysis has not been performed.

15.4.3 Control Rod Misoperation

The limiting control rod misoperation events have been analyzed as discussed in Sections 15.4.1 and 15.4.2.

15.4.4 Startup of Inactive Recirculation Loop at Incorrect Temperature

See the introduction to Section 15.4 for information regarding use of details from this analysis description which may not be applicable to the current fuel cycle.

15.4.4.1 Identification of Causes

Startup of an inactive recirculation loop at incorrect temperature would require multiple operator errors. The initial conditions given below assume startup of the idle recirculation pump with the operating pump at 90% rated drive flow. Procedures require warming the idle loop and running back the operating recirculation pump to 30% (minimum speed) before starting the idle loop (as well as meeting the differential temperature limit).

The startup of an inactive recirculation loop at incorrect temperature has been classified as a moderate frequency event.

15.4.4.2 Sequence of Events and System Operation

The initial conditions assumed for the startup of an inactive recirculation loop at incorrect temperature are as follows:

A. One drive loop is shutdown and filled with cold water (110°F). (Normal procedure requires warming this loop to prevent thermal shock to the pumps and piping.)

B. The active recirculation loop is operating at about 90% of rated drive flow and 130% of normal rated diffuser flow in the 10 active jet pumps.

C. The core is receiving 50% of its normal flow, while the remainder of the flow is reversed up the 10 inactive jet pumps.

D. Reactor power is 30% of 2527 MWt. (25.6% of 2957 MWt)

E. The drive pump suction valve is open and the discharge valve is shut until minimum pump speed is established, then the discharge valve is jogged open. The equalizer line valves are closed on Unit 2. There are no equalizer line valves on Unit 3.

15.4.4.3 Core and System Performance

(Begin Historical Information)

The transient response of the plant to the starting of an idle recirculation loop without warming the drive loop water is shown in Figures 15.4-1 and 15.4-2. The results are from the original GE analysis that is maintained for information only.

Neutron flux would show a fairly sharp, small peak (40%) shortly after the actual pump excitation due to the slight peak in core inlet flow which would occur (about 55%). Core flow would subsequently increase slowly to its final value (near 56%). Throughout the transient, diffuser flow in the startup loop jet pumps would remain reversed. For this reason, the cold water would not significantly affect the reactor since it would flow out the suction of the jet pumps, mix in the downcomer region, and finally reach the lower plenum and core inlet through the active, forward flowing jet pumps. Peak fuel surface heat flux of 34%would occur near the end of the transient. No thermal limits would be approached.

For GE methodology, the Idle Recirculation Loop Startup (IRLS) was considered generically for the application of ARTS. For the application of ARTS (power and flow dependent limits), the IRLS basis is that there is an initial 50°F ΔT between operating loops. This is the appropriate assumption for thermal limits calculations and is consistent with Technical Specifications. Reference 24 provides further details of this event.

(End Historical Information)

For Westinghouse, the event is described in Reference 43. For AREVA, the event was reviewed and determined to be a non-limiting event bounded by other events and no additional analyses were performed (Reference 49).

15.4.4.4 Barrier Performance

This transient is not analyzed for reload cores since the fuel-specific operating limit MCPR is determined for each reload core based on bounding events for the cycle. The operating limit MCPR is established to preclude violation of the fuel cladding integrity safety limit.

15.4.4.5 Radiological Consequences

The fuel cladding integrity safety limit would not be violated. A radiological consequence analysis has not been performed.

15.4.5 Recirculation Loop Flow Controller Failure with Increasing Flow

This event (the fast recirculation flow increase transient) is not considered a limiting event and as such is not normally included as an AOO to be performed during reload analyses. The analysis results indicated in Reference 24 show that this event is not limiting for EPU conditions. The CPR result of Reference 24 is bounded by the ARTS flow dependent MCPR limit, $MCPR_F$. The design basis flow increase event is a slow recirculation flow increase which is not terminated by scram. This event is evaluated on a cycle specific basis.

A slow increase in recirculation flow resulting from the flow control failure is analyzed as a part of the cycle specific reload safety analysis. The event may also be analyzed as a pressurization transient if the maximum combined steam flow limit as specified in the reload transient analysis input document is exceeded during simulation as a slow event (e.g., for certain EOOS conditions).

Reference 43 describes the impact of the ASD system on this event. The current slow flow increase analysis can be found in the cycle specific reload report.

15.4.5.1 Identification of Causes

The recirculation flow control system is designed to accept step load changes of any magnitude in the design flow control range of the plant. The maximum flow control range is typically from 58 to 100% power (see Figure 4.4-1). Therefore, the recirculation flow control system could accept positive step changes in load of about 42% of rated power without reactor trip or steam dump. The prevention of

reactor trip for positive step changes in demand power is accomplished by the speed rate limit of the individual recirculation loop adjustable speed drive (ASD). The speed rate limit is adjusted to prevent a reactor scram from the neutron flux increase resulting from the increased core flow.

Recirculation loop flow controller failure with increasing flow has been classified as a moderate frequency event. This transient would result from the failure of the primary/backup ASD controls and the independent basler over-frequency protection relays.

15.4.5.2 Sequence of Events and System Operation

The slower the increase in neutron flux, the more severe the transient would be, since correspondingly more time would be allowed for the cladding heat flux to build up before the neutron flux scram.

A failure would cause the ASD system to increase the flow demand up to an assumed rate of 37.5% pump speed per second to maximum demand signal. A slower rate on increase would allow a gradual increase in neutron flux and cladding heat flux until a high neutron flux scram occurred. Consequently, the fast RFCF flow run-up for either dual pump or a single pump due to an ASD flow controller failure is not evaluated on a cycle specific basis. The run-up rates for the slow run-up event are specified in the cycle specific reload analysis. The slow pump run-up event is evaluated as part of the determination of the flow dependent limits for MCPR and LHGR (References 43 and 49).

15.4.5.3 Core and System Performance

The core impact of various rates of increase was evaluated in Reference 43. It determines that the results a fast rate of increase were bounded by other fast transients. The results of the slow rate of increase were similar to the slow rate of increase events analyzed previously with the MG-Sets.

The flow dependent MCPR limits for a particular cycle can be found in the Core Operating Limits Report (in the Dresden Technical Requirements Manual).

15.4.5.4 Barrier Performance

The slow recirculation flow increase event is analyzed on a cycle-specific basis. This event may establish flow dependent operating limits. Verification that the maximum steam flow during this event is within the plant's maximum combined steam flow limit capability is also performed. If the steam flow limit is exceeded, the impact of the system pressurization is included in the analysis.

15.4.5.5 Radiological Consequences

The fuel cladding integrity safety limit would not be violated. A radiological consequence analysis has not been performed.

15.4.6 Chemical and Volume Control System Malfunction

This event is not applicable to Dresden Station.

15.4.7 Mislocated Fuel Assembly Accident

15.4.7.1 Identification of Causes and Frequency Classification

A mislocated fuel assembly is an assembly which is loaded in an incorrect core position and not subsequently identified and corrected prior to core operation. The fuel assembly could then be monitored incorrectly, possibly resulting in a high reactivity, or limiting assembly, being modeled during the cycle as a low reactivity, or nonlimiting assembly.

The mislocated fuel assembly is characterized as an infrequent event (i.e. infrequence incident) in AREVA methodology (Reference 44).

15.4.7.2 Sequence of Events and System Operation

In a mislocated position, a fuel assembly that was expected to be surrounded by relatively low reactivity assemblies may instead be surrounded by relatively high reactivity assemblies. Therefore, an undetected and uncorrected mislocation of a fuel assembly may result in a degradation of MCPR margin and LHGR margin.

15.4.7.3 Core and System Performance

For Westinghouse reload cores, relatively high reactivity assemblies are selected as candidates to be mislocated to an unmonitored core location that is intended for a relatively low reactivity assembly. It is assumed that the mislocation is undetected throughout the cycle and that the loading error could result in a reduction in the MCPR at any point during the cycle. The effect of each mislocation on core MCPR is determined by depleting the core with the 3-D simulator. Both the mislocated and misoriented bundle events are evaluated on a cycle-specific basis. A complete description of the Westinghouse analysis methodology and assumptions for the mislocated fuel assembly accident is discussed in Section 8.5.1 of Reference 32.

For AREVA reload cores, high reactivity fuel assemblies are shuffled into locations that are face adjacent to the limiting CPR and LHGR in the core. Comparisons of the cycle depletions between the original and modified core loadings are used to determine if the potential impact on CPR and LHGR thermal limits. If the MCPR safety limit and the FDL remain protected then no fuel failures would occur. If either of these is exceeded then further evaluation would be required to verify that the total number of potential rod failures would not result in releases that would exceed the requirements for an infrequent incident (i.e. small fraction of the limiting accident). The mislocated fuel assembly is evaluated for each reload core but the evaluation may utilize previous bounding or generic analyses. The NRC approved methodology is discussed in References 44 and 45.

15.4.7.4 Barrier Performance

The fuel-specific Operating Limit Minimum Critical Power Ratio (OLMCPR) is determined for each reload based on bounding events for the cycle. The OLMCPR is established to preclude violation of the fuel cladding integrity safety limit.

For AREVA reload cores, the CPR and LHGR limits will be verified or established to protect the underlying SLMCPR and FDL requirements or an evaluation will be performed to ensure that any potential fuel failures do not provide releases that could exceed the license limits for an infrequent incident. The results of the analysis are contained in the cycle specific reload analysis.

15.4.7.5 Radiological Consequences

The fuel cladding integrity safety limit would not be violated. A radiological consequence analysis has not been performed.

For AREVA reload cores, an evaluation will be performed to ensure that any potential fuel failures do not provide releases that could exceed the license limits for an infrequent incident if it is determined that the underlying SLMCPR or FDL may be exceeded. The results of the analysis is contained in the cycle specific reload analysis.

15.4.8 Misoriented Fuel Assembly Accident

See the introduction to Section 15.4 for information regarding use of details from this analysis description which may not be applicable to the current fuel cycle.

15.4.8.1 Identification of Causes and Frequency Classification

Only the 90° and 180° rotations are investigated; the 270° rotation is equivalent to the 90° rotation due to the symmetry of BWR bundles.

The misoriented assembly has a lower frequency of occurrence than moderate frequency but is evaluated as a moderate frequency event.

The misoriented fuel assembly is characterized as an infrequent event (i.e. infrequent incident) in AREVA methodology (Reference 44).

15.4.8.2 Sequence of Events and System Operation

Dresden Station is a D-lattice plant utilizing partially symmetrized fuel (40 mil offset). The water gaps are nonuniform. An undetected and uncorrected misorientation of the fuel assembly may result in larger than anticipated local peaking on the wide-wide side of the fuel assembly since the wide-wide side has the larger water gap, and hence, greater neutron thermalization. This may lead to a degradation of MCPR margin and LHGR margin.

15.4.8.3 Core and System Performance

For Westinghouse reload cores, the internal power distribution is the key parameter in establishing the impact of the rotation on MCPR. Consequently, the two dimensional lattice physics code is used to determine the impact on CPR; specifically, the internal power distribution factor required for the MCPR evaluation (e.g., R-factor) is calculated. Calculations are performed to compute R-factors for the case of four fresh assemblies, four depleted assemblies, and 3 depleted assemblies with one fresh assembly at the nominal orientation and misorientation. Based on the change in power distribution factor associated with the rotation, the change in CPR is calculated at rated power and flow conditions as a function of burnup. A complete description of the Westinghouse analysis methodology and assumptions for the misoriented fuel assembly accident is discussed in Section 8.5.2 of Reference 32.

For AREVA reload cores, the change in internal peaking within the assembly is the key parameter evaluated to determine the impact on CPR and LHGR. The NRC approved methodology is discussed in References 44 and 45. However, the specific approach used to evaluate the impact of a misorientation is dependent upon the CPR correlation used for the specific fuel design. For fuel designs monitored with the SPCB (Reference 46) correlation, the CASMO-4 lattice physics code is used to determine the impact on peaking and Feff. The ACE correlation (Reference 47) has the ability to predict the axial location of the MCPR within the assembly, so a more detailed evaluation is performed using the MICROBURN-B2 reactor simulator code for fuel designs monitored using this correlation. The CASMO-4/MICROBURN-B2 codes are NRC approved (Reference 48). If either SLMCPR for FDL is determined to be potentially exceeded then further evaluation would be required to verify that the total number of potential rod failures would not results in released that would exceed the requirements for an infrequent incident (i.e., a small fraction of the limiting accident). The misoriented fuel assembly is evaluated for each reload core but the evaluation may utilize previous bounding or generic analyses.

15.4.8.4 Barrier Performance

The fuel-specific Operating Limit Minimum Critical Power Ratio (OLMCPR) is determined for each reload based on bounding events for the cycle. The OLMCPR is established to preclude violation of the fuel cladding integrity safety limit.

For AREVA reload cores, the CPR and LHGR limits will be verified or established to protect the underlying SLMCPR and FDL requirements or an evaluation will be performed to ensure that any potential fuel failures do not provide releases that could exceed the license limits for an infrequent incident. See the cycle specific reload analysis.

15.4.8.5 Radiological Consequences

The fuel cladding integrity safety limit would not be violated. A radiological consequence analysis has not been performed.

For AREVA reload cores, an evaluation will be performed to ensure that any potential fuel failures do not provide releases that could exceed the license limits for an infrequent incident if it is determined that the underlying SLMCPR or FDL may be exceeded. See the cycle specific reload analysis.

15.4.9 Control Rod Ejection Accidents (PWR)

Control rod ejection accidents are not applicable to Dresden Station.

15.4.10 Control Rod Drop Accident

15.4.10.1 Identification of Causes and Frequency Classification

The control rod drop accident (CRDA) is defined as a power excursion caused by accidental removal of a control rod from the core at a more rapid rate than can be achieved by the use of the control rod drive mechanism. In the CRDA, a fully inserted control rod is assumed to fall out of the core after becoming disconnected from its drive and after the drive has been removed to the fully withdrawn or an intermediate position.

The CRDA is considered a limiting fault.

15.4.10.2 System Operation

The control rods are designed to minimize the probability of a rod sticking in the core. The blades of the control rods travel in gaps between the fuel channels with approximately 1/2-inch total clearance and are equipped with rollers or pads which make contact with the channel walls. Control rods of similar design are now in use in a number of operating reactors, and periodic inspections have revealed no tendency for blade distortion or swelling (that could potentially lead to control rod sticking) due to services in the reactor environment.

The control rod coupling to the drive shaft and other control rod drive improvements which have been made over early designs significantly reduce the probability of an accidental separation of a control rod from a drive (see Section 4.6.1.3). Couplings of this design have undergone extensive tests under simulated reactor conditions and also at conditions more extreme than those expected to be encountered in reactor service. They have been operated through thousands of cycles of scram operation and a separation has never occurred. Tests have shown that the coupling will not separate when subjected to pull forces up to at least 20 times greater than can be applied with a control rod drive.

Operating procedures require rod-following verification checks during startup and during major rod movements, weekly verification checks on fully withdrawn rods, and monthly verification checks on partially withdrawn rods to insure that any rod-from-drive separation would be detected. Procedures require full insertion of rods when following cannot be verified.

Operating procedures require that control rod movements follow preplanned patterns designed to flatten the power distribution. Flattening the power distribution tends to minimize the reactivity worth of individual rods, so that extensive fuel damage would not be expected if a control rod drop were to occur.

15.4.10.3 Core and System Performance

The Westinghouse methodology (Reference 33) for the CRDA event is comprised of two steps:

1. The determination of candidates for the limiting control rod using the steady-state 3D simulator (POLCA7).

2. The determination of the resulting energy deposition in the fuel using the transient 3D plant analysis code (RAMONA-3) for each of the potentially limiting cases identified in the POLCA scoping evaluation.

In determining candidates for the limiting control rod, consideration is given both to the total reactivity worth of each rod considered and to the nodal peaking factor with the control rod in its final location. This approach ensures that the impact of the dropped control rod on peak fuel enthalpy is not underestimated. The potentially limiting cases are those for which the failure threshold of 170 calories/gm could be achieved during a CRDA.

The transient analysis model is usually a full-core calculation to account for asymmetric effects. The initial conditions, such as core power, flow, inlet enthalpy, control rod pattern and vessel pressure are the same as the corresponding conditions in the POLCA case. While in principal the accident is evaluated throughout the range from cold critical to 10% power, it has been shown in practice that the accident is most limiting when initiated from a sufficiently subcooled condition to avoid saturation conditions during the transient. The transient 3D plant analysis makes use of a combination of bounding and nominal inputs. For example, the dropped rod velocity was assumed to be a bounding value of 3.11 ft/s. Control rod velocity limiter tests[3] have shown that 3.11 ft/s is the maximum rod drop velocity that can be achieved for control rods incorporating the velocity limiter design. Similarly, scram worth, scram velocity, scram delay, initial coolant density, and nodal peaking are set at bounding values. However, other parameters, which are described in Reference 33, are set at nominal values in the transient analysis and an uncertainty analysis is used to address the effect on peak fuel enthalpy of variation in these parameters.

For Westinghouse reloads, the CRDA acceptance criteria are:

1. Reactivity excursions should not result in a radially averaged fuel rod enthalpy greater than 280 cal/gm at any axial location in any fuel rod.

2. The maximum reactor pressure during any portion of the assumed excursion should be less than the value that will cause stresses to exceed the Service Limit C as defined in the ASME code. It was concluded in Reference 33, that satisfaction of Criterion 1 will assure also that the Service Class C pressure will be satisfied.

3. The number of fuel rods predicted to reach assumed thresholds and associated parameters, such as the mass of fuel reaching melting conditions, will be input to a radiological evaluation. The assumed failure thresholds are a radially averaged fuel rod enthalpy greater than 170 cal/gm at any axial location for zero or low power initial conditions, and fuel cladding dryout for rated power initial conditions.

The Westinghouse CRDA analysis results are reported in the applicable cycle-specific reload documents.

For AREVA reload cores, the CRDA is analyzed using the CASMO-4/MICROBURN-B2 methodology (Reference 48) in conjunction with the generic rod drop parametric analysis results (Section 7.1 of Reference 45). The purpose of the CRDA evaluation is to ensure that the following NRC requirements are met:

1. The reactivity excursion shall not exceed the licensing limit of a radially averaged rod enthalpy of 280 cal/gm at any axial location within the assembly.

2. The number of fuel rods predicted to reach the assumed fuel failure threshold and associated parameters such as the amount of fuel reaching melting conditions does not exceed the inputs used in the radiological evaluation.

3. The maximum pressure during any point of the accident shall be less than the value that will cause stresses to exceed ASME "Service Limit C". It has been determined that the CRDA event cannot challenge this pressure limit so cycle-specific verification is not required.

The AREVA analysis is typically performed conservatively assuming the reactor is at hot zero power conditions (isothermal temperature, non-voided, and xenon free). Additional conservatism is incorporated by assuming that adiabatic conditions remain during the power excursion (i.e. no direct moderator heating is credited during the analysis), and that the reactor remains at hot zero power conditions for the analyzed control rod withdrawal sequence. The cycle specific portion of the CRDA analysis includes the determination of limiting rod withdrawals based upon calculated high rod worth and analyzing these withdrawals with MICROBURN-B2. The results of these analyzed withdrawals are used with the generic parameterized analysis to determine the maximum deposited enthalpy. These enthalpies are used to verify that items 1 and 2 above are met. For the purposes of item 2 above, any rod that exceeds a threshold of 170 cal/gm is assumed to experience failure and the total number of failures must remain within the releases assumed in the dose assessment (as described in FSAR Section 15.4.10.5.4.1). AREVA supplied source terms for the ATRIUM 10XM fuel design has been evaluated by Exelon to verify that this fuel design remains within the dose analysis basis for Dresden (Reference 50).

The AREVA CRDA analysis results are reported in the cycle-specific reload safety analysis report.

15.4.10.4 Barrier Performance

Barrier performance and radiological consequences for CRDA are not analyzed for reload cores. The following discussion pertains to the initial licensing analyses which show typical results.

Fuel rod damage estimates are based upon the UO_2 vapor pressure data of Ackerman[4] and interpretation of all the available SPERT, TREAT, KIWI, and PULSTAR test results which show that the immediate fuel rod rupture threshold is about 425 cal/g. Two especially applicable sets of data come from the PULSTAR[5] and ANL-TREAT[6,7] tests.

The PULSTAR tests, which used UO_2 pellets of 6% enrichment with Zirconium-2 cladding, achieved maximum fuel enthalpies of about 200 cal/g with a minimum period of 2.83 milliseconds. The coolant flow was by natural convection. Film boiling occurred and there were local clad bulges; however, fuel pin integrity was maintained and there were no abnormal pressure rises.

The two ANL-TREAT tests used Zircaloy-clad UO_2 pins with energy inputs of 280 and 450cal/g. The final mean particle diameter was 60 mils and 30 mils, and the pressure rise rate was 30 psi/s and 600 psi/s for the 280 cal/g and 450 cal/g tests, respectively.

The ultimate degree of fuel fragmentation and dispersal of the two cases was not significantly different; however, the pressure rise rate in the higher energy test was increased by a factor of 20. This pressure rise very strongly implies that the dispersion rate in the higher energy test was significantly higher than that of the lower energy. This leads to the logical conclusion that, although a high degree of fragmentation occurs for fuel in the 200 to 300 cal/g range, the breakup and dispersal into the water is gradual and pressure rise rates are very modest. On the other hand, for fuel above the 400 cal/g range, the breakup and dispersal is prompt and much larger pressure rise rates are probable.

Based on the analysis of the above referenced data, it is estimated that 170 cal/gm is the threshold for eventual fuel cladding damage. Fuel melting is estimated to occur in the 220- to 280-cal/g range, and a minimum of 425 cal/g would be required to cause immediate rupture of the fuel rods due a UO_2 vapor pressures.

15.4.10.5 Radiological Consequences for the CRDA

Regulation 10 CFR 50.67, "Accident Source Term," provides a mechanism for power reactor licensees to voluntarily replace the traditional TID 14844 (Ref. 28) accident source term used in design-basis accident analyses with an “Alternative Source Term” (AST). The methodology of approach to this replacement is given in USNRC Regulatory Guide 1.183 (Ref. 29) and its associated Standard Review Plan 15.0.1 (Ref. 30).

Accordingly, Dresden Nuclear Power Station (DNPS), Units 2 and 3, has applied the AST methodology for several areas of operational relief in the event of a Design Basis Accident (DBA), without crediting the use of previously assumed safety systems. Amongst these systems are the Control Room Emergency Filter System (CREFS) and the Standby Gas Treatment System (SGTS).

In support of a full-scope implementation of AST as described in and in accordance with the guidance of Ref. 29, AST radiological consequence analyses are performed for the four DBAs that result in offsite exposure (i.e., Loss of Coolant Accident (LOCA), Main Steam Line Break (MSLB), Fuel Handling Accident (FHA), and Control Rod Drop Accident (CRDA)).

Implementation consisted of the following steps:

- Identification of the AST based on plant-specific analysis of core fission product inventory,
- Calculation of the release fractions for the four DBAs that could potentially result in control room and offsite doses (i.e., LOCA, MSLB, FHA, and CRDA),
- Analysis of the atmospheric dispersion for the radiological propagation pathways,
- Calculation of fission product deposition rates and removal mechanisms,
- Calculation of offsite and control room personnel Total Effective Dose Equivalent (TEDE) doses.

15.4.10.5.1 Regulatory Guide 1.183 Compliance

The analyses are prepared in accordance with the guidance provided by Regulatory Guide 1.183 (Ref. 29).

15.4.10.5.1.1 Dose Acceptance Criteria

The AST acceptance criteria for Control Room dose for postulated major credible accident scenarios such as those resulting in substantial meltdown of the core with release of appreciable quantities of fission products is provided by 10 CFR 50.67, which requires

"Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident."

This limit is applied by Regulatory Guide 1.183 to all of the accidents considered with AST.

The AST acceptance criteria for an individual located at any point on the boundary of the exclusion area (the Exclusion Area Boundary or EAB) are provided by 10 CFR 50.67 as 25 rem TEDE for any 2-hour period following the onset of the postulated fission product release.

The AST acceptance criteria for an individual located at any point on the outer boundary of the low population zone (LPZ) are provided by 10 CFR 50.67 as 25 rem TEDE during the entire period of passage of the radioactive cloud resulting from the postulated fission product release.

These limits are applied by Regulatory Guide 1.183 to events with a higher probability of occurrence for the CRDA, provide the following acceptance criteria:

- For the BWR CRDA, doses at the EAB and LPZ should not exceed 6.3 rem TEDE for the accident duration (2 hour dose for EAB and 24 hour dose for LPZ).

15.4.10.5.2 Computer Codes

New AST calculations for the CRDA were prepared to simulate the radionuclide release, transport, removal, and dose estimates associated with the postulated accident scenarios.

The RADTRAD computer code (Ref. 35) endorsed by the NRC for AST analyses was used in the calculations for the CRDA. The RADTRAD program is a radiological consequence analysis code used to estimate post-accident doses at plant offsite locations and in the control room. The CRDA assessment takes no credit for control room isolation, emergency ventilation or filtration of intake air for the duration of the accident event.

Offsite χ/Qs were calculated with the PAVAN computer code (Ref. 36), using the guidance of Regulatory Guide 1.145 (Ref. 37); control room χ/Qs were calculated with the ARCON96 computer code (Ref. 38). The PAVAN and ARCON96 codes generally calculate relative concentrations in plumes from nuclear power plants at offsite locations and control room air intakes, respectively.

All of these computer codes have been used by the NRC staff in their safety reviews.

15.4.10.5.3 Core Inventory

As with the AST LOCA analyses, the inventory of reactor core fission products for RADTRAD analysis is based on maximum full power operation at a power level of 3016 MWth, the Extended Power Uprate (EPU) thermal power of 2957 MWth plus a 2% instrument error per Reg Guide 1.49 (Ref. 39). The fission products used for the accidents are the 60 isotopes of the standard RADTRAD input library, determined by the code developer as significant in dose consequences.

The Westinghouse Optima2 core inventory was analyzed at a core average exposure of 39 GWD/MTU for a 24 month cycle. The AREVA ATRIUM 10XM core inventory was evaluated at a core average exposure of 39 GWD/MTU and core average enrichments between 3.9 and 4.5 weight percent U-235 (Reference 50).

15.4.10.5.3.1 Reactor Coolant Inventory

The reactor coolant fission product inventory for CRDA analysis is based on the Technical Specification concentration limits.

15.4.10.5.3.2 Release Fraction

Current design basis accident evaluations as modified by Regulatory Guide 1.183 (Ref. 29) were used to determine the specific releases of radioactive isotopes at the given stages of fuel pin failure and provide these releases as a percentage of the total release for each accident, as summarized in sections 15.4.10.5.4.1, below.

15.4.10.5.4 Dose Calculations

As per Regulatory Guide 1.183 (Ref. 29), Total Effective Dose Equivalent (TEDE) doses are determined as the sum of the CEDE and the Effective Dose Equivalent (EDE) using dose conversion factors for inhalation CEDE from Federal Guidance Report No. 11 (Ref. 41) and for external exposure EDE from Federal Guidance Report No. 12 (Ref. 42).

15.4.10.5.4.1 Control Rod Drop Accident (CRDA)

Table 15.4-3a lists key assumptions and inputs used in the CRDA analysis. The design basis CRDA involves the rapid removal of a highest worth control rod resulting in a reactivity excursion that encompasses the consequences of any other postulated CRDA. The core performance analysis shows that the energy deposition that results from this event is inadequate to damage fuel pellets or cladding. However, for the dose consequence analysis, the bounding analysis assumed 850 8x8 fuel rods are damaged, with melting occurring in 0.77 percent of the damaged rods, and this analysis is applicable for current 10x10 fuel design. A core average radial peaking factor of 1.70 was assumed in the analysis, Except for this conservatively increased (from 1.5) radial peaking factor, these parameters are unchanged from the existing design bases, as fuel damage assumptions are unchanged by application of AST methodology. These assumptions are conservatively applied to the evaluation of Westinghouse Optima2 fuel.

The post-CRDA consequences of the AREVA ATRIUM 10XM reload were analyzed assuming 1,000 fuel rods fail; this assumption bounds the fuel failure calculated by AREVA of 273 fuel rods (Reference 50).

Releases to the environment are possible via three pathways. 99.85 percent of the activity released from the damaged fuel is assumed to reach the turbine and condenser, and 0.15 percent is assumed to be released directly through the turbine gland seal system (i.e., pathway 1). Releases from the main turbine and condenser are to the turbine building at a rate of one percent by volume for a period of 24-hours (i.e., pathway 2). No credit is taken for turbine building holdup or dilution and the release from the turbine building is conservatively assumed to be at ground level.

The final assessed scenario is for a CRDA at higher power levels when the condenser vacuum is maintained by the steam jet air ejectors (SJAE). This component of the release is via the large charcoal delay beds of the augmented off-gas (AOG) System driven by the SJAE. This pathway would eliminate all iodine releases and greatly delay noble gases (i.e., pathway 3).

The doses evaluated are for the combination of releases from pathways 1 and 2 (Scenario 1), and the combination of releases from pathways 1 and 3 (Scenario 2).

The analysis assumptions for the transport, reduction, and release of the radioactive material from the fuel and the reactor coolant are consistent with the guidance provided in Appendix C of Regulatory Guide 1.183.

15.4.10.5.4.2 Atmospheric Dispersion Factors (χ/Qs)

Table 15.4-3b lists χ/Q values used for the control room dose assessments, as derived in Section 2.3.5-1 and applied for release points applicable to the CRDA.

Table 15.4-3c lists χ/Q values for the EAB and LPZ boundaries, as also derived in Section 2.3.5-1 and applied for release points applicable to the CRDA.

15.4.10.5.5 Summary and Conclusions

The radiological consequences of the postulated CRDA are given in 15.4-3d. As indicated, the control room, EAB, and LPZ calculated doses are within regulatory limits after AST implementation.

15.4.11 Thermal Hydraulic Instability Transient

This section covers events that result in a thermal hydraulic instability. Additional information regarding the transient and the system designed to respond to it, namely the Oscillation Power Range Monitor (OPRM) system, is contained in chapters 4 and 7.

15.4.11.1 Identification of Causes and Frequency Classifications

Events such as Reactor Recirculation (RR) pump trips and runbacks, turbine/generator runbacks, loss of feedwater heating, and RR flow controller failures can result in unplanned entry into the high power and low flow region of the power to flow map. Under these conditions, axially varying moderator density in the fuel channels can cause flow oscillations that increase in amplitude. Without manual or automatic suppression, such oscillations can cause the MCPR Safety Limit to be exceeded (Reference 20).

This event is controlled by a system designed for detection and suppression of oscillations in accordance with GDC 10 and 12. The system is the Oscillation Power Range Monitor (OPRM) system. It provides automatic protection for this event, when it is installed and fully functional. For operation prior to the installation of OPRM, or when OPRM is not fully functional, the operator controls the oscillations by scramming the reactor upon entry into the region of power to recirculation flow map where such oscillations are possible.

Anticipated stability-related neutron flux oscillations are those instabilities that result from normal operating conditions, including conditions resulting from anticipated operational occurrences. This category of events is equivalent to the standard terminology for the analysis of events of moderate frequency (Reference 21).

15.4.11.2 Sequence of Events and System Operation

For this event, the plant must be operating in mode 1.

A. As a result of some manual actions or equipment problems (e.g., RR pump runback, loss of feedwater heating), the core power and flow combination may be such that oscillations of neutron flux may be possible.

B. Due to forced flow being inadequate to control density wave transit time up the fuel channels, flux oscillations start and begin to increase in amplitude.

C.1 Without OPRM being installed, armed, and operational, the operator manually scrams the reactor upon recognition of the instability.

C.2 With the OPRM installed, armed, and operational, the operator may be able to take action based on pre-trip alarms to insert control rods or increase flow. If not able to because of the rate of increasing oscillations, the OPRM automatically scrams the reactor before the Safety Limit MCPR is violated.

15.4.11.3 Core and System Performance

The OPRM system contains 4 LPRMs per OPRM cell (using the Bockstanz-Lehmann LPRM assignment methodology described in Appendix D of Reference 22) and requires 1 LPRM input for the cell to be operable. The amplitude setpoint for oscillation magnitude and the number of confirmation counts are specified for the analysis. Since core thermal hydraulic instability is characterized by a consistent period for the oscillations, the OPRM logic includes a check for a set number of consecutive counts as well as a magnitude.

The specified system setpoints are used to determine the hot bundle oscillation magnitude. This information is used, along with empirical data applicable to the fuel in the core, to determine the fractional change of CPR (delta CPR/IMCPR, where IMCPR is initial MCPR).

The Initial (pre-oscillation) MCPR (IMCPR) is determined as the lower of the following:

1. The MCPR following a dual RR pump trip from rated power on the highest allowed flow control line, after the coastdown to natural circulation and after feedwater temperature reaches equilibrium. The assumption is that the core was operating at the Operating Limit MCPR prior to the dual pump trip.

2. The MCPR Operating Limit with the reactor at steady state conditions at 45% core flow on the highest allowed flow control line.

The Final MCPR (FMCPR) is determined using the IMCPR and CPR/IMCPR data (Reference 22).

The FMCPR is then verified to be greater than the Safety Limit MCPR. Alternatively, a minimum IMCPR can be determined for a given Safety Limit and checked against the cycle specific Operating Limit (Reference 23).

If the minimum IMCPR is greater than the Operating Limit determined from other cycle analyses, or the FMCPR is less than the Safety Limit MCPR, the system setpoint may be changed and the reload confirmation performed again. Alternatively, the Operating Limit MCPR may be changed, or the LPRM assignment scheme may be modified.

The above is confirmed for each cycle as part of the reload analysis when OPRM is fully installed and armed.

15.4.11.4 Barrier Performance

Since the successful completion of this analysis demonstrates that the MCPR Safety Limit is not exceeded, fuel-cladding integrity is not challenged.

15.4.11.5 Radiological Consequences

Since fuel-cladding integrity s not challenged, there are no radiological consequences warranting evaluation of this event.

15.4.12 References

1. "General Electric Standard Application for Reactor Fuel," General Electric Company, NEDE-24011-P-A-US.

2. Deleted

3. "Rod Drop Accident Analysis for Large Boiling Water Reactors," General Electric Company, NEDO-10527, March 1972.

4. Ackerman, R.J., Gilles, W.P., and Thorn, R.J., "High Temperature Vapor Pressure of UO2," Journal of Chemical Physics, December 1956, Volume 25, No. 6.

5. MacPhee, J., and Lumb, R.F., "Summary Report, PULSTAR Pulse Tests-II," WNY-020, February 1965.

6. Baker, L., Jr., and Tevebaugh, A.D., "Chemical Engineering Division Report, January through June 1964, Section V - Reactor Safety," ANL-6900.

7. Baker, L., Jr., and Tevebaugh, A.D., "Chemical Engineering Division Report, July through December 1964, Section V - Reactor Safety," ANL-6925.

8. Williamson, H.E., and Rowland, T.C., "Performance of Defective Fuel in the Dresden Nuclear Power Station," APED-3894, 1962.

9. L.C. Watson, et al., "Iodine Containment by Dousing in NPD-II," AECL-1130, October 1960.

10. H.R. Diffey, et al., "Iodine Cleanup in a Steam Suppression System," International Symposium on Fission Product Release and Transport Under Accident Condition, April 1956.

11. Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/ MICROBURN BWR Nuclear Design Methods," Revision 0, Supplement 1 and 2, December 1991, March 1992, and May 1992, respectively; NRC SER letter dated March 22, 1993.

12. NRC SER, Topical Report for Neutronics Methods for BWR Reload Design for Commonwealth Plants, Siegel (NRC) to Kovach (ComEd). February 27, 1992.

13. NRC SER, Commonwealth Edison Company Topical Report NFSR-0091 Benchmark CASMO/MICROBURN BWR Nuclear Design Methods, Patel (NRC) to Kovach (ComEd), March 22, 19993.

14. ComEd letter, Dresden Nuclear Power Station Units 2 and 3, Quad Cities Nuclear Power Station units 1 and 2, Revised Control Rod Sequencing Methods, NRC Docket 50-237/249, 50-254/265 and 50-373/374, P. Piet to T. Murley, January 27, 1993.

15. General Electric, Banked Position Withdrawal Sequence, NEDO-21231, January 1977.

16. Letter, Revised Control Rod Sequencing Methods for the Dresden and Quad Cities Nuclear Power Station, B. Siegel to T. Kovach, September 21, 1990.

17. “Control Room Drop Accident/MSLRM Removal”, Bechtel Calculation DR-357-M-004.

18. ANSI/ANS-5.1-1994, "American National Standard for Decay Heat Power in Light Water Reactors," August 23, 1994 (Appendix B).

19. NEDO-31400A, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure and Scram Function of the Main Steam Line Radiation Monitors," by GE Nuclear Energy Class I, October 1992.

20. Generic Letter 94-02

21. GE Document NEDO-31960-A Supplement

22. GE Document NEDO-32465-A

23. BNDG 96-011

24. "Dresden and Quad Cities Extended Power Uprate, Task T0900: Transient Analysis," GE-NE-A22-00103-10-01 Revision 0, October 2000.

25. Letter to U.S. Nuclear Regulatory Commission, "Notification of Intent to Perform Analysis Using Vendor Safety Analysis Codes", Letter RS-03-174, September 19, 2003.

26. Licensing Topical Report NEDO-33091-A, Revision 2, "Improved BPWS Control Rod Insertion Process", July 2004.

27. Calculation Note, BNDL: 98-003, Revision 3, "Acceptance Review of GE letter NSA 98-070 or Rod Drive Speed," July 8, 1998.

28. U.S. Atomic Energy Commission, Technical Information Document (TID) 14844, "Calculation of Distance Factors and Power and Test Reactor Sites," March 23, 1962.

29. U.S. Nuclear Regulatory Commission Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

30. U. S. Nuclear Regulatory Commission Standard Review Plan 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000.

31. "Westinghouse BWR Reload Licensing Methodology Basis for Exelon Generation Company Dresden Nuclear Power Station Units 2 and 3," WCAP-16588-P Revision 3, August 2006.

32. "Reference Safety Report for Boiling Water Reactor Reload Fuel," Westinghouse Topical Report CENPD-300-P-A, July 1996.

33. "Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification," Westinghouse Topical Report CENPD-284-P-A, July 1996.

34. Not used.

35. RADTRAD Code, "A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," Version 3.03.

36 PAVAN Code, "An Atmospheric Dispersion Program for Evaluating Design Bases Accidental Releases of Radioactive Materials from Nuclear Power Stations".

37. U.S. Nuclear Regulatory Commission Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants, "Revision 1, November 1982.

38. ARCON96 Code, "Atmospheric Relative Concentrations in Building Wakes".

39. U.S. Nuclear Regulatory Commission Regulatory Guide 1.49, "Power Levels of Nuclear Power Plants," Revision 1, December 1973.

40. No longer used.

41. Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion", 1988.

42. Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil", 1993.

43. NF-BEX-10-54, Evaluation of the Planned Implementation of Adjustable Speed Drives in Dresden Units 2 and 3 (TSD DQC-09-009, Rev. 1), Revision 0 dated May 13, 2010.

44. XN-NF-80-19(P)(A) Volume 4: Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads", Exxon Nuclear Company, June 1986. (Unit 3 only)

45. XN-NF-80-19(P)(A), Volume 1, Supplements 1 and 2, Revision 0 "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Method for Design and Analysis", Exxon Nuclear Company, March 1983. (Unit 3 only)

46. EMF-2209(P)(A), Revision 3, "SPCB Critical Power Correlation", AREVA NP, September 2009. (Unit 3 only)

47. ANP-10298P-A, Revision 1, "ACE/ATRIUM 10XM Critical Power Correlation", AREVA, March 2014. (Unit 3 only)

48. EMF-2158(P)(A), Revision 0, " Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CAMSO-4/MICROBURN-B2", Siemens Power Corporation, October 1999. (Unit 3 only)

49. FS1-0022549, Revision 1, "Disposition of Events Summary for the Introductions of ATRIUM 10XM fuel at Dresden," AREVA, February 2016. (Unit 3 only)

50. DRE02-0037, Revision 4, "Re-analysis of Control Rod Drop Accident (CRDA) Using Alternative Source Terms", October 2016.

Table 15.4-1

Deleted

Table 15.4-2

Table Deleted

Table 15.4-3

CRDA-Radiological Consequences
Key Inputs, Assumptions and Radiological Doses

Table 15.4.3a: Key CRDA Analysis Inputs and Assumptions

Input/Assumption	Value
Core Damage	850 fuel rods failed*
Percent of Damaged Fuel with Melt	0.77%
Radial Peaking Factor	1.7
Condenser Free Volume	55,000 cubic feet
Condenser Leak Rate	1% per day
Release Period	24 hours
CREV System Initiation	Not utilized
Charcoal Delay Bed Noble Gas Delay for SJAE pathway	14.6 days for Xe 19.4 hours for Kr

*A bounding value, per GE NEDO-31400A and OPTIMA2-TR050DR-RELOAD. For AREVA ATRIUM 10 XM, 1000 fuel rods failed is bounded by the dose consequence of 850 fuel rods failed for 8x8 fuel per Reference 50.

Table 15.4-3b: Control Room χ/Q Values for the CRDA Releases[1]

Time Period	χ/Q (sec/m3)
0 – 2 hrs	1.30E-03
2 – 8 hrs	1.06E-03
8 – 24 hrs	4.49E-04

Notes:
1 Zero velocity vent release χ/Q values for Release from Main Steam Isolation Valve Room based on ARCON96.

Table 15.4-3c: Offsite χ/Q (sec/m^3) Values for the CRDA Releases[1]

Time Period	EAB χ/Q (sec/m^3)	LPZ χ/Q (sec/m^3)
0 – 2 hrs	2.51E-4	2.63E-5
2 – 8 hrs	-	1.09E-5
8 – 24 hrs	-	7.02E-6

Notes:
1 Zero velocity vent release χ/Q values for Release from Main Steam Isolation Valve Room based on Regulatory Guide 1.145 methodology.

Table 15.4.3d

CRDA Radiological Consequence Analysis[1]
(scenario)

Scenario 1 (Main and Gland Seal Condenser Leakage)			
Location	**Duration**	**TEDE (rem)**	**Regulatory Limit TEDE (rem)**
Control Room	30 days	1.31	5
EAB	Maximum, 2 hours	0.300	6.3
LPZ	30 days	0.0353	6.3
Scenario 2 (Gland Seal Condenser Leakage and SJAE Release)			
Location	**Duration**	**TEDE (rem)**	**Regulatory Limit TEDE (rem)**
Control Room	30 days	1.07	5
EAB	Maximum, 2 hours	0.581	6.3
LPZ	30 days	0.0608	6.3

Notes:
1 The radiological consequences are based on the Westinghouse Optima2 core inventory described in Section 15.4.10.5.3. These consequences are bounding for the AREVA ATRIUM fuel design.

15.5 INCREASE IN REACTOR COOLANT INVENTORY

This section describes the evaluation of the plant response to an inadvertent initiation of the high pressure coolant injection system. The description of the plant response to an increase in feedwater flow is provided in Section 15.1.2.

AREVA ATRIUM 10XM methods and fuel are only applicable to Unit 3.

15.5.1 Inadvertent Initiation of High Pressure Coolant Injection During Power Operation

For Westinghouse reloads, the inadvertent initiation of the high pressure coolant injection system (IHPCI) is analyzed every cycle to confirm that the IHPCI transient will not be a limiting pressurization event, such as the Feedwater Controller Failure (maximum flow demand) transient. If the injection of cold water does not cause the water level to reach the main turbine high water level (L8) set-point, then the IHPCI event is considered a slow transient. In this case, the IHPCI event would usually be bounded by the Loss of Feedwater Heating event (Section 15.1.1), because the increase in inlet subcooling due to the IHPCI is typically less than that produced by the loss of feedwater heating event.

If the main turbine high water level is reached, then the IHPCI would become a pressurization event superimposed on a subcooling event. In this case, it is likely that IHPCI would not be bounded by the loss of feedwater heating event, and may be potentially more limiting than the Feedwater Controller Failure (FWCF) transient. The results are documented in the Westinghouse cycle-specific reload reports."

For AREVA reloads, the inadvertent initiation of the high pressure coolant injection system (IHPCI) is a potentially limiting event and is analyzed on a cycle-specific basis. The water injection causes an increase in the water level. In an effort to maintain the water level, the feedwater/level control system responds to the water level increase by decreasing the feedwater flow. While the decrease in feedwater flow offsets the HPCI injection flow, the level control system is not necessarily able to keep the water level from increasing. Prior to reaching the high level set point which causes a turbine trip, a trip of the HPCI turbine pump occurs which shuts off the HPCI injection flow and the reactor returns to a steady state condition. The results show that all power levels, the combination of the level control system and the HPCI turbine pump trip decreases the makeup water flow such that a high level trip does not occur. While a pressurization event does not occur, the core does experience a fairly long term power excursion. However, if a high level turbine trip occurs, this could become a potentially limiting event. The results are documented in the AREVA cycle-specific reload reports.

15.5.1.1 Identification of Causes and Frequency Classification

Inadvertent startup of the high pressure coolant injection (HPCI) system, i.e., operator error, is postulated for this analysis. Inadvertent HPCI initiation is postulated because it is the only emergency core cooling system capable of increasing reactor coolant inventory during power operation. This transient disturbance is classified as an incident of moderate frequency.

15.5.1.2 Sequence of Events and Systems Operation

This transient is similar to the loss of feedwater heating event (see Section 15.1.1). The HPCI pumps are inadvertently started and the cold water injection results in an increase in inlet subcooling, a decrease in moderator void coefficients, and a consequent increase in power.

The plant operating conditions and assumptions are identical to those for the loss of feedwater heating. The HPCI system introduces cold water through the feedwater sparger. The normal feedwater flow is correspondingly reduced by the water level controls. The increase in inlet subcooling due to the inadvertent HPCI start is typically less than that produced by the loss of feedwater heating event.

However, if the inadvertent HPCI event causes the water level to rise enough to result in a turbine trip on high water level (L8), the event becomes a fast transient resembling a FWCF event. In this case, the sequence of events and results are similar to the FWCF transient described in Section 15.1.2. The IHPCI event is analyzed at rated and off-rated power and flow conditions to cover the expected range of operation. The results are documented in the cycle-specific reload reports.

15.5.1.3 Core and System Performance

For Westinhouse reloads, the water level response during an IHPCI event is evaluated for each reload. If the main turbine high water setpoint is not reached, then the analysis performed for a loss of 145 °F feedwater heating bounds the results for the IHPCI event.

If it is determined that the main turbine high water setpoint can be reached prior to HPCI pump shut-off, then it is conservatively assumed that feedwater and HPCI continue to operate until the time of high water level turbine trip and feedwater pump trip. As a result of this conservative modeling, the IHPCI event can become a limiting pressurization transient similar to the FWCF event. It is also possible that the L8 high water level setpoint is reached after HPCI pump shut-off. This scenario results in a less limiting pressurization transient (compared to the FWCF transient or turbine trip without bypass). The event then resembles a turbine trip with bypass available described in Section 15.2.3. Results are provided in the cycle-specific reload reports.

If the reactor level control system is assumed to fail and the HPCI flow is added to the full feedwater flow, the core inlet subcooling is still less than during loss of feedwater heating transient. Thus, inadvertent operation of HPCI transient is bounded by the loss of feedwater heating transient, provided there is no turbine trip and resulting pressurization. The continuous mismatch in the reactor coolant inventory would cause the vessel level to increase until the HPCI pump turbine is tripped by redundant high level signals.

For AREVA reloads, the combination of the level control system and the HPCI turbine pump trip decreases the makeup water flow such that a high level trip does not occur and the reactor returns to a steady state condition. While a pressurization event does not occur, the core does experience a fairly long term power excursion. Results are provided in the cycle-specific reload reports.

15.5.1.4 Barrier Performance

For Westinghouse reloads, the water level response during an IHPCI event is evaluated for each reload and the results are documented in the cycle-specific reload documents.

The main turbine trip setpoint is not expected to be reached due to the difference between the HPCI and main steam high level trip setpoints. If the main turbine trip setpoint is not reached then the IHPCI event is bounded by the loss of feedwater heating event. If it is conservatively determined that feedwater and HPCI continue to operate until the time of high water level main turbine trip and feedwater pump trip, then the IHPCI event can become a limiting pressurization transient similar to the FWCF event. A non-limiting pressurization transient results if the high water level setpoint is reached after HPCI pump termination.

For AREVA reloads, the combination of the level control system and the HPCI turbine pump trip decreases the makeup water flow such that a high level trip does not occur and the reactor returns to a steady state condition. While a pressurization event does not occur, the core does experience a fairly long term power excursion (Reference 2).

For each cycle the MCPR operating limit is calculated to preclude violation of the fuel cladding integrity safety limit. See Section 15.5 for details.

15.5.1.5 Radiological Consequences

Since the fuel cladding integrity safety limit would not be violated, radiological consequence analysis has not been performed.

15.5.2 Reference

1. “Reference Safety Report for Boiling Water Reactor Reload Fuel,” Westinghouse Topical Report CENPD-300-P-A, July 1996.

2. ANP-3516P, Revision 0, “Dresden Unit 3 Cycle 25 Reload Safety Analysis,” AREVA, September 2016. (Unit 3 only)

15.6 DECREASE IN REACTOR COOLANT INVENTORY

This section covers events which involve an unplanned decrease in reactor coolant inventory. These events include inadvertent opening of a safety valve, relief valve, or safety relief valve (SRV); failure of an instrument line carrying reactor coolant outside primary containment; main steam line break outside primary containment; and the failure of reactor coolant pressure boundary piping inside primary containment.

The events and radiological consequences described in this section are not reanalyzed for the current fuel cycle since they continue to be bounded by analyses for previous fuel cycles. The conclusions of these events and radiological analyses are still valid; however, specific details contained in the descriptions and associated results and figures should be used only to understand the analysis and its conclusions. These specific details should not be used as sources of current fuel cycle design information.

15.6.1 Inadvertent Opening of a Safety/Relief Valve

The following evaluation of an inadvertent opening of a safety/relief valve shows that this event is not of safety significance. The following information is based on the NRC-approved evaluation of Dresden Unit 2 performed during the Systematic Evaluation Program (SEP).

The inadvertent opening of a safety valve, relief valve, or SRV would result in a decrease in reactor coolant inventory and a decrease in reactor coolant system pressure.

If an SRV or relief valve fails open, it discharges to the suppression pool. The safety valves discharge directly to drywell atmosphere. Although a drywell high-pressure reactor trip might occur if a safety valve fails open, the following analysis conservatively assumes a safety valve discharge would result in a sequence of events similar to a relief valve or SRV discharge.

15.6.1.1 Identification of Causes

The cause of an inadvertent opening of a safety valve, relief valve, or SRV is a malfunction of the valve.

15.6.1.2 Sequence of Events and System Operations

The following sequence of events is assumed for this analysis.

The normal functioning of plant instrumentation and controls is assumed for this incident; specifically, normal operation of the pressure regulator and vessel level control systems is assumed normal. On an inadvertent opening of the relief valve or SRV, the pressure regulator senses the pressure decrease and causes the turbine

control valves to close partially. No reactor trip occurs, and the reactor stabilizes at a power level near the initial power. The feedwater system would make up the continuing loss in reactor coolant inventory.

If the pressure regulator fails to respond, the decrease in the main steam line pressure causes the main steam isolation valves (MSIVs) to close. The pressure regulator failure event is discussed in Section 15.1.3.

If the feedwater system becomes unavailable due to a single failure or loss of offsite power, the high pressure coolant injection (HPCI) system would provide makeup water. High pressure coolant injection is automatically actuated on low-low water level.

If a relief valve or SRV opens and fails to reclose, the torus experiences an increase in temperature. Closure of the MSIVs could not halt the blowdown since the relief valves and SRV are upstream of the MSIVs.

15.6.1.3 Core and System Performance

Inadvertent opening of a safety valve, relief valve, or SRV is not limiting from a core performance standpoint.

This event would cause a negligible pressure reduction which could lead to partial closure of the turbine control valve by the pressure regulator. The net change in power level and coolant conditions within the fuel assemblies would be negligible, and operating thermal margins would be relatively unaffected. Therefore, minimum critical power ratio (MCPR) would not change significantly.

Refer to Section 6.2.1.3 for details regarding suppression pool temperature and pressure response to an opened relief valve or SRV.

15.6.1.4 Barrier Performance

The NRC has concluded,[1] based on the evaluation of plants similar to Dresden,[2,3] there would not be any fuel failure resulting from a stuck-open safety valve, relief valve, or SRV event since MCPR would not change significantly. Therefore, the transient resulting from an inadvertently opened safety valve, relief valve, or SRV would not have a significant effect on the reactor coolant pressure boundary and would not violate the fuel cladding integrity safety limit.

15.6.1.5 Radiological Consequences

The consequences of inadvertent safety valve, relief valve, or SRV actuation would not result in fuel failure. Discharge of normal coolant activity to the suppression pool via SRV or relief valve operation or to the drywell via safety valve operation would result. This activity would be contained in the primary containment. Any discharges to the environment would be made under controlled release conditions. During purging of the containment, the release would be in accordance with the

established ODCM limits; therefore, this event, at the worst, would result in an insignificant increase in the yearly integrated exposure level.

15.6.2 Break in Reactor Coolant Pressure Boundary Instrument Line Outside Containment

See the introduction to Section 15.6 for information regarding use of details from this analysis description which may not be applicable to the current fuel cycle.

A rupture of a reactor coolant pressure boundary instrument line outside containment could allow primary coolant and radioactivity contained therein to escape to the environment. The following section describes an instrument line break analysis performed by CECo during the initial licensing phase, the potential radiological consequences, and a subsequent analysis performed by the NRC for the SEP.

15.6.2.1 Identification of Causes

A postulated 1-inch reactor coolant pressure boundary instrument line break has been analyzed for the Dresden Unit 3 plant. Dresden Unit 3 is virtually identical to the Dresden Unit 2 design.

15.6.2.2 Sequences of Events and System Operation

The break in the reactor coolant pressure boundary instrument line was assumed to occur outside the primary containment but upstream of the flow check valve in a 1-inch pipe. A manually operated stop valve located outside the containment wall upstream of the break was not assumed to be closed until after the reactor was shut down and depressurized. The reactor was assumed to be shut down manually by the operator upon detection of the break.

Radiation levels in the reactor building ventilation duct would not be high enough to start the standby gas treatment system (SBGTS), so all of the radioactive materials escaping to the atmosphere would do so via the reactor building ventilation stack. The analysis showed that 70,000 pounds of water and 30,000 pounds of steam would be released to the reactor building. Air in the building would be exhausted to make room for the expanding steam, then all the steam not condensed in the reactor building would be transported out via the stack.

The leak is non-isolable (between the primary containment and first isolation valve outside the containment) within the first 4 hours until the reactor is manually shut down and depressurized. Then the manual valve can be closed and the ruptured line can be repaired.

Routine surveillance on the part of the operator (as described in the following list A through H) has been a sufficient program for the periodic testing and examination of the valves in these small diameter instrument lines.

Such leaks would be detected by one or a combination of the following:

A. Comparison of readings among several instruments monitoring the same process variable, such as reactor level, jet pump flow, steam flow, and steam pressure;

B. Annunciation of the failure of the affected control function, either high or low, in the control room;

C. Annunciation of a half-channel scram if the rupture occurred on a reactor protection system instrument line;

D. A general increase in the area radiation monitor readings throughout the reactor building;

E. Noise from the leakage audible either inside the turbine building or outside the reactor building on a normal tour;

F. Unexplained increase in floor drain collector tank water level as well as alarms on the corner room floor sumps;

G. Detection of the leak as soon as an access door to the reactor building is opened; and

H. Increases in area temperature monitor readings in the reactor building.

15.6.2.3 Barrier Performance

No core uncovering would occur and no fuel cladding perforations would occur.

15.6.2.4 Radiological Consequences

Calculations of doses due to the released radioactive materials included the following assumptions. Coolant activity consistent with a plant off-gas release rate of 100,000 μCi/s was assumed to be released to the environment. Although the release would occur at the top of the reactor building, it was assumed that downwash would result in an effective release height of 0 meters. Since no core uncovering or fuel cladding perforations would occur, only coolant activity would be released. Iodine in the 30,000 pounds of water that flashed to steam was assumed to be transported with the steam.

The iodine activity associated with the released liquid was 0.04 μCi/cc of I-131 and 0.3 μCi/cc of I-133. No further release of iodine from the water was assumed. Very stable (1 m/s) meteorological conditions were assumed, since these conditions represent the worst case for an equivalent ground level release. Calculated lifetime dose for the duration of the release is 0.3 rem, which is well below the reference doses of 10 CFR 100 and is in fact less than the annual dose permitted in the old 10 CFR 20 prior to January 1, 1994.

Subsequent to the preceding analysis, the NRC evaluated the instrument line break outside containment in accordance with Standard Review Plan (SRP) Section 15.6.2 using the assumptions listed in Table 15.6-1. The corresponding radiological consequences estimated by the NRC are shown in Table 15.6-2. The results show that use of the current Dresden Technical Specification value of dose equivalent I-131 limit of 0.2 μCi/g results in doses which are less than the guideline values of 10 CFR 100.

The specific activity of the primary coolant is limited by Technical Specification. In addition, there is no core uncovery and no perforations of the fuel during an instrument line break. Therefore, since only the coolant activity is released, the radiological dose calculations are independent of fuel type or design.

The reactor coolant release mass and flashed fraction in Table 15.6-1 envelop the releases for extended power uprate (EPU). Since only the coolant activity is released, the calculated doses in Table 15.6-2 remain valid for EPU.

15.6.3 Steam Generator Tube Failure

This section is not applicable to Dresden Station.

15.6.4 Steam System Line Break Outside the Containment

See the introduction to Section 15.6 for information regarding the use of detail from the analysis description which may not be applicable to the current fuel cycle.

A steam line break outside containment could allow primary coolant and radioactivity contained therein to escape to the environment. The following section describes an analysis for a steam line break outside containment and potential radiological consequences. Historical analyses performed by CECo during the initial licensing phase and subsequent analysis performed by the NRC are included as historical information only.

15.6.4.1 Identification of Causes and Frequency Classification

The postulated accident is a sudden, complete severance of one main steam line outside containment with subsequent release of steam and water containing fission products to the pipe tunnel and the turbine building. This large flow of steam to the turbine building would relieve through the blowout panels and lead to the formation of a large steam cloud which is presumed to drift to the site boundary.

This event is classified as a limiting fault, i.e., an event that is not expected to occur but is postulated because the consequences may result in the release of significant amounts of radioactive material.

The steam system line break outside containment is not reanalyzed for reload cores.

15.6.4.2 Sequence of Events and System Operation

To evaluate the overall consequences of the postulated severance of one of the four main steam lines, the sequence of events following the break was investigated in detail.

The initial conditions prior to the main steam line break were assumed as follows:

A.	Reactor power	2527 MWt
B.	Reactor pressure	1020 psia
C.	Reactor water level	normal

The sequence of events is assumed as follows:

Event	Time After Break (seconds)
Main steam line break	0
Main turbine control valves closure initiation	0.2
MSIV closure initiation	0.5
Reactor trip/control rod insertion started	1.0
Feedwater flow shut off	5.0

15.6.4.3 Core and System Performance

15.6.4.3.1 Main Steam Line Isolation Valve Closure

The steam blowdown flowrate through both ends of the postulated break would cause an increase in steam flow in each of the four lines to the maximum value allowed by critical flow considerations.

Blowdown through the main steam lines is limited by the main steam line flow restrictors. Design flowrate through the restrictors is 175% of rated steam flow. For conservatism, this analysis assumes a blowdown rate of 200%. The restrictors are sized in conjunction with isolation valve closure time so that core submergence is assured during blowdown and after termination of the accident. The increased pressure differential across the flow limiters would indicate the severance immediately and would initiate MSIV closure (all 8 valves) within 0.5 seconds after the accident. Multiple flow limiter pressure differential sensors are provided in the primary containment isolation system to accomplish this function.

Rapid depressurization in the steam lines downstream of the flow limiters would also initiate closure of the main turbine control valves within 0.2 seconds after the accident.

15.6.4.3.2 Reactor Core Shutdown

A reactor scram would be initiated by MSIV position switches, at approximately 10% closure of either MSIV in three or more steam lines, as described in Section 7.2.

Control rod insertion would begin within about 1 second after the line break with an MSIV total closure time of 3.5 to 5.5 seconds (0.5 seconds for detection plus 3 to 5 seconds for closure). The MSIVs are designed to close against reactor operating pressure. In addition, voids generated in the core caused by excess flow leaving the vessel would contribute sufficient negative reactivity to reduce reactor power

immediately. Finally, as an additional backup, reactor low water level, which would occur later during the blowdown, would also initiate a scram and isolate the reactor.

15.6.4.3.3 Feedwater Flow

Assuming that the reactor water level control system is in three-element control, the increased steam flow will result in a demand for the feedwater flow control valve to open fully. Within 1 second after the accident, the indicated high-water level in the reactor vessel would initiate closure of the feedwater control valve. The feedwater flow would then decrease linearly to shut off approximately 5 seconds after the accident. Following closure of the MSIVs, the reactor vessel water level would drop due to collapsing steam voids, thereby actuating the feedwater system to return the vessel water level to normal.

15.6.4.3.4 Reactor Coolant Blowdown

The two distinct intervals of blowdown are vapor blowdown before the coolant mixture flows into the steam line and coolant mixture blowdown.

The steam flowrate through the upstream side of the break would increase from the initial value of 675 lb/s in the line to 1350 lb/s (200% of initial) with critical flow occurring at the flow limiter. The steam flowrate was calculated using an ideal nozzle model.[4] The flow model predicting the behavior of the flow limiter is substantiated by tests conducted on a scale model over a variety of pressure, temperature, and moisture conditions. The steam flowrate through the downstream side of the break would consist of essentially equal flow components from the other three unbroken lines. The pipe resistance and local restrictions in the three unbroken lines would result in critical flow occurring not at their flow limiters but at the break area.

The steam flowrate in each of the three unbroken lines would increase from the initial value of 675 lb/s to 925 lb/s (140% of initial) based on a resistance coefficient K (which is fL/D) equal to 1.0 for this end of the break. Total break flow is shown in Figure 15.6-1. Therefore, the total steam flowrate leaving the vessel would be approximately 4100 lb/s, which would be in excess of the generation rate of 2700 lb/s.

The initial depressurization in the vessel would be at a rate of 25 psi/s, as shown in Figure 15.6-2, which would cause flashing of the moderator throughout the reactor. Steam bubbles generated within the system would cause the reactor water level to rise at a rate determined by the difference between the rate at which bubbles are formed and the rate at which they break the water surface. Steam bubbles rise by buoyancy at an average velocity of 2 ft/s[5,6] relative to liquid eventually separating from the mixture surface.

An analytical model was used to predict the rate of the reactor water level rise. In a portion of the range of interest (i.e., steam blowdown) this model has shown to be in reasonable agreement with level rise data obtained in a large vessel undergoing depressurization. The model predicts that the water level would rise at

approximately 2 ft/s. With the MSIVs closing at 5.5 seconds after the start of the accident, the water level would not reach the steam separators. If the MSIVs were conservatively assumed to close more slowly, in 10 seconds rather than the normal 3 to 5 seconds, the water level would flood the steam dryers and reach the vessel steam nozzles, and the blowdown would change from single-phase steam to mixture blowdown. However, the calculation of record uses 5.5 seconds in accordance with NRC guidance (Reference 39).

Historical analyses conservatively assume an MSIV closure time of 10 seconds due to the lack of NRC guidance at the time the analyses were performed. The mixture blowdown, beginning approximately 6 seconds after the accident, would adjust to an average value of 14,200 lb/s[7] for the time interval from 6 to 10.5 seconds when the isolation valves would be closed. No credit was taken for separator action which would reduce coolant loss. The corresponding blowdown energy content would be 550 Btu/lbm (assuming separator efficiency is zero) to obtain maximum coolant loss. Vapor fraction in the blowdown would actually be higher as would be determined by its separation rate from the mixture. Vessel depressurization stops when two-phase blowdown begins through the steam line and the system slowly pressurizes at 6 psi/s. Based on the conservative separation model, the estimated loss of mass for 10-second valve closure would be 66,000 pounds (approximately 21,000 pounds of steam and 45,000 pounds of water, as shown in Figures 15.6-1 and 15.6-2), which is well below the 140,000 pounds of fluid that must be lost before the core would be uncovered.

15.6.4.3.5 Steam-Water Mixture Impact Forces

The maximum differential pressure which would be generated by continuous water flow past the MSIVs is 1000 psi. This is below the differential pressure across the valve during hydrostatic testing.

The impact pressure from the steam-water mixture in the steam line has been evaluated as a function of steam quality, assuming instantaneous stoppage of a saturated water slug. Line friction was ignored and a driving pressure of 1000 psi was assumed.

By the time two-phase flow occurs in the steam line, the flow limiters would be effective in restricting flow. The most realistic case is that a continuous slug would be present in the steam line. For this case, the maximum impulse overpressure would be 70 psi. The resultant total transient differential pressure across the valve would be 1070 psi, which is well below the piping design pressure of 1250 psig.

Should a short discrete slug be present instead, the total transient differential pressure could rise to 1300 psi. This value is within the test pressure and not sufficiently in excess of the design pressure to jeopardize the integrity of the piping. Reaction forces resulting from these transient differential pressures are within those generated during hydrostatic testing.

These results are highly dependent upon the velocity of sound in the mixture present in the pipe. This calculation is based on F.E. Tippet's model[8] for calculating this characteristic of the system.

15.6.4.3.6 Effect of Main Steam Isolation Valve Closure Time

A parametric analysis was performed to determine the effect of assuming various closure times for the MSIVs. The results are shown in Table 15.6-3.

It would be necessary to lose approximately 140,000 pounds of water and steam before the top of the core would be exposed. The values given in Table 15.6-3 are for a constant mixture blowdown rate after approximately 6 seconds, with no credit being taken for any incoming control rod drive water. It is evident from Table 15.6-3 that if the MSIVs were closed in 10.5 seconds, the core would not be uncovered even for the limiting condition of zero steam separation in the separators.

15.6.4.3.7 Core Cooling

As a result of the steam line break outside the containment, the recirculation pumps would continue to function for at least 6 - 7 seconds maintaining near rated core flow and core cooling even if it were conservatively assumed that the feedwater pumps stop completely and the separators are 100% efficient such that rapid depressurization continued until isolation. The vessel pressure would need to be reduced approximately 150 psi before the core inlet plenum would flash and cavitation would occur in the recirculation pumps. When the lower plenum saturates, core inlet flow increases due to the swell caused by flashing and provides an additional 1 - 2 seconds of core cooling. Even after core inlet flow decreases, the blowdown tests described in the Dresden Plant Design and Analysis Report (PDAR), Amendment 5, indicate that several seconds of boiling heat transfer would continue before the fuel rods dry out. Thus, even for the conservative limiting case of no mixture blowdown, effective core cooling would be maintained throughout the blowdown for a 10.5-second MSIV closure, and the fuel cladding would not exceed saturation temperature by more than 15°F.

After the MSIVs close, the reactor would be cooled by operation of either the isolation condenser or the HPCI system in conjunction with the shutdown cooling system.

15.6.4.4 Barrier Performance

Since the separators would not be 100% efficient, the depressurization rate would be reduced by two-phase mixture blowdown before the lower plenum saturates, and the core flow would be maintained by the jet and drive pumps until the MSIVs close. It is estimated that the minimum flow through the core would drop by less than 10% in this case. The core heat flux, in the meantime, would drop continuously during the blowdown, and would be reduced to about 40% of rated at the end of 8 seconds. Even at the full power level of 2527 MWt, a 50% reduction in core flow could be tolerated before a minimum critical heat flux ratio (MCHFR) of unity would occur.

The MCHFR throughout the transient was calculated using a digital computer code. The code calculates the thermal hydraulic response of a single nuclear reactor coolant channel, consisting of an array of cylindrical fuel rods surrounded by channel walls. The code includes provisions for pressure losses, fluid expansion, inlet flow variations, axial power shape, and various heat transfer modes. The MCHFR throughout the transient was calculated for the case of normal ac power available and the degraded case of simultaneous loss of normal ac power, which would cause a recirculation pump trip. The core flowrates both with and without recirculation pumps were calculated using the five-node digital computer code used to calculate internal forces. These results are shown on Figure 15.6-3 which includes the sweep time of the core. The calculated MCHFR would not go below 2.0 (Figure 15.6-3) throughout the transient even if the isolation valves were closed at the longest time; therefore, core integrity is maintained throughout the accident and no fuel damage should result. (MCHFR has been replaced in later analyses by MCPR. Since these quantities are not interchangeable, the terminology in this analysis has not been revised.)

15.6.4.5 Radiological Consequences for the MSLB

Regulation 10 CFR 50.67, "Accident Source Term," provides a mechanism for power reactor licensees to voluntarily replace the traditional TID 14844 (Ref. 41) accident source term used in design-basis accident analyses with an "Alternative Source Term" (AST). The methodology of approach to this replacement is given in USNRC Regulatory Guide 1.183 (Ref. 42) and its associated Standard Review Plan 15.01.1 (Ref. 43).

Accordingly, Dresden Nuclear Power Station (DNPS), Units 2 and 3, has applied the AST methodology for several areas of operational relief in the event of a Design Basis Accident (DBA), without crediting the use of previously assumed safety systems. Amongst these systems are the Control Room Emergency Filter (CREF) and the Standby Gas Treatment System (SGTS).

In support of a full-scope implementation of AST as described in and in accordance with the guidance of Ref. 42, AST radiological consequence analyses are performed for the four DBAs that result in offsite exposure (i.e., Loss for Coolant Accident (LOCA), Main Steam Line Break (MSLB), Fuel Handling Accident (FHA), and Control Rod Drop Accident (CRDA)).

Implementation consisted of the following steps:

- Identification of the AST based on plant-specific analysis of core fission product inventory,
- Calculation of the release fractions for the four DBAs that could potentially result in control room and offsite doses (i.e., LOCA, MSLB, FHA, and CRDA),
- Analysis of the atmospheric dispersion for the radiological propagation pathways,
- Calculation of fission product deposition rates and removal mechanisms,
- Calculation of offsite and control room personnel Total Effective Dose Equivalent (TEDE) doses.

15.6.4.5.1 Regulatory Guide 1.183 Compliance

The analyses are prepared in accordance with the guidance provided by Regulatory Guide 1.183 (Ref. 42).

15.6.4.5.1.1 Dose Acceptance Criteria

The AST acceptance criteria for Control Room dose for postulated major credible accident scenarios such as those resulting in substantial meltdown of the core with release of appreciable quantities of fission products is provided by 10 CFR 50.67, which requires.

"Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident".

This limit is applied by Regulatory Guide 1.183 to all of the accidents considered with AST.

The AST acceptance criteria for an individual located at any point on the boundary of the exclusion area (the Exclusion Area Boundary or EAB) are provided by 10 CFR 50.67 as 25 rem TEDE for any 2-hour period following the onset of the postulated fission product release.

The AST acceptance criteria for an individual located at any point on the outer boundary of the low population zone (LPZ) are provided by 10 CFR 50.67 as 25 rem TEDE during the entire period of passage of the radioactive cloud resulting from the postulated fission product release.

These limits are applied by Regulatory Guide 1.183 to events with a higher probability of occurrence for the MSLB, provide the following acceptance criteria:

- For the BWR MSLB for the case of an accident assuming fuel damage or a pre-incident Iodine spike, doses at the EAB and LPZ should not exceed 25 rem TEDE for the accident duration (2 hours dose for EAB and 30 day dose for LPZ). For MSLB accidents assuming normal equilibrium Iodine activity , doses should not exceed 2.5 rem TEDE for the accident duration.

15.6.4.5.2 Computer Codes

New AST calculations for the MSLB were prepared to simulate the radionuclide release, transport, removal, and dose estimates associated with the postulated accidents.

While the RADTRAD computer code (Ref.48) endorsed by the NRC for AST analyses was used in the calculations for the LOCA, CRDA and FHA, the MSLB was analyzed using the Regulatory Guide 1.183 methodology in a spreadsheet. The MSLB assessment takes no credit for control room isolation, emergency ventilation or filtration of intake air for the duration of the accident event.

15.6.4.5.3 Source Term

15.6.4.5.3.1 Core Inventory

Not applicable for the MSLB

15.6.4.5.3.2 Reactor Coolant Inventory

The reactor coolant fission product inventory for MSLB analysis is based on the Technical Specification limits in terms of Dose Equivalent I-131 (the concentration of I-131 that alone would produce the same dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present), using inhalation Committed Effective Dose Equivalent (CEDE) dose conversion factors from Federal Guidance Report No.11 (Ref. 54). Cesium, as Cesium Iodide, and Noble Gas releases are also considered, but the iodine isotopes are the only significant dose contributors.

15.6.4.5.3.3 Release Fraction

Not applicable for the MSLB

15.6.4.5.4 Methodology

15.6.4.5.4.1 Dose Calculations

As per Regulatory Guide 1.183 (Ref. 42), Total Effective Dose Equivalent (TEDE) doses are determined as the sum of the CEDE and the Effective Dose Equivalent (EDE) using dose conversion factors for inhalation CEDE from Federal Guidance Report No. 11 (Ref. 54) and for external exposure EDE from Federal Guidance Report No. 12 (Ref 55).

15.6.4.5.4.2 Main Steam Line Break (MSLB)

15.6-4a lists the key assumptions and inputs used in the analysis. The postulated MSLB accident assumes a double-ended break of one main steam line outside the primary containment with displacement of the pipe ends that permits maximum blowdown rates. However, the break mass released is taken for the dose calculations as a bounding maximized value for all current Boiling Water reactor plants of 140,000 pounds of water, as provided in Standard review Plan 15.6.4 for a GESSAR-251 plant. This value bounds for dose calculation purposes the historic UFSAR values, ensuring that the dose consequences are maximized and that the releases bound any other credible pipe break. Two activity release cases corresponding to the pre-accident spike and maximum equilibrium concentration allowed by Technical Specifications of 4.0 µCi/gm and 0.2 µCi/gm dose equivalent I-131 respectively were assumed, with inhalation CEDE dose conversion factors from Federal Guidance Report 11 conservatively used for normalized Dose Equivalent I-131 determination. The released activity assumptions are consistent with the guidance provided in Appendix D of Regulatory Guide 1.183.

The analysis assumes an instantaneous ground level release. For the control room dose calculations, the released reactor coolant and steam is assumed to expand to a hemispheric volume at atmospheric pressure and temperature (consistent with an assumption of no Turbine Building credit). This hemisphere is then assumed to move at a speed of 1 meter per second downwind past the control room intake. No credit is taken for buoyant rise of the steam cloud or for decay, and dispersion of the activity of the plume was conservatively ignored. For offsite locations, the buoyant rise of the steam cloud is similarly ignored, and the ground level dispersion is based on the conservative and simplified Regulatory Guide 1.5 (Ref. 56) methodology.

The radiological consequences following an MSLB accident were determined using a Microsoft EXCEL spreadsheet. The following significant assumptions were made:

- Iodine activity distribution in the coolant Iodine activity distribution in the coolant as follows:

Iodine Isotope	Activity (μCi/cc)
I-131	0.067
I-132	0.38
I-133	0.40
I-134	0.53
I-135	0.49

- Release from the break to the environment is assumed instantaneous. No holdup in the Turbine Building or dilution by mixing with Turbine Building air volume is credited.
- The steam cloud is assumed to consist of the portion of the liquid reactor coolant release that flashed to steam.
- The activity of the cloud is based on the total mass of water released from the break. This assumption is conservative because it considers the maximum release of fission products.
- Flashing fraction of liquid water was released was assumed as 40%. However, all activity in the water is assumed to be released.

15.6.4.5.4.3 Atmospheric Dispersion Factors (χ/Os)

For the control room dose calculations, the released reactor coolant and resultant flashed steam is assumed to expand to a hemispheric volume at atmospheric pressure and temperature (consistent with an assumption of no Turbine Building credit). This hemisphere is then assumed to move at a speed of 1 meter per second downwind past the control room intake. No credit is taken for buoyant rise of the steam cloud or for decay, and dispersion of the activity of the plume was conservatively ignored.

For offside locations, the buoyant rise of the steam cloud is similarly ignored, and the ground level dispersion is based on the conservative and simplified Regulatory Guide 1.5 (Ref.56) methodology. Table 15.6-41b lists χ/Q values for the EAB and LPZ boundaries.

15.6.4.5.5 Summary and Conclusions

The radiological consequences of the postulated MSLB are given in Table 15.6-4c. As indicated, the control room, EAB and LPZ calculated doses are within regulatory limits after AST implementation.

15.6.5 Loss-of-Coolant Accidents Resulting from Piping Breaks Inside Containment

See the introduction to Section 15.6 for information regarding the use of details from this analysis description which may not be applicable to the current fuel cycle.

A loss-of-coolant accident (LOCA) resulting from piping breaks inside containment would result in the heating and pressurization of containment, a challenge to the emergency core cooling system (ECCS), and the potential release of radioactive material to the environment. The response of the containment to a LOCA is discussed in Section 6.2.1.3.2. The fuel thermal response and ECCS performance are described in Section 6.3.3.

AREVA ATRIUM 10XM methods and fuel are only applicable to Unit 3.

15.6.5.1 Identification of Causes and Frequency Classification

The full range of LOCAs has been analyzed from a small rupture where the makeup flow is greater than the coolant loss rate to a highly improbable circumferential recirculation line break. The initial power level assumed was 2578 MWt.

Section 6.3.3 includes discussion on the GE LOCA analysis for the uprated power at 2957 MWt. Section 6.3.3 includes discussion on the AREVA LOCA analysis for ATRIUM 10XM fuel and on the Westinghouse LOCA analysis for SVEA-96 Optima2 fuel at the extended power uprate (2957 MWt).

This event is classified as a limiting fault, i.e., an event that is not expected to occur but is postulated because the consequences may result in the release of significant amounts of radioactive material.

15.6.5.2 Sequence of Events and Systems Operation

The postulated LOCA results from a rupture of primary system piping. The loss of inventory produces core depressurization and decreasing water level in the vessel.

The HPCI, automatic depressurization system (ADS), LPCI, and core spray systems would act to cool the core following the accident.

Reactor scram occurs on low water level or high drywell pressure. The ECCS is automatically actuated on either low-low water level or high-drywell pressure. A simultaneous loss of offsite power is assumed with the break together with the worst single failure in the ECCS. The assumed simultaneous loss of offsite power causes the recirculation pumps to coast down and delays initiation of ECCS until the diesel generators attain speed and are loaded.

The course of the accident depends on the break size and location.

For a small break, failure of HPCI is the worst single failure, and core cooling is provided by the ADS, LPCI, and the core spray systems. For these breaks, the vessel depressurizes relatively slowly due to the small break size. The ADS automatically actuates to reduce pressure so that the low pressure cooling system can function.

For larger breaks, failure of the LPCI injection valve is the most severe single failure since the vessel depressurizes faster and the HPCI system is not available due to the low-system pressure. The core uncovers as coolant inventory is lost through the break. After several minutes, injection flow from the core spray refloods the core.

The coolant lost through the rupture is condensed by the pressure suppression pool, thus reducing primary containment pressure. Energy is removed from the pressure suppression pool by the containment cooling system.

15.6.5.3 Core and System Performance

The analyses for loss-of-coolant accidents were originally performed using calculational models and techniques different from those that are currently used.

Section 6.3.3 discusses the fuel thermal response, the ECCS performance, and the current analysis models, according to 10 CFR 50, Appendix K, and the results comply with the NRC 10 CFR 50.46 criteria.

Section 6.2.1 discusses the containment and coolant blowdown responses.

In the case of a LOCA, the reactor coolant system inventory loss would result in a high drywell pressure and reduced reactor vessel pressure. The concurrent high drywell pressure and low reactor vessel pressure provide an initiation signal which brings a coolant injection system into operation. During the early phase of the LOCA depressurization transient, core cooling is provided by the existing coolant inventory. In the latter stage of system depressurization and after depressurization has been achieved, the core spray provides core cooling and supplies liquid to refill the lower portion of the reactor vessel and reflood the core. The reflood process provides sufficient heat removal to terminate the core temperature transient.

15.6.5.4 Barrier Performance {HISTORICAL}

The break of a pipe in the primary system within the containment is considered the maximum credible accident because of the large potential for fission product release. A break in the "second line of defense" against fission product release in the primary system could lead to violation of the "first line of defense," the fuel clad. This, in turn, means that the last major barrier to fission product release, the containment, assumes an increased importance.

15.6.5.4.1 Fission Product Release from the Fuel

Calculations performed during initial licensing show that about 45% of the fuel rods in the core might experience cladding perforation, based on a 1500°F perforation temperature, but no fuel would melt. A maximum of 1% of the noble gas activity and 0.5% of the halogen activity contained in a fuel rod is in the plenums and could be released if the cladding were perforated. Negligible solid or particulate activity would be released from the perforated rods. The amount of the total reactor fission product inventory released from the fuel would be about 0.45% of the noble gases and about 0.225% of the halogens. The release would occur as the cladding is perforated.

For Westinghouse SVEA-96 Optima2 fuel, a maximum cladding temperature of 1385°F was evaluated to not result in cladding rupture.

15.6.5.4.2 Fission Product Release to the Drywell {HISTORICAL}

The fallout and plateout of fission products within the reactor vessel and piping would reduce the amount of fission products available for transport to the drywell. Of the halogens that would be released from the fuel, 5% are estimated to be organic halides, principally methyl iodide.

Because organic halogens are less soluble in water and more difficult to filter than uncombined halogens, a conservatively large fraction of halogens was assumed to be organic. Fuel melting experiments[12-14] have shown that 0.1% - 3% of the released halogens are organic. For the LOCA analysis, 5% of the halogens released from the fuel are assumed to be organic. This assumption is conservative by a factor of 1.5 to 50.

No organic halogens are assumed to fall out or plate out. Of the remaining 95% (which are inorganic), 50% would plate out on metal surfaces. The amount of fallout and plateout in the reactor vessel and piping assumed in the analysis is as follows:

Fallout and Plateout

Fission Product Group	Percent
Noble gases	0
Halogens, organic	0
Halogens, inorganic	50

The pressure suppression pool contains approximately 112,000 cubic feet of water for absorption of halogens. The containment air-to-water volume ratio is about 2.5. All the organic halogens are assumed to remain airborne; although, at an air-to-water ratio of 2.5 about half would be expected to be absorbed in water.[15] In Oak Ridge Reactor (ORR) in-pile UO_2 melting experiments, the condensation of the steam in the gas stream removed essentially all halogens from the gas stream.[16] The inorganic halogen partition factor[17,18] would be greater than 10^4.

These experiments, including both steam condensation in vapor suppression systems and in air, correspond to the conditions accompanying a LOCA. The initial blowdown through the suppression pool would be mostly air with the trailing phases of blowdown essentially all steam. Most fission product release would accompany the final steam release and would be efficiently scrubbed by the condensing steam. Airborne inorganic halogen and solid fission products in the drywell would be rapidly removed by the containment spray and steam condensation then mixed with water in the suppression pool. For the accident analysis, a partition factor of 10^2 for inorganic halogens was used. Inorganic halogens are assumed to be reevolved from the water as leakage from the containment reduces the inventory of airborne halogens. The assumption of a high fraction of organic halogens with no absorption in water and the conservative water-to-air partition factor for inorganic halogens results in a conservatively high fraction of halogens remaining airborne available for leakage from the containment. The inventory of airborne fission products in the drywell which could leak into the reactor building is shown in Table 15.6-5.

The UFSAR licensing basis prior to extended power uprate utilizes the TID-14844 methodology, which establishes the source term based on rated core thermal power. The impact of extended power uprate on the radiological consequences is discussed at the end of relevant sections.

15.6.5.4.3 Fission Product Release from Drywell to the Reactor Building {HISTORICAL}

The primary containment leakage rates were calculated assuming that the primary containment leaks 0.5% of the contained free volume per 24 hours at 25 psig; the turbulent rough passage equation[19] was used for interpolation to lower pressures. The long term primary containment pressure is shown in Section 6.2.1.3. The corresponding containment leakage for case d, shown in Figure 6.2-19, represents a highly conservative condition of operation of only one of the two core spray system loops and part of one containment cooling loop, i.e., one LPCI pump and one heat exchanger (with two containment cooling service water pumps) in service.

If fission products leak from the drywell, drywell high pressure or reactor building high-radiation signals would isolate secondary containment as described in Section 6.2.3. The analysis assumed all the noble gases and halogens released into the reactor building remain airborne.

The airborne fission product inventory in the reactor building, which was evaluated considering the leakage from the drywell to the reactor building, radioactive decay, fallout and plateout, and an air change rate of 100% of the reactor building volume per day, is shown in Table 15.6-6.

The UFSAR licensing basis prior to extended power uprate utilizes the TID-14844 methodology, which establishes the source term based on rated core thermal power. The impact of extended power uprate on the radiological consequences is discussed at the end of relevant sections.

15.6.5.4.4 Fission Product Release from Reactor Building to Atmosphere {HISTORICAL}

THIS SECTION IS MAINTAINED FOR HISTORICAL INFORMATION ONLY.

The halogens which leak from the pressure suppression containment into the reactor building are exhausted by the SBGTS through a high efficiency filter and an activated charcoal adsorber. The reactor building exhaust air is treated to reduce the humidity so that the activated charcoal adsorber is effective for removal of organic halogens. Tests on adsorber efficiencies have shown that inorganic halogens are removed by charcoal filters with efficiencies greater than 99.99%.[20,21]These tests have also shown that organic halogens are removed at a relative humidity less than 30% with adsorber efficiencies from 99.9% to 99.9999%.[21-25] The activated charcoal adsorber on the EVESR at Vallecitos Atomic Power Laboratory retained organic halogens produced at power operation with efficiency from 99.8% to 99.9% at a relative humidity of 10 to 15%. The experimental results were used as the basis of the original SBGTS design. The system is designed to provide the necessary residence time in the adsorbers. Thus, the analysis assumption of 99% efficiency for the removal of inorganic and organic halogens in the SBGTS is conservative by approximately 4 orders of magnitude.

The compounding of conservative assumptions used in the LOCA analysis results in calculated doses from halogens that are 20 to 1000 times higher than the doses which would actually be expected. LOCA discharge rates to the chimney (stack) are shown in Table 15.6-7.

The UFSAR licensing basis utilizes the TID-14844 methodology, which establishes source term based on rated core thermal power. The impact of extended power uprate on the radiological consequences is discussed at the end of relevant sections.

15.6.5.5 Radiological Consequences for the LOCA

Regulation 10 CFR 50.67, "Accident Source Term," provides a mechanism for power reactor licensees to voluntarily replace the traditional TID 14844 (Ref. 41) accident source term used in design-basis accident analyses with an “Alternative Source Term” (AST). The methodology of approach to this replacement is given in USNRC Regulatory Guide 1.183 (Ref. 42) and its associated Standard Review Plan 15.0.1 (Ref. 43).

The AST methodology has been applied to justify that a Design Basis Accident (DBA) can be accommodated without crediting the use of previously assumed safety systems. Amongst these systems are the Control Room Emergency Filter (CREF) and the Standby Gas Treatment System (SGTS).

In support of a full-scope implementation of AST as described in and in accordance with the guidance of Ref. 42, AST radiological consequence analyses are performed for the four DBAs that result in offsite exposure (i.e., Loss of Coolant Accident (LOCA), Main Steam Line Break (MSLB), Fuel Handling Accident (FHA), and Control Rod Drop Accident (CRDA)).

Implementation consisted of the following steps:

- Identification of the AST based on plant-specific analysis of core fission product inventory,
- Calculation of the release fractions for the four DBAs that could potentially result in control room and offsite doses (i.e., LOCA, MSLB, FHA, and CRDA),
- Analysis of the atmospheric dispersion for the radiological propagation pathways,
- Calculation of fission product deposition rates and removal mechanisms,
- Calculation of offsite and control room personnel Total Effective Dose Equivalent (TEDE) doses,
- Evaluation of suppression pool pH to ensure that the particulate iodine deposited into the pool during a DBA LOCA does not re-evolve and become airborne as elemental iodine.

15.6.5.5.1 Regulatory Guide 1.183 Compliance

The analyses are prepared in accordance with the guidance provided by Regulatory Guide 1.183 (Ref. 42).

15.6.5.5.1.1 Dose Acceptance Criteria

The AST acceptance criteria for Control Room dose for postulated major credible accident scenarios such as those resulting in substantial meltdown of the core with release of appreciable quantities of fission products is provided by 10 CFR 50.67, which requires

"Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident."

This limit is applied by Regulatory Guide 1.183 to all of the accidents considered with AST.

The AST acceptance criteria for an individual located at any point on the boundary of the exclusion area (the Exclusion Area Boundary or EAB) are provided by 10 CFR 50.67 as 25 rem TEDE for any 2-hour period following the onset of the postulated fission product release.

The AST acceptance criteria for an individual located at any point on the outer boundary of the low population zone (LPZ) are provided by 10 CFR 50.67 as 25 rem TEDE during the entire period of passage of the radioactive cloud resulting from the postulated fission product release.

These limits are applied by Regulatory Guide 1.183 to events with a higher probability of occurrence to provide acceptance criteria.

15.6.5.5.2 Computer Codes

New AST calculations for the LOCA were prepared to simulate the radionuclide release, transport, removal, and dose estimates associated with the postulated accident scenario.

The RADTRAD computer code (Ref. 48) endorsed by the NRC for AST analyses was used in the calculations for the LOCA. The RADTRAD program is a radiological consequence analysis code used to estimate post-accident doses at plant offsite locations and in the control room.

Offsite χ/Qs were calculated using the guidance of Regulatory Guide 1.145 (Ref. 50); control room χ/Qs were calculated with the ARCON96 computer code (Ref. 51).

All of these computer codes and methodologies have been used by the NRC staff in their safety reviews.

15.6.5.5.3 Source Terms

15.6.5.5.3.1 Core Inventory

The inventory of reactor core fission products for RADTRAD the AST LOCA analyses is based on maximum full power operation at a power level of 3016 MWth, the Extended Power Uprate (EPU) thermal power of 2957 MWth plus a 2% instrument error per Reg Guide 1.49 (Ref. 52). The fission products used for the accidents are the 60 isotopes of the standard RADTRAD input library, determined by the code developer as significant in dose consequences. The post-LOCA consequences for the Westinghouse Optima2 fuel were analyzed at a core average exposure of 39 GWD/MTU in Reference 57. The post-LOCA consequences for the AREVA ATRIUM 10XM reload were evaluated at a core average exposure of 39 GWD/MTU and between core average enrichments of 3.9 and 4.5 weight percent U-235 in Reference 58.

15.6.5.5.3.2 Reactor Coolant Inventory

Reactor coolant activity is not applicable for the LOCA assessment.

15.6.5.5.3.3 Release Fraction

Current design basis accident evaluations as modified by Regulatory Guide 1.183 (Ref. 42) were used to determine the specific releases of radioactive isotopes at the given stages of fuel pin failure and provide these releases as a percentage of the total release for the accident, as summarized in sections 15.6.5.5.5.2, below.

15.6.5.5.4 Methodology

15.6.5.5.4.1 Dose Calculations

As per Regulatory Guide 1.183 (Ref. 42), Total Effective Dose Equivalent (TEDE) doses are determined as the sum of the CEDE and the Effective Dose Equivalent (EDE) using dose conversion factors for inhalation CEDE from Federal Guidance Report No. 11 (Ref. 54) and for external exposure EDE from Federal Guidance Report No. 12 (Ref. 55). Breathing rates and occupancy factors are given in Table 15.6-9.

15.6.5.5.4.2 Loss of Coolant Accident (LOCA)

The LOCA radiological assessment was performed in accordance with the guidance of Regulatory Guide 1.183. The key inputs used in this analysis are included in Tables 15.6-10 through 15.6-13. These inputs and assumptions are grouped into three main categories (i.e., release, transport, and removal). The initial source term parameters are given in Table 15.6-9.

LOCA Release Inputs

Key parameters used in the release pathway modeling for the LOCA analysis are given in Table 15.6-11. The primary containment is assumed to leak at 3.0 v%/day (the sum of primary-to-secondary leakage and leakage through the MSIVs) for the entire 30-day duration of the accident. No primary containment leakage, with the exception of MSIV leakage, has been identified which bypasses the secondary containment and is released unfiltered to the atmosphere.

The analysis assumes that the leak rate through the MSIVs to the environment is 150 scfh at 43.9 psig for the entire 30-day duration of the accident, split into 60 scfh through one line with the MSIV failed, 60 scfh through another intact line, and 30 scfh through a second intact line.

The analysis assumes an engineered safety feature systems liquid leakage rate outside of containment of 2 gpm day (Table 15.6-11). Ten percent of the activity in the leakage is assumed to become airborne. This is consistent with Regulatory Guide 1.183. Although the engineered safety feature systems leakage rate may realistically be assumed to begin approximately 15 minutes following the accident, with the actuation of the drywell sprays, the present analysis conservatively assumes leakage to begin at the onset of the accident and to continue throughout the 30-day duration of the postulated accident. (Prior to 15 minutes post accident, there is no engineered safety feature systems recirculation of suppression pool water, therefore no leakage is assumed since an ECCS failure is an implicit assumption of the core damage leading to the AST.)

The Regulatory Guide 1.183 accident isotopic release specification allows deposition of iodine in the suppression pool. Essentially all of the iodine is assumed to remain in solution as long as the pool pH is maintained at or above a level of 7. Station procedures will direct operators, upon detection of symptoms indicating that core damage is occurring (e.g., primary containment high radiation), to manually initiate the SLC System. The calculation results demonstrate the buffering effect of the boron solution maintains the suppression pool pH above 7 for the 30-day duration of the postulated LOCA (Table 15.6-17). Maintaining suppression pool pH at or above a level of 7, as an assumption in support of radiological consequence analysis, is a change to the design and licensing bases.

LOCA Transport Inputs

Prior to the LOCA, the reactor building fans are running and maintaining the reactor building at a negative pressure. At the beginning of the LOCA event, the reactor building exhaust fans are tripped and the reactor building (i.e., secondary containment) is then exhausted by the SGT System continuing the building's negative pressure thus precluding unfiltered exfiltration.

In the analysis, the accident activity was assumed to enter the control room unfiltered for the first 40 minutes of the LOCA at a nominal CREV System filtered ventilation flow rate. After 40 minutes, the CREV System is manually initiated and filtered flow and unfiltered leakage is then assumed to enter the control room envelope. Flow rates are given in Table 15.6-12.

LOCA Removal Inputs

Key parameters identifying radionuclide removal processes are given in Table 15.6-13. The activity of elemental iodine and aerosols released from the core into the drywell is reduced by deposition (i.e., plate-out) and settling in the drywell utilizing the natural deposition values identified in the RADTRAD code. No credit is assumed for natural deposition of elemental or organic iodine, or for suppression pool scrubbing.

Containment leakage into the reactor building is collected by the SGT System which exhausts the reactor building, via filters, and reduces releases. The deposition removal mechanisms are characteristics of the AST methodology and represent a change in the plant design and licensing basis.

Main steam line pipe deposition was modeled using the RADTRAD code with removal coefficients based on gravitational settling. Two-node treatment is used for each steam line in which flow occurs. The first node is from the reactor vessel to the inboard MSIV. The second node is from the inboard MSIV to the outboard MSIV. No credit is taken for holdup or plate-out in the main steam lines beyond the outboard MSIV. Additionally, no credit is taken for holdup and plate-out in the main condenser. Main steam line deposition was based on using the shortest line (i.e., most rapid transport) for the worst case line (i.e., the one with the assumed failed inboard isolation valve).

Removal efficiencies for the SGT System and the CREV System filters are given in Table 15.6-13.

The analysis assumptions for the transport, reduction, and release of the radioactive material from the fuel and the reactor coolant are consistent with the guidance provided in Appendix C of Regulatory Guide 1.183.

15.6.5.5.4.3 Atmospheric Dispersion Factors (χ/Qs)

The station release points and control room intake are shown in Figure 1. Table 15.6-15 lists χ/Q values used for the control room dose assessments. For release points applicable to the LOCA, the zero velocity vent release χ/Q values were calculated with the ARCON96 computer code, as derived in Section 2.3.5-1. The elevated release χ/Q values (i.e., Station Chimney release) are calculated using Regulatory Guide 1.145 methodology and include an initial up to 30-minute fumigation period, as also derived in Section 2.3.5-1.

Table 15.6-16 lists χ/Q values for the EAB and LPZ boundaries. These χ/Q values are calculated using Regulatory Guide 1.145 methodology, as derived in Section 2.3.5-1.

15.6.5.5.5 Summary and Conclusions

The radiological consequences of the postulated LOCA are given in Table 15.6-14. As indicated, the control room, EAB, and LPZ calculated doses are within regulatory limits after AST implementation.

15.6.6 References

1. Letter from W.A. Paulsen (NRC) to L. Del George (CECo), January 4, 1982, Dresden 2 SEP Topic XV-15, Inadvertent Opening of a BWR Safety/Relief Valve.

2. Letter from D. Crutchfield (NRC) to W. Counsil (Northeast Nuclear Energy Co.), October 28, 1981, SEP Topic XV-15 for Millstone 1 (Docket 50-245).

3. Letter from D. Crutchfield (NRC) to I. Finfrock (Jersey Central Power & Light Co.), December 4, 1981, SEP Topic XV-15 for Oyster Creek (Docket 50-219).

4. Moody, F.J., "Maximum Flowrate of a Single Component, Two Phase Mixture," Journal of Heat Transfer. Trans., ASME, Series C, Vol. 87, p. 134.

5. J.F. Wilson, et al., "The Velocity of Rising Steam in a Bubbling Two Phase Mixture," ANS Transactions, Vol. 5, No. 1, p. 151, 1962;

6. Moody, F.J., "Liquid-Vapor Action in a Vessel During Blowdown," APED-5177, June 1966.

7. Moody, F.J., "Maximum Two Phase Vessel Blowdown from Pipes," ASME Paper No. 65-WA/HT-1.

8. Tippet, F.E., "Transient Flow of Steam-Water Mixture," Excerpt from Water Wall Rupture in a High Pressure Reactor - Hydraulic and Heat Transfer Effect," HW-40388, December 1955.

9. Watson, L.C., et al., "Iodine Containment by Dousing in NPD-II," AECL 1130, October 1960.

10. Diffey, H.R., et al., "Iodine Cleanup in a Steam Suppression System," International Symposium on Fission Product Release and Transport Under Accident Conditions, April 1956.

11. LOCA Analysis Report for Duane Arnold Station, NEDO-21082-02-1A, July 1977.

12. Collins, D.A., et al., "International Symposium on Fission Product Release and Transport Under Accident Conditions," Paper 59, 1965, Oak Ridge, Tennessee.

13. Parker, et al., "Fission Product Release," SIFTOR Draft, Chapter 18, Volume II.

14. Collins, R.D., and Hillary, "International Symposium on Fission Product Release and Transport Under Accident Conditions," Oak Ridge, Tennessee, Paper 44, April 1965.

15. Diffey, et al., "International Symposium on Fission Product Release and Transport Under Accident Conditions", Oak Ridge, Tennessee, Paper 41, April, 1965.

16. Miller, et al., "International Symposium on Fission Product Release and Transport Under Accident Conditions", Oak Ridge, Tennessee, Paper 12, April 1965.

17. Allen, T.L., and Keefer, R.M., "The Formation of Hypoiodus Acid and Hydrated Iodine Cation by the Hydrolysis of Iodine," JACS 77, No. 11, June 1955.

18. Watson, Bancroft, and Hoelke, "Iodine Containment by Dousing in NPD-II", AECL-1130, 1960.

19. Maccary, R.L., et al., "Leakage Characteristics of Steel Containment Vessels and the Analysis of Leakage Rate Determinations," TID-20583, May 1964.

20. Keiholts, G.W., and Barton, C.J., "Behavior of Iodine in reactor Containment Systems," ORNL-NSIC-4, p. 64, February 1965.

21. Adams and Browning, "International Symposium on Fission Product Release and Transport Under Accident Conditions," Oak Ridge, Tennessee, Paper 46, April 1965.

22. Collins, R.D. and Hillary, "International Symposium on Fission Product Release and Transport Under Accident Conditions," Oak Ridge, Tennessee, Paper 44, April 1965.

23. Collins and Eggleton, "Behavior of Iodine in Reactor Containment Systems," ORNL-NSIC-4, p. 65, February 1965.

24. Adams and Browning, "Behavior of Iodine in Reactor Containment Systems," ORNL-NISC-4, p. 65, February 1965.

25. Collins, D.A., et al., "International Symposium on Fission Product Release and Transport Under Accident Conditions," Oak Ridge, Tennessee, Paper 45, April 1965.

26. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.

27. Nuclear Regulatory Commission, Standard Review Plan 6.4,for the Review of Safety Analysis Reports for Nuclear Power Plants, Section 6.4, "Habitability Systems" NUREG-0800 Rev. 1, December 1978.

28. Nuclear Regulatory Commission, Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," Rev. 2, June 1974.

29. ORNL NSIC-5, "U.S. Containment Technologies," Oak Ridge National Laboratory and Bechtel Corp., August 1965.

30. Nuclear Regulatory Commission, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Section 15.4.9, "Spectrum of Rod Drop Accidents (BWR)," NUREG-0800 Rev. 1, December 1978.

31. "Technological Basis for Models of Spray Washout of Airborne Contaminants in Containment Vessels," NUREG/CR-009, A.K. Posta, R.R. Sherry, P.S. Tam, October 1978.

32. Murphy, K.G. and Campe,K.M., "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19," 13th AEC Air Cleaning Conference.

33. Slade, D.H., ed., Meteorology and Atomic Energy, TID, 24190, 1968.

34. Walker, D.H., Nassano, R.N., Capo, M.A., 1976, "Control Room Ventilation Intake Selection for the Floating Nuclear Power Plant," 14 ERDA Air Cleaning Conference.

35. Hatcher, R.N., Meroney, R.N., Peterka, J.A., and Kothari, K., "Dispersion in the Wake of a Model Industrial Complex," NUREG-0373, 1978.

36. Meroney, R.N. and Yang, B.T.,, "Wind Tunnel Study on Gaseous Mixing Due to Various Stack Heights and Injection Rates Above an Isolated Structure," FDDL Report CER 71-72RNM-BTY16, Colorado State University, 1971.

37. Start, G.E., Cate, J.H., Dickson, C.R., Ricks, N.R., Ackerman, G.H. and Sagendorf, J.F., "Rancho Seco Building Wake Effects on Atmospheric Diffusion," NOAA Technical Memorandum, ERL ARL-69, 1977.

38. Morewitz, H.A., Johnson, R.P., Nelson, C.T., Vaughn, E.V., Guderjahn, C.A., Hillard, R.K., McCormack, J.D., and A.K., "Attenuation of Airborne Debris from Liquid-Metal Fast Breeder Reactor Accident," Nuclear Technology, Volume 46, December 1979.

39. Nuclear Regulatory Commission, Regulatory Guide 1.5, “Assumptions Used For Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors”, 3/10/71.

40. Calculation DRE97-0150, “Control Room Habitability Following a Main Steam Line Break”, Revision 2.

41. U. S. Atomic Energy Commission, Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 23, 1962.

42. U. S. Nuclear Regulatory Commission Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

43. U. S. Nuclear Regulatory Commission Standard Review Plan 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000.

44. Not used

45. Not used

46. Not used

47. Not used

48. RADTRAD Code, "A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," Version 3.03.

49. PAVAN Code, "An Atmospheric Dispersion Program for Evaluating Design Bases Accidental Releases of Radioactive Materials from Nuclear Power Stations."

50. U. S. Nuclear Regulatory Commission Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, November 1982.

51. ARCON96 Code, "Atmospheric Relative Concentrations in Building Wakes".

52. U. S. Nuclear Regulatory Commission Regulatory Guide 1.49, "Power Levels of Nuclear Power Plants," Revision 1, December 1973.

53. GE Task Report No. GE-NE-A22-00103-64-01, Revision 0, Project Task Report: "Dresden and Quad Cities Asset Enhancement Program – Task T0802: Radiation Sources and Fission Products", August 2000.

54. Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion", 1988.

55. Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil", 1993.

56. Regulatory (Safety) Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors," 3/10/71.

57. DRE05-0048, Revision 2, "Dresden Units 2 & 3 Post-LOCA EAB, LPZ, and CR Dose – AST Analysis."

58. DRE05-0048, Revision 3, "Dresden Units 2 & 3 Post-LOCA EAB, LPZ, CR Dose – AST Analysis."

Table 15.6-1

ANALYSIS ASSUMPTIONS USED FOR RADIOLOGICAL CONSEQUENCES OF INSTRUMENT LINE BREAK OUTSIDE CONTAINMENT AT DRESDEN UNIT 2

A.	Mass of reactor coolant in vessel mixing volume (lbm)	590,000
B.	Reactor water cleanup system cleanup rate (gal/min)	600
C.	Condensate demineralizer cleanup rate (carryover fraction x feedwater flowrate) (gal/min)	263
D.	Iodine spiking factor	500
E.	Flash fraction (%)	37
F.	Duration of accident (hr)	4
G.	X/Q ground level values (s/m^3)	
	0–2 hr, EAB	2.6×10^{-4}
	0–4 hr, LPZ	1.1×10^{-5}
H.	Reactor coolant concentration ($\mu Ci/g$)	0.2

I. Discharge rate of reactor coolant from break:

Time after break (hr)	Discharge rate (lbm/hr)
0 – 0.5	96,000
0.5 – 1.0	87,000
1.0 – 1.5	69,000
1.5 – 2.0	53,000
2.0 – 2.5	37,000
2.5 – 3.0	25,000
3.0 – 3.5	14,000
3.5 – 4.0	9,000

J. No credit for standby gas treatment system filtration

K. Reactor water cleanup system continues to function during the accident

L. No cleanup from condensate demineralizer following break

Note: The reactor coolant mass discharged from the break and the flash fraction in this table envelop the releases for extended power uprate.

Table 15.6-2

RADIOLOGICAL CONSEQUENCES OF THE INSTRUMENT LINE BREAK OUTSIDE CONTAINMENT

	Thyroid Dose (rem)	Whole Body Dose (rem)
0 – 2 hr, EAB	128	0.02
0 – 4 hr, LPZ	9	0.002

Note: Since the reactor coolant releases in Table 15.6-2 are enveloping and the released coolant activity is at the Technical Specification level (with iodine spiking factor of 500), the doses in this table are valid for extended power uprate.

Table 15.6-3

EFFECT OF MAIN STEAM LINE ISOLATION VALVE CLOSURE TIME

Steam Line Isolation Valve Closure Time[1]	Net Mass of Water and Steam Lost from Pressure Vessel (lb)	
	With Feedwater	Without Feedwater
3.5 seconds	3,000	13,000
10.5 seconds	37,000	66,000 (for rated steam flowrate)[2]
		76,200 (for increased steam flowrate)[3]

Notes:

1. Includes 0.5-second detection time.
2. The net mass lost is comprised of 21,000 lbs. of steam and 45,000 lbs. of water.
3. The net mass lost is comprised of 17,000 lbs. of steam and 59,200 lbs. of water.

Table 15.6-4

MSLB – Radiological Consequences
Key Inputs, Assumptions and Radiological Doses

Table 15-6-4a: Key MSLB Accident Analysis Inputs and Assumptions

Input/Assumption	Value
Mass Release	140,000 lb_m of reactor coolant
Pre-Accident Spike Iodine Concentration	4 μCi/gm I-131 equivalent
Maximum Equilibrium Iodine Concentration	0.2 μCi/gm I-131 equivalent
Transport model for Control Room	Steam cloud moves past the Control Room intake at 1 m/sec
Control Room Filtration	No Credit Taken

.

Table 15.6-4b:

Offsite χ/Q (sec/m^3) Values for the MSLB Releases

Time Period	EAB χ/Q (sec/m3)	LPZ χ/Q (sec/m^3)
0 - 2 hrs	4.40E-4 [1]	5.5E-5 [1]

Notes:
Based on Regulatory Guide 1.5 methodology with Pasquill F atmospheric conditions and 1 meter/second wind speed.

Table 15.6-4c:

MSLB Accident Radiological Consequence Analysis

E.	**F.**	**4.0 μCi/gm Dose Equivalent I-131 TEDE (rem)**	**0.2 μCi/gm Dose Equivalent I-131 TEDE (rem)**	**Regulatory Limit TEDE (rem)**
Control Room	30-day integrated dose	3.77	1.89E-1	5
EAB	Worst 2-hour integrated dose	1.70	8.48E-2	25 (4.0 μCi/gm) 2.5 (0.2 μCi/gm)
LPZ	30-day integrated dose	2.12E-1	1.06E-2	25 (4.0 μCi/gm) 2.5 (0.2 μCi/gm)

Table 15.6-5

POST-LOCA PRIMARY CONTAINMENT AIRBORNE FISSION PRODUCT INVENTORY

This table is maintained for historical information only.

Time After Accident	Noble Gases (Ci)	Halogens (Ci)
30 minutes	1.6×10^6	8.1×10^4
1 hour	1.4×10^6	7.7×10^4
3 hours	1.3×10^6	6.7×10^4
10 hours	1.1×10^6	4.2×10^4
1 day	8.3×10^5	3.1×10^4
3 days	6.8×10^5	2.6×10^4
10 days	2.2×10^5	6.5×10^3
25 days	3.4×10^4	1.3×10^3

Table 15.6-6

POST-LOCA REACTOR BUILDING AIRBORNE FISSION PRODUCT INVENTORY

This table (15.6-6) is maintained for historical information only. The bases for this table are described in Section 15.6.5.4.3.

Time After Accident	Noble Gases (Ci)	Halogens (Ci)
30 minutes	1.4×10^{2}	7.0×10^{0}
1 hour	2.3×10^{2}	1.3×10^{1}
3 hours	6.3×10^{2}	3.2×10^{1}
10 hours	1.4×10^{3}	5.7×10^{1}
1 day	1.9×10^{3}	7.1×10^{1}
3 days	1.8×10^{3}	6.6×10^{1}
10 days	3.1×10^{0}	9.0×10^{-2}
25 days	less than 10^{-10}	8.0×10^{-9}

Table 15.6-7

POST-LOCA DISCHARGE RATES TO CHIMNEY

This table (15.6-7) is maintained for historical information only. The bases for this table are described in Section 15.6.5.4.4.

Time After Accident	Noble Gases (Ci/s)	Halogens (Ci/s)
30 minutes	1.5×10^{-3}	8.1×10^{-7}
1 hour	2.7×10^{-3}	1.5×10^{-6}
3 hours	7.3×10^{-3}	3.7×10^{-6}
10 hours	1.7×10^{-2}	6.6×10^{-6}
1 day	2.2×10^{-2}	8.2×10^{-6}
3 days	2.1×10^{-2}	7.8×10^{-6}
10 days	3.7×10^{-5}	1.1×10^{-8}
25 days	less than 10^{-12}	less than 10^{-14}

Table 15.6-8
RADIOLOGICAL EFFECTS OF THE LOSS-OF-COOLANT ACCIDENT

This table is retained for historical information only

	First 2-Hour Dose						Total Accident Dose					
Distance (mi)	VS-2	MS-2	N-2	N-10	U-2	U-10	VS-2	MS-2	N-2	N-10	U-2	U-10
Whole Body Passing Cloud Dose (rem)												
½	1.7×10^{-5}	1.7×10^{5}	1.0×10^{-5}	2.8×10^{-6}	2.2×10^{-5}	3.2×10^{-6}	8.5×10^{-4}	9.0×10^{-4}	9.0×10^{-4}	1.4×10^{-4}	1.1×10^{-3}	1.6×10^{-4}
1	1.2×10^{-5}	1.2×10^{-5}	1.4×10^{-5}	2.1×10^{-6}	1.0×10^{-5}	1.0×10^{-6}	6.0×10^{-4}	6.0×10^{-4}	7.0×10^{-4}	1.0×10^{-4}	5.0×10^{-4}	8.5×10^{-5}
6				4.1×10^{-7}		1.8×10^{-7}	1.6×10^{-4}	1.8×10^{-4}	1.1×10^{-4}	2.0×10^{-5}	3.8×10^{-5}	9.0×10^{-6}
9	(1)			1.8×10^{-7}		7.8×10^{-8}	9.0×10^{-5}	9.5×10^{-5}	3.8×10^{-5}	9.0×10^{-5}	1.2×10^{-5}	3.6×10^{-6}
12				1.2×10^{-7}		4.7×10^{-8}	6.5×10^{-5}	9.5×10^{-5}	2.2×10^{-5}	6.0×10^{-6}	6.5×10^{-6}	2.3×10^{-6}
Lifetime Thyroid Dose (rem)												
½	a(2)	a	1.0×10^{-6}	2.5×10^{-8}	5.0×10^{-6}	5.5×10^{-7}	a	3.1×10^{-8}	7.5×10^{-6}	1.8×10^{-6}	3.7×10^{-4}	4.0×10^{-5}
1	a	2.2×10^{-5}	2.4×10^{-6}	2.3×10^{-7}	2.1×10^{-6}	2.8×10^{-7}	a	1.6×10^{-6}	1.8×10^{-4}	1.6×10^{-5}	1.5×10^{-4}	4.1×10^{-5}
6				7.5×10^{-8}		2.9×10^{-8}	a	3.6×10^{-5}	2.9×10^{-5}	5.5×10^{-6}	1.2×10^{-5}	2.1×10^{-6}
9				3.4×10^{-8}		7.2×10^{-8}	8.5×10^{-9}	3.5×10^{-5}	1.2×10^{-5}	2.4×10^{-6}	4.7×10^{-6}	9.0×10^{-7}
12				2.2×10^{-8}		8.0×10^{-9}	7.5×10^{-8}	3.1×10^{-5}	8.0×10^{-6}	1.6×10^{-6}	3.1×10^{-6}	6.0×10^{-7}
Whole Body Fallout Dose (rem)												
½	a	a	9.5×10^{-10}	1.2×10^{-10}	1.1×10^{-8}	6.0×10^{-9}	a	a	1.6×10^{-7}	2.1×10^{-8}	1.9×10^{-6}	1.0×10^{-6}
1	a	a	2.4×10^{-9}	1.1×10^{-9}	4.6×10^{-9}	3.1×10^{-9}	a	2.1×10^{-9}	4.0×10^{-7}	1.9×10^{-7}	8.0×10^{-7}	5.5×10^{-7}
6				3.7×10^{-10}		3.2×10^{-10}	a	4.7×10^{-8}	6.5×10^{-8}	6.5×10^{-8}	6.0×10^{-8}	5.5×10^{-8}
9				1.6×10^{-10}		1.3×10^{-10}	a	4.6×10^{-8}	2.8×10^{-8}	2.8×10^{-8}	2.4×10^{-8}	3.3×10^{-8}
12				1.1×10^{-10}		a	a	4.0×10^{-8}	1.6×10^{-8}	1.8×10^{-8}	1.6×10^{-8}	1.5×10^{-8}

(Sheet 1 of 2)

Table 15.6-8 (Continued)

OFFSITE RADIOLOGICAL EFFECTS OF THE LOSS-OF-COOLANT ACCIDENT

	First 2-Hour Dose						Total Accident Dose					
Distance (mi)	VS-2	MS-2	N-2	N-10	U-2	U-10	VS-2	MS-2	N-2	N-10	U-2	U-10
Whole Body Fallout (Washout) Dose (rem)												
½		4.0×10^{-8}						1.5×10^{-5}				
1		3.4×10^{-8}						6.0×10^{-6}				
6								6.0×10^{-7}				
9								2.4×10^{-7}				
12								1.5×10^{-7}				

LEGEND

	Meteorology	Wind Speed (mph)
VS-2	Very stable	2
MS-2	Moderately stable	2
N-2	Neutral	2
N-10	Neutral	10
U-2	Unstable	2
U-10	Unstable	10

Notes:
1. First 2-hour dose is zero since time of cloud travel is greater than 2 hours.
2. The symbol "a" means less than 1×10^{-10}.

Table 15.6-9

Table 15.6-9 Personnel Dose Inputs	
Input/Assumption	**Value**
Onsite Breathing Rate	3.47E-04 m^3/sec
Offsite Breathing Rate	0-8 hours: 3.47E-04 m^3/sec 8-24 hours: 1.75E-04 m^3/sec 1-30 days: 2.32E-04 m^3/sec
Control Room Occupancy Factors	0-1 day: 1.0 1-4 days: 0.6 4-30 days: 0.4

Table 15.6-10

<table>
<tr><th colspan="2">Table 15.6-10 Key Analysis Inputs and Assumptions</th></tr>
<tr><th colspan="2">Release Inputs - LOCA Radionuclide Source Term</th></tr>
<tr><th>Input/Assumption</th><th>Value</th></tr>
<tr><td>Core Fission Product Inventory</td><td>ORIGEN-2*
Only the 60 nuclides considered by RADTRAD are utilized in the analysis</td></tr>
<tr><td>Core Power Level</td><td>3016 MWt</td></tr>
<tr><td>Core Burnup</td><td>39 GWD/MTU core average</td></tr>
<tr><td>Fission Product Release Fractions for LOCA</td><td>RG 1.183, Table 1
BWR Core Inventory Fraction Released Into Containment
<table>
<tr><th>Group</th><th>Gap Release Phase</th><th>Early In-vessel Phase</th><th>Total</th></tr>
<tr><td>Noble Gases</td><td>0.05</td><td>0.95</td><td>1.0</td></tr>
<tr><td>Halogens</td><td>0.05</td><td>0.25</td><td>0.3</td></tr>
<tr><td>Alkali Metals</td><td>0.05</td><td>0.20</td><td>0.25</td></tr>
<tr><td>Tellurium Metals</td><td>0.00</td><td>0.05</td><td>0.05</td></tr>
<tr><td>Ba, Sr</td><td>0.00</td><td>0.02</td><td>0.02</td></tr>
<tr><td>Noble Metals</td><td>0.00</td><td>0.0025</td><td>0.0025</td></tr>
<tr><td>Cerium Group</td><td>0.00</td><td>0.0005</td><td>0.0005</td></tr>
<tr><td>Lanthanides</td><td>0.00</td><td>0.0002</td><td>0.0002</td></tr>
</table></td></tr>
<tr><td>Fission Product Release Timing
(Per RG 1.183, the release phases are modeled sequentially)</td><td>RG 1.183, Table 4
LOCA Release Phases
<table>
<tr><th></th><th colspan="2">BWRs</th></tr>
<tr><th>Phase</th><th>Onset</th><th>Duration</th></tr>
<tr><td>Gap Release</td><td>2 min</td><td>0.5 hr</td></tr>
<tr><td>Early In-Vessel</td><td>0.5 hr</td><td>1.5 hr</td></tr>
</table></td></tr>
</table>

* ORIGEN-S was used to calculate the core inventory for AREVA ATRIUM.

Table 15.6-11 Key LOCA Analysis Inputs and Assumptions	
Release Inputs - Primary and Secondary Containment Parameters	
Input/Assumption	**Value**
Drywell Free Volume	1.58E+05 cubic feet
Surface Area in Drywell	32,250 square feet
Suppression Pool Water Volume	110,000 cubic feet
Primary Containment Total Leak Rate (includes MSIV leakage)	3.0% per day for the duration of the accident
Total MSIV leak rate @ 43.9 psig containment pressure	150 scfh for the duration of the accident (60 scfh max per line)
Secondary Containment Volume	4.5E+06 cubic feet
Fraction of Secondary Containment Available for Mixing	0.5
SGT System Flow Rate (with 10% margin)	4400 cfm
Secondary Containment Drawdown Time	0
Secondary Containment Bypass	0
ESF Systems Leak Rate Outside of Primary Containment (includes factor of 2 margin)	2 gpm
Release Location ESF/Containment Leakage MSIV Leakage	 Station Chimney (elevated release) MSIV Room (ground release)
Release Duration ESF/Containment Leakage MSIV Leakage	 0-30 days 0 minutes to 30 days

Table 15.6-12

Table 15.6-12 Key LOCA Analysis Inputs and Assumptions	
Transport Inputs - Control Room Parameters	
Input/Assumption	Value
Nuclide Release Locations	See Figure 1
CREV System Initiation (manual)	40 minutes after LOCA initiation
Control Room Free Volume	81,000 cubic feet
CREV System Air Intake Flow Rate (normal and accident)	2000 cfm +/- 10%
Control Room Unfiltered Inleakage Rate	60,000 cfm during normal operation 395 cfm during CREV operation

Table 15.6-13

Table 15.6-13 Key LOCA Analysis Inputs and Assumptions	
Removal Inputs	
Input/Assumption	Value
Aerosol Natural Deposition Coefficients Used in the Drywell	Credit is taken for natural deposition of aerosols based on equations for the Power's model in NUREG/CR 6189 and input directly into RADTRAD as natural deposition time dependent lambdas.
Elemental Iodine Removal in the Drywell	Based on Standard Review Plan 6.5.2 methodology, credit is taken until a DF of 200 is reached.
Main Steam Lines Deposition	Two-node treatment, each well-mixed, is used for each steam line in which flow occurs. The first node is from the reactor vessel to the inboard MSIV. The second node is from the inboard MSIV to the outboard MSIV. Gravitational settling applied to aerosols on horizontal pipe projected areas. For Elemental Iodine Deposition, a DF of 2 or elemental removal efficiency of 50% is used per AEB 98-03, Appendix B.
Main Steam Line and Condenser Holdup Credit for MSIV Leakage	No credit is taken for holdup or plate-out in the main steam lines beyond the outboard MSIVs. No credit is taken for holdup and plate-out in the main condenser.
SGT System Filter Efficiency	HEPA: Particulate aerosol 98% Charcoal: Elemental and organic iodine 80%
CREV System Filter Efficiency	HEPA: Particulate aerosol 99% Charcoal: Elemental and organic iodine 99%

Table 15.6-14

Table 15.6-14 LOCA Radiological Doses[1]			
Location	Duration	TEDE (rem)	Regulatory Limit TEDE (rem)
Control Room	30 days	4.83	5
EAB	Maximum, 2 hours	1.64	25
LPZ	30 days	0.781	25

1 Radiological doses are based on the Westinghouse Optima2 core inventory described in Section 15.6.5.5.3.1. These doses are also bounding for the AREVA ATRIUM 10XM fuel design.

2 The doses here include the external cloud shine, control room filter shine, and inhalation doses from radioactivity drawn into the control room.

Table 15.6-15

15.6-15 Control Room χ/Q (sec/m^3) Values for the Different Release and Intake Combinations[1, 2]		
Time Period	**LOCA Chimney**	**LOCA MSIV**
0 - 2 hrs	6.42E-06	1.30E-03
2 - 8 hrs	2.87E-06	1.06E-03
8 - 24 hrs	1.92E-06	4.49E-04
1 - 4 d	8.03E-07	2.96E-04
4 - 30 d	2.29E-07	2.44E-04

Notes:

1. Ground level χ/Q values are based on ARCON 96; elevated release χ/Q values are based on Regulatory Guide 1.145 methodology.
2. Control room intake χ/Q values are applicable for control room inleakage.

Table 15.6-16a

Table 15.6-16a Elevated Release χ/Q (sec/m^3) Values Using RG 1.145 Methodology for the EAB and LPZ		
Time Period	**EAB χ/Q (sec/m^3)**	**LPZ χ/Q (sec/m^3)**
0 - 0.5 hrs	8.74E-5	-
0.5 - 720 hrs	6.74E-6	-
0 - 0.5 hrs	-	1.55E-5
0.5 - 2 hrs	-	8.30E-6
2 - 8 hrs	-	3.57E-6
8 - 24 hrs	-	2.34E-6
1 - 4 d	-	9.39E-7
4 - 30 d	-	2.53E-7

Table 15.6-16b

Table 15.6-16b: Ground Level Release χ/Q (sec/m^3) Values Using RG 1.145 Methodology for the EAB and LPZ		
Time Period	**EAB χ/Q (sec/m^3)**	**LPZ χ/Q (sec/m^3)**
0 - 2 hrs	2.51E-4	-
0 - 2 hrs	-	2.63E-5
2 - 8 hrs	-	1.09E-5
8 - 24 hrs	-	7.02E-6
1 - 4 d	-	2.70E-6
4 - 30 d	-	6.86E-7

Table 15.6-17

Suppression Pool pH Results
(SLC System Sodium Pentaborate Inventory = 3769.4 lb_m)

Suppression Pool pH Results (SLC System Sodium Pentaborate Inventory = 3769.4 lb_m)	
Time	**pH**
0-24 hrs; All Sodium Pentaborate Is in Suppression Pool	-
30 days	8.15

15.7 RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT

The events described in Sections 15.7.1, 15.7.2, and 15.7.4 are not reanalyzed for the current fuel cycle because they continue to be bounded by generic analyses or analyses for previous fuel cycles. These events, including the associated assumptions and conclusions, continue to be part of the plant's licensing basis. The conclusions of these analyses are still valid; however, specific details contained in the descriptions and associated figures should be used only to understand the analysis and its conclusions. These specific details should not be used as sources of current fuel cycle design information.

15.7.1 Radioactive Gas Waste System Leak or Failure

The accident analysis is based upon the isotope inventories of the off-gas treatment system equipment as listed in Table 15.7-1. The isotope inventories in the off-gas system are based upon the following parameters:

A. Reactor rated at 2527 MWt;

B. 18.5 standard ft^3/min air inleakage;

C. 100,000 μCi/s diffusion gas mixture after a 30-minute delay; and

D. 12 activated carbon beds - 74,000 pounds of activated carbon.

The analysis assumes the following equipment characteristics with respect to the retention of daughter products prior to the failure of the off-gas equipment:

A. Off-gas condenser - 100% but washed out;

B. Water separator - 100% but washed out;

C. Holdup pipe - 60% but washed out;

D. Prefilter - 100%;

E. Activated carbon beds - 100%; and

F. Postfilter - 100%.

These assumptions generally give conservative daughter inventories or do not have a significant effect on daughter inventories. For example, 100% washout in the off-gas condenser removes daughter products from the prefilter, but this represents less than 1 minute of delay, compared to 6 hours of delay experienced in the holdup pipe when the recombiner is in operation. Washout of 60% in the holdup pipe is conservative compared to 60 - 99% that has been measured in the EVESR facility at Vallecitos.

The iodine inventories of Table 15.7-1 are based upon the iodine activities measured at Dresden 2 in the reactor water, at the condensate pump discharge, and off-gas after being discharged from the 30-minute holdup. The iodine

inventories are also based upon standard plant iodine source term at 100,000 μCi/s diffusion gas mixture after 30-minute decay. The off-gas treatment system has a 6-hour holdup and several other features that are expected to reduce the amount of iodine reaching the prefilter and charcoal.

Additionally, the following assumptions are used with respect to equipment failures:

A. Activated carbon beds - Activated carbon beds are contained in activated carbon vessels (4 feet in diameter by 21 feet tall, dished heads, and 350 psig design pressure). The vessels are contained in a single vault which is not accessible during operation because of the radioactivity level. Therefore, no failure of the vessels due to operator error can be postulated; the only postulated vessel failure that could result in a loss of activated carbon would be the failure of the concrete structure surrounding the vessel, allowing the vessel to fall out of the vertical position. A circumferential failure of the vessel could result from concrete falling on the vessel under one of two conditions:

   1. Bending load - the vessel being supported in the center and loaded on each end. This could possibly result in a tear around 50% of the circumference.

   2. Shearing load - the vessel being supported at the bottom and loaded above near the same point.

   In either case, no more than 10 - 15% of the activated carbon would be displaced from the vessel. Iodine is strongly bonded to the activated carbon and is not expected to be removed by exposure to the air. A 1% loss of iodine is a conservative estimate.

   Measurements made at Kernkraftwerk RWE-Boyenwerk GmbH (KRB) indicate that off-gas is about 30% richer in krypton than air is. Therefore, if this carbon is exposed to air, it would eventually reach equilibrium with the noble gases in the air. However, the first few inches of activated carbon would blanket the underlying activated carbon from the air.

   A 10% loss of noble gas from a failed vessel is conservative because the fraction of activated carbon exposed to the air is small.

   The activated carbon adsorbers operate essentially at room temperature and are designed to limit the temperature of the activated carbon to well below its charcoal ignition temperature, thus precluding overheating or fire and consequent escape of radioactive materials. The adsorbers are located in the shielded room, maintained at a constant temperature by an air conditioning system that removes the decay heat generated in the adsorbers. The maximum centerline temperature of the activated carbon is less than 10°F above room temperature when the flow is stopped. The decay heat of 50 Btu/hr is sufficiently small compared to the thermal mass of the activated carbon vault so that even if the vault cooling is lost, the temperature rise would not be sufficient to cause activated carbon ignition.

The activated carbon is maintained at 77°F by the vault air conditioning system. Due to the thermal inertia of the steel activated carbon vessels and the massive concrete vault walls, temperature changes caused by failure of the vault air conditioning system would be sufficiently slow so that the resulting changes in activated carbon adsorption coefficient would not result in a rapid release of adsorbed radioactivity. In order to maintain consistent operation of the system, a redundant vault air conditioning system is supplied to allow for maintenance and operational convenience. During a plant outage, when the condenser is not maintained at vacuum, there is no gas flow in the activated carbon and holdup is very high even if the activated carbon reaches ambient temperature.

B. Prefilter - Due to the short length of the vessel, heavy wall thickness due to the design pressure, and collapsible nature of the filter media of the prefilter vessel (24 inches in diameter, 4 feet high, and 350 psig design pressure), a failure mechanism that would result in emission of filter media or daughter products from this vessel cannot be postulated. A 1% release is used to illustrate the consequences of loss from this vessel.

C. Holdup pipe - Pipe rupture and depressurization of the pipe is considered. The pipe normally operates at less than 15.6 psia and depressurizes to 14.7 psia. The loss is conservatively taken as assumed plateout or washout of 60% in calculating the holdup pipe inventory.

To provide an estimate of hypothetical radiological doses from off-gas system equipment failures, certain percentages of the activity contained in the most significant off-gas system components are assumed to be released to the environment under very stable 1 m/s meteorological conditions with an effective release height of zero meters. Percentages of primary activities released from components and the corresponding estimated radiological exposures are presented in Table 15.7-2 and are compared to the limits in 10 CFR 20 and 10 CFR 100.

This accident analysis indicates there is no undue hazard to the health and safety of the public resulting from installation and operation of the off-gas treatment system. A radiological evaluation has confirmed that with off-gas input radioactivity increased to 350, 000µ(i)s, failure of equipment in the off-gas treatment system would result in only a fraction of the offsite doses presently permitted.

15.7.2 Postulated Liquid Releases Due to Liquid Tank Failures

Tanks at grade referenced in Section 11.2 may contain 3 curies of activity maximum. Presuming seismic damage to all tanks and structures which could contain these wastes, the wastes could flow to the nearby discharge structure and then to the river. River flow varies from at least 3000 ft^3/s (98% of the time) to at least 6000 ft^3/s (48% of the time). These values are used as the assumed maximum and minimum flowrates for river concentration calculations. Assuming entry into the river over a period of 1 hour, activity concentration in the river could vary from 1×10^{-5} µCi/cc maximum to 5×10^{-6} µCi/cc minimum.

The Effluent Concentration Limit (ECL) for an unidentified mixture is 10^{-6} μCi/cc. Thus, in the worst case of all tanks being full, at maximum activity concentration, and failing simultaneously, river concentration over an hour would be greater than the ECL for the mixture. However since the release period and subsequent human intake are for periods much shorter than a year, the doses attendant with maximum releases expected for the seismic event can be reduced by the ratio of the release period to a year. The 10 CFR 20 ECL for an unidentified mixture of 1 x 10^{-6} μCi/cc is equivalent to 50 mrem/yr dose limit assuming a human intake of 2200 ml/day for a year.

$$\text{dose potential} = \frac{(1 \times 10^{-5}\ \mu\text{Ci/cc})\,(1\ \text{hour})}{(1 \times 10^{-6}\ \mu\text{Ci/cc})\,(8760\ \text{hr/yr})}$$
$$= 0.0011 \text{ of yearly dose rate}$$

The pool of the Dresden dam would provide some additional volume for mixing with the river water. Tritium content at a maximum could be 4 curies to give a maximum river concentration of 1.3 x 10^{-5} μCi/cc. This would still be 2 orders of magnitude below its ECL of 1 x 10^{-3} μCi/cc.

Therefore the actual maximum dose expected from this event would be 2 orders of magnitude less than the 50 mrem/yr allowed by 10 CFR 20 even with no consideration for downstream dilution factors, decay times, or unidentified versus identified mixtures.

Thus, the dose associated with the postulated seismic events described above is not considered to be at a level to cause concern.

The remaining waste tankage is located below grade in the radwaste building. The building is a concrete structure located in rock. Although seismic occurrences may cause tank damage, radwaste liquids will accumulate in the basement of the radwaste building. Escape of the radwaste liquids from the basement would have to be through cracks and fissures in concrete walls and surrounding rock. Passage through the rock would be accompanied by some degree of filtration and ion exchange. Maximum activity possibly present in basement tanks is 530 curies. If the liquids escaped, they would take a substantial period of time to get through the rock. Assuming this time to be 100 hours would make the consequence similar to the case above.

In both of the above cases, river activity would accumulate in a discrete time interval, very short when compared to the year interval upon which 10 CFR 20 ECL limits are based.

Further it may be pointed out that, although the activities released to the river would result from a tank failure condition indicating applicability of 10 CFR 100 limits, the actual expected concentrations are sufficiently low to allow 10 CFR 20 limits to be applied. This total failure analysis also eliminates the requirement for a retention sump/curbing around the radwaste tanks.

Radiological consequences due to equipment failures and operator errors, as described in Section 11.2, are minimal and certainly less severe than those due to the seismic damage.

15.7.3 Design Basis Fuel Handling Accidents During Refueling

See the introduction to Section 15.7 for information regarding use of details from this analysis description which may not be applicable to the current fuel cycle.

15.7.3.1 Identification

During a refueling operation the primary containment (drywell-suppression chamber) and the reactor vessel are open; the secondary containment (reactor building) serves as the major barrier to the release of radioactive materials. The accident is assumed to occur when a fuel assembly is accidentally dropped onto the top of the core during fuel handling operations.

15.7.3.2 Designed Safeguards

The reactor core is designed to remain subcritical with one control rod fully withdrawn and all other control rods fully inserted, even if it is assumed that a fresh fuel assembly is dropped into an empty fuel space in an otherwise fully constituted core. At least two control rods adjacent to the empty fuel space would have to be withdrawn for a nuclear excursion to occur.

With the reactor mode switch in STARTUP, a rod withdrawal interlock prevents any withdrawal whenever the travel limit switch indicates that the platform is over the reactor core.

With the reactor mode switch in REFUEL or STARTUP, a rod withdrawal interlock prevents any withdrawal whenever the travel limit switch in combination with the hoist load switch on the refueling platform indicates that the platform is carrying fuel over the reactor core.

With the reactor mode switch in REFUEL, a rod withdrawal interlock prevents the withdrawal of more than one control rod. When any one rod position indicator shows that a rod is withdrawn from the fully inserted position, the interlock prevents the withdrawal of any other rod.

When any rod position indicator shows a rod is withdrawn, an interlock prevents the movement of the refueling platform toward a position over the reactor core while the hoists are carrying fuel.

Each fuel hoist is equipped with a load limit switch and two independent travel limit switches to prevent damage due to upward movement. To drop the fuel assembly, either: (1) the assembly bail, the fuel grapple, or the grapple cable would have to break, (2) the grapple opens due to malfunction or (3) the bundle was never fully latched and the friction force holding the bundle is overcome by gravity. Section 9.1.4 provides additional details regarding refueling platform controls and interlocks which would prevent the occurrence of such an event

15.7.3.3 Procedural Safeguards

Procedures require the reactor control operator to observe rod position instrumentation and to be in communication with the refueling operator during all fuel loading operations. The reactor is verified to be subcritical by observing the source range monitor (SRM) count rate which is maintained at greater than 3 cps in accordance with the Technical Specifications, when fuel is being moved in the core except when two or fewer bundles are located in each of the four quadrants.

15.7.3.4 Accident Analysis

The description in this section is for a reactor core and fuel assemblies prior to extended power uprate (EPU). The impact of EPU is discussed at the end of this section.

Dropping a fuel assembly onto the reactor core is assumed to occur under nonoperating conditions for a 8x8, 9x9, or 10x10 fuel array. The key assumption of this postulated occurrence is the inadvertent mechanical damage to the fuel rod cladding as a consequence of the fuel assembly being dropped on the core while in the cold condition. Therefore, fuel densification considerations do not affect the accident analysis results.

15.7.3.4.1 Methods, Assumptions, and Conditions

The assumptions and analyses applicable to this type of fuel handling accident are as follows:

A. The fuel assembly is dropped 34 feet to impact the core (from the maximum height allowed by the fuel handling equipment).

B. The entire amount of potential energy, including the energy of the entire fuel assembly falling to its side from a vertical position (referenced to the top of the reactor core), is available for application to the fuel assemblies involved in the accident. This assumption neglects the dissipation of some of the mechanical energy of the falling fuel assembly in the water above the core and requires that the assembly separates from the grapple head.

C. None of the energy associated with the dropped fuel assembly is absorbed by the fuel material (uranium dioxide).

D. All fuel rods, including tie rods, are assumed to fail by 1% strain in compression, the same mode as ordinary fuel rods. For the fuel designs considered here, there is no propensity for preferential failure of tie rods.

E. As stated above, the fuel handling accident (FHA) evaluates the consequences of the inadvertent dropping of a fuel assembly onto other fuel assemblies already loaded in the core. However, the current regulatory guidance, Standard Review Plan (SRP) 15.7.4 (Reference 11), assumes the dropped mass to include the weight of the refueling mast. As such General Electric performs the FHA evaluation with a 619 lb. mast, the approximate weight of the NF-500 (Reference 7).

F. It is assumed that 50% of the energy is absorbed by the dropped fuel assembly and that the remaining 50% is absorbed by the struck fuel assemblies in the core during each impact.

G. It is assumed that the refueling mast adds kinetic energy (when included in the analysis), and dissipates all of this energy during impact. Upon impact, half the energy is absorbed by the dropped fuel assembly.

H. The current GE analysis for the Fuel Handling Accident (FHA) as found in GESTAR (Reference 7) addresses the consequences of a Fuel Handling Accident for GE 7x7, 8x8, 9x9, and 10x10 fuel arrays. As described in GESTAR, the radiological consequences based upon the 172 failed rods associated with the GE14 fuel design (10x10 array) bound the consequences for all GE non-7x7 array configurations. The Alternative Source Term analysis conservatively assumes that the fission product release source term would be higher with the failure of 111 fuel rods with 7x7 fuel array than with the GE-14 10x10 fuel array with172 damaged fuel rods. The fission product inventory in the fuel rod gap of the damaged 111 fuel rods is assumed to be instantaneously released because of the FHA. The impact of the transition to Westinghouse SVEA-96 Optima2 fuel on the fuel handling accident consequences has been determined in accordance with the methodology described in Reference 18. The analysis for the Westinghouse SVEA-96 Optima2 fuel determined that a total of 116 fuel rods would be damaged – 96 rods in the dropped assembly and 20 rods in the impacted assembly. Since the number of failed fuel rods is significantly fewer than estimated for the GE14 design and the rods that fail do not increase the release, the radiological consequences developed based upon the 7x7 fuel design key input and assumptions will be bounding for the Westinghouse SVEA-96 Optima 2 fuel design. For the transition to the ATRIUM 10XM fuel design for Unit 3, AREVA determined that a total of 162 rods would fail for the postulated fuel handling accident. This number of failed rods is less than the 172 rod failures associated with the GE-14 10x10 FHA analysis which itself was bounded by the 111 GE 7x7 rod failures assumed by the dose analysis for the accident. The AREVA core inventory for the ATRIUM 10XM fuel design have been evaluated, with a conservative assumption that 179 fuel rods in the ATRIUM 10XM bundle fail, and the evaluation confirmed that this fuel design is bounded by GE-14 dose consequences, Reference 34.

I. Dresden Unit 1 fuel is assumed to be bounded by the Dresden Unit 2/3 fuel due to the significant weight difference and the lower exposures associated with Dresden Unit 1 fuel. Unit 1 fuel will never be loaded into Units 2 or 3, but the FHA over the core is the bounding event for a bundle drop in the spent fuel pool. Since Unit 1 fuel is stored in the Unit 2/3 pools, Unit 1 fuel must be addressed as above.

15.7.3.4.2 Radiological Consequences for the FHA

Regulation 10 CFR 50.67, "Accident Source Term," provides a mechanism for power reactor licensees to voluntarily replace the traditional TID 14844 (Ref. 19) accident source term used in design-basis accident analyses with an “Alternative Source Term” (AST). The methodology of approach to this replacement is given in USNRC Regulatory Guide 1.183 (Ref. 20) and its associated Standard Review Plan 15.0.1 (Ref. 21).

Accordingly, Dresden Nuclear Power Station (DNPS) Units 2 and 3, has applied the AST methodology for several areas of operational relief in the event of a Design Basis Accident (DBA), without crediting the use of previously assumed safety systems. Amongst these systems are the Control Room Emergency Filter (CREF) and the Standby Gas Treatment System (SGTS).

In support of a full-scope implementation of AST as described in and in accordance with the guidance of Ref. 20, AST radiological consequence analyses are performed for the DBA that result in offsite exposure for a Fuel Handling Accident (FHA).

Implementation consisted of the following steps:

- Identification of the AST based on plant-specific analysis of core fission product inventory,
- Calculation of the release fractions for the DBA that could potentially result in control room and offsite doses (FHA),
- Analysis of the atmospheric dispersion for the radiological propagation pathways,
- Calculation of fission product deposition rates and removal mechanisms,
- Calculation of offsite and control room personnel Total Effective Dose Equivalent (TEDE) doses.

15.7.3.4.2.1 Regulatory Guide 1.183 Compliance

The analyses are prepared in accordance with the guidance provided by Regulatory Guide 1.183 (Ref. 20).

15.7.3.4.2.1.1 Dose Acceptance Criteria

The AST acceptance criteria for Control Room dose for postulated major credible accident scenarios such as those resulting in substantial meltdown of the core with release of appreciable quantities of fission products is provided by 10 CFR 50.67, which requires

“Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.”

This limit is applied by Regulatory Guide 1.183 to all of the accidents considered with AST.

The AST acceptance criteria for an individual located at any point on the boundary of the exclusion area (the Exclusion Area Boundary or EAB) are provided by 10 CFR 50.67 as 25 rem TEDE for any 2-hour period following the onset of the postulated fission product release.

The AST acceptance criteria for an individual located at any point on the outer boundary of the low population zone (LPZ) are provided by 10 CFR 50.67 as 25 rem TEDE during the entire period of passage of the radioactive cloud resulting from the postulated fission product release.

These limits are applied by Regulatory Guide 1.183 to events with a higher probability of occurrence for the FHA to provide the following acceptance criteria:

- For the FHA, doses at the EAB and LPZ should not exceed 6.3 rem TEDE for the accident duration (2 hour dose for EAB and 30 day dose for LPZ).

15.7.3.4.2.2 Computer Codes

New AST calculations were prepared for the FHA to simulate the radionuclide release, transport, removal, and dose estimates associated with the postulated accident.

The RADTRAD computer code (Ref. 26) endorsed by the NRC for AST analyses was used in the calculations for analyzing the FHA. The RADTRAD program is a radiological consequence analysis code used to estimate post-accident doses at plant offsite locations and in the control room. The FHA assessment takes no credit for SGTS operation, control room isolation, emergency ventilation or filtration of intake air for the duration of the accident event.

Offsite χ/Qs were calculated with the PAVAN computer code (Ref. 27), using the guidance of Regulatory Guide 1.145 (Ref. 28). Control room χ/Qs were calculated with the ARCON96 computer code (Ref. 29). The PAVAN and ARCON96 codes calculate relative concentrations in plumes from nuclear power plants at offsite locations and control room air intakes, respectively.

All of these computer codes have been used by the NRC staff in its safety reviews.

15.7.3.4.2.3 Source Terms

15.7.3.4.2.3.1 Core Inventory

As with the AST LOCA analyses, the inventory of reactor core fission products for RADTRAD analysis is based on maximum full power operation at a power level of 3016 MWth, the Extended Power Uprate (EPU) thermal power of 2957 MWth plus a 2% instrument error per Reg Guide 1.49 (Ref. 30). The fission products used for the accidents are the 60 isotopes of the standard RADTRAD input library, determined by the code developer as significant in dose consequences. These were extracted from Appendix D of the GE task report No. GE-NE-A22-00103-64-01 (Ref. 31), corresponding to a fuel exposure of 1600 effective full power days (EFPD) and a 24 month fuel cycle. For Unit 3, the post-FHA consequences for the Westinghouse Optima2 fuel were evaluated at 39 GWD/ MTU and the post-FHA consequences of the AREVA ATRIUM 10XM fuel were evaluated using a core average exposure of 39 GWD/MTU and core average enrichment between of 3.9 and 4.5 weight percent U-235, Reference 34.

15.7.3.4.2.3.2 Reactor Coolant Inventory

Not applicable for the FHA

15.7.3.4.2.3.3 Release Fraction

Current design basis accident evaluations as modified by Regulatory Guide 1.183 (Ref. 20) were used to determine the specific releases of radioactive isotopes at the given stages of fuel pin failure and provide these releases as a percentage of the total release for each accident, as summarized in section 15.7.3.4.2.4.2, below.

15.7.3.4.2.4 Methodology

15.7.3.4.2.4.1 Dose Calculations

As per Regulatory Guide 1.183 (Ref. 20), Total Effective Dose Equivalent (TEDE) doses are determined as the sum of the CEDE and the Effective Dose Equivalent (EDE) using dose conversion factors for inhalation CEDE from Federal Guidance Report No. 11 (Ref. 32) and for external exposure EDE from Federal Guidance Report No. 12 (Ref. 33).

15.7.3.4.2.4.2 Fuel Handling Accident (FHA)

Table 15.7-4a lists the key assumptions and inputs used in the analysis. The postulated FHA involves the drop of a fuel assembly on top of the reactor core during refueling operations. The analysis assumes that 111 fuel pins in a 7x7 fuel rod array are damaged. A radial peaking factor of 1.7 was assumed in the analysis, consistent with core operating limit report bases, as suggested in Regulatory Guide 1.183. A post-shutdown 24-hour decay period was used to determine the release activity inventory. This assumption is consistent with plant procedures, but is conservative when compared to plant refueling outage history. The analysis assumes that gap activity in the affected rods was released instantaneously into the water in the reactor well. The analysis assumes the fuel bundle is dropped 34 feet for the fuel damage assessment, but assumes a water depth of only 19 feet above the damaged assemblies for decontamination factor determination using guidance in Regulatory Guide 1.183 and consistent with the limits in the Technical Specifications. The analysis assumes that the activity in the reactor building environment is released within two hours, from the reactor building through the reactor building vent stack, as a zero velocity vent release with no further credit for reactor building holdup or dilution. No credit is taken for the control room emergency ventilation system or standby gas treatment system operation.

The analysis assumptions for the transport, reduction, and release of the radioactive material from the fuel and the reactor coolant are consistent with the guidance provided in Appendix B of Regulatory Guide 1.183.

15.7.3.4.2.4.3 Atmospheric Dispersion Factors (χ/Qs)

Table 15.7-4b lists χ/Q values used for the control room dose assessments, as derived in Section 2.3.5.1 and applied for the release point (reactor building vent stack) applicable to the FHA, for a zero velocity vent release

Table 15.7-4c lists χ/Q values for the EAB and LPZ boundaries, as also derived in Section 2.3.5.1 and applied for the release point (reactor building vent stack) applicable to the FHA, for a zero velocity vent release.

15.7.3.4.2.5 Summary and Conclusions

The radiological consequences of the postulated FHA are given in Table 15.7-4d. As indicated, the control room, EAB, and LPZ calculated doses are within regulatory limits after AST implementation.

15.7.4 Deleted

15.7.5 References

1. Miller, et al., "International Symposium on Fission Product Release and Transport Under Accident Conditions," Paper 12, April 1955, Oak Ridge, Tennessee.

2. Allen, T.L., and Keefer, R.M., "The Formation of Hypoiodus Acid and Hydrated Iodine Cation by the Hydrolysis of Iodine," JACS 77, No. 11, June 1955.

3. Watson, Bancroft and Hoelke, AECL-1130, "Iodine Containment by Dousing in NPD-11," 1960.

4. Safety Evaluation by the Division of Reactor Licensing, U.S. Atomic Energy Commission, August 31, 1966.

5. Safety Evaluation by the Division of Reactor Licensing, U.S. Atomic Energy Commission, October 17, 1969.

6. Safety Evaluation by the Division of Reactor Licensing, U.S. Atomic Energy Commission, November 18, 1970.

7. NEDE-24011-P-A, General Electric Standard Application for Nuclear Fuel, General Electric Company.

8. NFS letter NSF:BPS:83-071, H.E. Bliss to Messers, Onsite and Offsite Reviews of Refueling Accident for 9x9 LTA is at Dresden, date April 19, 1983.

9. Russel, C.R., Reactor Safeguards, The MacMillian Co., New York, New York, 1962.

10. EMF-96-173(P), "Dresden/Quad Cities Fuel Handling Accident Analysis for ATRIUM-9B Fuel," October 1996, NFS NDIT 9600173.

11. USNRC Standard Review Plan 15.7.4, "Radiological Consequences of Fuel Handling Accidents," Revision 1, July 1981.

12. Not used.

13. Regulatory Guide 1.25, Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors, Revision 0, March 23, 1972..

14. TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites, U.S. Atomic Energy Commission Technical Information Document, March 23, 1962.

15. ICRP Publication 2, “Report of Committee II, Permissable Dose for Internal Radiation,” 1959.

16. G. Burley, “Evaluation of Fission Product Release and Transport for Fuel Handling Accident,” Radiological Safety Branch, Division of Reactor Licensing, October 5, 1971.

17. Calculation No. DRE00-0011, Rev. 0, Dresden Units 2 and 3, Effect of Reduced Pool Water Level on FHA Radiological Consequences.

18. “Reference Safety Report for Boiling Water Reactor Reload Fuel,” Westinghouse Topical Report CENPD-300-P-A, July 1996.

19. U. S. Atomic Energy Commission, Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 23, 1962.

20. U. S. Nuclear Regulatory Commission Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

21. U. S. Nuclear Regulatory Commission Standard Review Plan 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000.

22. Not used.

23. Not used.

24. Not used.

25. Not used.

26. RADTRAD Code, "A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," Version 3.03.

27. PAVAN Code, "An Atmospheric Dispersion Program for Evaluating Design Bases Accidental Releases of Radioactive Materials from Nuclear Power Stations."

28. U. S. Nuclear Regulatory Commission Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, November 1982.

29. ARCON96 Code, "Atmospheric Relative Concentrations in Building Wakes".

30. U. S. Nuclear Regulatory Commission Regulatory Guide 1.49, "Power Levels of Nuclear Power Plants," Revision 1, December 1973.

31. GE Task Report No. GE-NE-A22-00103-64-01, Revision 0, Project Task Report: “Dresden and Quad Cities Asset Enhancement Program – Task T0802: Radiation Sources and Fission Products”, August 2000.

32. Federal Guidance Report No. 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion”, 1988.

33. Federal Guidance Report No. 12, “External Exposure to Radionuclides in Air, Water, and Soil”, 1993.

34. DRE02-0036, Revision 3, “Re-analysis of Fuel Handling Accident (FHA) Using Alternative Source Terms.”

Table 15.7-1

AMBIENT OFF-GAS TREATMENT SYSTEM INVENTORY ACTIVITIES (μCi)

	Preheater	Recombiner	Condenser	Separator	Holdup Pipe	Condenser	Moisture Separator	Reheater	Prefilter	Charcoal Vessel Train	First Charcoal Vessel	After-filter
Residence time	0.8 s	0.9 s	50 s	5.1 s	4 hr(1)	178 s	6.5 s	14.5 s	43.5 s	Kr 19.4 hr Xe 14.6 days	Kr 1.6 hr Xe 1.2 days	43.5 s
Operating time	0	0	0	0	0	0	0	0	1 yr	10 yr	10 yr	1 yr
Solid daughter capture	0	0	100%	100%	60%	0	0	0	100%	100%	100%	100%
Solid daughter washout	0	0	100%	100%	100%	0	0	0	0	0	0	0
Isotope												
Kr-83M	2.80×10^{3}	3.50×10^{3}	1.75×10^{5}	1.78×10^{4}	2.61×10^{7}	1.38×10^{5}	5.00×10^{3}	1.11×10^{4}	3.33×10^{4}	7.42×10^{6}	3.35×10^{6}	2.49×10^{1}
Kr-85M	4.56×10^{3}	5.69×10^{3}	2.84×10^{5}	2.90×10^{4}	6.07×10^{7}	5.36×10^{5}	1.95×10^{4}	4.35×10^{4}	1.30×10^{5}	5.51×10^{7}	1.54×10^{7}	6.11×10^{3}
Kr-85	6.54×10^{0}	8.18×10^{0}	4.09×10^{2}	4.17×10^{1}	1.18×10^{5}	1.48×10^{3}	5.40×10^{1}	1.20×10^{2}	3.61×10^{2}	5.87×10^{5}	4.84×10^{4}	3.67×10^{2}
Kr-87	1.63×10^{4}	2.04×10^{4}	1.02×10^{6}	1.03×10^{5}	1.18×10^{8}	3.98×10^{5}	1.43×10^{4}	3.19×10^{4}	9.53×10^{4}	1.44×10^{7}	8.43×10^{6}	2.32×10^{0}
Rb-87	0	0	0	0	0	0	0	0	0	1.99×10^{-3}	1.17×10^{-3}	0
Kr-88	1.53×10^{4}	1.91×10^{4}	9.54×10^{5}	9.71×10^{4}	1.74×10^{8}	1.25×10^{6}	4.54×10^{4}	1.01×10^{5}	3.03×10^{5}	1.00×10^{8}	3.33×10^{7}	2.44×10^{3}

Table 15.7-1 (continued)

AMBIENT OFF-GAS TREATMENT SYSTEM INVENTORY ACTIVITIES (μCi)

	Preheater	Recombiner	Condenser	Separator	Holdup Pipe	Condenser	Moisture Separator	Reheater	Prefilter	Charcoal Vessel Train	First Charcoal Vessel	After-Filter
Rb-88	4.40×10^{0}	1.61×10^{1}	1.64×10^{4}	1.61×10^{2}	9.71×10^{7}	6.02×10^{5}	2.34×10^{4}	5.24×10^{4}	5.87×10^{6}	1.00×10^{8}	3.33×10^{7}	9.86×10^{4}
Kr-89	1.39×10^{5}	1.73×10^{5}	7.91×10^{6}	7.30×10^{5}	3.93×10^{7}	0	0	0	0	0	0	0
Rb-89	4.18×10^{1}	1.69×10^{2}	1.62×10^{5}	1.40×10^{3}	2.36×10^{7}	5.06×10^{1}	1.72×10^{0}	3.81×10^{0}	3.44×10^{2}	0	0	0
Sr-89	0	0	4.43×10^{-1}	0	4.64×10^{4}	4.30×10^{2}	1.57×10^{1}	3.51×10^{1}	1.52×10^{7}	0	0	0
Y-89M	0	0	1.71×10^{-1}	0	4.64×10^{4}	4.30×10^{2}	1.57×10^{1}	3.51×10^{1}	1.52×10^{7}	0	0	0
Kr-90	2.87×10^{5}	3.52×10^{5}	1.08×10^{7}	5.90×10^{5}	5.21×10^{6}	0	0	0	0	0	0	0
Rb-90	4.56×10^{2}	1.84×10^{3}	1.29×10^{6}	6.05×10^{3}	3.13×10^{6}	0	0	0	0	0	0	0
Sr-90	0	0	0	0	3.45×10^{1}	2.80×10^{-1}	1.06×10^{-2}	2.38×10^{-2}	5.07×10^{4}	0	0	0
Y-90	0	0	0	0	7.25×10^{-1}	1.21×10^{-2}	4.45×10^{-4}	9.94×10^{-4}	5.02×10^{4}	0	0	0
Kr-91	1.64×10^{5}	2.16×10^{5}	2.91×10^{6}	2.80×10^{4}	6.61×10^{4}	0	0	0	0	0	0	0
Rb-91	7.12×10^{2}	2.80×10^{3}	1.02×10^{6}	7.16×10^{2}	3.96×10^{4}	0	0	0	0	0	0	0
Sr-91	9.10×10^{-3}	3.87×10^{-2}	4.54×10^{2}	2.52×10^{-2}	9.81×10^{3}	0	0	0	0	0	0	0
Y-91	0	0	0	0	1.00×10^{1}	7.05×10^{1}	2.57×10^{0}	5.74×10^{0}	2.85×10^{6}	0	0	0
Kr-92	1.92×10^{4}	1.96×10^{4}	7.53×10^{4}	5.02×10^{-1}	2.23×10^{-1}	0	0	0	0	0	0	0
Rb-92	1.00×10^{3}	3.51×10^{3}	1.09×10^{5}	1.60×10^{-1}	1.34×10^{-1}	0	0	0	0	0	0	0

Table 15.7-1 (continued)

AMBIENT OFF-GAS TREATMENT SYSTEM INVENTORY ACTIVITIES (μCi)

	Preheater	Recombiner	Condenser	Separator	Holdup Pipe	Condenser	Moisture Separator	Reheater	Prefilter	Charcoal Vessel Train	First Charcoal Vessel	After-filter
Sr-92	1.86×10^{-2}	1.63×10^{-1}	3.22×10^{2}	0	8.57×10^{-2}	4.04×10^{-4}	1.46×10^{-5}	3.26×10^{-5}	3.14×10^{-2}	0	0	0
Y-92	0	0	3.67×10^{-1}	0	3.01×10^{-2}	3.60×10^{-4}	1.31×10^{-5}	2.93×10^{-5}	6.80×10^{-2}	0	0	0
Kr-93	1.22×10^{3}	1.11×10^{3}	2.69×10^{3}	0	1.37×10^{-5}	0	0	0	0	0	0	0
Rb-93	6.09×10^{1}	2.04×10^{2}	4.74×10^{3}	0	8.24×10^{-6}	0	0	0	0	0	0	0
Sr-93	2.42×10^{-2}	2.22×10^{-1}	2.87×10^{2}	0	8.24×10^{-6}	0	0	0	0	0	0	0
Y-83	0	0	1.17×10^{-1}	0	1.88×10^{-6}	0	0	0	0	0	0	0
Zr-93	0	0	0	0	0	0	0	0	0	0	0	0
Nb-93M	0	0	0	0	0	0	0	0	0	0	0	0
Kr-94	2.93×10^{0}	1.98×10^{0}	1.98×10^{0}	0	0	0	0	0	0	0	0	0
Rb-94	2.87×10^{-1}	8.04×10^{-1}	5.80×10^{0}	0	0	0	0	0	0	0	0	0
Sr-94	0	6.63×10^{-3}	2.31×10^{0}	0	0	0	0	0	0	0	0	0
Y-94	0	0	3.25×10^{-2}	0	0	0	0	0	0	0	0	0
Kr-95	2.05×10^{-1}	1.39×10^{-1}	1.38×10^{-1}	0	0	0	0	0	0	0	0	0
Rb-95	2.30×10^{-2}	6.36×10^{-2}	3.96×10^{-1}	0	0	0	0	0	0	0	0	0
Sr-95	0	0	2.36×10^{-1}	0	0	0	0	0	0	0	0	0

Table 15.7-1 (continued)

AMBIENT OFF-GAS TREATMENT SYSTEM INVENTORY ACTIVITIES (μCi)

	Preheater	Recombiner	Condenser	Separator	Holdup Pipe	Condenser	Moisture Separator	Reheater	Prefilter	Charcoal Vessel Train	First Charcoal Vessel	After-filter
Y-95	0	0	6.45×10^{-3}	0	0	0	0	0	0	0	0	0
Zr-95	0	0	0	0	0	0	0	0	0	0	0	0
Nb-95M	0	0	0	0	0	0	0	0	0	0	0	0
Kr-97	0	0	0	0	0	0	0	0	0	0	0	0
Rb-97	0	0	0	0	0	0	0	0	0	0	0	0
Sr-97	0	0	0	0	0	0	0	0	0	0	0	0
Y-97	0	0	0	0	0	0	0	0	0	0	0	0
Zr-97	0	0	0	0	0	0	0	0	0	0	0	0
Nb-97	0	0	0	0	0	0	0	0	0	0	0	0
Xe-131M	9.58×10^{0}	1.20×10^{1}	5.89×10^{2}	6.10×10^{1}	1.71×10^{6}	2.11×10^{4}	7.70×10^{2}	1.72×10^{3}	5.15×10^{3}	1.01×10^{8}	1.20×10^{7}	2.21×10^{3}
Xe-133M	1.70×10^{2}	2.12×10^{2}	1.06×10^{4}	1.08×10^{3}	2.98×10^{8}	3.59×10^{4}	1.31×10^{3}	2.92×10^{3}	8.77×10^{3}	5.61×10^{7}	1.77×10^{7}	9.97×10^{1}
Xe-133	4.36×10^{3}	5.45×10^{3}	2.73×10^{5}	2.78×10^{4}	7.76×10^{7}	9.49×10^{5}	3.46×10^{4}	7.73×10^{4}	2.32×10^{5}	3.03×10^{9}	5.19×10^{8}	3.50×10^{4}
Xe-135M	2.57×10^{4}	3.21×10^{4}	1.58×10^{6}	1.58×10^{5}	4.16×10^{7}	1.20×10^{2}	4.11×10^{0}	9.09×10^{0}	2.67×10^{1}	8.20×10^{2}	8.20×10^{2}	2.91×10^{-4}
Xe-135	1.53×10^{4}	1.91×10^{4}	9.65×10^{5}	9.73×10^{4}	2.37×10^{8}	2.63×10^{6}	9.57×10^{4}	2.13×10^{5}	6.40×10^{5}	7.00×10^{8}	6.23×10^{8}	0
Cs-136	0	0	0	0	7.68×10^{-2}	1.24×10^{3}	4.52×10^{1}	1.01×10^{2}	2.19×10^{8}	2.11×10^{3}	1.87×10^{3}	0

Table 15.7-1 (continued)

AMBIENT OFF-GAS TREATMENT SYSTEM INVENTORY ACTIVITIES (μCi)

	Preheater	Recombiner	Condenser	Separator	Holdup Pipe	Condenser	Moisture Separator	Reheater	Prefilter	Charcoal Vessel Train	First Charcoal Vessel	After-filter
Xe-137	1.68×10^{5}	$2.10 \times 10^{\wedge 5 \wedge}$	$9.71 \times 10^{\wedge 6 \wedge}$	$9.10 \times 10^{\wedge 5 \wedge}$	$6.82 \times 10^{\wedge 7 \wedge}$	0	0	0	0	0	0	0
Cs-137	0	0	1.98×10^{-1}	1.70×10^{-3}	3.60×10^{2}	3.04×10^{0}	1.11×10^{-1}	2.47×10^{-1}	5.31×10^{5}	0	0	0
Ba-137M	0	0	1.47×10^{-2}	0	3.64×10^{2}	3.04×10^{0}	1.11×10^{-1}	2.47×10^{-1}	5.31×10^{5}	0	0	0
Xe-130	8.29×10^{4}	1.11×10^{5}	5.43×10^{6}	5.42×10^{5}	1.29×10^{8}	1.22×10^{2}	4.12×10^{0}	9.12×10^{0}	2.67×10^{1}	7.42×10^{2}	7.42×10^{2}	0
Cs-138	1.30×10^{1}	5.17×10^{1}	5.23×10^{4}	4.96×10^{2}	7.64×10^{7}	3.21×10^{4}	1.13×10^{3}	2.52×10^{3}	4.83×10^{5}	7.42×10^{2}	7.42×10^{2}	0
Xe-139	2.94×10^{5}	3.63×10^{5}	1.21×10^{7}	7.54×10^{5}	8.38×10^{6}	0	0	0	0	0	0	0
Cs-139	1.44×10^{2}	5.78×10^{2}	4.50×10^{5}	2.37×10^{3}	5.03×10^{6}	1.74×10^{-2}	5.66×10^{-4}	1.25×10^{-3}	6.87×10^{-2}	0	0	0
Ba-139	3.94×10^{-2}	5.52×10^{-2}	1.16×10^{3}	5.79×10^{-1}	4.26×10^{6}	1.26×10^{4}	4.53×10^{2}	1.01×10^{3}	5.00×10^{5}	0	0	0
Xe-140	2.24×10^{5}	2.68×10^{5}	4.71×10^{6}	9.14×10^{4}	3.08×10^{5}	0	0	0	0	0	0	0
Cs-140	9.45×10^{2}	3.74×10^{3}	1.61×10^{6}	2.51×10^{3}	1.85×10^{5}	0	0	0	0	0	0	0
Ba-140	0	1.78×10^{-3}	2.18×10^{1}	2.78×10^{-3}	1.65×10^{3}	1.36×10^{1}	4.97×10^{-1}	1.11×10^{0}	1.22×10^{5}	0	0	0
La-140	0	0	1.03×10^{-2}	0	5.51×10^{1}	9.09×10^{-1}	3.34×10^{-2}	7.45×10^{-2}	1.23×10^{5}	0	0	0
Xe-141	3.37×10^{3}	3.09×10^{3}	7.45×10^{3}	0	3.81×10^{-5}	0	0	0	0	0	0	0
Cs-141	4.04×10^{1}	1.41×10^{2}	1.03×10^{4}	0	2.28×10^{-5}	0	0	0	0	0	0	0
Ba-141	7.10×10^{-3}	6.67×10^{-2}	2.01×10^{2}	0	2.28×10^{-5}	0	0	0	0	0	0	0

Table 15.7-1 (continued)

AMBIENT OFF-GAS TREATMENT SYSTEM INVENTORY ACTIVITIES (μCi)

	Preheater	Recombiner	Condenser	Separator	Holdup Pipe	Condenser	Moisture Separator	Reheater	Prefilter	Charcoal Vessel Train	First Charcoal Vessel	After-filter
La-141	0	0	1.80×10^{-2}	0	1.07×10^{-5}	0	0	0	0	0	0	0
Ca-141	0	0	0	0	0	0	0	0	0	0	0	0
Xe-142	2.06×10^{2}	1.71×10^{2}	2.90×10^{2}	0	2.56×10^{9}	0	0	0	0	0	0	0
Cs-142	2.43×10^{1}	7.23×10^{1}	6.30×10^{2}	0	0	0	0	0	0	0	0	0
Ba-142	7.14×10^{-2}	6.06×10^{-1}	2.56×10^{2}	0	0	0	0	0	0	0	0	0
La-142	0	0	8.09×10^{-1}	0	0	0	0	0	0	0	0	0
Xe-143	1.49×10^{0}	1.01×10^{0}	1.01×10^{-0}	0	0	0	0	0	0	0	0	0
Cs-143	2.06×10^{-1}	5.50×10^{-1}	2.75×10^{0}	0	0	0	0	0	0	0	0	0
Ba-143	3.35×10^{-3}	2.62×10^{-2}	3.25×10^{0}	0	0	0	0	0	0	0	0	0
La - 143	0	0	8.93×10^{-2}	0	0	0	0	0	0	0	0	0
Ca - 143	0	0	0	0	0	0	0	0	0	0	0	0
Pr - 143	0	0	0	0	0	0	0	0	0	0	0	0
Xe - 144	2.99×10^{2}	3.48×10^{2}	4.16×10^{3}	2.74×10^{1}	5.54×10^{1}	0	0	0	0	0	0	0
Cs - 144	7.01×10^{1}	2.12×10^{2}	4.52×10^{3}	2.06×10^{1}	3.22×10^{1}	0	0	0	0	0	0	0
Ba - 144	1.12×10^{0}	9.45×10^{0}	4.13×10^{3}	2.33×10^{0}	3.22×10^{1}	0	0	0	0	0	0	0

Table 15.7-1 (continued)

AMBIENT OFF-GAS TREATMENT SYSTEM INVENTORY ACTIVITIES (μCi)

	Preheater	Recombiner	Condenser	Separator	Holdup Pipe	Condenser	Moisture Separator	Reheater	Prefilter	Charcoal Vessel Train	First Charcoal Vessel	After-filter
La - 144	3.91×10^{-3}	8.17×10^{-2}	1.48×10^{3}	5.94×10^{-2}	3.22×10^{1}	0	0	0	0	0	0	0
Ce - 144	0	0	0	0	1.37×10^{-2}	0	0	0	0	0	0	0
Pr - 144	0	0	0	0	1.23×10^{-2}	0	0	0	0	0	0	0
N - 13	5.78×10^{3}	7.21×10^{3}	3.51×10^{5}	3.45×10^{4}	5.98×10^{6}	8.16×10^{-3}	2.68×10^{-3}	5.90×10^{-3}	1.71×10^{-2}	3.33×10^{-1}	1.32×10^{-1}	3.85×10^{-5}
N - 16	5.22×10^{7}	5.94×10^{7}	5.44×10^{8}	1.30×10^{6}	1.88×10^{6}	0	0	0	0	0	0	0
N - 17	4.43×10^{3}	4.76×10^{3}	2.62×10^{2}	3.52×10^{0}	2.61×10^{0}	0	0	0	0	0	0	0

Table 15.7-1 (continued)

AMBIENT OFF-GAS TREATMENT SYSTEM INVENTORY ACTIVITIES (μCi)

	Preheater	Recombiner	Condenser	Separator	Holdup Pipe	Condenser	Moisture Separator	Reheater	Prefilter	Charcoal Vessel Train	First Charcoal Vessel	After-filter
O - 19	7.12×10^5	9.69×10^5	2.41×10^7	1.13×10^6	7.99×10^6	0	0	0	0	0	0	0
Iodine	0	0	0	0	0	0	0	0	1.34×10^7	1.34×10^7	1.34×10^7	0
Kr + Xe (Gas)	1.49×10^6	1.82×10^6	5.89×10^7	4.17×10^6	9.80×10^8	5.96×10^6	2.16×10^5	4.83×10^5	1.45×10^6	4.08×10^9	2.23×10^9	4.61×10^4
Solid daughters	3.51×10^3	1.33×10^4	4.74×10^8	1.37×10^4	2.10×10^8	6.48×10^5	2.50×10^4	5.85×10^4	2.60×10^8	1.00×10^8	3.33×10^7	9.86×10^4
Kr gas	6.69×10^5	8.10×10^5	2.41×10^7	1.59×10^6	4.23×10^8	2.32×10^6	8.42×10^4	1.86×10^5	5.61×10^6	1.87×10^8	6.05×10^7	8.95×10^3
Xe gas	8.24×10^5	1.01×10^6	3.48×10^7	2.58×10^6	5.57×10^8	3.64×10^6	1.32×10^5	2.95×10^5	8.86×10^5	3.89×10^9	1.17×10^9	3.72×10^4
Total	5.44×10^7	6.21×10^7	6.32×10^8	6.65×10^6	1.21 + 9	6.61×10^6	2.42×10^5	5.39×10^5	2.75×10^8	4.18×10^9	2.28×10^9	1.45×10^5

Note:

1. Four-hour residence time in holdup pipe is conservative. Actual holdup time is about 6 hours.

Table 15.7-2

RADIOLOGICAL EXPOSURE DUE TO OFF-GAS TREATMENT SYSTEM COMPNENT FAILURE

Component Failed	Primary Activity Released	Percentage Released	Resultant Exposure (800 meters)	Regulatory Limits	
				10 CFR 20	10 CFR 100
First Activated Carbon Bed	Iodine	1%	12.4 mrem	* Note 1	300,000 mrem
12 Activated Carbon Beds	Noble Gas	10%	1.0 mrem	* Note 1	25,000 mrem
Prefilter	Particulates	1%	18.5 mrem	* Note 1	25,000 mrem
Holdup Pipe	Particulates	20%	20.3 mrem	* Note 1	25,000 mrem

* Note 1 - The total limit for annual exposure of all types of radiations (noble gases, particulates, iodines) is 100 mrem/year TEDE. The 100 mrem TEDE is the annual 10 CFR 20 limit for dose to the public.

Table 15.7-3

REACTOR BUILDING AIRBORNE FISSION PRODUCT INVENTORY
(Original Analysis, Retained for Historical Purpose)

Time	Noble Gases (Ci)	Halogens (Ci)
1 minute	4.9×10^{3}	1.0×10^{2}
30 minutes	4.8×10^{3}	1.0×10^{2}
1 hour	4.6×10^{3}	1.0×10^{2}
3 hours	4.2×10^{3}	9.8×10^{1}
10 hours	2.9×10^{3}	7.8×10^{1}
1 day	1.4×10^{3}	5.4×10^{1}
3 days	1.4×10^{2}	1.8×10^{1}
10 days	5.6×10^{-2}	5.8×10^{0}
25 days	0	8.6×10^{-5}

Table 15.7-4 FHA – Radiological Consequences
Key Inputs, Assumptions and Radiological Doses

Table 15.7-4a Key FHA Analysis Inputs and Assumptions

Input/Assumption	Value
Core Damage	111 fuel pins 7x7 rod array)
Radial Peaking Factor	1.7
Fuel Decay Period	24 hours
Fuel Pool Water Iodine Decontamination Factor	DF = 92 (19 feet depth)
Release Period	2 hours
Release Location	Reactor building vent stack Unfiltered, zero velocity vent release, 302 feet from intake
CREV System Initiation	No Credit Taken

Table 15.7-4b

Control Room χ/Q Values for the FHA Releases

Time Period	χ/Q (sec/m^3) [1]
0 - 2 hrs	6.44E-04

Notes: 1. Zero velocity vent release χ/Q values for normal reactor building exhaust stack release based on ARCON96.

Table 15.7-4c

Offsite χ/Q (sec/m^3) Values for the FHA Releases

Time Period	EAB χ/Q (sec/m^3)	LPZ χ/Q (sec/m^3)
0 - 2 hrs	2.51E-4 [1]	2.63E-5 [1]

Notes: 1. Zero velocity vent release χ/Q values for normal reactor building exhaust stack release based on Regulatory Guide 1.145 methodology.

Table 15.7-4d

FHA Radiological Consequence Analysis[1]
[with 19 feet water coverage]

Location	Duration	TEDE (rem)	Regulatory Limit TEDE (rem)
Control Room	30 days	1.99	5
EAB	Maximum, 2 hours	0.967	6.3
LPZ	30 days	0.101	6.3

1 Radiological consequences are based on core inventory described in Section 15.7.3.4.2.3.1. Radiological consequences are also bounding for the Westinghouse SVEA-96 Optima2 fuel design as well as the AREVA ATRIUM 10XM fuel design for Unit 3.

Table 15.7-5

RADIOLOGICAL EFFECTS OF THE FUEL HANDLING ACCIDENT - 7x7 FUEL
(Original Analysis, Retained for Historical Purpose)

Distance (mi)	First 2-Hour Dose						Total Accident Dose					
	VS-2(1)	MS-2(2)	N-2(3)	N-10(4)	U-2(5)	U-10(6)	VS-2(1)	MS-2(2)	N-2(3)	N-10(4)	U-2(5)	U-10(6)
Whole Body Passing Cloud Dose (rem)												
1/2	3.5×10^{-4}	3.5×10^{-4}	3.8×10^{-4}	5.9×10^{-5}	4.7×10^{-4}	6.8×10^{-5}	2.3×10^{-3}	2.3×10^{-3}	2.5×10^{-3}	3.8×10^{-4}	3.1×10^{-3}	4.4×10^{-4}
1	2.3×10^{-4}	2.4×10^{-4}	2.9×10^{-4}	4.3×10^{-5}	2.1×10^{-4}	3.3×10^{-5}	1.5×10^{-3}	1.5×10^{-3}	1.9×10^{-3}	2.8×10^{-4}	1.4×10^{-3}	2.2×10^{-4}
5	-	-	-	7.9×10^{-6}	-	3.5×10^{-6}	4.1×10^{-4}	4.6×10^{-4}	2.8×10^{-4}	5.2×10^{-5}	9.5×10^{-5}	2.2×10^{-5}
9	(7)	-	-	3.4×10^{-6}	-	1.4×10^{-6}	2.3×10^{-4}	2.5×10^{-4}	9.6×10^{-5}	2.2×10^{-5}	2.9×10^{-5}	8.9×10^{-6}
12	-	-	-	2.2×10^{-6}	-	8.8×10^{-7}	1.6×10^{-4}	1.7×10^{-4}	5.3×10^{-5}	1.4×10^{-5}	1.6×10^{-5}	5.7×10^{-6}
Lifetime Thyroid Dose (rem)												
1/2	(8)	(8)	2.6×10^{-5}	6.6×10^{-7}	1.3×10^{-4}	1.4×10^{-5}	(8)	9.0×10^{-8}	2.1×10^{-4}	5.3×10^{-6}	1.1×10^{-3}	1.2×10^{-4}
1	(8)	-	6.4×10^{-5}	6.0×10^{-6}	5.5×10^{-5}	7.4×10^{-6}	(8)	4.6×10^{-6}	5.1×10^{-4}	4.8×10^{-5}	4.4×10^{-4}	6.0×10^{-5}
5	-	-	-	2.0×10^{-6}	-	7.5×10^{-7}	(8)	1.1×10^{-4}	8.3×10^{-5}	1.6×10^{-5}	3.4×10^{-5}	6.1×10^{-6}
9	-	-	-	8.7×10^{-7}	-	3.2×10^{-7}	2.5×10^{-8}	1.0×10^{-4}	3.5×10^{-5}	7.0×10^{-6}	1.4×10^{-5}	2.6×10^{-6}
12	-	-	-	5.8×10^{-7}	-	2.1×10^{-7}	2.2×10^{-7}	8.9×10^{-5}	2.3×10^{-5}	4.7×10^{-6}	8.8×10^{-6}	1.7×10^{-6}
Whole Body Fallout Dose (rem)												
1/2	(8)	(8)	1.6×10^{-8}	2.0×10^{-9}	1.8×10^{-7}	9.7×10^{-8}	(8)	(8)	4.4×10^{-7}	5.6×10^{-8}	5.1×10^{-6}	2.8×10^{-6}
1	(8)	(8)	3.8×10^{-8}	1.8×10^{-8}	7.4×10^{-8}	5.0×10^{-8}	(8)	5.6×10^{-9}	1.1×10^{-6}	5.1×10^{-7}	2.1×10^{-6}	1.4×10^{-6}
5	-	-	-	5.9×10^{-9}	-	5.1×10^{-9}	(8)	1.3×10^{-7}	1.8×10^{-7}	1.7×10^{-7}	1.6×10^{-7}	1.4×10^{-7}
^^9	-	-	-	2.6×10^{-9}	-	2.2×10^{-9}	(8)	1.2×10^{-7}	7.4×10^{-8}	7.4×10^{-8}	6.5×10^{-8}	6.1×10^{-8}
12	-	-	-	1.7×10^{-9}	-	1.4×10^{-9}	(8)	1.1×10^{-7}	4.9×10^{-8}	4.9×10^{-8}	4.2×10^{-8}	4.0×10^{-8}
Whole Body Fallout (Washout) Dose (rem)												
1/2		1.5×10^{-6}						4.2×10^{-5}				
1		5.5×10^{-7}						1.6×10^{-5}				
5		-						1.6×10^{-6}				
9		-						6.4×10^{-7}				
12		-						4.1×10^{-7}				

Notes:

1. Very stable meteorological conditions with 2-mph wind speed
2. Moderately stable meteorological conditions with 2-mph wind speed.
3. Neutral meteorological conditions with 2-mph wind speed
4. Neutral meteorological conditions with 10-mph wind speed
5. Unstable meteorological conditions with 2-mph wind speed.
6. Unstable meteorological conditions with 10-mph wind speed
7. First 2-hour dose is zero since time of cloud travel is greater than 2 hours.
8. Less than 1×10^{-10}.

Table 15.7-6

REACTOR BUILDING AIRBORNE FISSION PRODUCT INVENTORY
Assuming 111 Failed Fuel Rods
(Original Analysis, Retained for Historical Purpose)

Time	Noble Gases (Ci)		Halogens (Ci)
1 minute	5.91E+03		1.21E+02
30 minutes	5.79E+03		1.21E+02
1 hour	5.55E+03		1.21E+02
3 hours	5.07E+03		1.18E+02
10 hours	3.50E+03		9.41E+01
1 day	1.69E+03		6.52E+01
3 days	1.69E+02		2.17E+01
10 days	6.76E-02		7.00E+00
25 days	0		1.04E-04

Note: The numbers in this table were calculated by multiplying the values from the original FSAR by 111/92.

Table 15.7-7

FUEL HANDLING ACCIDENT RELEASE RATE TO ATMOSPHERE
Assuming 111 Failed Fuel Rods
(Original Analysis, Retained for Historical Purpose)

Time	Noble Gases (Ci)		Halogens (Ci)
1 minute	6.76E-02		1.45E-05
30 minutes	6.64E-02		1.45E-05
1 hour	6.39E-02		1.45E-05
3 hours	5.79E-02		1.21E-05
10 hours	4.10E-02		1.10E-05
1 day	1.93E-02		7.60E-06
3 days	1.93E-03		2.53E-06
10 days	7.84E-07		8.08E-07
25 days	0		1.21E-11

Note: The numbers in this table were calculated by multiplying the values from the original FSAR by 111/92.

Table 15.7-8

RADIOLOGICAL EFFECTS OF THE FUEL HANDLING ACCIDENT — 7x7 FUEL
Assuming 111 Failed Fuel Rods
(Original Analysis, Retained for Historical Purpose)

First 2-Hour Dose						
Distance (mi)	VS-2(1)	MS-2(2)	N-2(3)	N-10(4)	U-2(5)	U-10(6)
Whole Body Passing Cloud Dose (rem)						
1/2	4.22E-04	4.22E-04	4.58E-04	7.12E-05	5.67E-04	8.20E-05
1	2.78E-04	2.90E-04	3.50E-04	5.19E-05	2.53E-04	3.98E-05
5	-	-	-	9.53E-06	-	4.22E-06
9	(7)	-	-	4.10E-06	-	1.69E-06
12	-	-	-	2.65E-06	-	1.06E-06
Lifetime Thyroid Dose (rem)						
1/2	(8)	(8)	3.14E-05	7.96E-07	1.57E-04	1.69E-05
1	(8)	-	7.72E-05	7.24E-06	6.64E-05	8.93E-06
5	-	-	-	2.41E-06	-	9.05E-07
9	-	-	-	1.05E-06	-	3.86E-07
12	-	-	-	7.00E-07	-	2.53E-07
Whole Body Fallout Dose (rem)						
1/2	(8)	(8)	1.93-08	2.41E-09	2.17E-07	1.17E-07
1	(8)	(8)	4.58E-08	2.17E-08	8.93E-08	6.03E-08
5	-	-	-	7.12E-09	-	6.15E-09
9	-	-	-	3.14E-09	-	2.65E-09
12	-	-	-	2.05E-09	-	1.69E-09
Whole Body Fallout (Washout) Dose (rem)						
1/2		1.81E-06				
1		6.64E-07				
5		-				
9		-				
12		-				

Table 15.7-8 (Continued)

RADIOLOGICAL EFFECTS OF THE FUEL HANDLING ACCIDENT – 7x7 FUEL
Assuming 111 Failed Fuel Rods

Total Accident Dose						
Distance (mi)	VS-2(1)	MS-2(2)	N-2(3)	N-10(4)	U-2(5)	U-10(6)
Whole Body Passing Cloud Dose (rem)						
1/2	2.78E-03	2.78E-03	3.02E-03	4.58E-04	3.74E-03	5.31E-04
1	1.81.E-03	1.81E-03	2.29E-03	3.38E-04	1.69E-03	2.65E-04
5	4.95E-04	5.55E-04	3.38E-04	6.27E-05	1.15E-04	2.65E-05
9	2.78E-04	3.02E-04	1.16E-04	2.65E-05	3.50E-05	1.07E-05
12	1.93E-04	2.05E-04	6.39E-05	1.69E-05	1.93E-05	6.88E-06
Lifetime Thyroid Dose (rem)						
1/2	(8)	1.09E-07	2.53E-04	6.93E-06	1.33E-03	1.45E-04
1	(8)	5.55E-06	6.15E-04	5.79E-05	5.31E-04	7.24E-05
5	(8)	1.33E-04	100E-04	1.93E-05	4.10E-05	7.36E-06
9	3.02E-08	1.21E-04	4.22E-05	8.45E-06	1.69E-05	3.14E-06
12	2.65E-07	1.07E-04	2.78E-05	5.67E-06	1.06E-05	2.05E-06
Whole Body Fallout Dose (rem)						
1/2	(8)	(8)	5.31E-07	6.76E-08	6.15E-06	3.38E-06
1	(8)	6.76E-09	1.33E-06	6.15E-07	2.53E-06	1.69E-06
5	(8)	1.57E-07	2.17E-07	2.05E-07	1.93E-07	1.69E-07
9	(8)	1.45E-07	8.93E-08	8.93E-08	7.84E-08	7.36E-08
12	(8)	1.33E-07	5.91E-08	5.91E-08	5.07E-08	4.83E-08
Whole Body Fallout (Washout) Dose (rem)						
1/2		5.07E-05				
1		1.93E-05				
5		1.93E-06				
9		7.72E-07				
12		4.95E-07				

Table 15.7-8 (Continued)

RADIOLOGICAL EFFECTS OF THE FUEL HANDLING ACCIDENT – 7x7 FUEL
Assuming 111 Failed Fuel Rods

Notes:

1. Very stable meterological conditions with 2-mph wind speed.
2. Moderately stable meteorological conditions with 2-mph wind speed.
3. Neutral meteorological conditions with 2-mph wind speed.
4. Neutral meteorological conditions with 10-mph wind speed.
5. Unstable meteorological conditions with 2-mph wind speed.
6. Unstable meteorological conditions with 10-mph wind speed.
7. First 2-hour dose is zero since time of cloud travel is greater than 2 hours.
8. Less than 1 x 10^{-10}.

Table 15.7-9

FUEL HANDLING ACCIDENT IN THE SPENT FUEL POOL OR IN CONTAINMENT
EAB, LPZ AND CONTROL ROOM DOSES FOLLOWING EPU
(Original Analysis, Retained for Historical Purpose)

Location	**Organ**	**Dose (Rem)**	**Regulatory Dose Limit (rem)**
EAB			
	Thyroid	3.84	75
	Whole Body	0.18	6.25
LPZ			
	Thyroid	0.46	75
	Whole Body	0.024	6.25
Control Room			
	Thyroid	10.2	30
	Whole Body	0.015	5
	Beta	0.58	30

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

This section covers the events, which result in an anticipated transient without scram (ATWS). Anticipated transient without scram events are beyond design basis accidents. Anticipated transients without scram are those low probability events in which an anticipated transient occurs and is not followed by an automatic reactor shutdown (scram) when required. The failure of the reactor to scram quickly during these transients can lead to unacceptable reactor coolant system pressures and to fuel damage. Mitigation of the lack of scram must involve insertion of negative reactivity into the reactor, thereby terminating the long-term aspects of the event.

The occurrence of a common-mode failure, which completely disables the reactor scram function, is a very low probability event. Therefore, no significant risk to public safety is presented by the combination of an infrequent event and a common-mode failure, which prevents scram. Thus, attention is focused on those transient situations, which have a relatively high expected frequency of occurrence at a power condition at which serious plant disturbance might result.

AREVA ATRIUM 10XM methods and fuel are only applicable to Unit 3.

GE Performed the ATWS analysis for operation at Extended Power Uprate (EPU, at 2957 MWt). The GE analysis shows that a Pressure Regulator Failure Open to maximum steam demand (PRFO) is the limiting ATWS event. Westinghouse demonstrated that PRFO remains limiting with the ATWS analysis performed to support transition to the SVEA-96 Optima2 fuel. AREVA demonstrated that PRFO remains limiting with the ATWS analysis performed to support transition to ATRIUM 10XM fuel (Reference 7).

Westinghouse performed a plant-unique ATWS analysis as part of the fuel licensing analysis supporting the introduction of SVEA-96 Optima2 fuel. The analysis was performed for two equilibrium cores (SVEA-96 Optima2 and GE14) using the same cycle design specifications at EPU conditions and for a transition core with 2/3 GE14 fuel and 1/3 SVEA-96 Optima2 fuel. The results of the analysis are reported in Reference 6. Additional calculations for the long-term ATWS were also performed and reported in Reference 6. Long-term effects refer to the containment response and the general plant response with emphasis on the heat transfer to the suppression pool. The time delay from the symptom to initiating operator action credited for the SVEA-96 Optima2 fuel transition are equal to or conservatively longer than the current licensing basis long-term ATWS. The results confirm that for the limiting pressure regulator open to maximum demand (PRFO), all acceptance criteria are met after the introduction of SVEA-96 Optima2 fuel.

Westinghouse has performed the limiting ATWS analysis for PRFO using bounding ATWS inputs for Dresden. Westinghouse has analyzed the limiting ATWS event for the current plant configuration including the affect of using 45 a/o enriched B-10 and 40 gpm SLC system flow while considering applicable core design configurations. The Westinghouse calculation in Reference 6 has established the licensing basis for the reload of Westinghouse fuel and replaces the GE ATWS analysis in Reference 4. Westinghouse ATWS analysis for Dresden has addressed all ATWS acceptance criteria as described below:

- Peak vessel bottom pressure – The Westinghouse reload analysis considers the effects of core design changes and confirms on a cycle-specific basis that the peak vessel pressure meets the 1500 psig ASME acceptance criterion.

- Peak Cladding Temperature – Westinghouse has confirmed in Reference 6 that the PCT for the limiting ATWS event is bounded by the applicable LOCA analysis for the Westinghouse fuel.

- Peak Cladding Oxidation – Westinghouse has confirmed in Reference 6 that the peak cladding oxidation for the limiting ATWS event is bounded by the applicable LOCA analysis for the Westinghouse fuel.

- Peak suppression pool temperature – The Westinghouse calculation in Reference 6 confirmed that the peak suppression temperature is less than the acceptance criterion. This analysis is cycle independent.

- Peak containment pressure – The Westinghouse calculation in Reference 6 is cycle independent and provides bases that represent the current plant configuration. The Westinghouse calculation in Reference 6 confirmed that the peak containment pressure is less than the design limits. This analysis is cycle independent.

The PRFO would be the most severe postulated event from virtually all aspects when accompanied by a lack of scram. Other significant ATWS events, which are postulated to occur are described in the following subsections. Other transients such as closure of all main isolation valves (MSIVs), inadvertent opening of a relief or safety relief valve, and feedwater failure to maximum demand are less severe and are bounded by PRFO event described in section 15.8.7. The following transients have been originally analyzed:

A. Closure of main steam isolation,

B. Loss of normal ac power,

C. Loss of normal feedwater flow,

D. Turbine generator trip, and

E. Loss of condenser vacuum,

F. Pressure regulator failure - open to maximum demand.

The descriptions of ATWS events for closure of MSIVs, loss of normal ac power, loss of normal feedwater flow, turbine generator trip, and loss of condenser vacuum are based upon analyses [1,2], which utilized setpoints and initial conditions and differ from those currently in effect for Dresden. Specific details contained in the descriptions and associated figures should be used only to understand the analysis and its conclusions, not as sources of current design information.

Section 15.8.7 includes description of the limiting PRFO event that is based on the Westinghouse analysis in Reference 6.

AREVA confirms on a cycle-specific basis that the peak vessel pressure meets the 1500 psig ASME acceptance criterion. The PCT and cladding oxidation for the limiting ATWS are bounded by the applicable LOCA analysis results. The suppression pool temperature and containment pressure meet the acceptance criteria and/or design limits (Reference 8).

15.8.1 Closure of Main Steam Isolation Valves

See the introduction to Section 15.8 for information regarding the use of details from this analysis description, which may not be applicable to the current plant design.

15.8.1.1 Identification of Causes

Closure of all MSIVs is caused by any of a number of plant conditions, such as low-low reactor water level, main steam line high flow, main steam low pressure, main steam line high radiation, or main steam tunnel high temperature.

15.8.1.2 Sequence of Events and Systems Operations

This transient would be initiated with the closure of the MSIVs. Closure of the MSIVs would produce an immediate increase in reactor pressure, which would result in a reduction in moderator voids and a rapid increase in reactor power. In the absence of normal scram, the fuel temperature would rise, and the negative Doppler reactivity would limit the power. The opening of relief valves would tend to curtail the increases in reactor pressure and power. The reactor pressure would reach the ATWS mitigation system setpoint of 1250 psig for Unit 2 and 1200 psig for Unit 3 approximately 5 seconds after the start of the event. The ATWS mitigation system would initiate a recirculation pump trip (RPT) and would initiate alternate rod insertion (ARI). The RPT and ARI would introduce negative reactivity into the core. The reactor pressure would peak in about 12 seconds and then decrease to a value slightly above the relief valve setpoints. The standby liquid control (SBLC) system can be initiated if ARI is unavailable.

15.8.1.3 Core and System Performance

A. Reactor Shutdown by RPT and ARI

The projected vessel pressure as a function of time for this event is shown in Figure 15.8-1. In this case, the reactor pressure would rise to the ATWS setpoint, which would trip the recirculation pumps. This would cause a rapid reduction of core flow and a corresponding increase in core moderator voids, which would reduce core power. The resulting neutron flux behavior is shown in Figure 15.8-2. The ATWS signal would also initiate opening of valves on the scram air header, which would result in insertion of the control rods. This transient would result in a peak reactor pressure of 1476 psig, which would satisfy the overpressure limit of 1500 psig. NEDE-25026[1] used 1500 psig as a pressure guideline limit.

B. Reactor Shutdown by RPT and SBLC (No ARI)

In the event that the control rod insertion (via ARI) is unavailable for shutdown of the reactor, the SBLC system would be used as an alternative method of achieving reactor shutdown. This system injects an aqueous solution of sodium pentaborate into the reactor vessel. When the boron concentration in the reactor coolant reaches approximately 165 ppm, sufficient negative reactivity would be available to bring the core to hot standby. The SBLC pumps would continue injection so that sufficient boron is in the core to achieve cold shutdown, with an adequate margin to allow for uneven mixing. The SBLC system is actuated manually.

15.8.1.3.1 Reactor Water Level Response

The projected reactor water level response to the event with utilization of ARI is shown in Figure 15.8-3. The reactor water level would remain at near-normal level from event initiation until hot shutdown. Thereafter, adequate water inventory would be maintained either by the feedwater system or by the high pressure coolant injection (HPCI) system.

15.8.1.3.2 Containment and Suppression Pool Response

A. Reactor Shutdown by RPT and ARI

After the MSIVs are closed, the reactor power would be dissipated by the relief valve discharge of steam into the suppression pool. The steam discharged into the suppression pool would heat the suppression pool. The reactor would be shut down by ARI and steam would continue to flow into the pool due to decay heat. The operator would place the containment cooling system heat exchanger in the suppression pool cooling mode. The suppression pool temperature would continue to increase until the decay heat input decreases below the heat removal capacity of the containment cooling heat exchangers. The peak temperature and pressure of the suppression pool would further be limited by the operation of the isolation condenser, in addition to the containment cooling system heat exchangers, which would remove decay heat. Containment pressure and temperature response is shown on Figures 15.8-4 and 15.8-5.

B. Reactor Shutdown by RPT and SBLC (No ARI)

Achievement of reactor shutdown using SBLC rather than ARI would result in a peak containment pressure less than the design pressure. Containment pressures and suppression pool temperatures will be higher using SBLC rather than ARI due to the greater amount of relief valve steam flow entering the suppression pool.

15.8.1.3.3 Long-Term Response

For ATWS considerations, the reactor condition of concern would be hot shutdown rather than cold shutdown, because the key factor would be stopping thermal power generation during the event. The power generated prior to reaching hot shutdown has the most significant potential impact on the plant. Consequently, the time required to achieve hot shutdown would be the important parameter for ATWS. After hot shutdown is achieved, further action would be required to bring the reactor to cold shutdown conditions.

15.8.1.3.4 Operator Actions

In case of an apparent ATWS, certain manual actions would be required to be performed by the operator if automatic features do not function as designed. Possible operator actions would include trip of the recirculation pump, manual initiation of ARI, actuation of SBLC, or manually lowering the water level to top of active fuel.

Certain alarms and indications would be provided to the operator to support performance of the required manual actions within the time limits. Annunciator windows would alarm when the reactor water level or reactor pressure reaches the ATWS setpoints. At the beginning of the ATWS event, the recirculation pumps would be signaled to trip, the ARI would be automatically initiated, and the operator would receive the annunciation indicating that an ATWS has occurred. The operator would then have sufficient time to perform the required actions.

Operator actions that would be required in the event of an ATWS are set forth in plant procedures. Plant procedures specify that upon receipt of an automatic scram signal, if the reactor has not achieved shutdown using the control rods and reactor power is above a specified point, the operator is to actuate ARI. This action would insert the control rods. The recirculation pumps would be tripped manually. Manual initiation of ARI and RPT is described in greater detail in Section 7.8.3.

In the event that control rod insertion is unavailable for shutting the reactor down, the SBLC system would be manually actuated to inject an aqueous solution of sodium pentaborate into the reactor vessel.

Operator actions would involve actuation of the containment cooling system to cool the suppression pool. Operator actions are also required to bring the reactor from hot shutdown to cold shutdown.

15.8.1.4 Barrier Response

During the MSIV closure transient without scram the reactor fuel would experience a rapid power spike. Since heat removal through the fuel surface would follow the relatively slow heat transfer characteristics of the fuel, a significant rise in fuel enthalpy would be encountered.

The analysis of the event shows that the amount of cladding oxidation (less than 1% by volume) would be far less than the 17% guideline (per NEDE-25026[1]), and peak fuel enthalpy would be less than the Regulatory Guide 1.77 limit of 280 cal/g. Few (if any) fuel rod perforations would be experienced.

15.8.1.5 Radiological Consequences

The radiological consequences would be minimal due to the small (if any) number of fuel rod perforations.

15.8.2 Loss of Normal AC Power

See the introduction to Section 15.8 for information regarding the use of details from this analysis description which may not be applicable to the current plant design.

15.8.2.1 Identification of Causes

The loss of normal ac power would generally be caused by large grid disturbances, which in turn would deenergize buses that supply power to auxiliary equipment such as the recirculation pumps, condensate pumps, and circulating water pumps.

15.8.2.2 Sequence of Events and Systems Operations

When auxiliary power is lost, the circulating water pumps, feedwater pumps, and recirculation pumps would begin coasting down immediately. The reduction in core flow would begin to reduce the reactor power. A turbine generator trip would occur at the start of the event due to a general grid disturbance and would increase the reactor pressure. The safety-relief valves would open momentarily, limiting the pressure rise in the vessel. The peak vessel pressure experienced in this event would be less than in the MSIV closure event. The short-term response would be much less severe than the MSIV ATWS event for the following reasons:

A. The recirculation pumps would trip at time zero which would result in lower core flow rather than tripping when the reactor pressure reaches the ATWS mitigation system setpoint.

B. The feedwater pumps would trip at time zero which would result in reduced core inlet subcooling and, hence, in a lower reactor power.

15.8.2.3 Core and System Performance

A. Reactor Shutdown by ARI

The reactor would achieve shutdown by utilizing ARI. The HPCI system would restore reactor water level to the normal range. Cold shutdown would be reached by performing the normal manual actions.

B. Reactor Shutdown by SBLC (No ARI)

In the event that insertion of the control rods via ARI is not achievable the SBLC system would be utilized as an alternative method of effecting neutronic power shutdown. The vessel water level would be restored by HPCI. Containment pressures and suppression pool temperatures will be higher using SBLC rather than ARI due to the greater amount of steam flow entering the suppression pool.

The operator actions associated with this event would be similar to those described in Section 15.8.1.3.4.

15.8.2.4 Barrier Performance

As in the MSIV closure event, cladding oxidation would be less than 1% by volume. Peak fuel enthalpy would be less than 280 cal/g. Very few (if any) fuel rod perforations would be experienced.

15.8.2.5 Radiological Consequences

Radiological consequences would be minimal due to the small (if any) number of fuel rod perforations.

15.8.3 Loss of Normal Feedwater Flow

See the introduction to Section 15.8 for information regarding the use of details from this analysis description which may not be applicable to the current plant design.

15.8.3.1 Identification of Causes

Inadvertent trip of all feedwater pumps or water level controller failure (zero demand) would be a potential cause for loss of all normal feedwater flow to the vessel. Loss of normal ac power, as described above, would also be a potential cause of this event.

15.8.3.2 Sequence of Events and Systems Operations

The short-term effects of this event would be less severe than those of the MSIV closure event.

Reactor core flow would be reduced when the feedwater flow reduction occurs, which would drop power gradually until a low water level scram is initiated. Gradual vessel water inventory reduction would occur until vessel isolation is initiated. The HPCI systems would be initiated automatically to restore proper water level until the event is terminated. If the reactor water level reaches the low-low level, the ATWS logic would initiate ARI and RPT.

15.8.3.3 Core and System Performance

A. Reactor Shutdown by RPT and ARI

Shutdown would be achieved by using ARI. The recirculation pumps would also trip. The peak temperature reached in the pool would be slightly less than the temperature for the MSIV closure event. The peak vessel pressure experienced in this event would be less than in the MSIV closure event.

B. Reactor Shutdown by RPT and SBLC (No ARI)

In the event that insertion of the control rods via ARI is not achievable, the SBLC system would be utilized as an alternative method of achieving reactor shutdown.

The cold shutdown condition would be achieved by normal manual actions similar to those performed during the MSIV closure event. Without the ARI function, the plant long-term response to the transient would be similar to the loss of normal ac power transient (SBLC without ARI).

The operator actions associated with this event would be similar to those described in Section 15.8.1.3.4.

15.8.3.4 Barrier Performance

This event would result in cladding oxidation of less than 1% by volume. Peak fuel enthalpy would be less than 280 cal/g. Very few (if any) fuel rod perforations would be experienced.

15.8.3.5 Radiological Consequences

Radiological consequences would be minimal due to the small (if any) number of fuel rod perforations.

15.8.4 Turbine Generator Trip

See the introduction to Section 15.8 for information regarding the use of details from this analysis description which may not be applicable to the current plant design.

15.8.4.1 Identification of Causes

Loss of generator electrical load would initiate fast closure of the turbine control valves to provide overspeed protection for the unit. A variety of equipment protection signals would lead to closure of the turbine stop valves directly. Both the turbine control valve fast closure and turbine stop valve closure would have a similar effect on the reactor. Normally, a scram would be initiated almost simultaneously with the start of either the control valve fast closure or the stop valve closure. However, the scram is postulated not to occur.

15.8.4.2 Sequence of Events and Systems Operations

The fast closure of the turbine control valves would cause an abrupt reactor pressure rise which would be limited to well below design pressures by the action of the turbine bypass and the safety relief valves.

If the dome pressure reaches the ATWS setpoint, a RPT and ARI would be initiated.

15.8.4.3 Core and System Performance

A. Reactor Shutdown by RPT and ARI

The neutron flux, vessel pressure, cladding oxidation, and fuel enthalpy peaks experienced in this event would be less than those in the MSIV closure event.

The long-term response of the plant is not analyzed, as it would be similar to the MSIV closure ATWS event. However, because of the availability of turbine bypass to the condenser, the steam flow into the suppression pool would be considerably less than the MSIV closure event. Long-term heat removal would be provided by the steam bypass to main condenser. Reactor coolant inventory would be maintained using the feedwater system.

B. Reactor Shutdown by RPT and SBLC (No ARI)

In the event that insertion of the control rods via ARI is not achievable, the SBLC system would be utilized as an alternative method of achieving reactor shutdown. The peak suppression pool temperature will be less than the temperature reached in the MSIV closure event (SBLC without ARI).

The operator actions associated with this event would be similar to those described in Section 15.8.1.3.4.

15.8.4.4 Barrier Performance

This event would result in cladding oxidation of less than 1% by volume. Peak fuel rod enthalpy would be less than 280 cal/g. Very few (if any) fuel rod perforations would be experienced.

15.8.4.5 Radiological Consequences

Radiological consequences are minimal due to the small (if any) number of fuel rod perforations.

15.8.5 Loss of Condenser Vacuum

See the introduction to Section 15.8 for information regarding the use of details from this analysis description which may not be applicable to the current plant design.

15.8.5.1 Identification of Causes

The reduction or loss of vacuum in the main condenser can be caused by loss of circulating water pumps or ineffective operation of the vacuum support equipment. The long-term results of this event would be similar to the MSIV closure event unless enough vacuum can be maintained to preserve bypass flow. Preserving the bypass flow to the condenser would permit decay heat removal through the condenser instead of relying upon the suppression pool and the shutdown cooling systems.

15.8.5.2 Sequence of Events and Systems Operations

Loss of condenser vacuum would trip the turbine stop valves closed (which would normally scram the reactor). If the event is severe enough, then the steam bypass valves would close. These actions would occur normally over a period of several minutes or at worst, 20 to 30 seconds. The initial sequence of events would be the same as a turbine-generator trip since all systems would function in the same way.

15.8.5.3 Core and System Performance

A. Reactor Shutdown by RPT and ARI

   The loss of condenser vacuum event would result in short-term peak values that would be less severe than in the MSIV closure event. In this event, ATWS logic would be rapidly activated by the high pressure transient. The long-term response of this event (assuming vacuum continues to deteriorate) would be similar to the response for the MSIV closure event. The peak vessel pressure experienced in this event would be less than in the MSIV closure event.

B. Reactor Shutdown by RPT and SBLC (No ARI)

In the event that insertion of control rods via ARI is not achievable, the SBLC system would be utilized as an alternative method of achieving reactor shutdown.

The operator actions associated with this event would be similar to those described in Section 15.8.1.3.4.

15.8.5.4 Barrier Performance

This event would result in cladding oxidation of less than 1% by volume. Peak fuel rod enthalpy would be less than 280 cal/g. Very few (if any) fuel rod perforations would be experienced.

15.8.5.5 Radiological Consequences

The radiological consequences would be minimal due to the small (if any) number of full rod perforations.

15.8.6 Increased Steam Flow Evaluation

See the introduction to Section 15.8 for information regarding the use of details from this analysis description, which may not be applicable to the current plant design.

(Begin Historical Information)

Plant efficiency has been improved due to a combination of changes to the Reactor Water Cleanup (RWCU) system and feedwater heater performance. As a result, steam flow at the licensed thermal power of 2527 MWt is expected to exceed the original rated steam flow rate. Thus, the ATWS events were evaluated assuming a maximum steam flow rate of 9.90 Mlbm/hr which corresponds to a maximum feedwater flow rate of 9.87 Mlbm/hr. The acceptance criteria for the evaluation are:

1. Reactor Coolant Pressure Boundary remaining below emergency pressure limits (i.e., 1500 psig).
2. Containment pressure remaining below design limits, and suppression pool remaining below local saturation temperature.
3. Maintain a coolable geometry.
4. Radiological release remaining within 10CFR100 allowable limits.
5. Equipment necessary to mitigate the postulate ATWS functioning in the environment (pressure, temperature, humidity, and radiation) predicted for the ATWS event.

The evaluation addressed ATWS-RPT setpoint of 1250 psig, a main steam flow rate of 9.9 Mlbm/hr, and use of non-GE fuel with different void and Doppler coefficients. That evaluation (referred to as the Bounding Assessment in the following discussion) addresses the first four criteria.[3]

The ATWS/MSIV Closure (MSIVC) event is the basis for the assessment of peak vessel pressure. The Bounding Assessment was based on previous generic analyses and concluded that the peak reactor vessel pressure for an ATWS would increase but would remain less than the 1500 psig acceptance criterion.

The Bounding Assessment discusses containment impact based on the ATWS/Inadvertently Open Relief Valve (IORV) event. This is bounding for other ATWS events, including ATWS/MSIVC. The Bounding Assessment for an ATWS mitigated by ARI (ATWS-ARI) concluded that the peak bulk pool temperature of 149 F previously found for Dresden remains bounding for the suppression pool. The Bounding Assessment for an ATWS mitigated by SLCS (ATWS-SLCS) concluded that the peak bulk pool temperature is estimated to increase to 192 F based on previous generic studies and Dresden-specific parameters (including a main steam flow rate of 9.9 Mlbm/hr, and use of non GE fuel with different void and Doppler coefficients). Based on previous generic ATWS analyses of quenching and pool mixing, the Bounding Assessment concludes that local saturation temperature will not be exceeded for a peak pool temperature of 192 F. The Bounding Assessment cites a previous value of containment pressure of 12.7 psig and estimated that the higher estimated peak bulk pool temperature (for ATWS-SLCS) would increase containment pressure by less than 3 psi. Based on these results from the Bounding Assessment, the containment pressure is maintained less than design (i.e., 62 psig), and the suppression pool is maintained below the local saturation temperature. Therefore, Dresden continues to meet the second acceptance criterion.

The Bounding Assessment reported that a specific assessment of the impact of the changes on fuel integrity (i.e., maintaining a coolable geometry) was not necessary due to the large margins in previous generic analyses. Also, radiological consequences would remain well below the 10CFR100 guidelines. Therefore, Dresden continues to meet the third and fourth acceptance criteria.

The Bounding Assessment does not address the fifth criteria other than noting that a previous generic assessment had concluded that operability of ATWS mitigation equipment would not be impaired by the ATWS event. Based on the results above for the changes in steam flow, etc., it is concluded that local environmental conditions are not adversely changed for an ATWS mitigated by ARI because the peak bulk pool temperature of 149 F previously found for Dresden remains bounding for the suppression pool. For an ATWS mitigated by SLCS (a backup shutdown method needed only if ARI fails), the Bounding Assessment concluded that there would be an increase in the peak bulk temperature of the suppression pool (Torus). This would increase ambient air temperature in the vicinity of the suppression pool slightly. Because of the large physical separation between the Torus (in the Reactor Building basement) and the SLCS pumps (on the fourth floor of the Reactor Building), it is concluded that the SLCS environment is not impacted. The Recirc Pump Trip (RPT) function would not be impacted by the increased pool temperature because RPT would function long before the suppression pool temperatures would reach its peak.

(End Historical Information)

15.8.7 Pressure Regulator Failure — Open to Maximum Demand

See the introduction to Section 15.8 for information regarding the use of details from this analysis description, which may not be applicable to the current plant design.

15.8.7.1 Identification of Causes

Pressure regulator failure to the maximum demand is caused by the malfunction of the normal pressure regulator.

15.8.7.2 Sequence of Events and Systems Operations

This transient would be initiated by the failure of the pressure regulator, which generates a maximum demand signal for the turbine-generator. This signal yields the opening of all the turbine bypass valves and forces the turbine control valves to the fully open position. This causes the reactor vessel to depressurize and to void the core, which in turn further reduces the reactor power and the vessel pressure. The MSlVs start to close once the low steamline pressure is detected at about 13 seconds. The reactor pressure would reach the ATWS mitigation system setpoint of 1250 psig approximately 21 seconds after the start of the event (Reference 4). The ATWS mitigation system would initiate a recirculation pump trip (RPT) and would initiate alternate rod insertion (ARI). The RPT and ARI would introduce negative reactivity into the core. The reactor pressure would peak in about 27 seconds (Reference 4). The standby liquid control (SBLC) system can be initiated if ARI is unavailable. The reactor water level is reduced and maintained at an elevation consistent with Emergency Operation procedures (EOP). The water level is restored to the normal level once the hot shutdown boron weight (HSBW) is injected into the vessel.

15.8.7.3 Core and System Performance

In the event that the control rod insertion (via ARI) is unavailable for shutdown of the reactor, the SBLC system would be used as an alternate method of achieving reactor shutdown. Westinghouse ATWS evaluations (Reference 6) confirm that the analysis meets the ATWS acceptance criteria for SVEA-96 Optima2 fuel. The AREVA cycle-specific ATWS peak vessel pressure evaluation are performed to confirm the ATWS peak pressure criteria are met. The evaluations use an ATWS mitigation system setpoint of 1200 psig (Reference 8).

15.8.7.4 Barrier Response

This event would result in cladding oxidation of less than 1% by volume. Peak fuel rod enthalpy would be less than 280 cal/g. Very few (if any) fuel rod perforations would be experienced.

15.8.7.5 Radiological Consequences

The radiological consequences would be minimal due to the small (if any) number of fuel rod perforations.

15.8.8 References

1. "Studies of ATWS for Dresden 2, 3 and Quad Cities 1, 2 Nuclear Power Stations," General Electric Company, NEDE-25026, December 1976.

2. "Main Steam Isolation Closure Event with ATWS/RPT and ARI for Dresden 2, 3 and Quad Cities 1, 2 Nuclear Generating Plants," General Electric Company, NSE-45-0880, August 1980.

3. "Bounding ATWS Assessment for Dresden 2/3," General Electric, GENE A13-00419-01, Revision 1, July 1998.

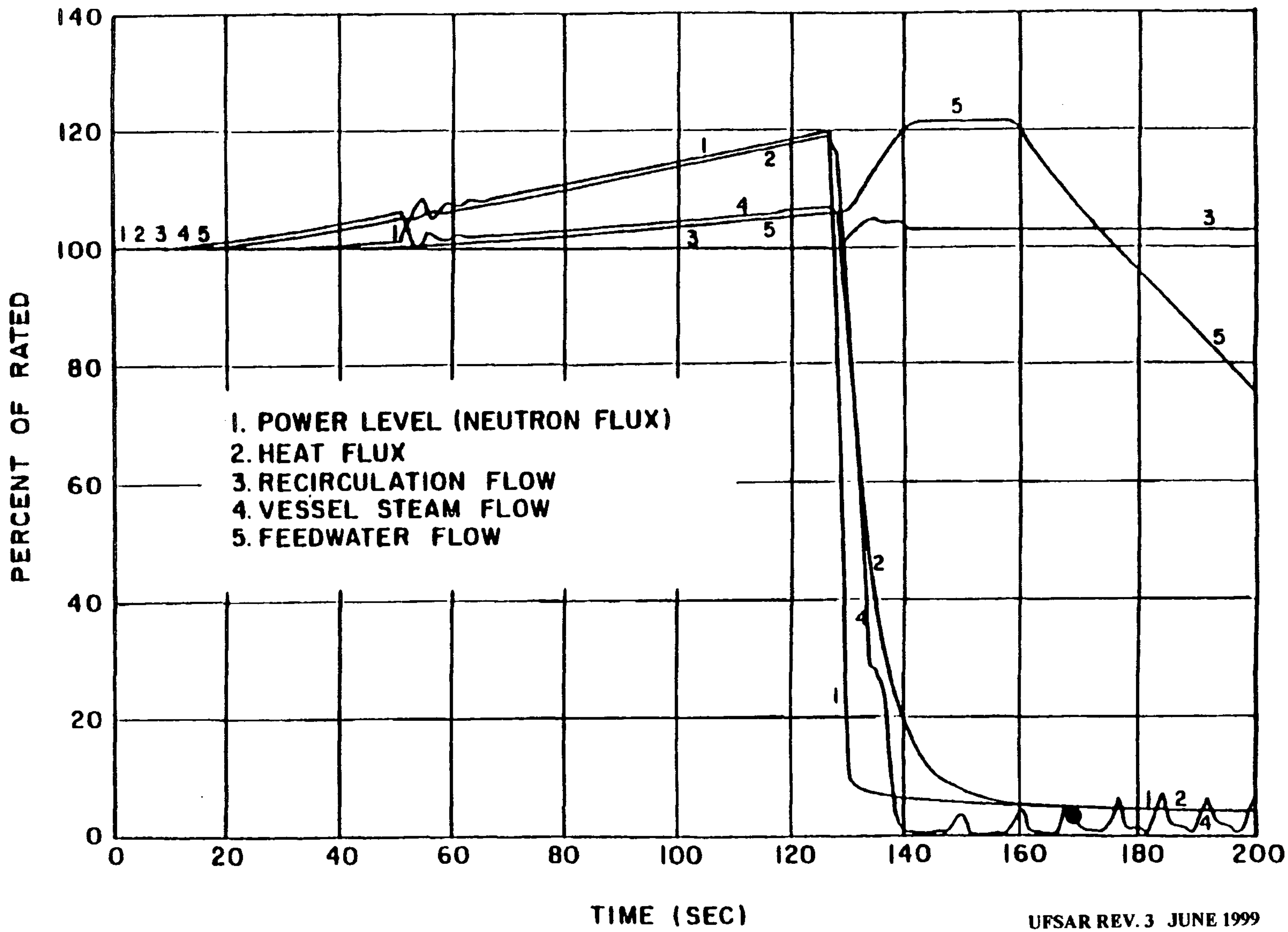
4. Dresden and Quad Cities Extended Power Uprate, Task T0902: Anticipated Transient Without Scram, GE-NE-A22-00103-11-01.

5. Safety Analysis Report for Dresden 2 & 3 Extended Power Uprate, NEDC32962P, Revision 2, August 2001.

6. “ATWS Analysis for the Introduction of SVEA-96 Optima2 Fuel at Dresden Units 2&3” Westinghouse Report OPTIMA2-TR051DR-ATWS, August 2006.

7. ANP-3338P, Revision 1, "Applicability of AREVA BWR Methods to the Dresden and Quad Cities Reactors Operating at Extended Power Uprate", AREVA, August 2015. (Unit 3 only)

8. ANP-3516P, Revision 0, "Dresden Unit 3 Cycle 25 Reload Safety Analysis," AREVA, September 2016. (Unit 3 only)

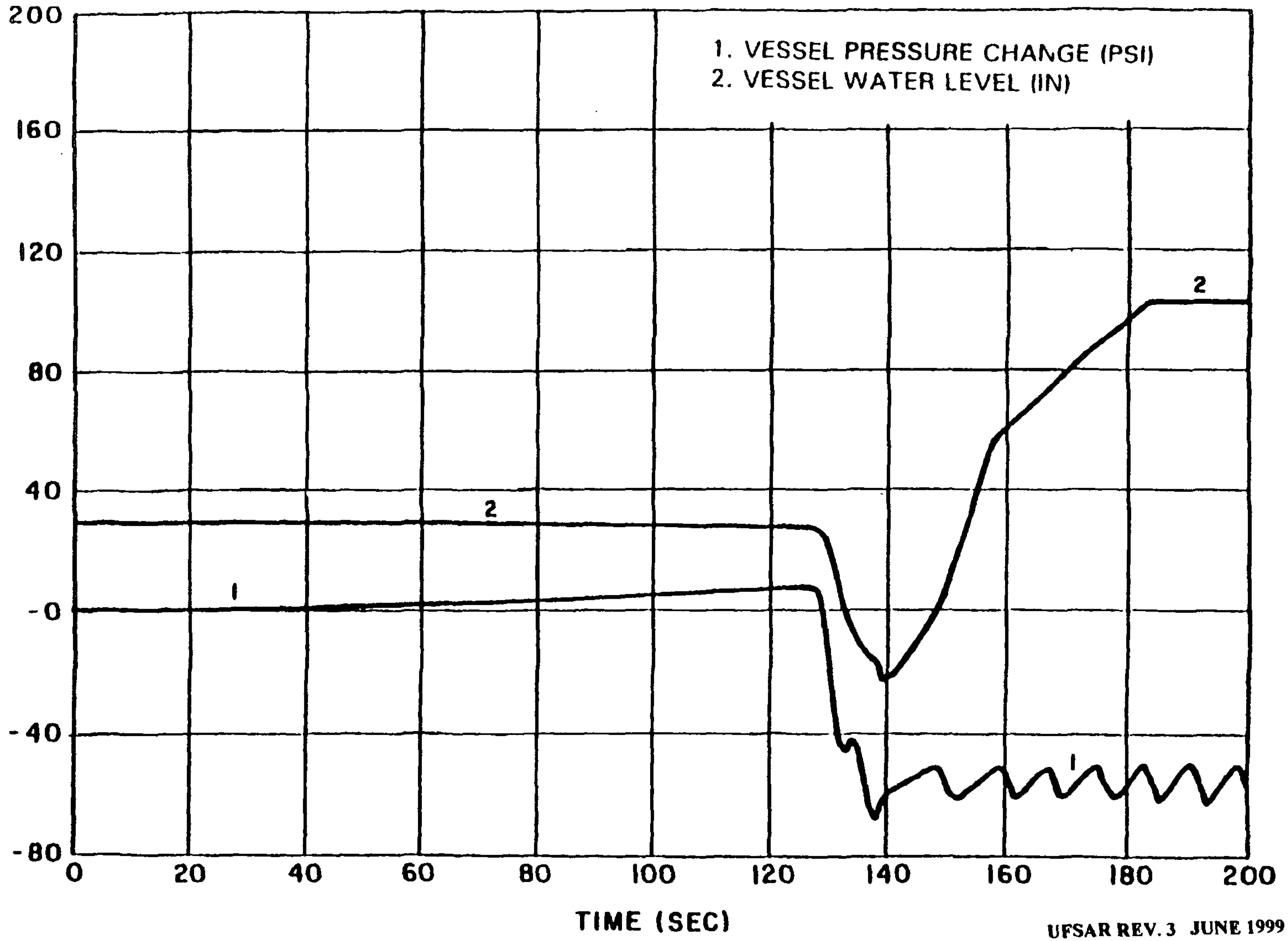


UFSAR REV. 3 JUNE 1999

DRESDEN STATION
UNITS 2 & 3

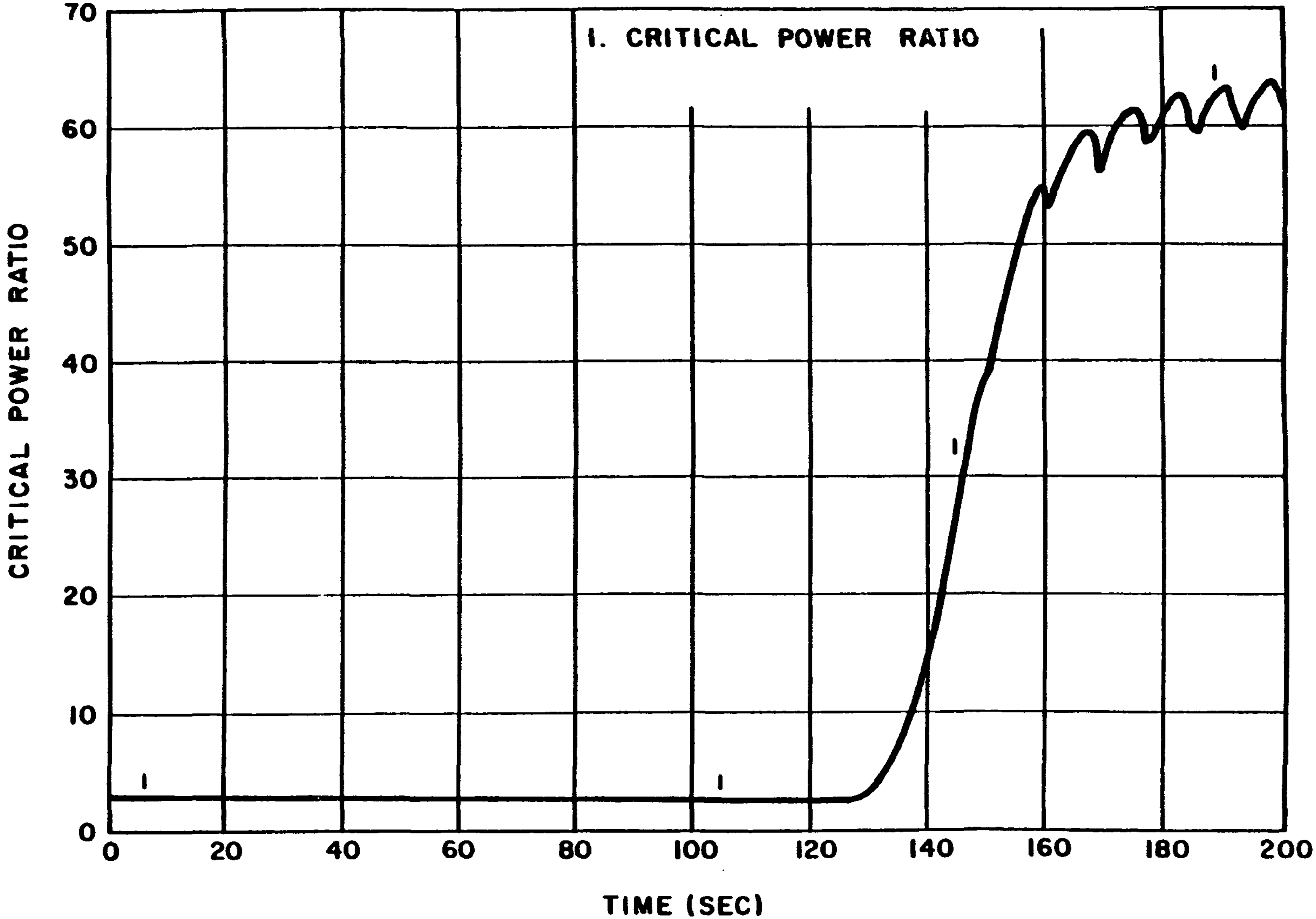
LOSS OF FEEDWATER HEATING
(POWER AND FLOWS, TYPICAL)

FIGURE 15.1-1



UFSAR REV. 3 JUNE 1999

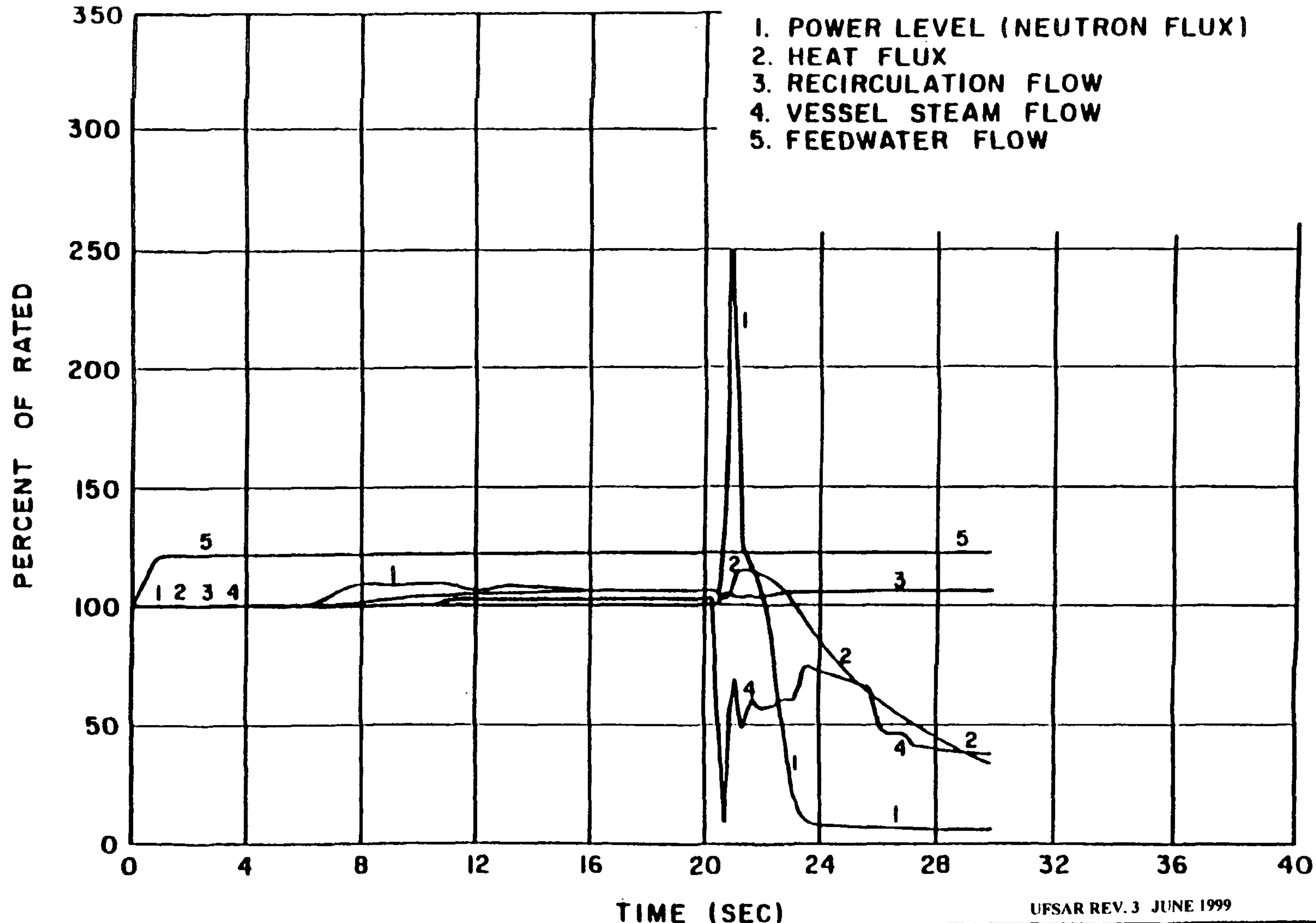
DRESDEN STATION UNIT 2 & 3
LOSS OF FEEDWATER HEATING (POWER AND FLOWS, TYPICAL)
FIGURE 15.1-2



DRESDEN STATION
UNITS 2 & 3

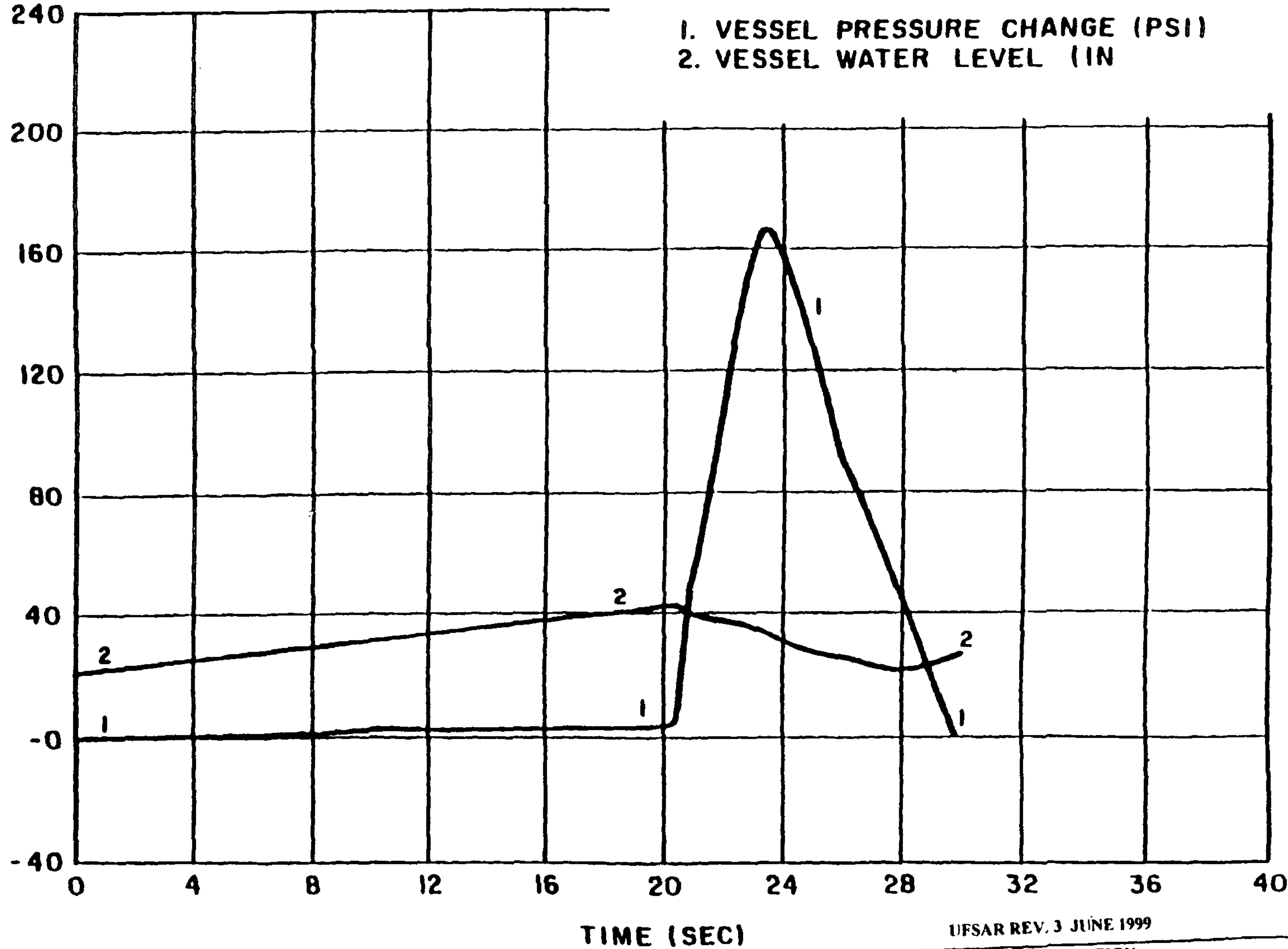
LOSS OF FEEDWATER HEATING
(TYPICAL CPR)

FIGURE 15.1-3



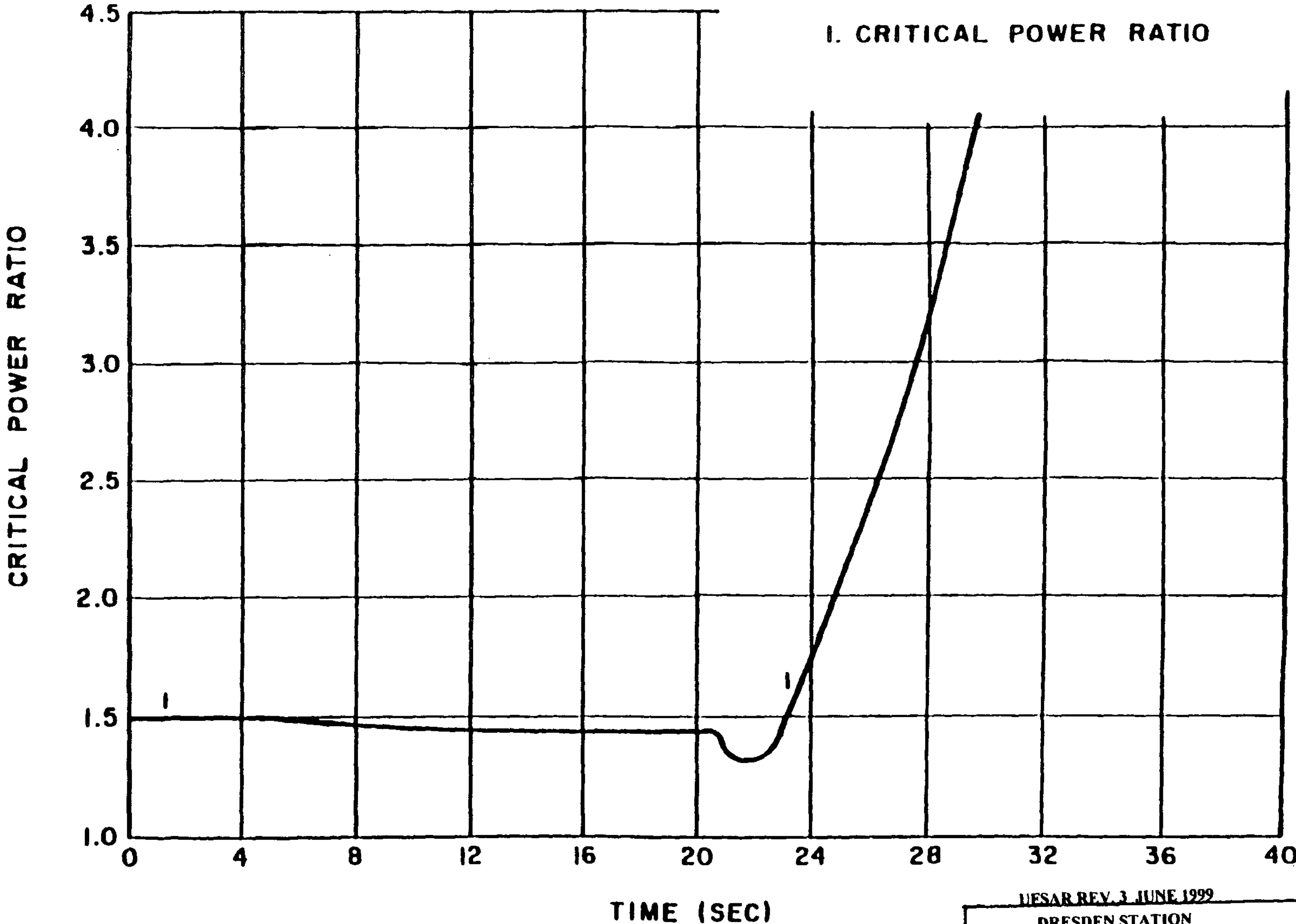
UFSAR REV. 3 JUNE 1999

DRESDEN STATION UNITS 2 & 3
INCREASE IN FEEDWATER FLOW (POWER AND FLOWS, TYPICAL)
FIGURE 15.1-4



UFSAR REV. 3 JUNE 1999

DRESDEN STATION UNITS 2 & 3
INCREASE IN FEEDWATER FLOW (VESSEL PRESSURE AND LEVEL, TYPICAL)
FIGURE 15.1-5

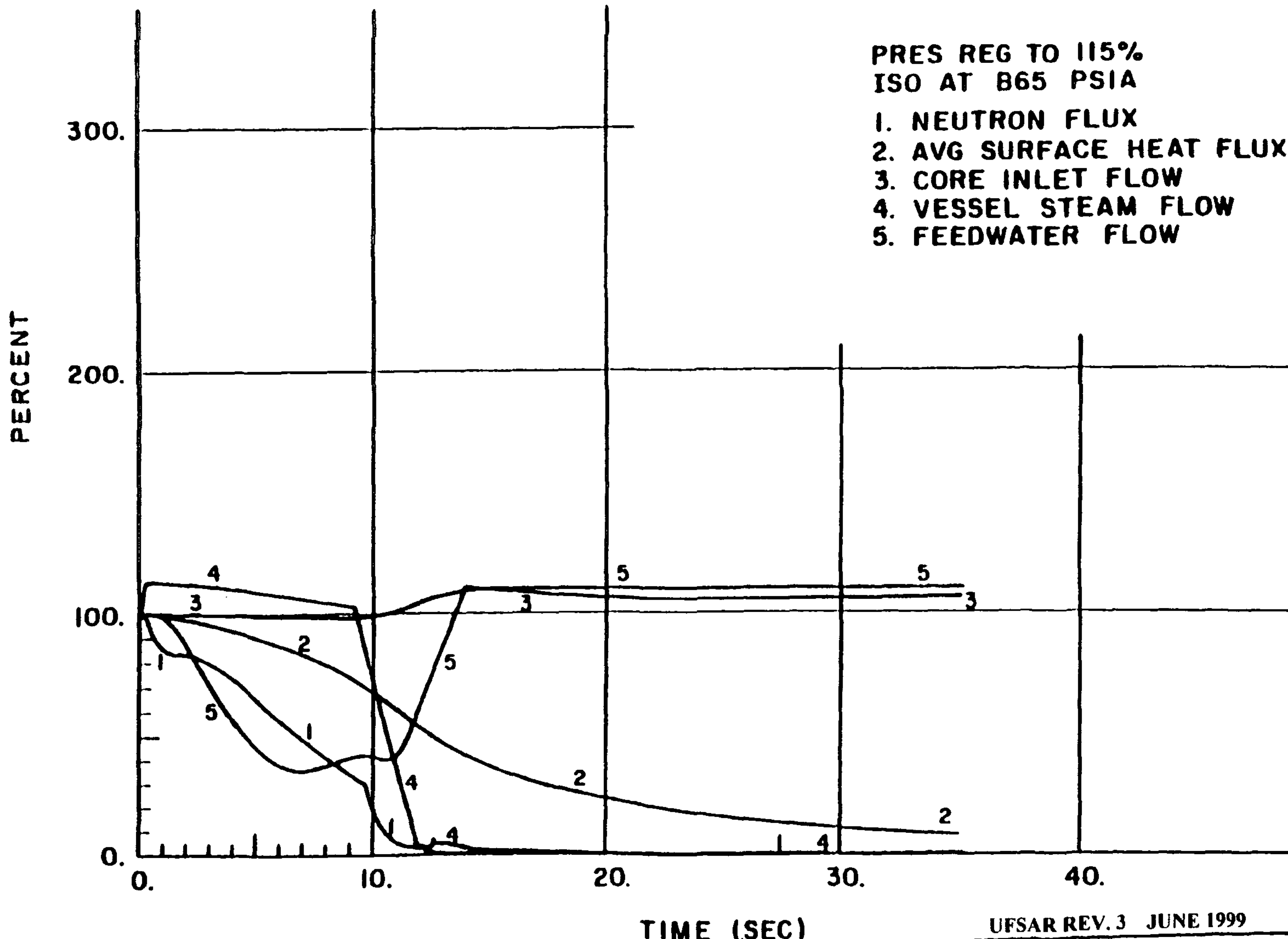


UFSAR REV. 3 JUNE 1999

DRESDEN STATION
UNITS 2 & 3

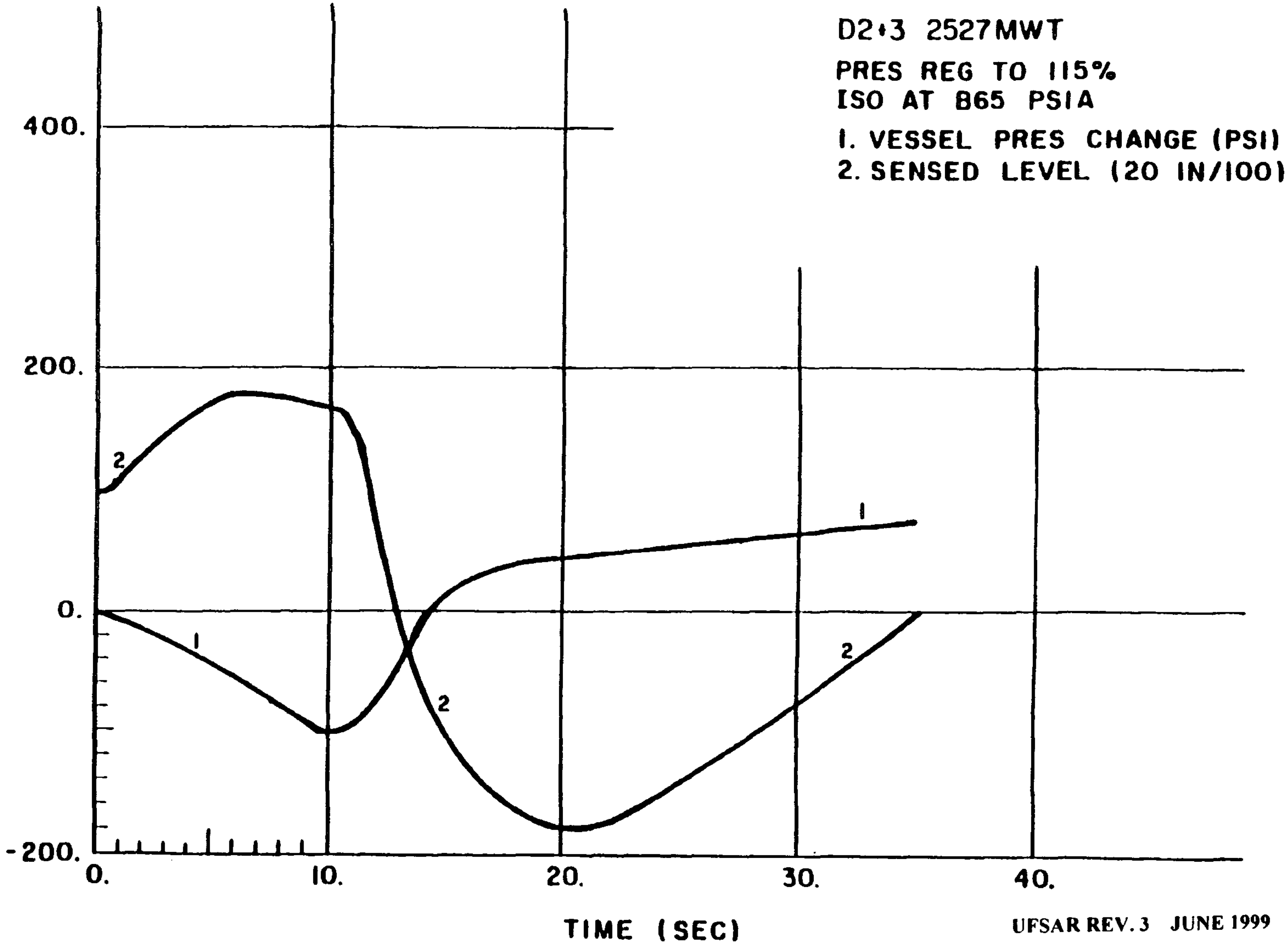
INCREASE IN FEEDWATER FLOW
(CPR, TYPICAL)

FIGURE 15.1-6



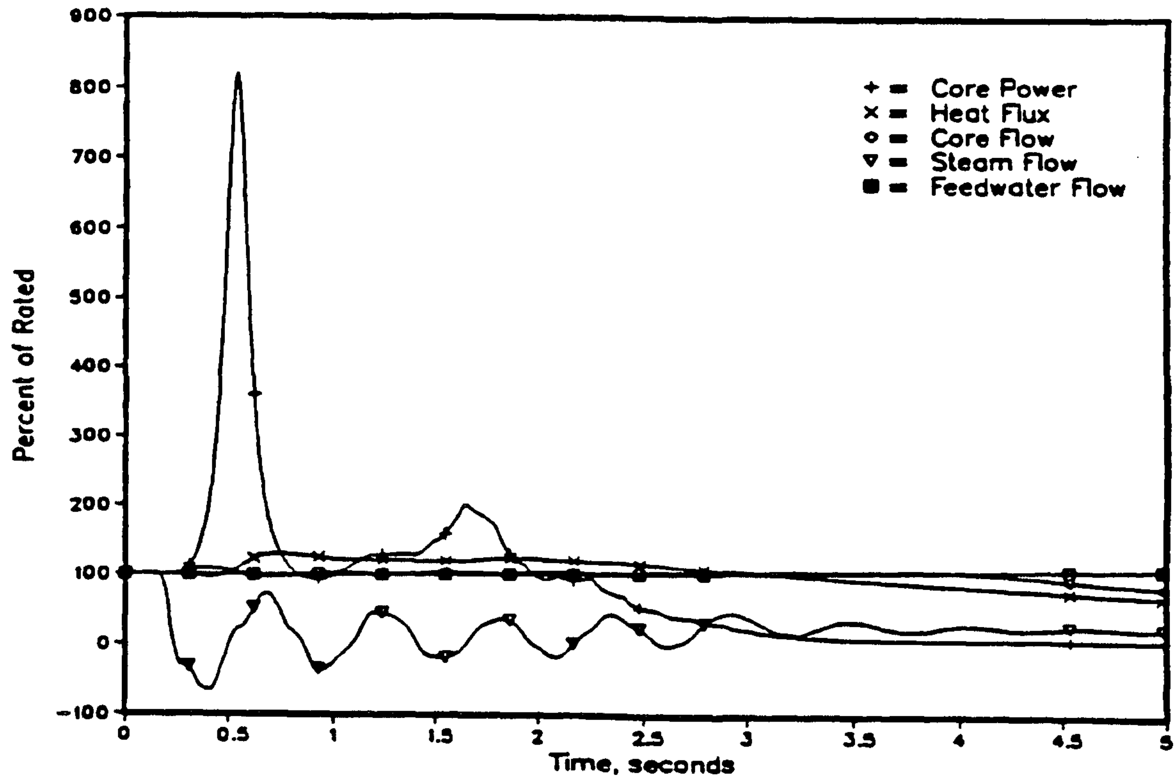
UFSAR REV. 3 JUNE 1999

DRESDEN STATION UNITS 2 & 3
FAILURE OF PRESSURE REGULATOR IN DIRECTION OF MAXIMUM OUTPUT (115%) (POWER AND FLOWS, TYPICAL)
FIGURE 15.1-7



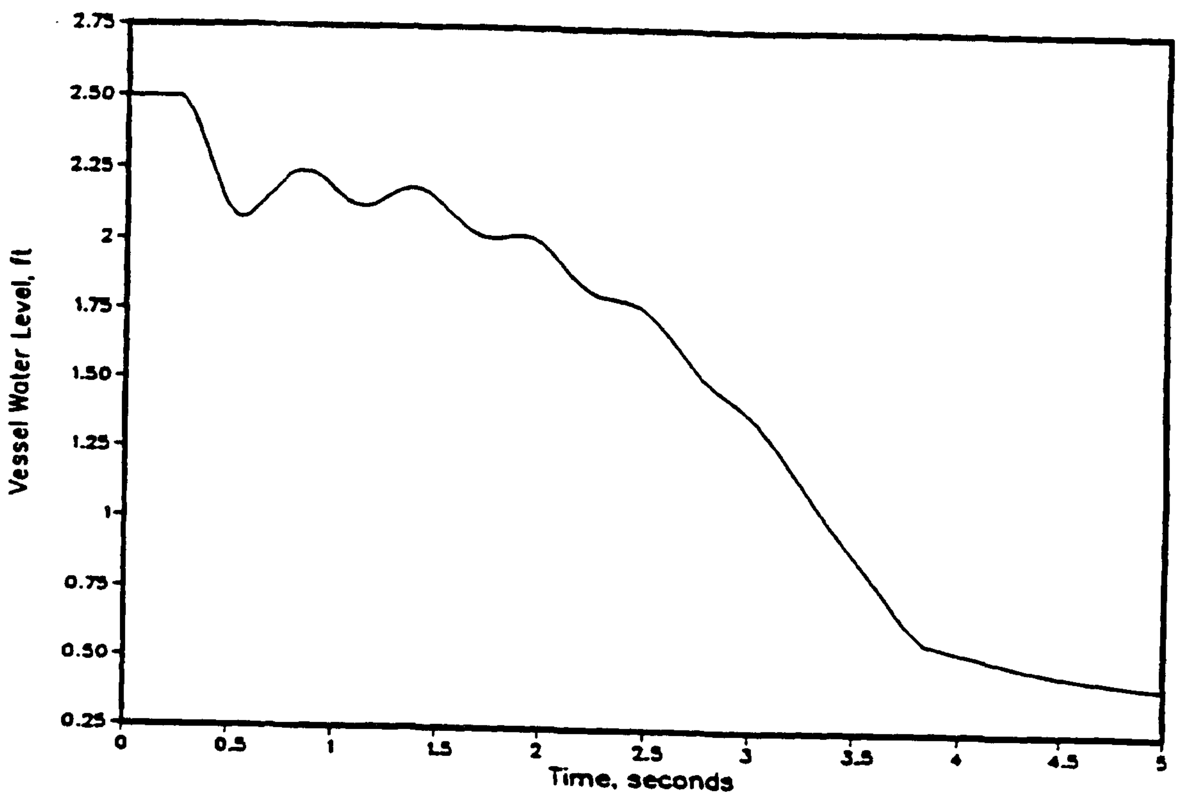
UFSAR REV. 3 JUNE 1999

DRESDEN STATION UNITS 2 & 3
FAILURE OF PRESSURE REGULATOR IN DIRECTION OF MAXIMUM OUTPUT (115%) (VESSEL PRESSURE AND LEVEL, TYPICAL)
FIGURE 15.1-8



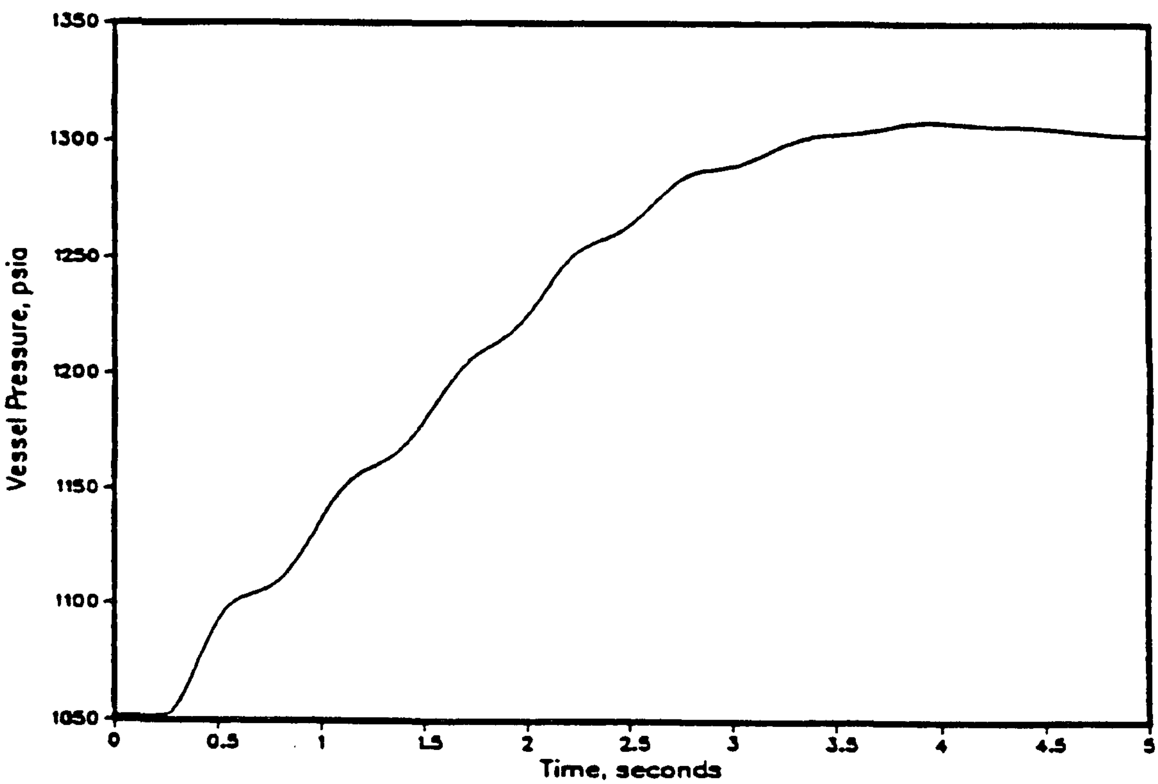
UFSAR REV. 3 JUNE 1999

DRESDEN STATION UNITS 2 & 3
GENERATOR LOAD REJECTION WITHOUT BYPASS AT 100/100-KEY PARAMETERS, TYPICAL
FIGURE 15.2-1



UFSAR REV.3 JUNE 1999

DRESDEN STATION UNITS 2 & 3
GENERATOR LOAD REJECTION WITHOUT BYPASS AT 100/100-VESSEL LEVEL, TYPICAL
FIGURE 15.2-2

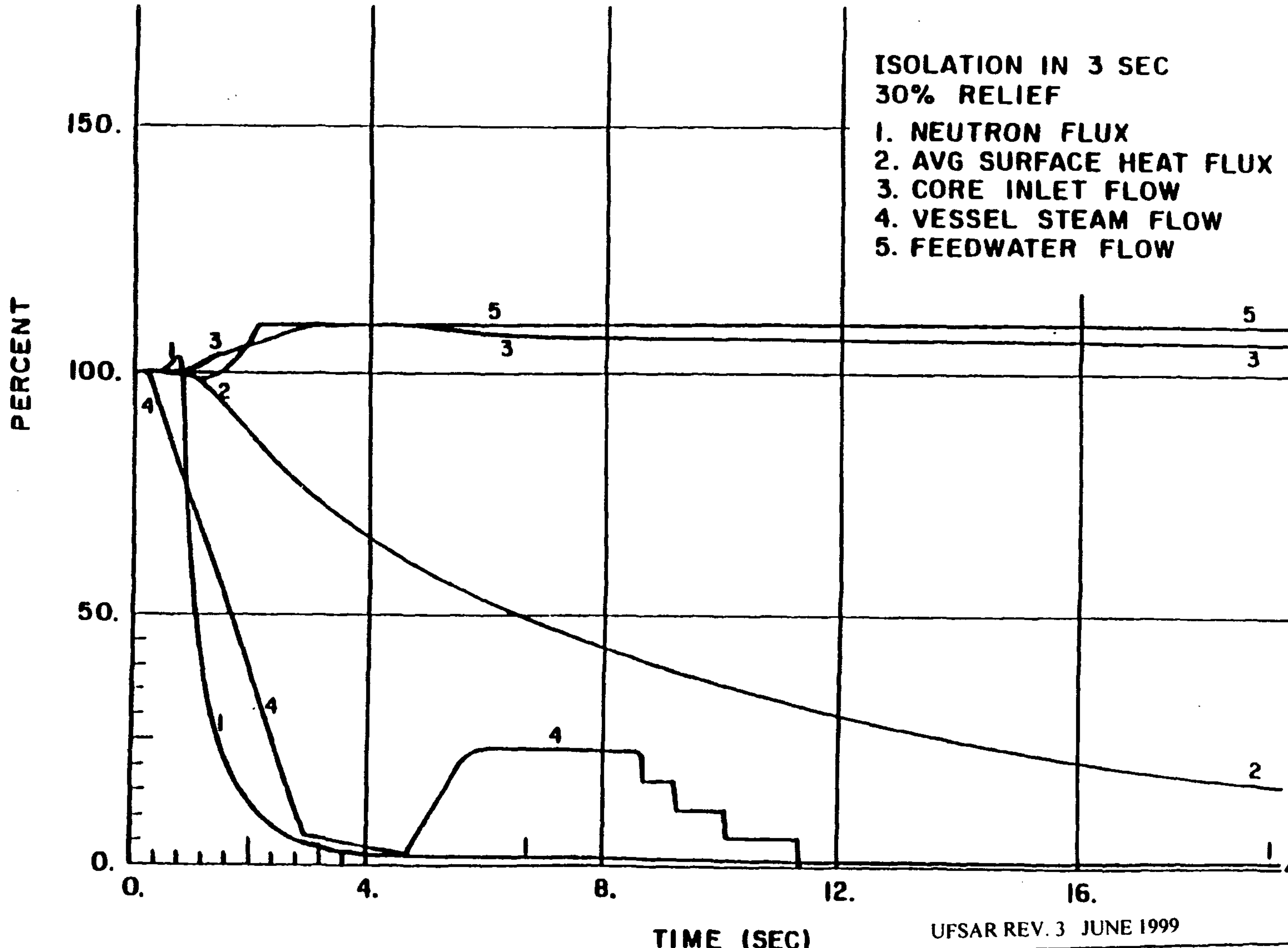


UFSAR REV.3 JUNE 1999

DRESDEN STATION
UNITS 2 & 3

GENERATOR LOAD REJECTION WITHOUT
BYPASS AT 100/100-VESSEL PRESSURE, TYPICAL

FIGURE 15.2-3

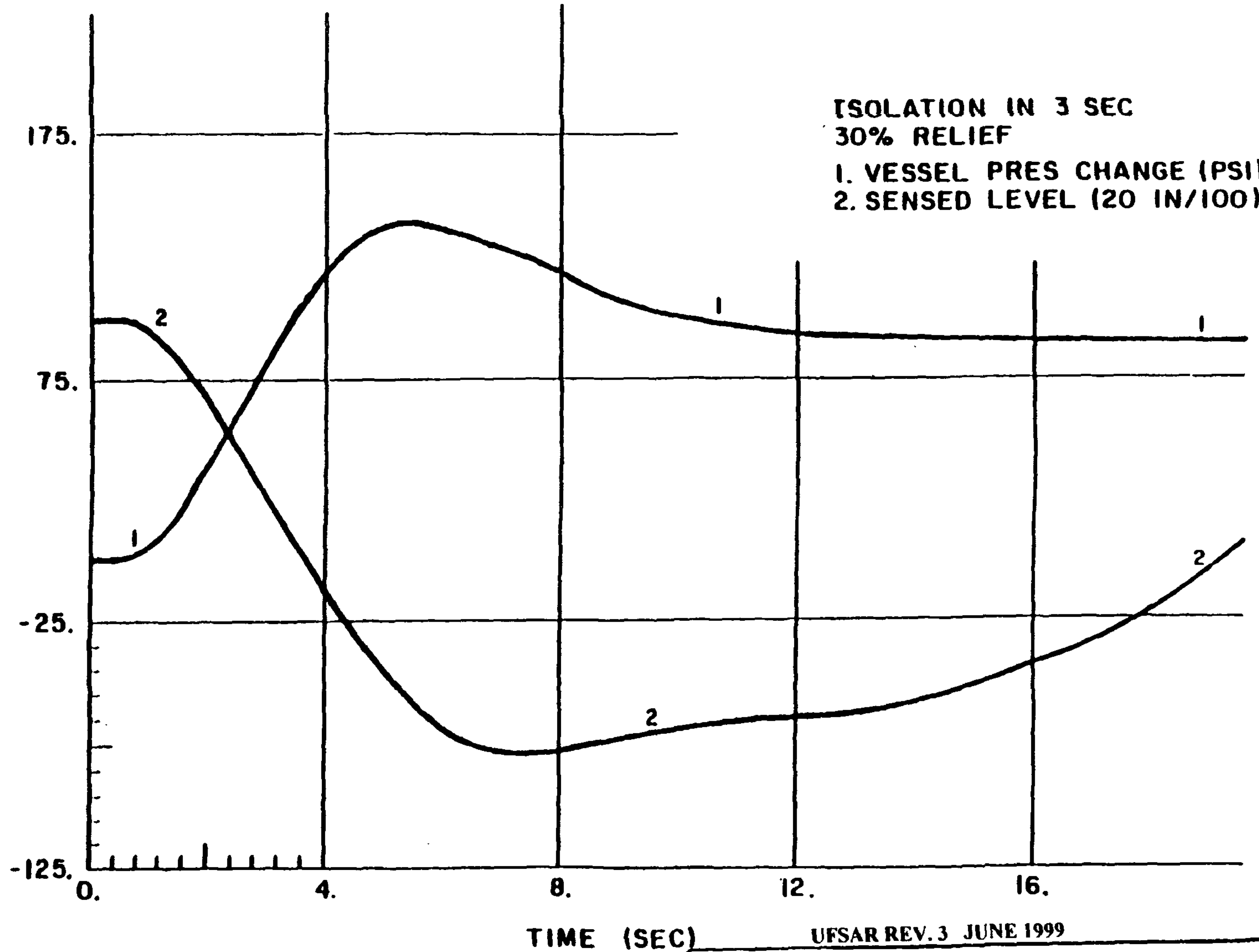


UFSAR REV. 3 JUNE 1999

DRESDEN STATION
UNITS 2 & 3

INADVERTENT MSIV CLOSURE WITH DIRECT SCRAM (POWER AND FLOW), TYPICAL

FIGURE 15.2-4

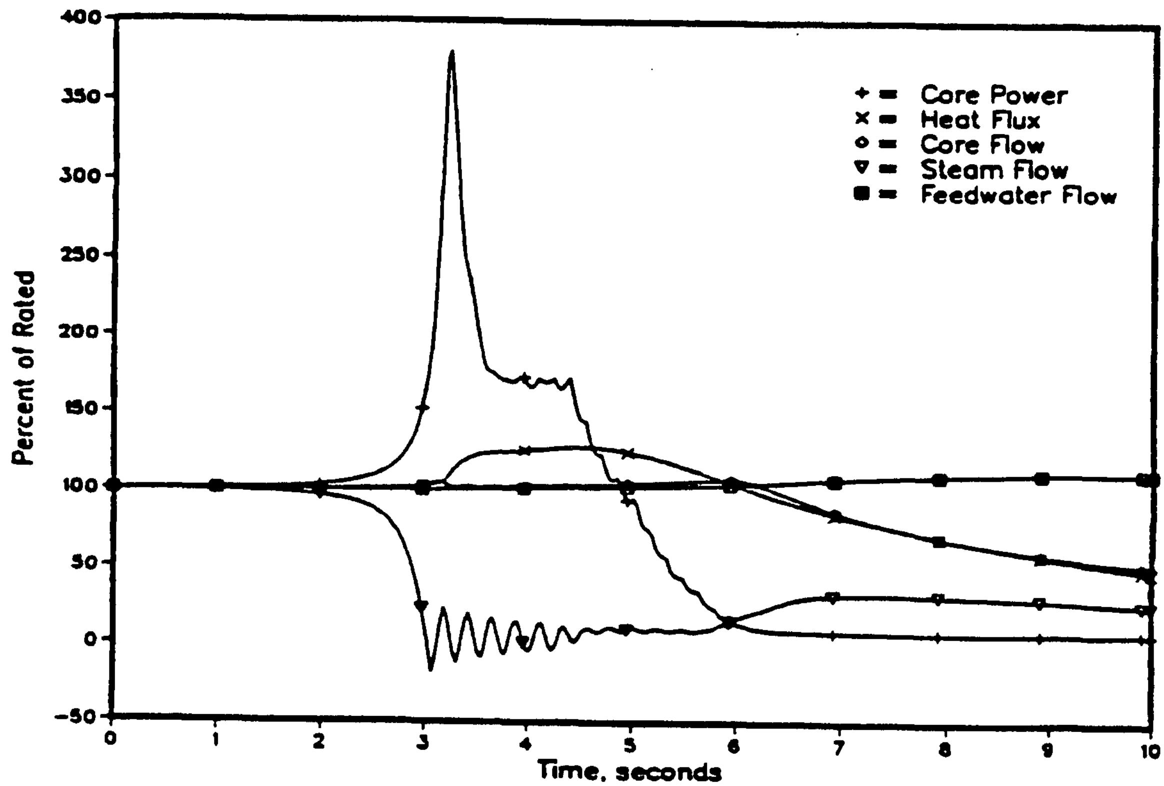


UFSAR REV. 3 JUNE 1999

DRESDEN STATION
UNITS 2 & 3

INADVERTENT MSIV CLOSURE WITH DIRECT SCRAM (PRESSURE AND LEVEL), TYPICAL

FIGURE 15.2-5

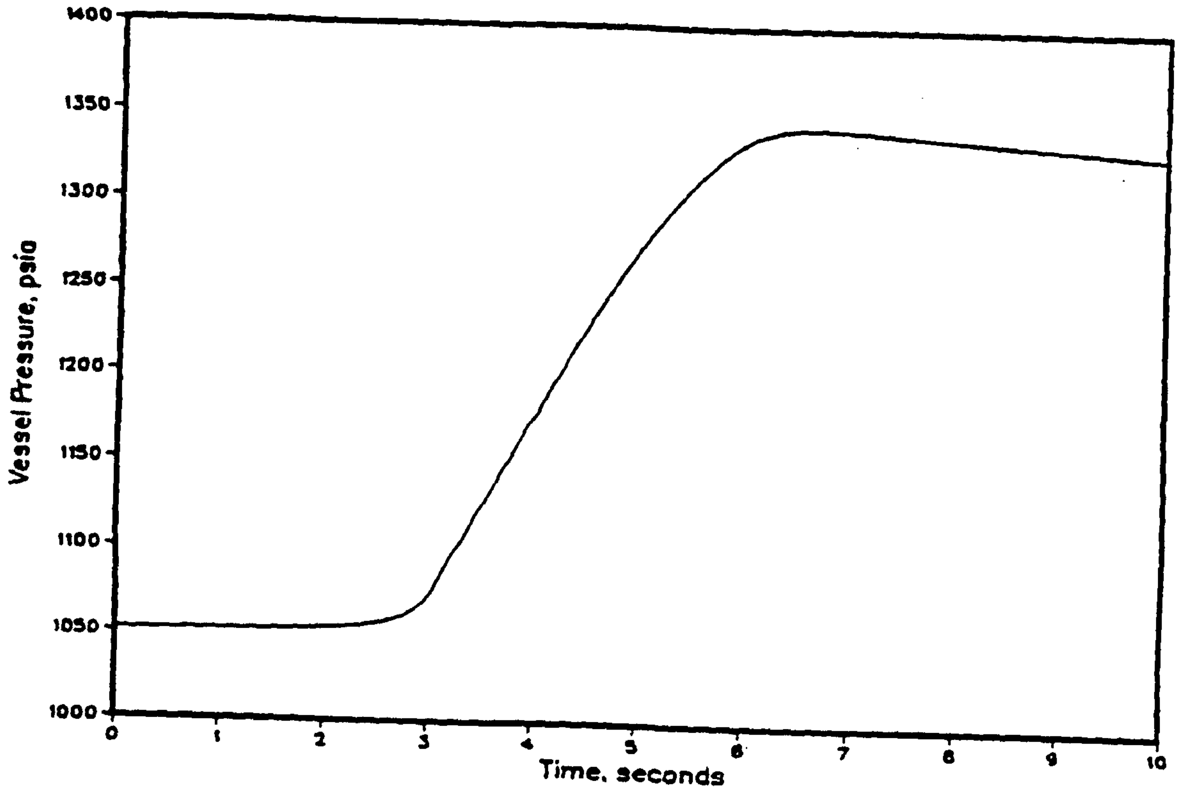


UFSAR REV. 3 JUNE 1999

DRESDEN STATION
UNITS 2 & 3

INADVERTENT MSIV CLOSURE WITHOUT
DIRECT SCRAM AT 100/100-KEY PARAMETERS, TYPICAL

FIGURE 15.2-6

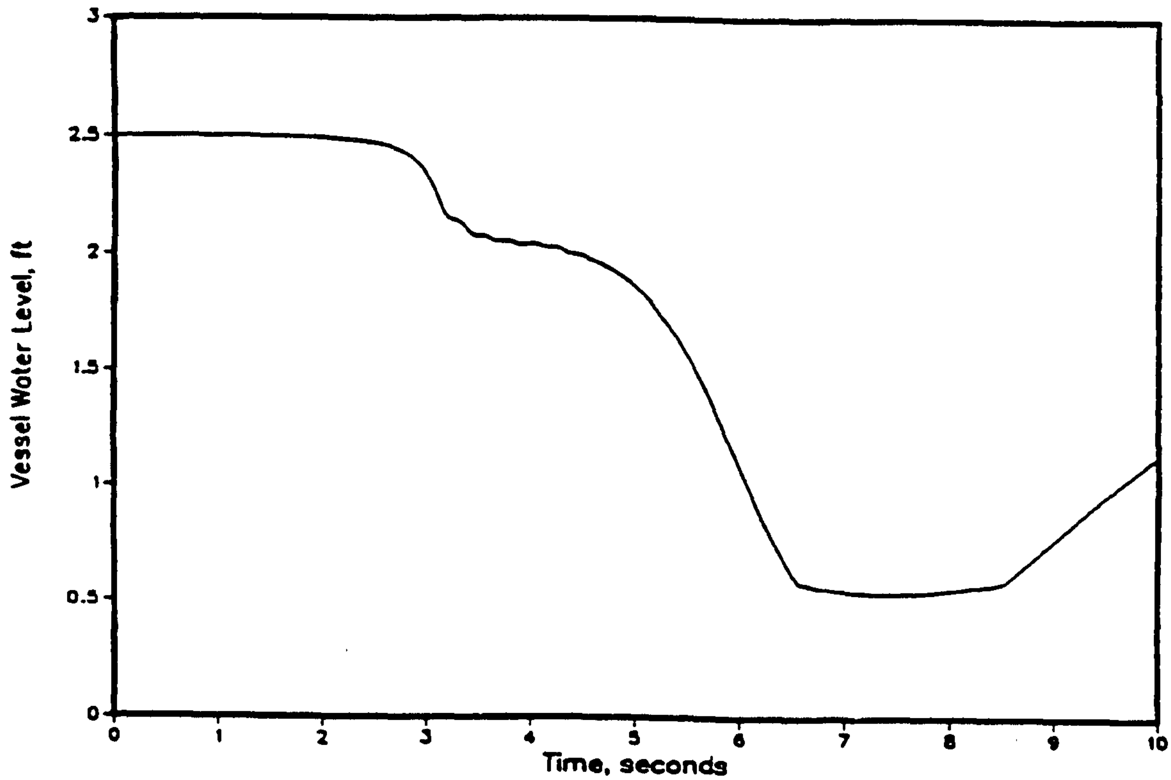


UFSAR REV. 3 JUNE 1999

DRESDEN STATION
UNITS 2 & 3

INADVERTENT MSIV CLOSURE WITHOUT
DIRECT SCRAM AT 100/100-
VESSEL PRESSURE RESPONSE, TYPICAL

FIGURE 15.2-7

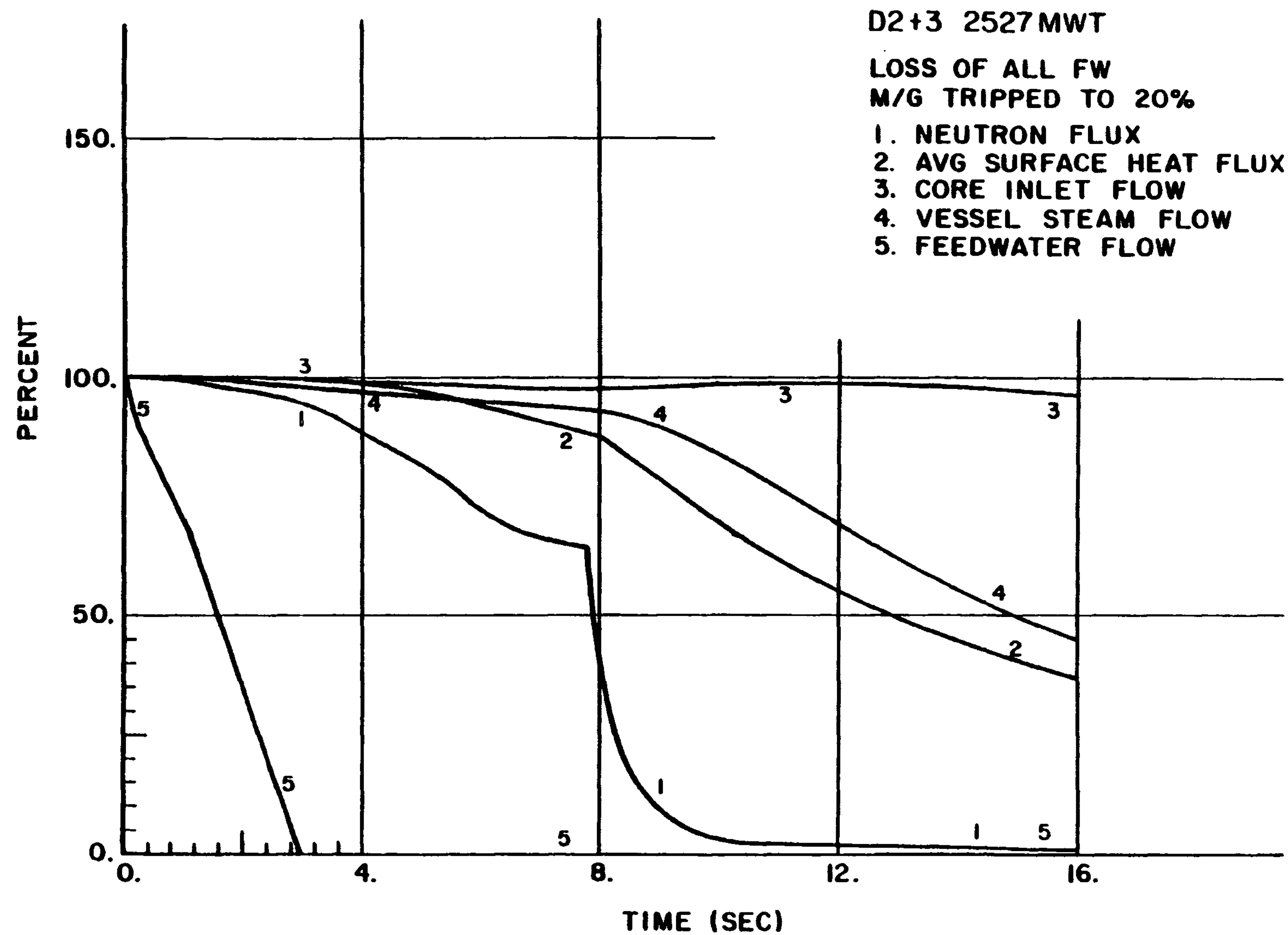


UFSAR REV. 3 JUNE 1999

DRESDEN STATION
UNITS 2 & 3

INADVERTENT MSIV CLOSURE WITHOUT DIRECT SCRAM AT 100/100-
VESSEL WATER LEVEL, TYPICAL

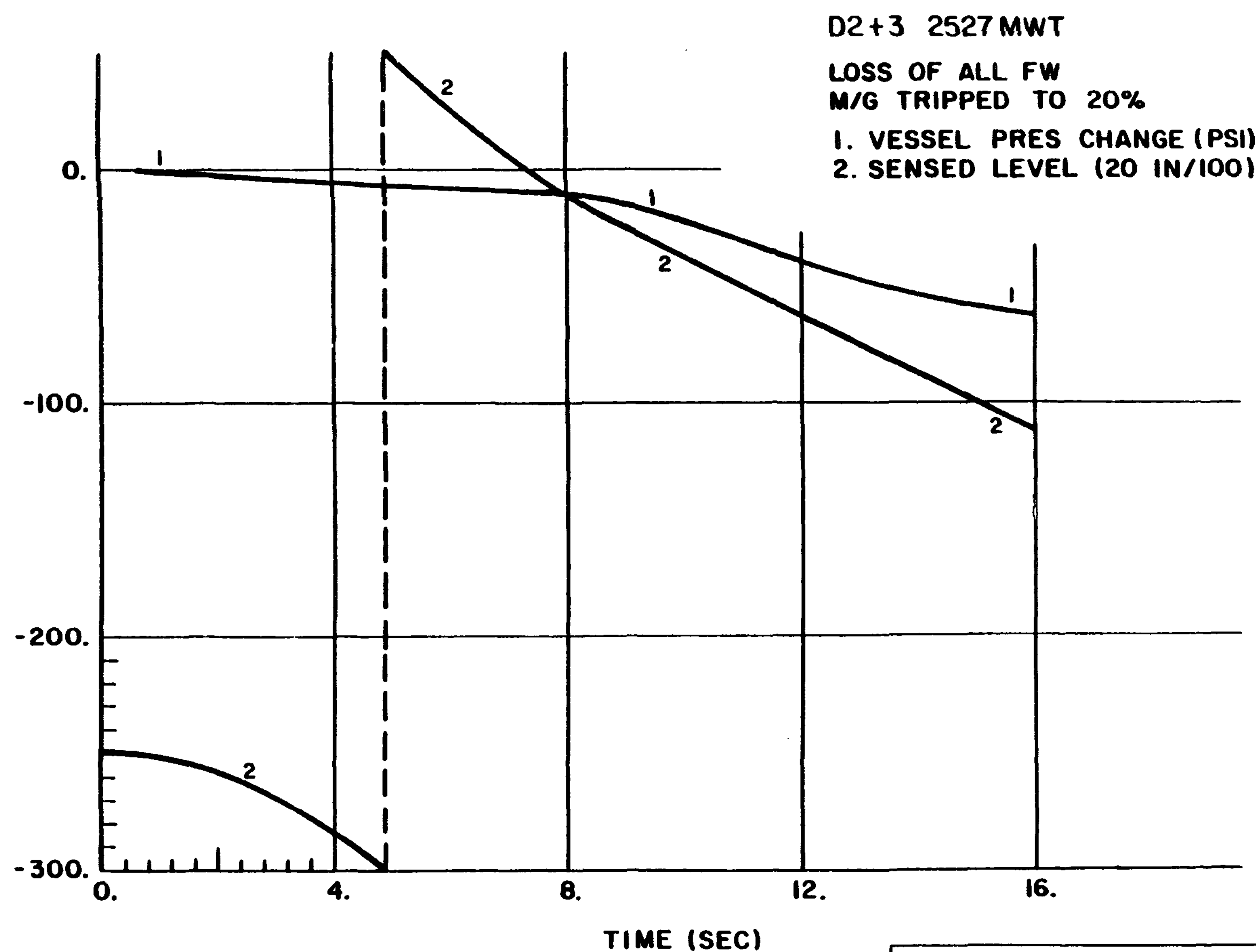
FIGURE 15.2-8



DRESDEN STATION
UNITS 2 & 3

LOSS OF FEEDWATER FLOW
(POWER AND FLOW)

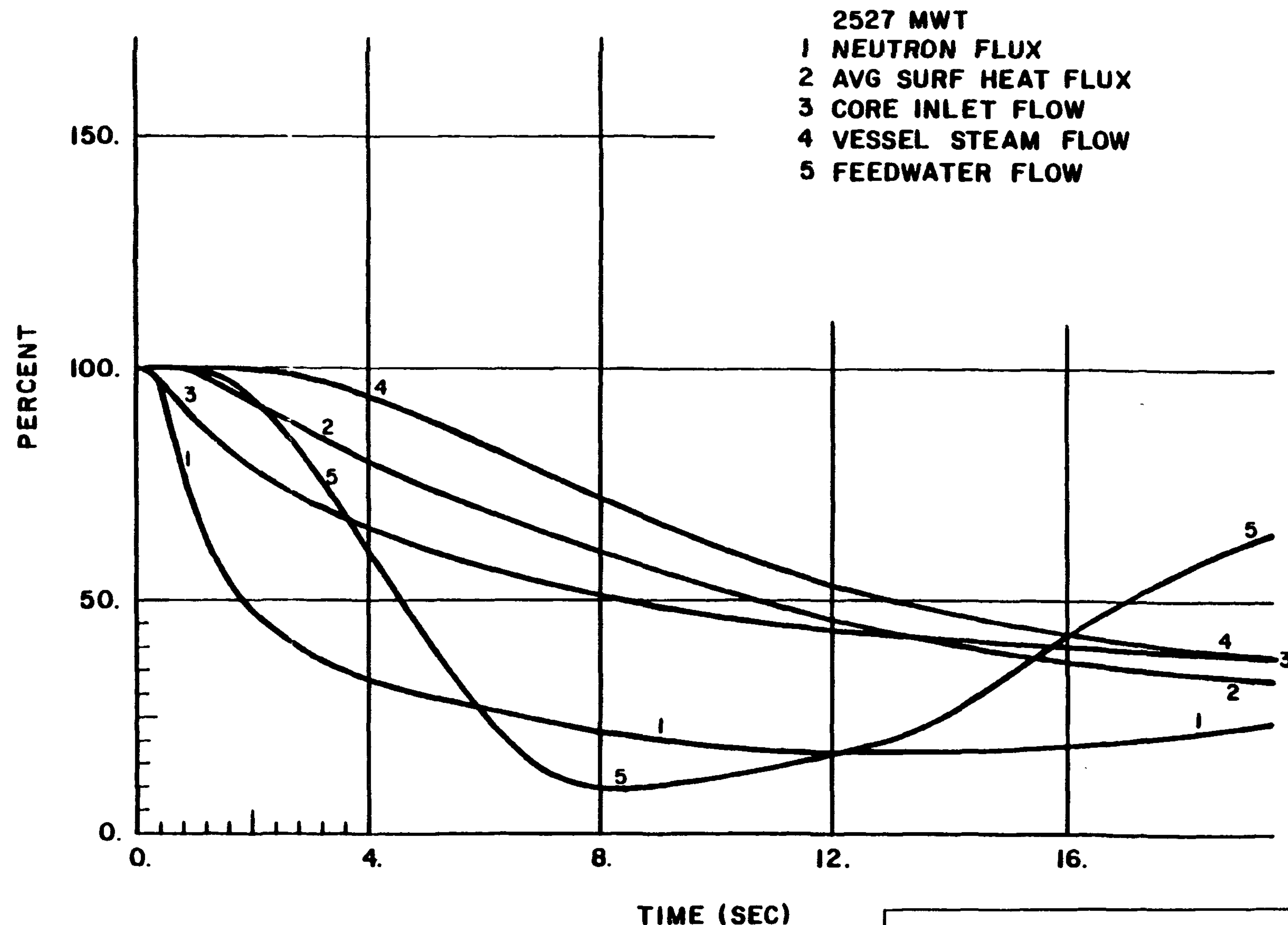
FIGURE 15.2-9



DRESDEN STATION
UNITS 2 & 3

LOSS OF FEEDWATER FLOW
(PRESSURE AND LEVEL)

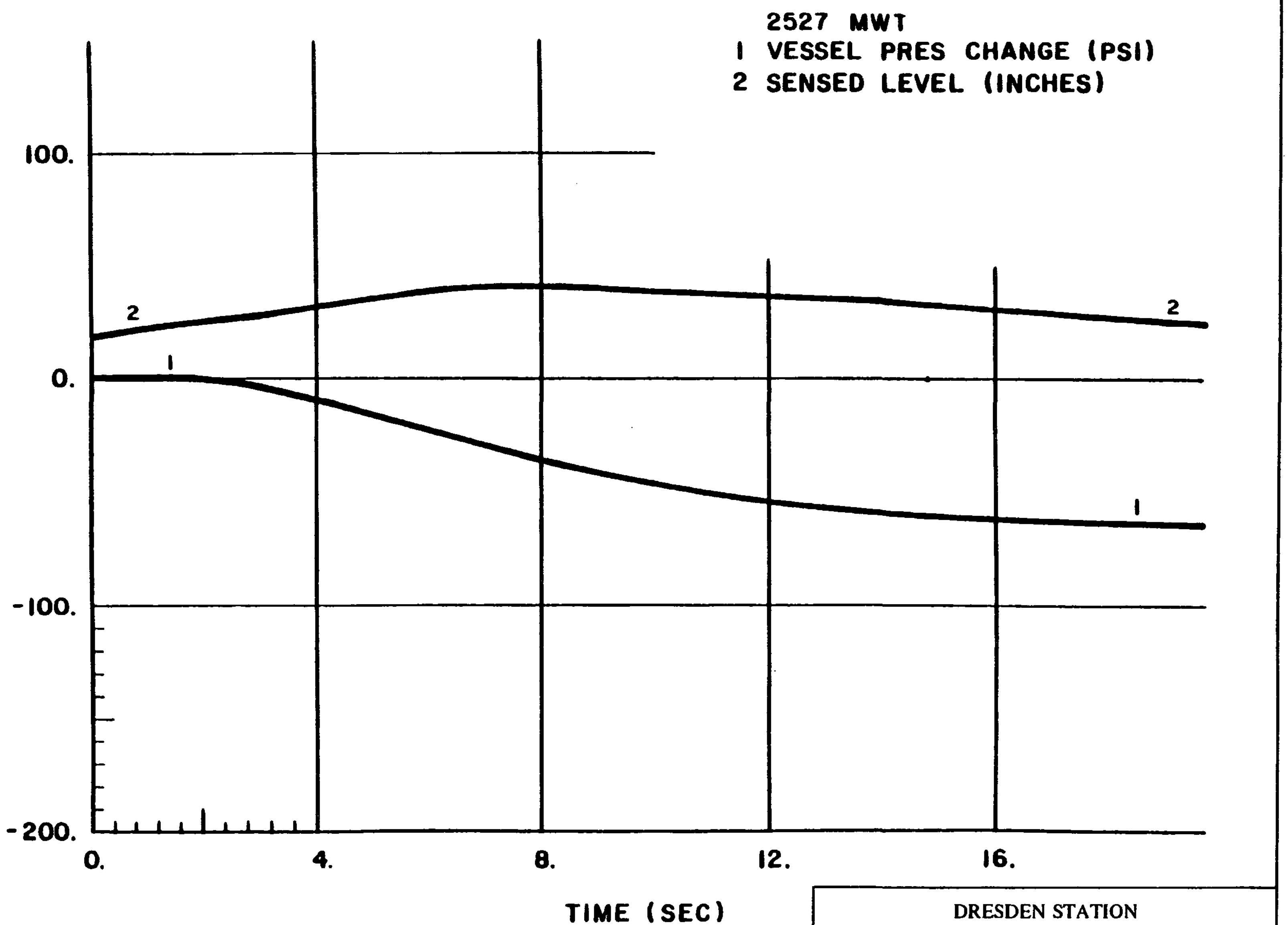
FIGURE 15.2-10



DRESDEN STATION
UNITS 2 & 3

TRIP OF TWO RECIRCULATION
M-G SET DRIVE MOTOR BREAKERS
TRANSIENT ANALYSIS

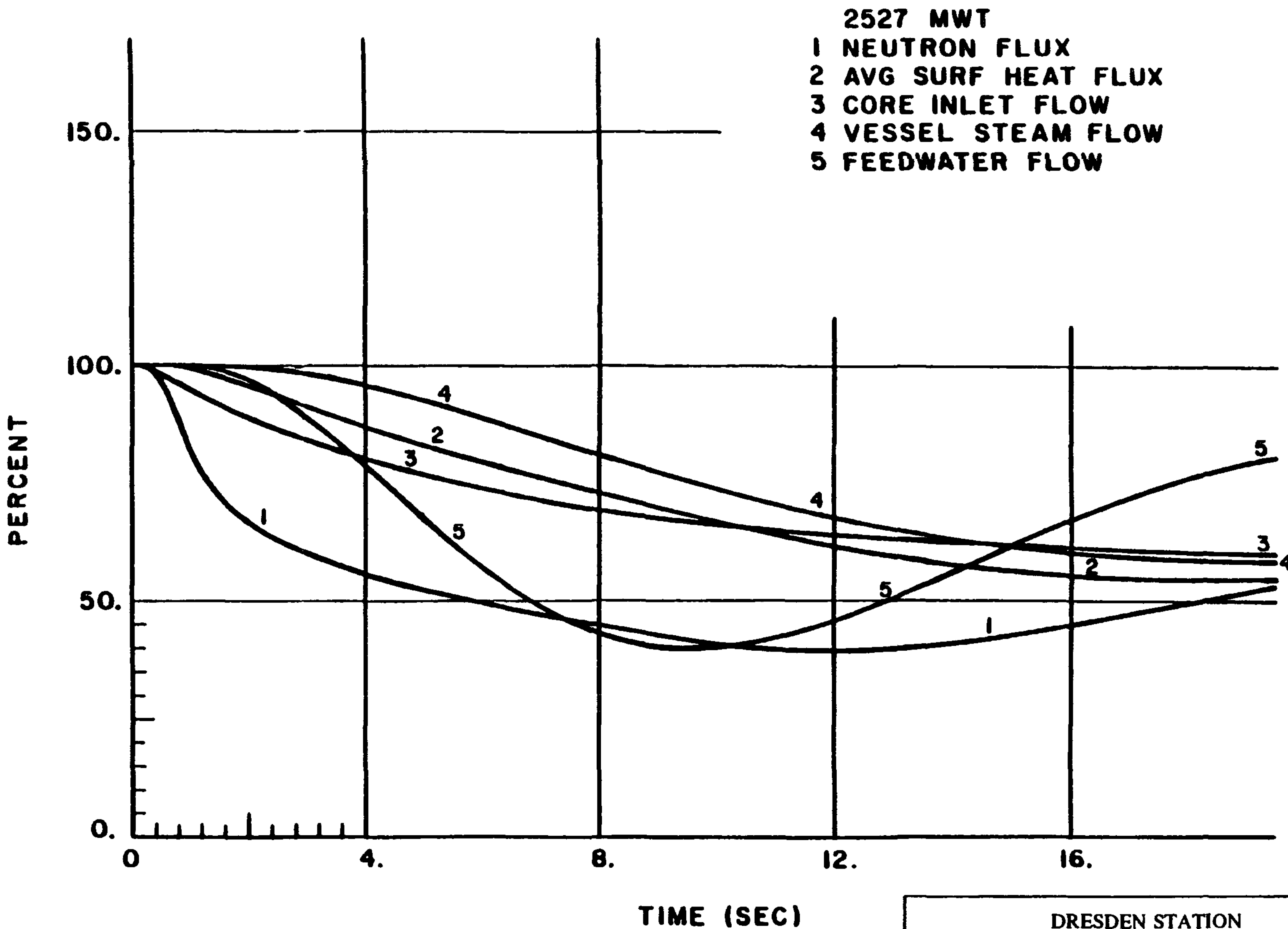
FIGURE 15.3-1



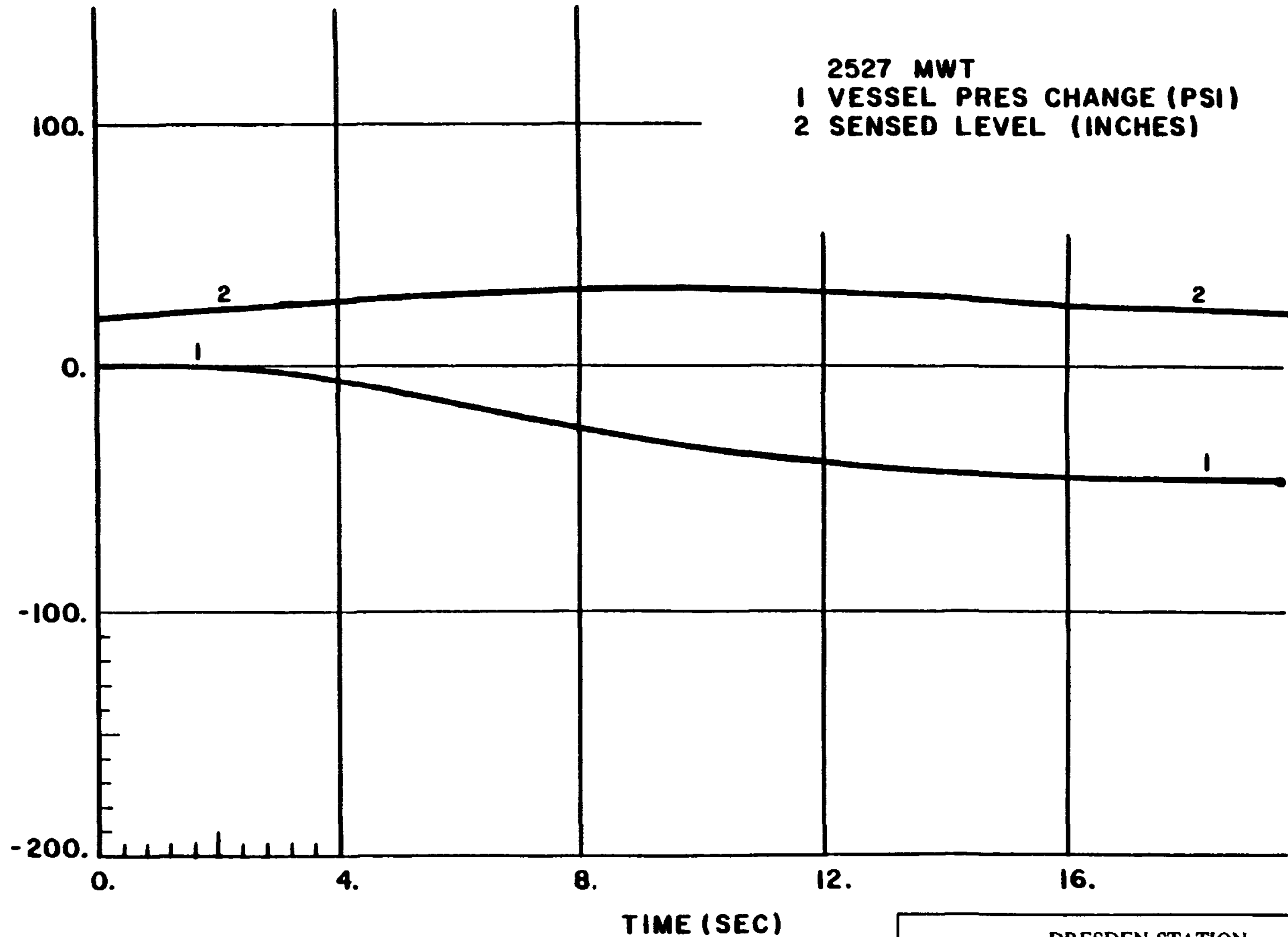
DRESDEN STATION
UNITS 2 & 3

TRIP OF TWO RECIRCULATION
M-G SET DRIVE MOTOR BREAKERS
TRANSIENT ANALYSIS

FIGURE 15.3-2



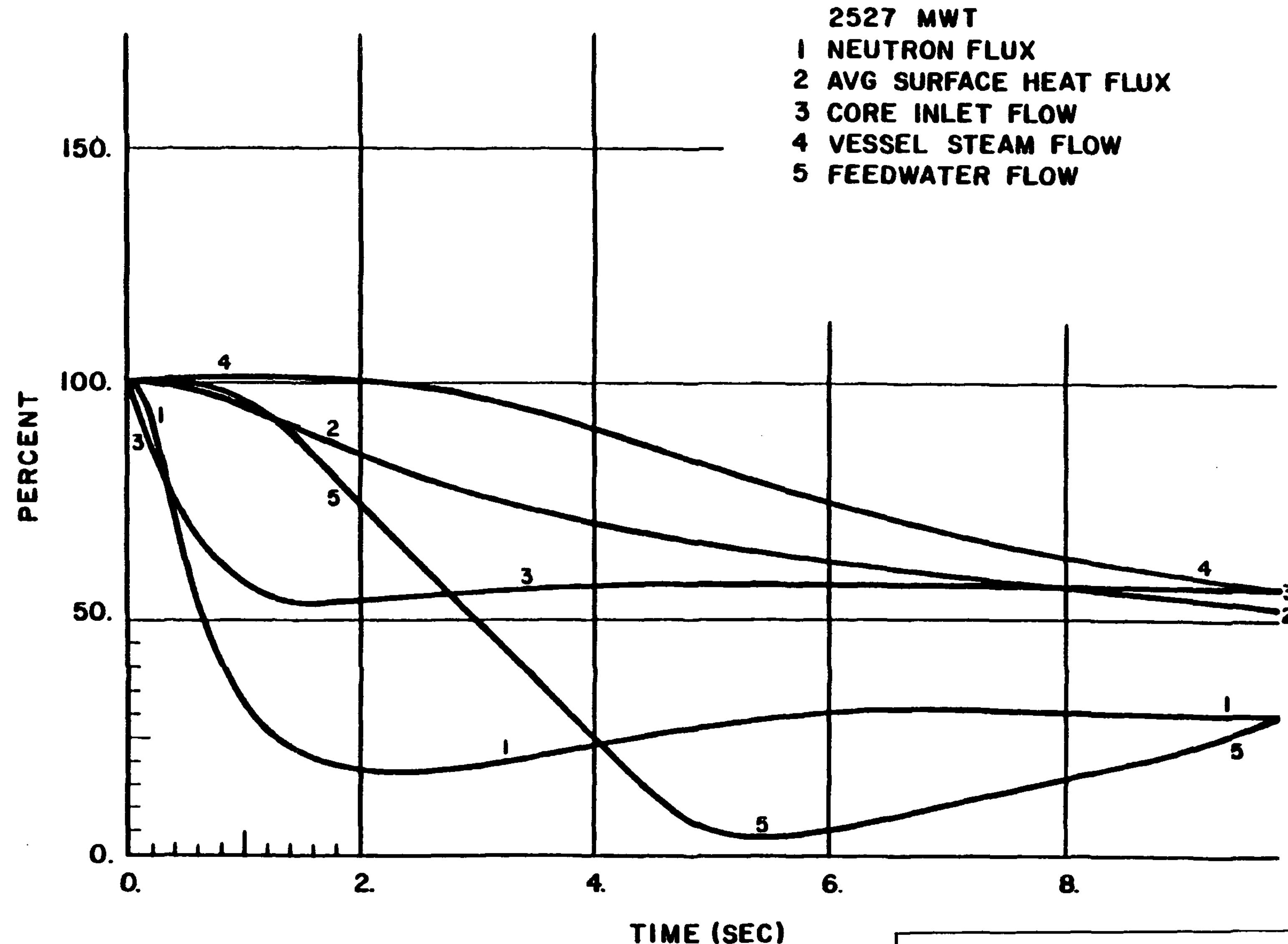
DRESDEN STATION UNITS 2 & 3
TRIP OF ONE RECIRCULATION M-G SET DRIVE MOTOR BREAKER TRANSIENT ANALYSIS
FIGURE 15.3-3



DRESDEN STATION
UNITS 2 & 3

TRIP OF ONE RECIRCULATION
M-G SET DRIVE MOTOR BREAKER
TRANSIENT ANALYSIS

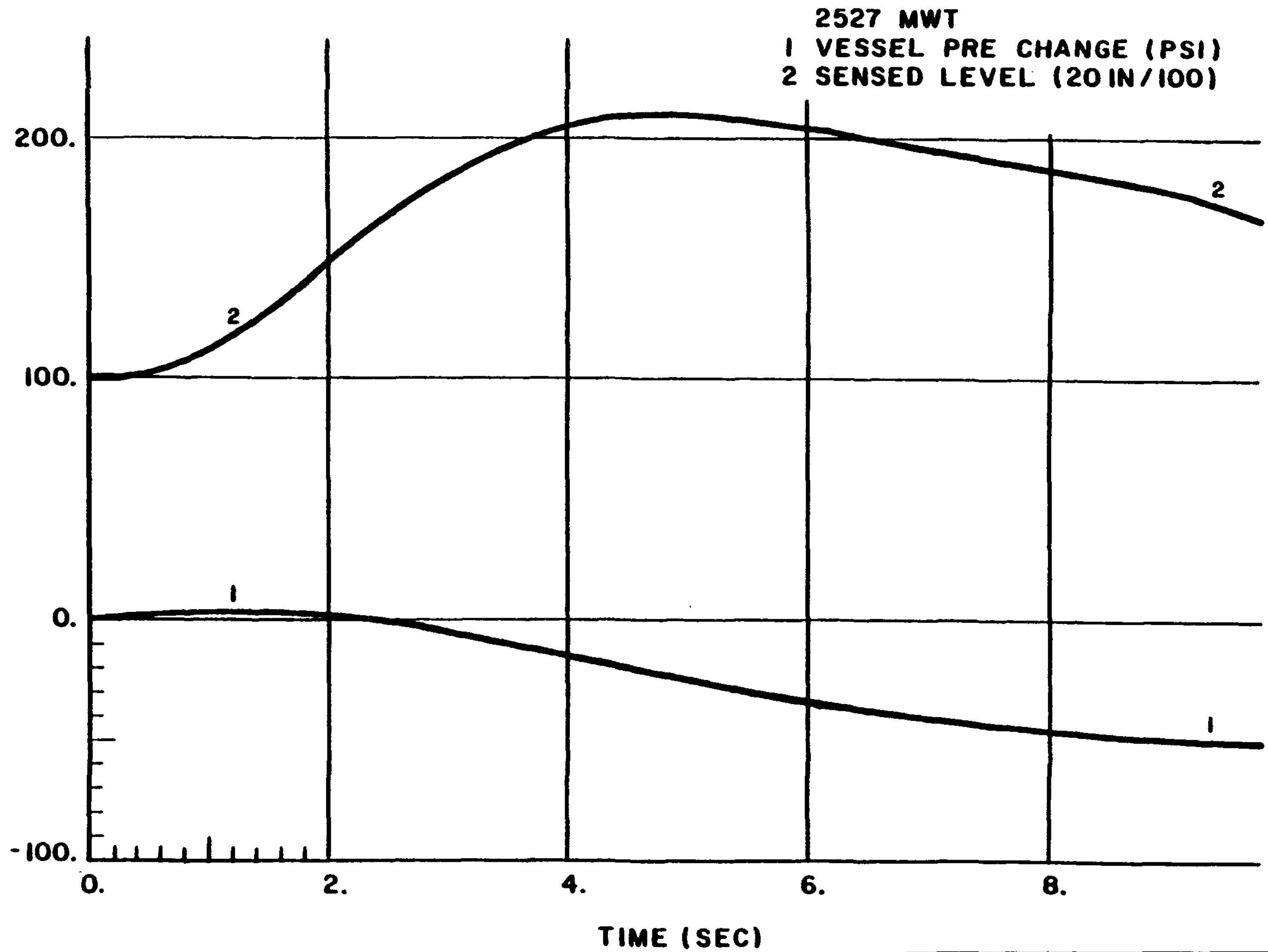
FIGURE 15.3-4



DRESDEN STATION
UNITS 2 & 3

SEIZURE OF ONE RECIRCULATION PUMP
SHAFT TRANSIENT ANALYSIS

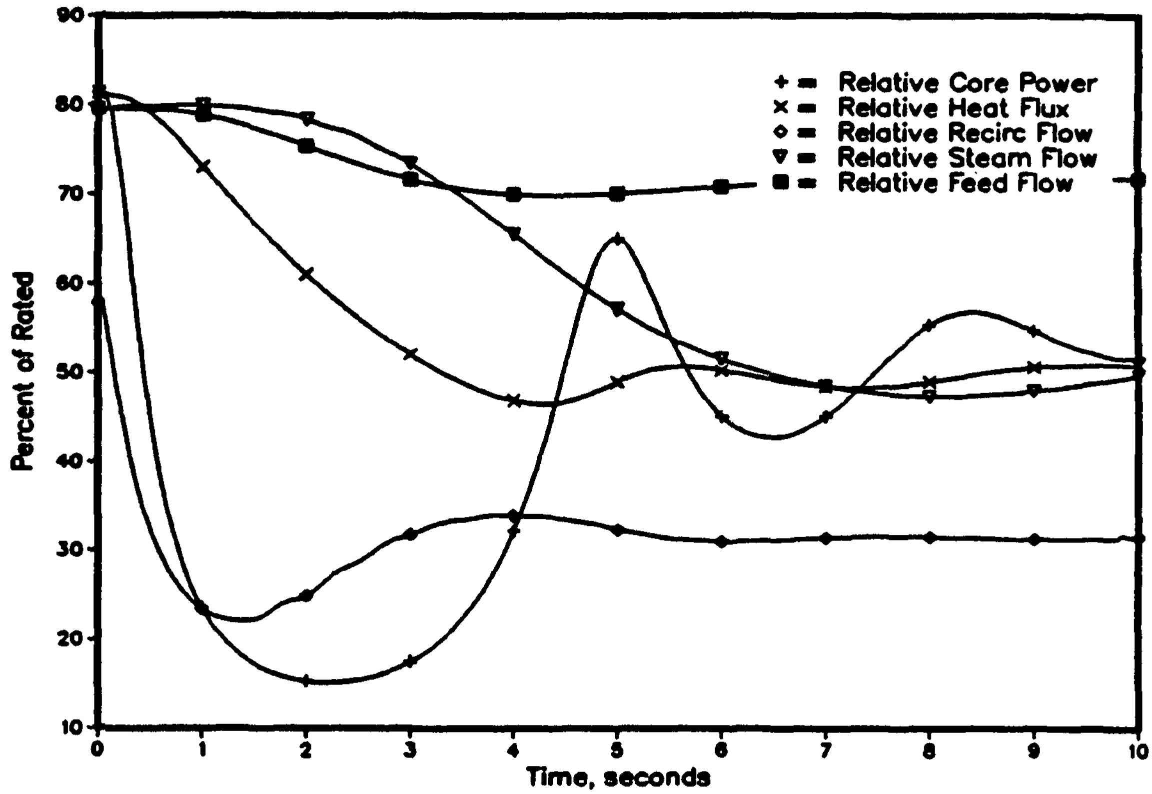
FIGURE 15.3-5



DRESDEN STATION
UNITS 2 & 3

SEIZURE OF ONE RECIRCULATION PUMP
SHAFT TRANSIENT ANALYSIS

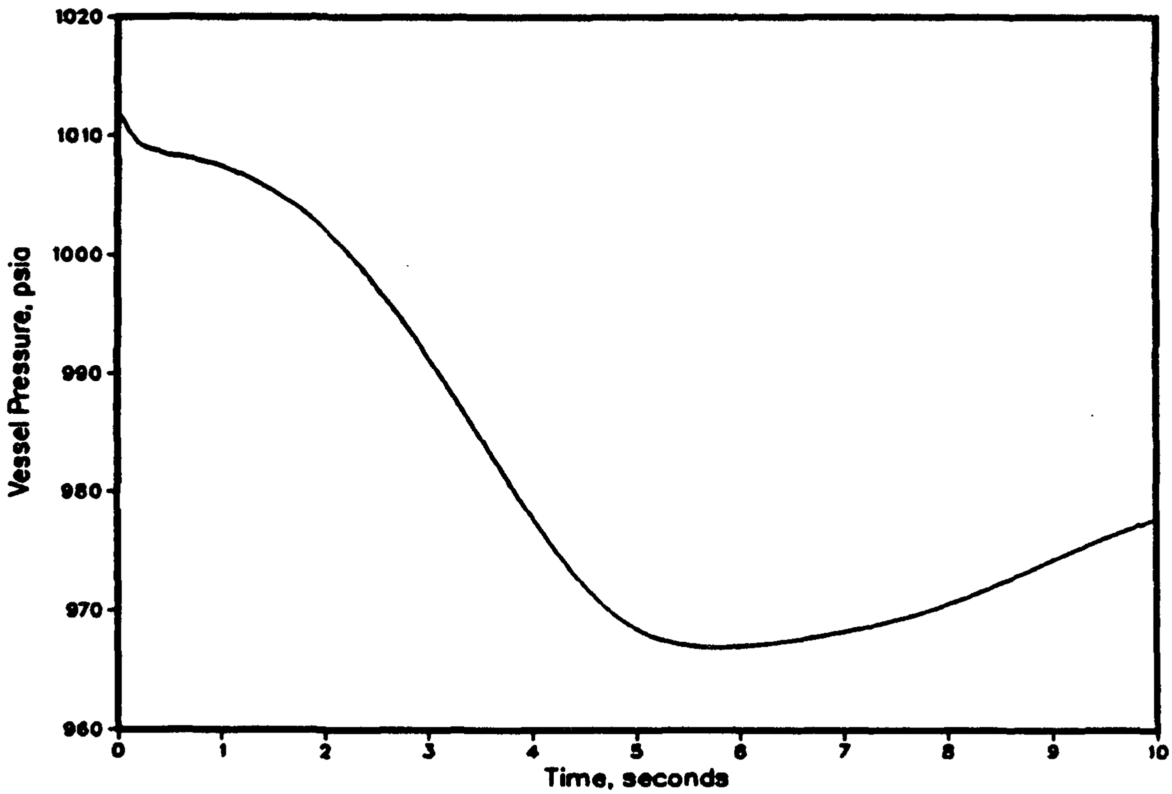
FIGURE 15.3-6



DRESDEN STATION
UNITS 2 & 3

SINGLE-LOOP OPERATION -
PUMP SEIZURE, KEY PARAMETERS

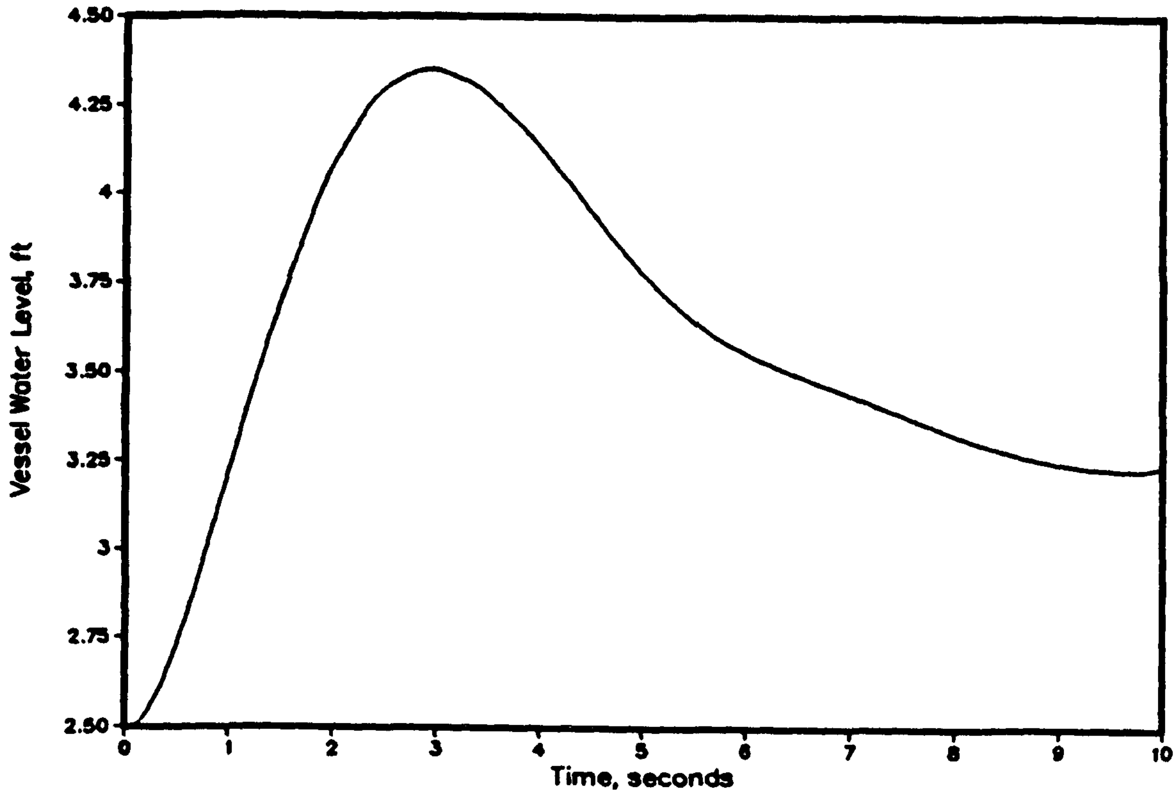
FIGURE 15.3-7



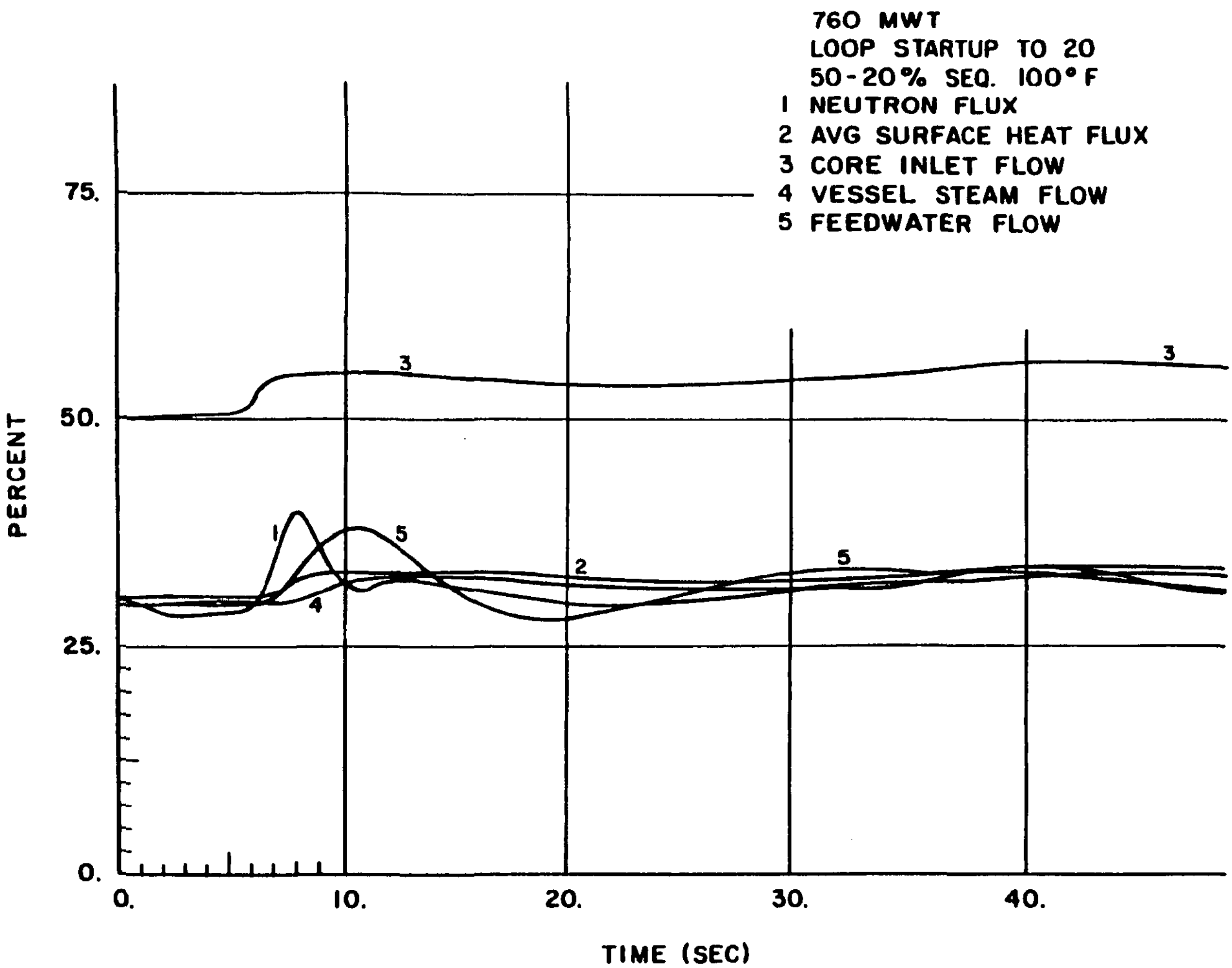
DRESDEN STATION
UNITS 2 & 3

SINGLE-LOOP OPERATION -
PUMP SEIZURE, VESSEL PRESSURE

FIGURE 15.3-8



DRESDEN STATION UNITS 2 & 3
SINGLE-LOOP OPERATION - PUMP SEIZURE, VESSEL LEVEL
FIGURE 15.3-9

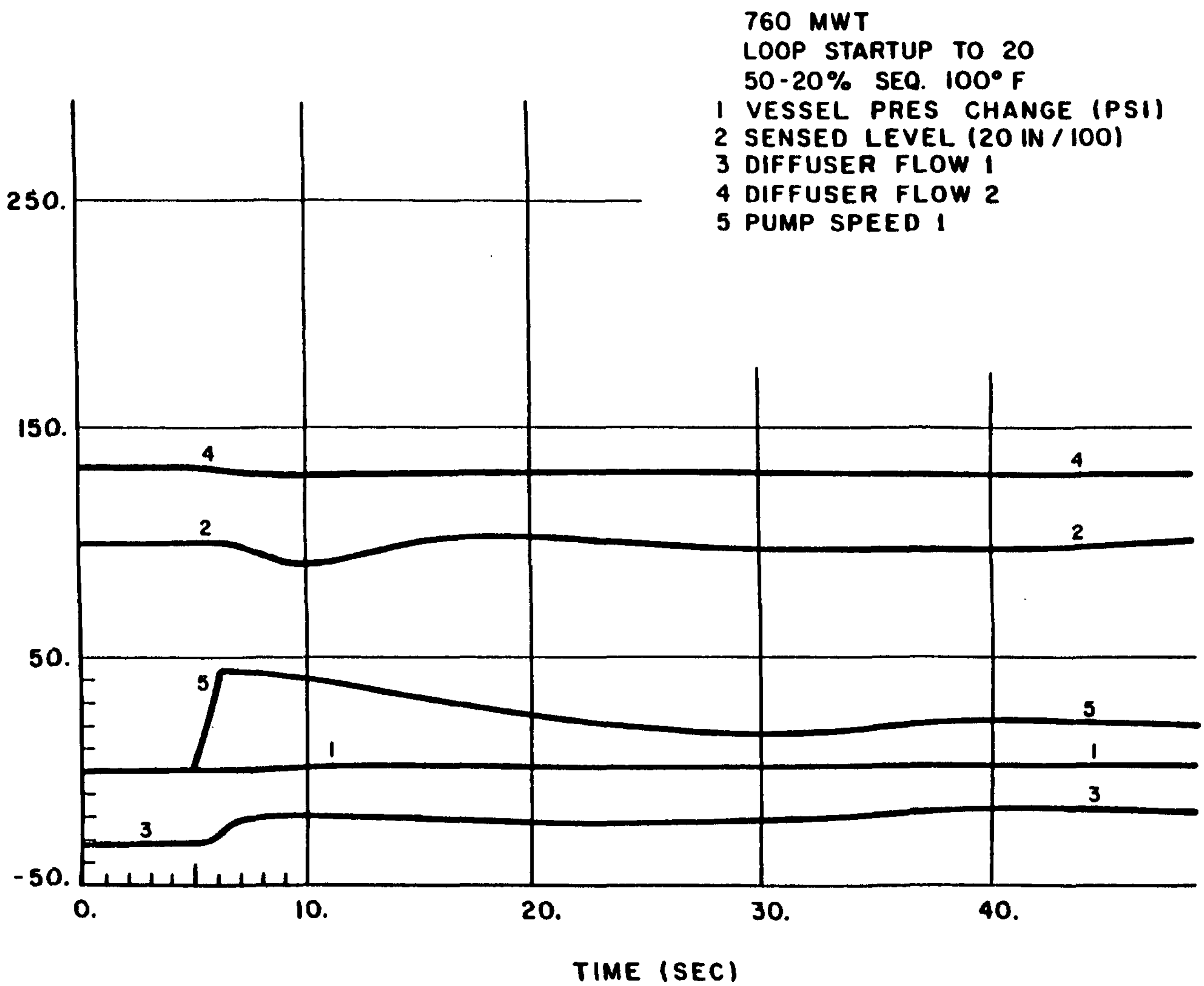


HISTORICAL

This figure contains historical information only.

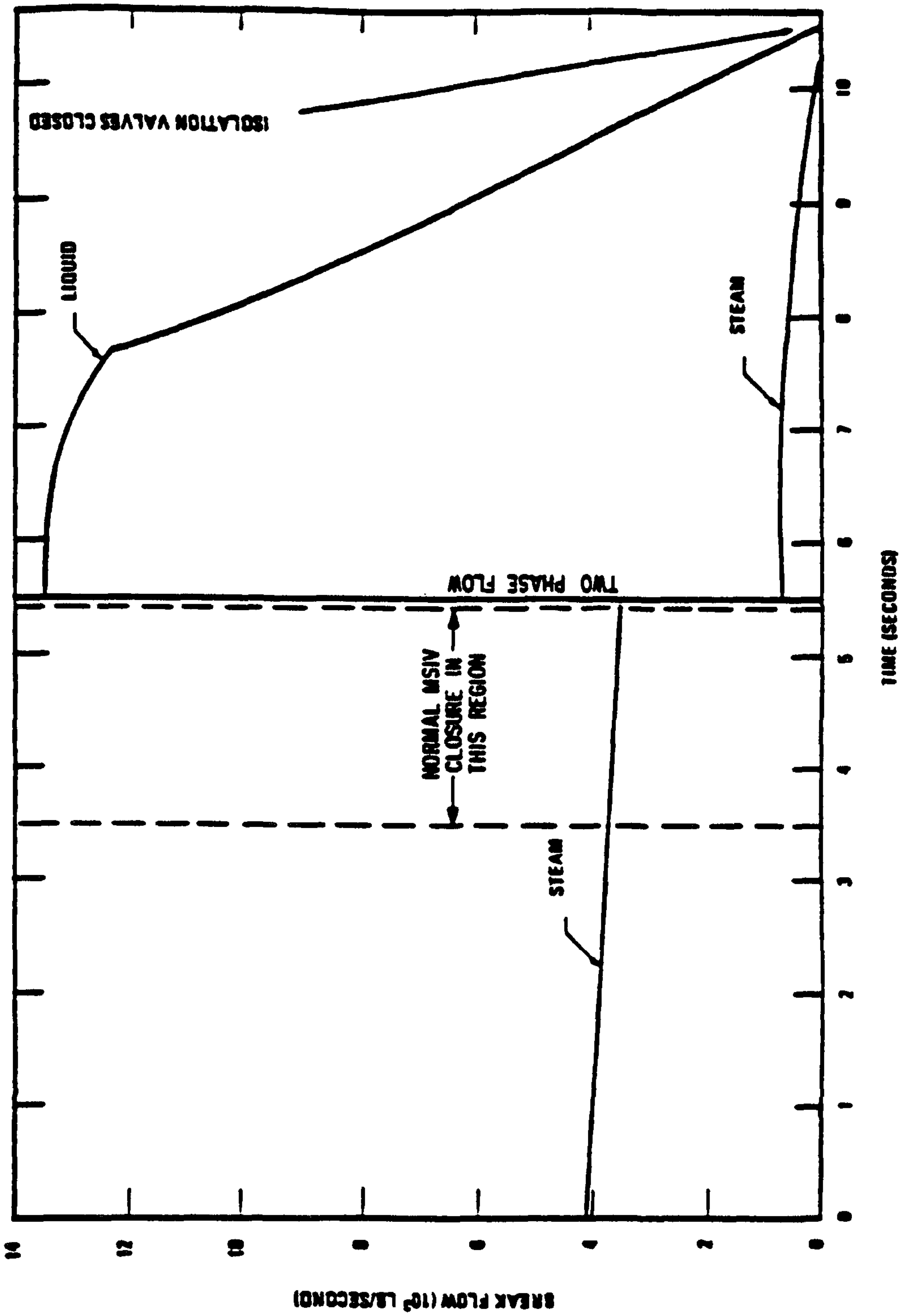
DRESDEN STATION UNITS 2 & 3
COLD LOOP STARTUP TRANSIENT (POWER AND FLOW)
FIGURE 15.4-1

REVISION 5
JANUARY 2003

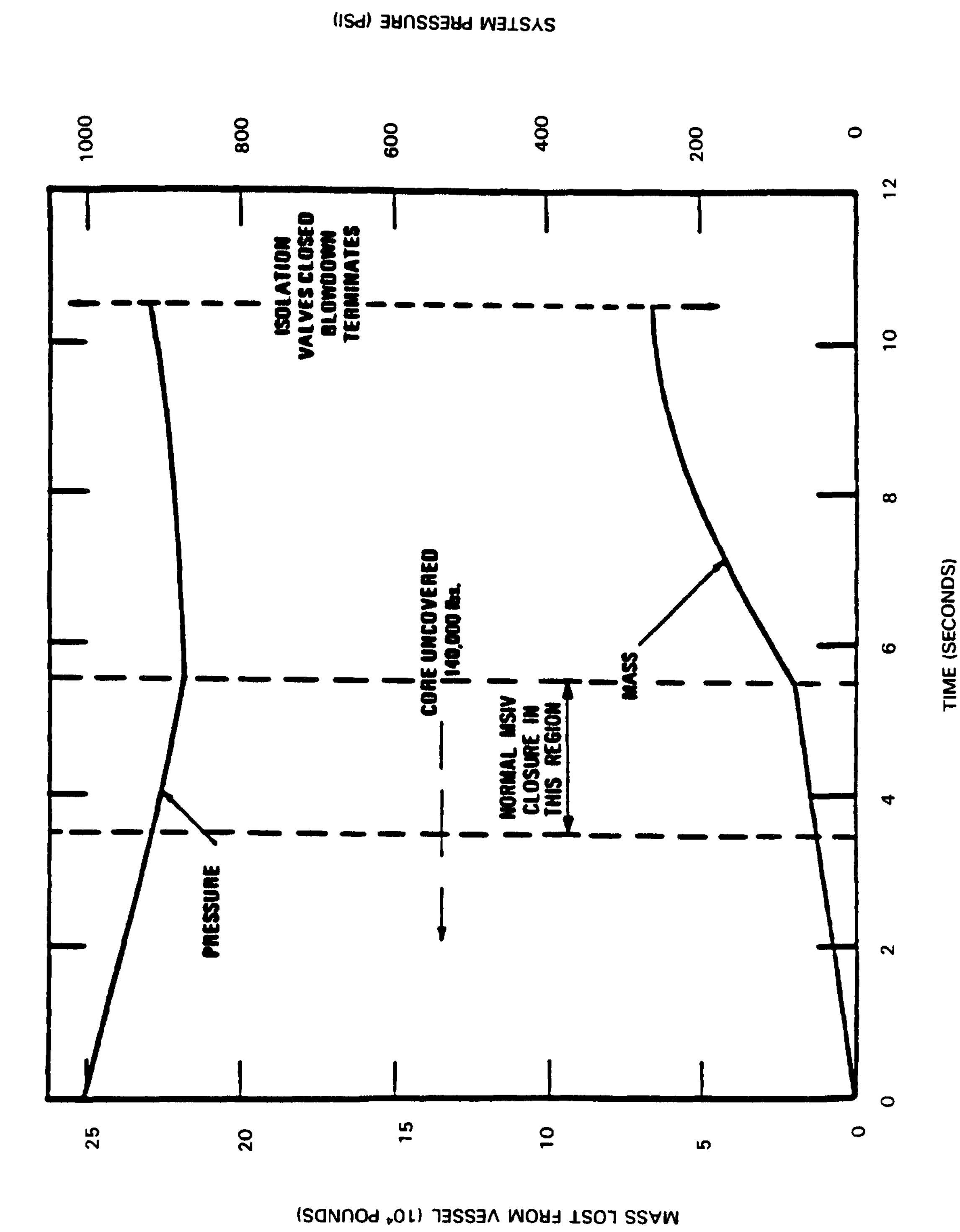


<u>HISTORICAL</u>
This figure contains historical information only.

DRESDEN STATION UNITS 2 & 3
COLD LOOP STARTUP TRANSIENT (PRESSURE AND LEVEL)
FIGURE 15.4-2



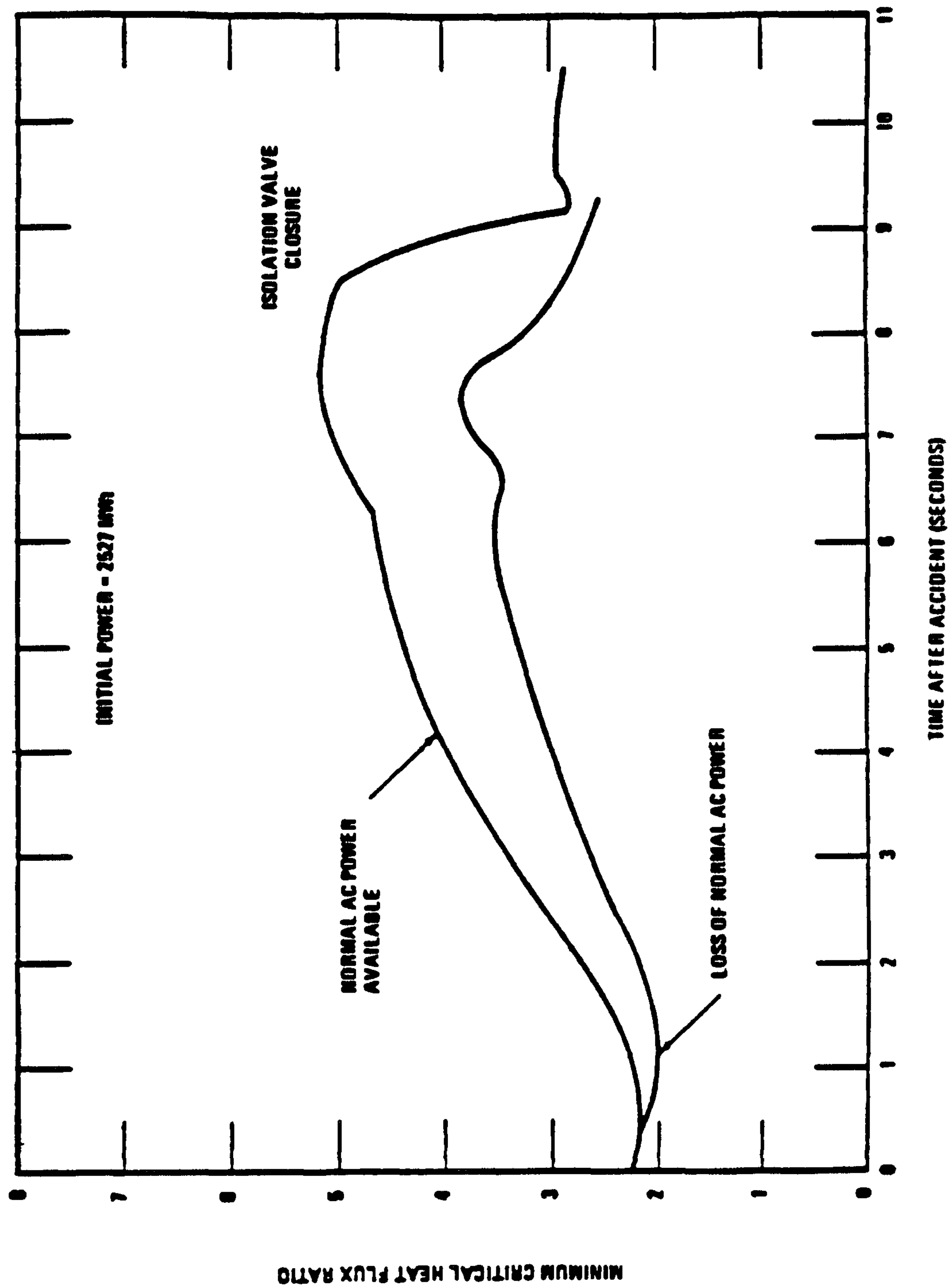
DRESDEN STATION UNITS 2 & 3
COOLANT LOSS FOR MAIN STEAM LINE BREAK OUTSIDE DRYWELL, 10 SECOND MSIV CLOSURE TIME
FIGURE 15.6-1



DRESDEN STATION
UNITS 2 & 3

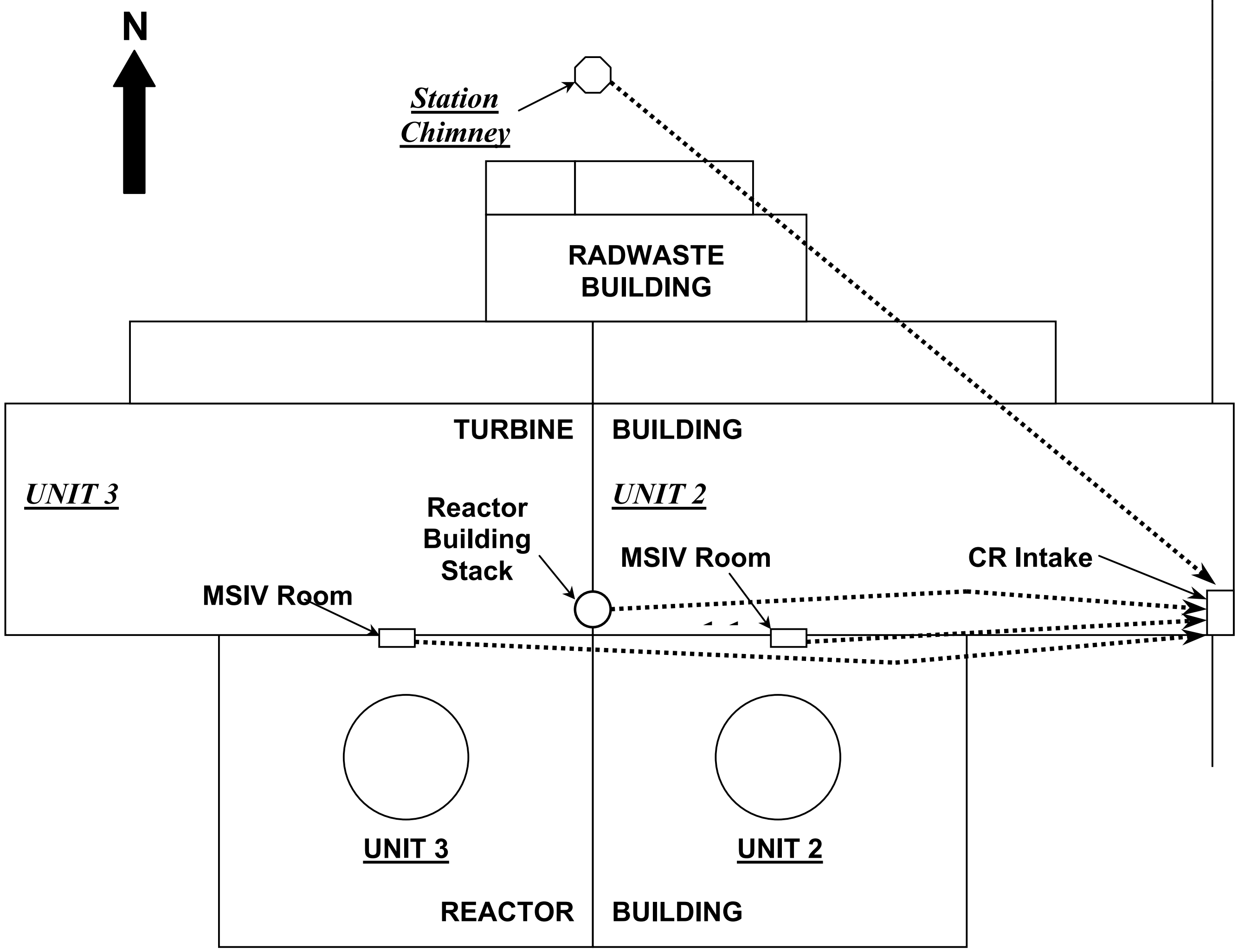
VESSEL PRESSURE AND MASS CHANGES FOR MAIN STEAM LINE BREAK OUTSIDE DRYWELL 10 SECOND MSIV CLOSURE TIME

FIGURE 15.6-2

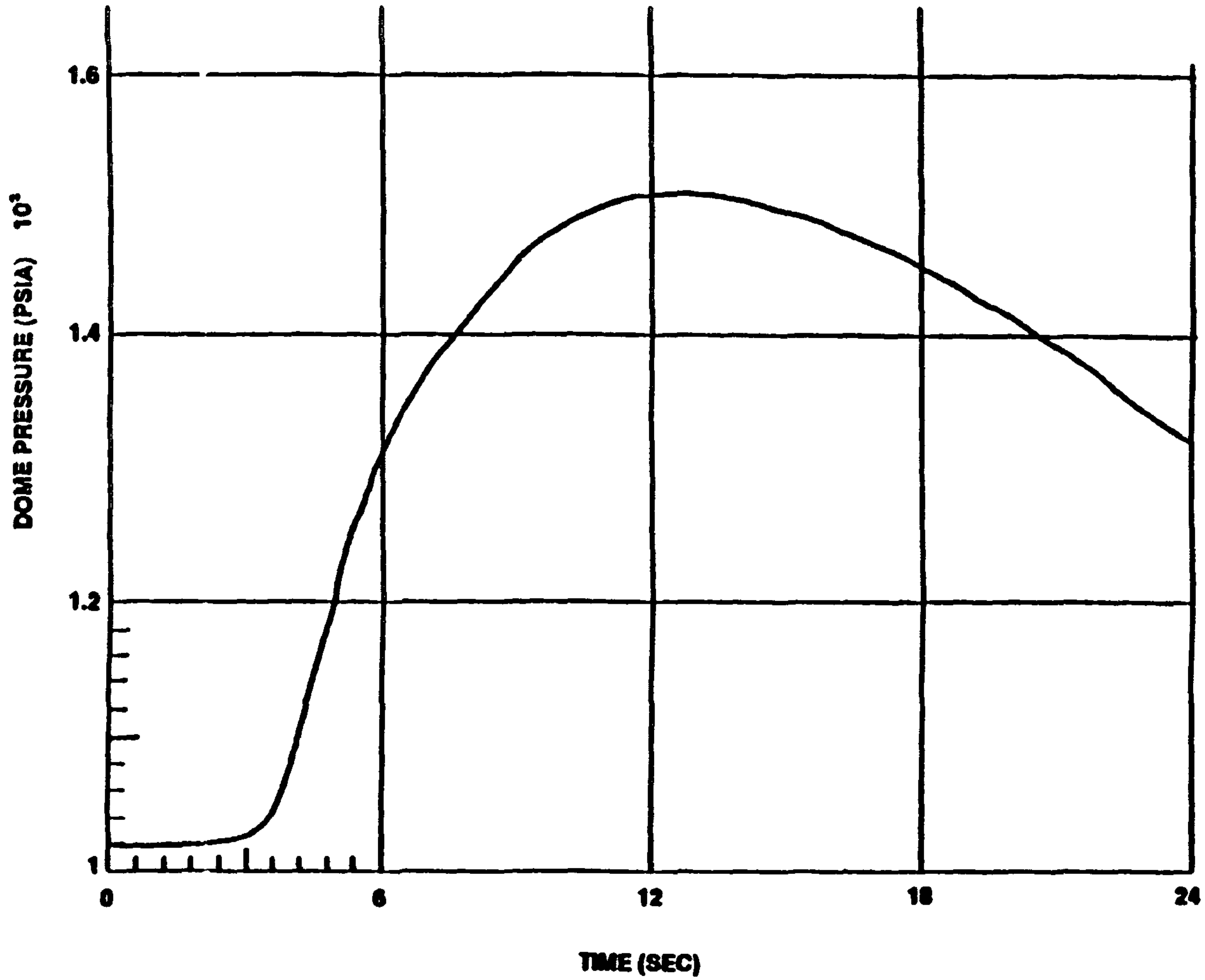


DRESDEN STATION UNITS 2 & 3
MCHFR RESPONSE TO MAIN STEAM LINE BREAK OUTSIDE DRYWELL
FIGURE 15.6-3

Figure 15.6-4: Layout of Intakes and Release Points



DBA	Release Paths
LOCA	Station chimney and MSIV room
CRDA	Station chimney and MSIV room
FHA	Reactor building stack
MSLB	Steam cloud, transported over CR intake

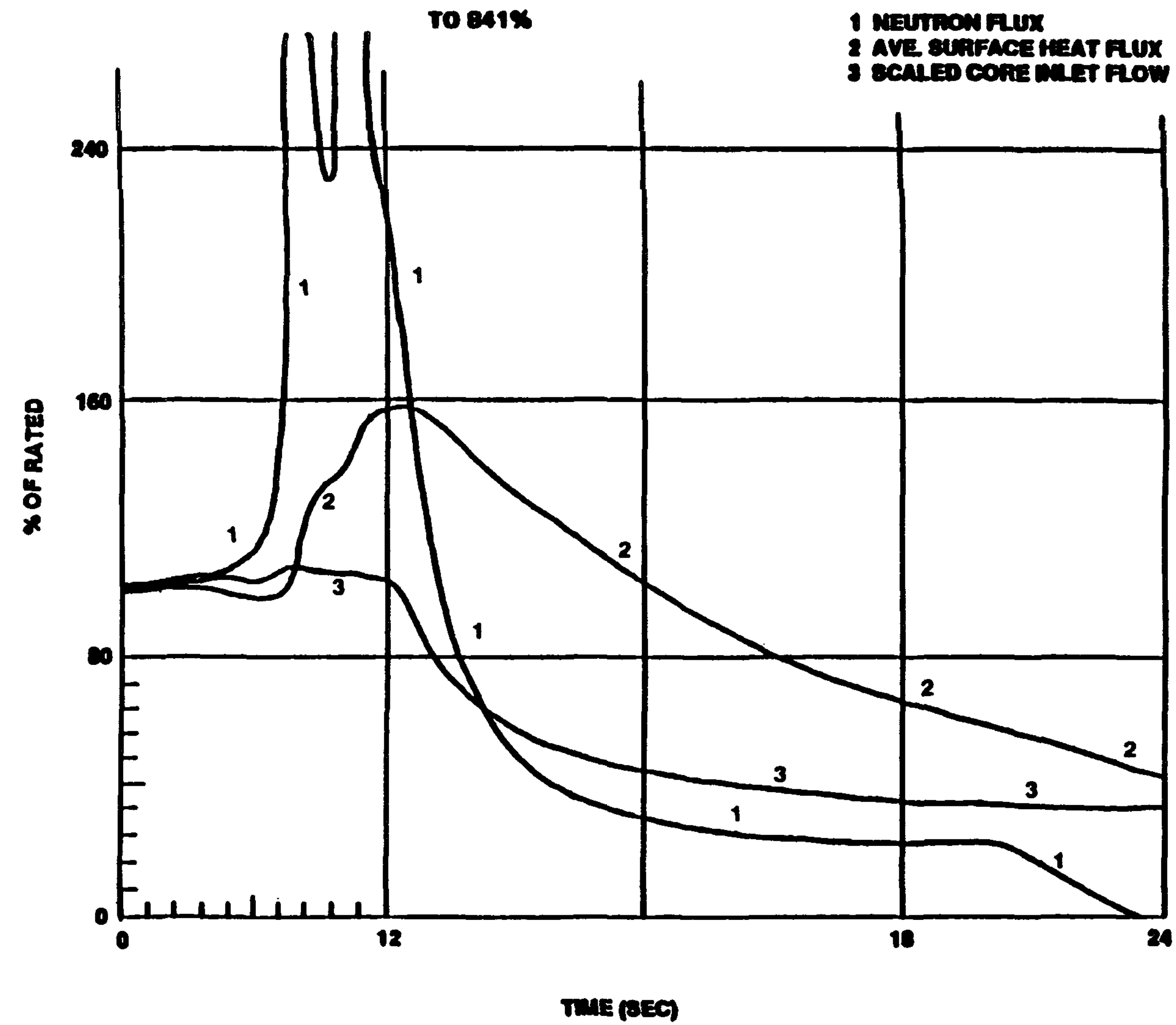


See the introduction to Section 15.8 for information regarding use of details from this analysis which may not be applicable to the current plant design.

DRESDEN STATION
UNITS 2 & 3

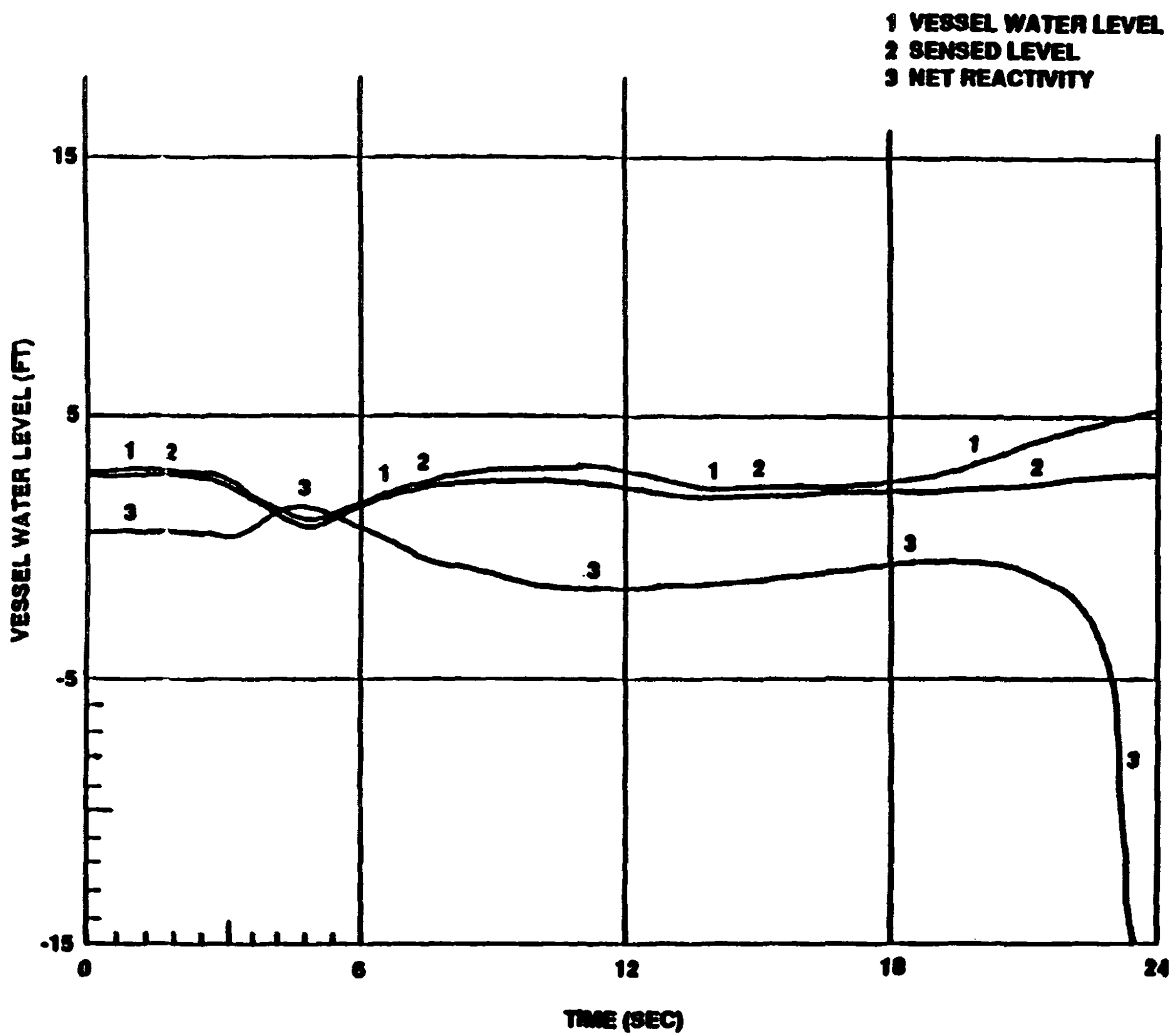
ATWS-MSIV CLOSURE TRANSIENT:
REACTOR PRESSURE RESPONSE

FIGURE 15.8-1



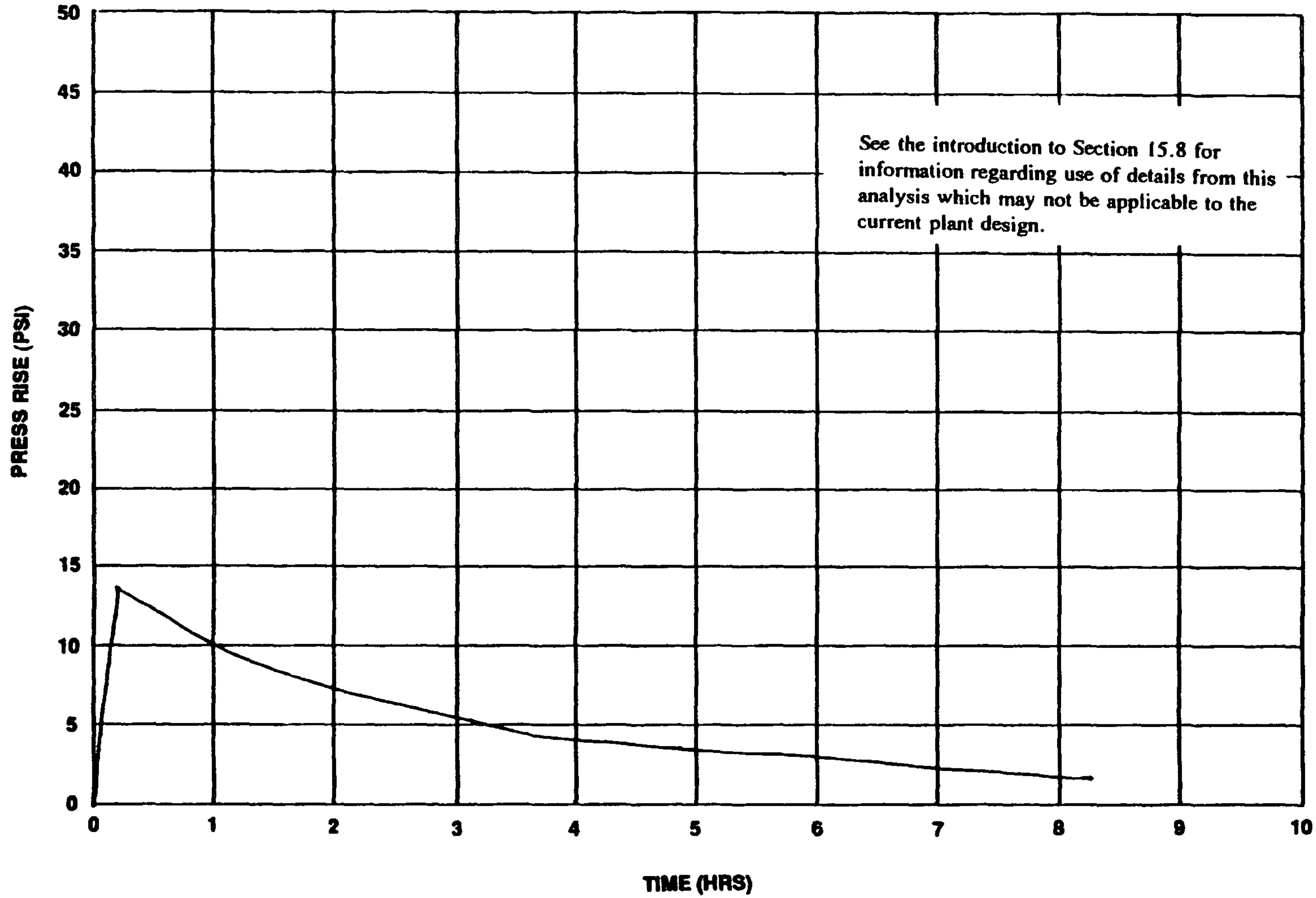
See the introduction to Section 15.8 for information regarding use of details from this analysis which may not be applicable to the current plant design.

DRESDEN STATION UNITS 2 & 3
ATWS-MSIV CLOSURE TRANSIENT: NEUTRON FLUX RESPONSE
FIGURE 15.8-2



See the introduction to Section 15.8 for information regarding use of details from this analysis which may not be applicable to the current plant design.

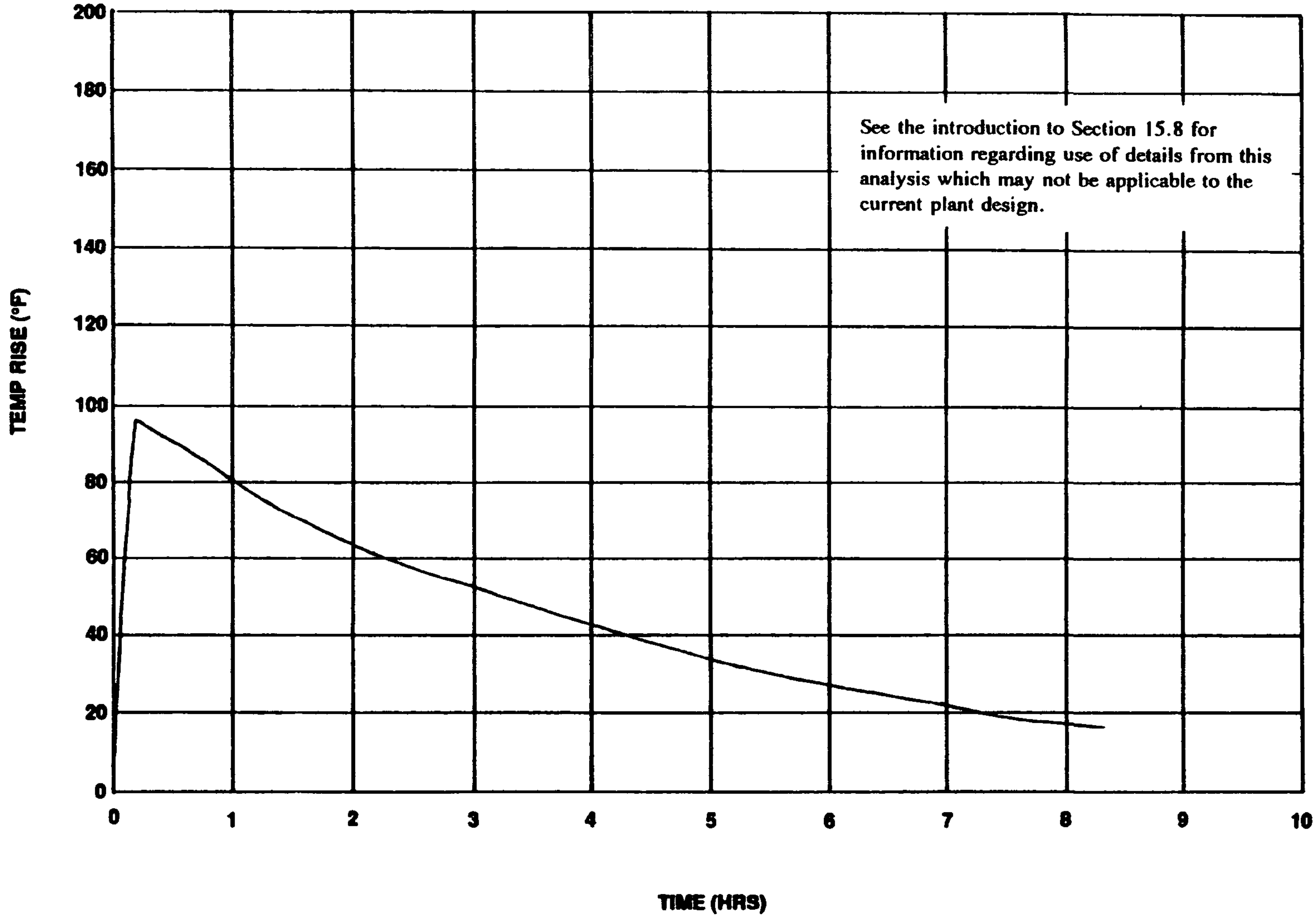
DRESDEN STATION UNITS 2 & 3
ATWS-MSIV CLOSURE TRANSIENT: REACTOR LEVEL RESPONSE
FIGURE 15.8-3



DRESDEN STATION
UNITS 2 & 3

ATWS-MSIV CLOSURE TRANSIENT:
CONTAINMENT PRESSURE RESPONSE

FIGURE 15.8-4



DRESDEN STATION
UNITS 2 & 3

ATWS-MSIV CLOSURE TRANSIENTS:
CONTAINMENT TEMPERATURE RESPONSE

FIGURE 15.8-5