SECTION 14.0

PLANT SAFETY ANALYSIS

This section describes the Plant Safety Analysis for PBAPS Units 2 and 3. Original analyses have been updated as required as part of the Power Rerate Project which increased the licensed rated thermal power to 3458 MWt (References 1 and 2) and as part of the TPO power rerate, which increased thermal power to 3514 MWt (References 18 and 19). The analysis was again updated as part of the Extended Power Uprate (EPU), which increased the licensed rated thermal power to 3951 MWt (Reference 29) and again as part **Uncertainty Recapture (MUR)** uprate which increased the licensed of an additional Measurement rated thermal power to 4016 MWt (Reference 34). In addition, the original analyses have been supplemented to reflect changes to the licensed reactor operating domain including Increased Core Flow and Final Feedwater Temperature Reduction (ICF/FFWTR) and ARTS/MELLLA, and MELLLA+ (References 3, 4, and 33).

The results of the abnormal operational transient analyses presented in the following sections provide information regarding the general sequence of events, features, system behaviors, and trends and characteristics of each event. The information provided allows for an evaluation of the potential impact of changes to plant systems on the results of a particular analysis. The specific results presented (i.e., peak neutron flux and pressure, ΔCPR, etc.) for the abnormal operational transients are based on a representative equilibrium core comprised of GNF2 fuel.

For each core reload, a cycle specific safety analysis is performed utilizing the methods described in GESTAR (Reference 6). The safety analysis reevaluates the limiting transient events to establish the required core thermal operating limits. Those events that are limiting, or near limiting, are identified as such in the following sections. The results of these analyses are documented in the cycle specific Supplemental Reload Licensing Report (SRLR). Information from SRLR is used in the development of the cycle specific Core Operating Limits Report (COLR).

In Section 14.5, the limiting abnormal operational transients in each event category (see Section 14.4.2) are discussed in detail with specific analytical results presented. For analyses that account for reactor power uncertainty, the analysis is performed at 4030 MWt when deterministic methods are used. For statistically based analysis methods, uncertainty in reactor power is incorporated by the methodology. In general, the non-limiting events are discussed in a qualitative manner only.

The accident analyses are performed in a conservatively bounding fashion and are applicable to all cycles for both units. Paragraph 14.8.4 discusses the containment response following a LOCA. Reference 8 provides the sequence of events of the Mark I Containment-related phenomena for the postulated LOCA. The plant-unique definition of specific containment loading conditions that would result from a postulated LOCA at PBAPS Units 2 and 3 is presented in References 1, 9, and 29.

14.1 ANALYTICAL OBJECTIVE

The objective of this Plant Safety Analysis is to evaluate the ability of the plant to operate without undue hazard to the health and safety of the public.

Previous sections of this report provide the objective, design basis, and description of each major system and component. Systems that have unique requirements arising from considerations of nuclear safety are evaluated in the safety evaluation portions of those sections of the report. The safety evaluations consider the effects of failures within the system being investigated. Systems essential to safety are capable of performing their functions in adverse circumstances.

Definitions for key terms used in this section are presented in Section 1.0, "Introduction and Summary." A list of references is included at the end of the appropriate subsections.

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14.0 PLANT SAFETY ANALYSIS REFERENCES (CONTINUED)

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18. NEDC-33064P Safety Analysis Report for Peach Bottom Atomic Power Station Units 2 & 3 Thermal Power Optimization.

19. NEDC-32938P Licensing Topical Report: Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization.

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22. G-080-VC-399, "GNF2 Fuel Design Cycle-Independent Analyses for Exelon Peach Bottom Atomic Power Station Units 2 and 3," GEH-0000-0107-7348, Rev. 1, September 2010.

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24. GEH Report 0000-0135-9000-R0, “Peach Bottom Atomic Power Station Units 2 and 3 TRACG Implementation for Reload Licensing Transient Analysis,” August 2011.

25. G-080-VC-313, "Project Task Report, Peach Bottom Atomic Power Station Unit 2 and 3, SIL 636 Evaluation," GE-NE-0000-0011-4483, Rev. 0 Class III, March 2003.

14.0 PLANT SAFETY ANALYSIS REFERENCES (CONTINUED)

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14.2 UNACCEPTABLE SAFETY RESULTS FOR ABNORMAL OPERATIONAL TRANSIENTS

1. Release of radioactive material to the environs that exceeds the limits of 10CFR20.

2. Any fuel failure calculated as a result of the transient analyses.

3. Nuclear system stress exceeding that allowed for transients by applicable ASME and ANSI codes.

14.3 UNACCEPTABLE SAFETY RESULTS FOR ACCIDENTS

1. Radioactive material release exceeding the limits of 10CFR50.67.

2. Catastrophic failure of the fuel barrier as a result of exceeding mechanical or thermal limits.

3. Nuclear system stresses exceeding that allowed for accidents by applicable ASME and ANSI codes.

4. Containment stresses exceeding that allowed for accidents by applicable ASME and ANSI codes when containment is required.

5. Overexposure to radiation of station personnel in the control room.

14.4 APPROACH TO SAFETY ANALYSIS

14.4.1 General

Two basic groups of events pertinent to safety, abnormal operational transients and accidents, are investigated separately. The unacceptable safety results requires that no damage to the fuel occurs and that no nuclear system process barrier damage results from any abnormal operational transient. Thus, analysis of this group of events evaluates the plant features that protect the first two radioactive material barriers. Analysis of the events in the second group, accidents, evaluates situations that require functioning of the engineered safeguards, including containment. Tables 14.9.6 and 14.9.7 display the overall results of these analyses.

In considering the various abnormal operational transients and accidents, the full spectrum of conditions in which the core may exist is considered. This is accomplished by investigating the differing safety aspects of the four BWR operating states, as described in Appendix G, "Plant Nuclear Safety Operational Analysis." In general, only the most severe event of a given type is described in detail.

Table 14.4.1 provides historical results for selected transients. Key limiting transients are evaluated on a cycle-specific basis; while the results are not reported in this format, operating limits are established to ensure unacceptable safety results are avoided. Additional information regarding the transient evaluations performed in support of the uprate to EPU power may be found in letter to NRC Document Control Desk from Kevin F. Borton, dated September 28, 2012, Peach Bottom Atomic Power Station Units 2 and 3, License Amendment Request - Extended Power Uprate. **The results from the additional analyses performed in support of the MUR uprate to the current power level of 4016 MWt may be found in the letter to the NRC Document Control Desk from James Barstow dated February 17, 2017, Peach Bottom Atomic Power Station, Units 2 and 3,License Amendment Request Measurement Uncertainty Recapture Power Uprate.**

14.4.2 Abnormal Operational Transients

Figure 14.4.1 shows, in block form, the general method of identifying and evaluating abnormal operational transients. Six nuclear system parameter variations are listed as potential initiating causes of threats to the fuel and the nuclear system process barrier; the parameter variations are as follows:

1. Nuclear system pressure increase.

2. Reactor vessel water (moderator) temperature decrease.

3. Positive reactivity insertion.

4. Reactor vessel coolant inventory decrease.

5. Reactor core coolant flow decrease.

6. Reactor core coolant flow increase.

These parameter variations, if uncontrolled, could result in excessive damage to the reactor fuel or damage to the nuclear system process barrier, or both. A nuclear system pressure increase threatens to rupture the nuclear system process barrier from internal pressure. A pressure increase also collapses the voids in the moderator, causing an insertion of positive reactivity that threatens fuel damage from overheating. A reactor vessel water (moderator) temperature decrease results in an insertion of positive reactivity as density increases. Positive reactivity insertions are possible from causes other than nuclear system pressure or moderator temperature changes; such reactivity insertions threaten fuel damage caused by overheating. Both a reactor vessel coolant inventory decrease and a reduction in the flow of coolant through the core threaten to overheat the fuel as the coolant becomes unable to adequately remove the heat generated in the core. An increase in coolant flow through the core reduces the void content of the moderator, resulting in an increased fission rate.

These six parameter variations include all of the effects within the nuclear system caused by abnormal operational transients that threaten the integrities of the reactor fuel or nuclear system process barrier. The variation of any one parameter may cause a change in another listed parameter; however, for analysis purposes, threats to barrier integrity are evaluated by groups according to the parameter variation originating the threat. For example, positive reactivity insertions resulting from sudden pressure increases are evaluated in the group of threats stemming from nuclear system pressure increases.

Abnormal operational transients are defined in subsection 1.2, "Definitions." The following types of single failures and operator errors are identified:

1. The opening or closing of any single valve (a check valve is not assumed to close against normal flow).

2. The starting or stopping of any single component.

3. The malfunction or maloperation of any single control device.

4. Any single electrical failure.

5. Any single operator error.

Operator error is defined as an active deviation from written operating procedures or nuclear plant standard operating practices. A single operator error is the set of actions which is a direct consequence of a single erroneous decision. The set of actions is limited as follows:

1. Those actions that could be performed by not more than one person.

2. Those actions that would have constituted a correct procedure had the initial decision been correct.

3. Those actions that are subsequent to the initial operator error and have an effect on the designed operation of the plant, but are not necessarily directly related to the operator error.

The following are examples of single operator errors:

1. An increase in power above the established flow control power limits by control rod withdrawal in the specified sequences.

2. The selection and complete withdrawal of a single control rod out of sequence.

3. An incorrect calibration of an APRM.

4. Manual isolation of the main steam lines due to operator misinterpretation of an alarm or indication.

The five types of single errors or single malfunctions are applied to the various plant systems with a consideration for a variety of plant conditions to discover events that directly result in any of the listed undesired parameter variations. Once discovered, each event is evaluated for the threat it poses to the integrities of the radioactive material barriers. Generally, the most severe event of a group of similar events is described.

The following performance improvement and equipment out-of-service (EOOS) features are also included in the abnormal operational transient analyses: the Final Feedwater Temperature Reduction (FFWTR) option used in conjunction with Increased Core Flow (ICF)

at the end of cycle, the Turbine Bypass System out-of-service (TBSOOS) contingency mode of operation, the End-of-Cycle Recirculation Pump Trip out-of-service (EOC-RPTOOS) contingency mode of operation, the Power Load Unbalance out-of-service (PLUOOS) contingency mode of operation, and the Pressure Regulator out-of-service (PROOS) contingency mode of operation. Additional details related to the combination of equipment OOS options with ARTS, MELLL, MELLLA+, and ICF operation can be found in References 14.0.18, 14.0.19, and 14.0.26.

14.4.3 Accidents

Accidents are defined as hypothesized events that affect one or more of the radioactive material barriers and which are not expected during the course of plant operations. The following types of accidents are considered:

1. Mechanical failure of various components leads to the release of radioactive material from one or more barriers. The components referred to here are not components that act as radioactive material barriers. Examples of mechanical failures are breakage of the coupling between a CRD and the control rod, failure of a crane cable, and failure of a spring used to close an isolation valve.

2. The fuel barrier overheats. This includes overheating as a result of reactivity insertion or loss of cooling. Other radioactive material barriers are not considered susceptible to failure due to any potential overheating situation.

3. The arbitrary rupture of any single pipe up to and including complete severance of the largest pipe in the nuclear system process barrier. Such rupture is assumed only if the component is subjected to significant pressure.

Figure 14.4.2 shows, in block form, the method of identifying and evaluating accidents. For analysis purposes, accidents are categorized as follows:

1. Accidents that result in radioactive material release from the fuel with the nuclear system process barrier, primary containment, and secondary containment initially intact.

2. Accidents that result in radioactive material release directly to the primary containment.

3. Accidents that result in radioactive material release directly to the secondary containment with the primary containment initially intact.

4. Accidents that result in radioactive material release directly to the secondary containment with the primary containment not intact.

5. Accidents that result in radioactive material release outside the secondary containment.

The effects of the various accident types are investigated, with a consideration for a variety of plant conditions, to examine events that result in the release of radioactive material. The accidents resulting in potential radiation exposures greater than any other accident considered under the same general accident assumptions are designated design basis accidents and are described in detail.

To incorporate additional conservatism into the accident analyses, consideration is given to the effects of an additional, unrelated, unspecified fault in some active component or piece of equipment. Such a fault is assumed to result in the maloperation of a device which is intended to mitigate the consequences of the accident. The assumed result of such an unspecified fault is restricted to such relatively common events as an electrical failure, instrument error, motor stall, breaker freeze-in, or valve maloperation. Highly improbable failures, such as pipe breaks, are not assumed to occur coincident with the assumed accident. The additional failures to be considered are in addition to failures caused by the accident itself.

In the analyses of the design basis accidents, a variety of single additional failures is considered by making analysis assumptions that are sufficiently conservative to include the range of effects from any single additional failure. Thus, no single additional failure of the types to be considered could worsen the computed radiological effects of the design basis accidents.

14.4.4 Barrier Damage Evaluations

14.4.4.1 Fuel Damage

Subsection 3.7, "Thermal and Hydraulic Design," describes the various fuel failure mechanisms and establishes fuel damage limits for various plant conditions. Avoidance of unacceptable safety results 1 and 2, for abnormal operational transients, is determined by demonstrating that abnormal operational transients do not result in an MCPR less than the Technical Specification Safety Limit MCPR. If MCPR does remain above the Safety Limit MCPR, no fuel failures result from the transient, and thus the

radioactivity released from the plant cannot be increased over the operating conditions existing prior to the transient. It should be noted that maintaining MCPR greater than the Safety Limit MCPR is a sufficient, but not necessary, condition to assure that no fuel damage occurs.

For situations in which fuel damage is sustained, the extent of damage is determined by correlating fuel energy content, cladding temperature, fuel rod internal pressure, and cladding mechanical characteristics. These correlations are substantiated by fuel rod failure tests and are discussed in subsection 3.7, "Thermal and Hydraulic Design," and Section 6.0, "Core Standby Cooling Systems."

Avoidance of unacceptable safety result 2 for accidents is shown by demonstrating that fuel clad temperature remains below 2,200°F. The selection of this temperature is discussed in Section 6.0, "Core Standby Cooling Systems."

Avoidance of unacceptable result 2 for abnormal operational transients and accidents is shown by demonstrating that peak fuel enthalpy does not exceed 280 cal/g. The selection of this value is discussed in subsection 3.6, "Nuclear Design."

14.4.4.2 Nuclear System Process Barrier Damage

Avoidance of unacceptable safety result 3 for abnormal operational transients and unacceptable safety result 3 for accidents is assessed by comparing peak internal pressure with the overpressure transient allowed by the applicable industry code. The only significant areas of interest for internal pressure damage are the high-pressure portions of the nuclear system primary barrier, the reactor vessel, and the high-pressure lines attached to the reactor vessel. The overpressure criteria is based on the most restrictive maximum pressure design requirements of either the ASME Boiler and Pressure Vessel Code, Section III for the reactor vessel or ANSI B31.1 "Pressure Piping, Power Piping." The ASME Boiler and Pressure Vessel Code, Section III, specifies a pressure transient of 10 percent over the design pressure (110 percent x 1,250 psig = 1,375 psig); ANSI B31.1 specifies that the piping design be based on the maximum internal pressure of the system, including system pressure surges (1,375 psig). Therefore, the nuclear system process barrier satisfies the overpressure requirements, provided that the peak nuclear system internal pressure does not exceed 1,375 psig.

An analysis, which is discussed in subsection 3.6, "Nuclear Design," is used to evaluate whether nuclear system process barrier damage occurs as a result of reactivity accidents. If

peak fuel enthalpy remains below 280 cal/g, no nuclear system fuel barrier damage results from nuclear excursion accidents.

14.4.4.3 Containment Damage

Avoidance of unacceptable safety results 1 and 4 for accidents requires that the primary and secondary containments retain their integrities for certain accident situations. Containment integrity is maintained as long as internal pressures remain below the maximum allowable values. The containment maximum allowable internal pressures are as follows:

Drywell (primary containment): 62 psig

Suppression Chamber (primary containment): 62 psig

Secondary Containment: 0.25 psig

Damage to radioactive material barriers as a result of accident-initiated fluid impingement and jet forces is considered in the other portions of this Updated FSAR where the mechanical design features of systems and components are described. Design basis accidents are used to determine the sizing and strength requirements of many of the essential nuclear system components. A comparison of the accidents considered in this section with those used in the mechanical design of equipment reveals that either the applicable accidents are the same or that the accident in this section results in less severe stresses than those assumed for mechanical design.

14.4.5 Licensing Basis Versus Emergency Procedure Guidelines

The NRC Staff review of the BWR Owners' Group Emergency Procedure Guidelines (EPGs), Revision 4 (NEDO-31331, March 1987) is documented in an NRC Safety Evaluation Report (SER), dated September 12, 1988. In this SER, the NRC Staff concluded that the guidelines are acceptable for implementation; however, the NRC stated that each BWR Licensee who wishes to incorporate Revision 4 of the EPGs should assure that the EPGs will not impact its licensing bases.

To ensure that implementation of EPGs, Revision 4, did not conflict with the PBAPS licensing-based analyses, a review of the UFSAR Section 14 analyzed transients was conducted. This review (documented in Reference 14.0.11) identified the Pressure Regulator Failure - Open Direction event as the only analyzed transient in which operator action, as instructed by the Transient Response Implementation Procedures (TRIP), could interfere with the analyzed transient response as discussed in UFSAR Section 14. For the Pressure Regulator Failure - Open Direction transient, it

was concluded that procedural actions are sufficient to make the end result for the transient the same as that analyzed in UFSAR Section 14.

This review was performed based on the following position regarding Licensing Basis versus EPGs:

> Implementation of the TRIP procedures for events within the design basis does not conflict with the licensing basis, as long as the successful implementation of operator actions makes the end result for the transient response the same as or more conservative than that analyzed in the UFSAR.

The revision 4 BWROG EPGs already included, to a considerable degree, operator actions required to prevent or mitigate what are termed severe accidents (considered to begin with the onset of core damage). Subsequent to the issue of the Revision 4 EPGs, the preparation of Individual Plant Examinations (IPE) for nuclear plants, in conjunction with industry research and Fukushima nuclear accident insights, have provided additional severe accident mitigation information that was not available during the development of the Revision 4 EPGs. To ensure the EPG strategies reflected the most current severe accident mitigation information available, a review of the EPG strategies to the additional available severe accident mitigation information, was conducted. As a result of this review, the BWROG developed and subsequently revised the Emergency Procedure and Severe Accident Guidelines (EPG/SAG). The SAG, together with the modified set of symptomatic instructions attempt to cover all possible mechanistic accident sequences. The EPGs contain strategies applicable while adequate core cooling is maintained, and the SAG contains strategies applicable after the core cannot be adequately cooled. EPG/SAG has superseded EPG, Revision 4 as the basis for the Emergency Operating Procedures (EOPs) at PBAPS. No NRC SER exists for the Emergency Procedure and Severe Accident Guidelines (EPG/SAG).

To ensure that the implementation of EPG/SAG did not conflict with the PBAPS licensing - based analysis, a review of the SAR analyzed accidents, transients, and “special” events was conducted. The conclusions of this review are contained in the 10CFR50.59 Reviews prepared for the Plant Specific Technical Guidelines (PSTGs) and Plant Specific Severe Accident Management Guidelines (PSSAMGs).

The following exceptions have been taken to the EPG/SAG Guidelines:

- EPG/SAG allows for bypassing the Mechanical Vacuum Pump high radiation interlocks when the main condenser is available as a heat sink during ATWS conditions in the RPV Control Guideline

and Level/Power Control Guideline. This guidance has not been incorporated into the associated EOPs.

- EPG/SAG directs the operation of containment sprays down to 0 psig containment pressure in the Primary Containment Control Guideline and RPV Control Severe Accident Guideline. Containment sprays are allowed down to 2 psig in the associated Transient Response Implementation Plan (TRIP) procedures.

- EPG/SAG directs emergency blowdown after RPV water level drops to the Minimum Stream Cooling RPV Water Level (MSCRWL) in the Alternate Level Control Guideline. The MSCRWL is below the top of active fuel. Emergency blowdown is performed at the top of active fuel in the associated EOP.

These exceptions, and other plant unique variations from the EPG/SAG guidelines, have been evaluated in the 10CFR50.59 Reviews, the Plant Specific Technical Guidelines (PSTGs), and the Plant Specific Severe Accident Management Guidelines (PSSAMGs) which support the transition from EPG, Revision 4 to EPG/SAG.

TABLE 14.4.1

PLANT SAFETY ANALYSIS

RESULTS OF ABNORMAL OPERATIONAL TRANSIENTS

Undesired Parameter Variation	Event Causing Transient	Results: Scram Caused by	Results: ΔCPR[3]	Results: Peak Nuclear System Pressure
Nuclear system pressure increase	Electrical load rejection without bypass(*)	Turbine control valve fast closure	0.36	1,264 psig
Nuclear system pressure increase	Turbine trip without bypass(*)	Turbine stop valve closure	<0.36	1,261 psig
Nuclear system pressure Increase	MSIV closure	High Flux	N/A[4]	1,335 psig
Reactor water temperature decrease	Inadvertent pump start (HPCIS)	None	N/A[4]	Not applicable[1]
Reactor water temperature decrease	Feedwater controller failure - maximum demand(* **)	Turbine stop valve closure	0.37	1,254 psig
Reactor water temperature decrease	Loss of feedwater heating	None	0.12 uncorrected	Not applicable[1]
Positive reactivity insertion	Continuous rod withdrawal during power range operation	None	0.27 uncorrected	Not applicable[1]
Positive reactivity insertion	Continuous rod withdrawal during reactor startup	High neutron flux	N/A[4]	Not applicable[1]
Coolant inventory decrease	Pressure regulator failure - open	MSIV closure or Turbine stop valve closure	N/A[4]	<1,250 psig
Coolant inventory decrease	Open relief valve or safety valve	None	N/A[4]	Not applicable[2]
Coolant inventory decrease	Loss of feedwater flow[5]	Reactor vessel low water level	N/A[4]	Not applicable[2]
Coolant inventory decrease	Loss of auxiliary power	Loss of power to RPS	<0.30	<1,250 psig

(*) EOC-RPT out-of-service
(**) MELLA, TSVOOS, FFWTR

TABLE 14.4.1 (Continued)

		Results		
Undesired Parameter Variation	Event Causing Transient	Scram Caused by	Δ CPR	Peak Nuclear System Pressure
Core flow decrease	Recirculation flow control failure – decreasing flow	None	N/A[4]	Not applicable[2]
Core flow decrease	Trip of one recirculation pump	None	N/A[4]	Not applicable[2]
Core flow decrease	Trip of two recirculation Adjustable Speed Drives	None	N/A[4]	Not applicable[2]
Core flow decrease	Recirculation pump seizure	None	N/A[4]	1,135 psig
Core flow increase	Recirculation flow control failure - fast increase	High neutron flux	0.23	1,061 psig
Core flow increase	Recirculation flow control failure - slow increase	None	N/A[4]	No increase above rated
Core flow increase	Startup of idle recirculation pump	None	N/A[4]	Not applicable[1]

(1) This transient results in no significant change in nuclear system pressure.
(2) This transient results in a depressurization.
(3) ΔCPR is based on initial CPR which results in MCPR = Safety Limit CPR.
(4) ΔCPR is non-limiting for this event.
(5) The reactor vessel low water level (L3) scram analytical limit value may be slightly different for various events due to a steam flow induced process measurement error. However, as described in Reference 14.0.23, the impact of this error is not significant and the event descriptions or conclusions are not affected.

TABLE 14.4.2

PLANT SAFETY ANALYSIS

RESULTS OF DESIGN BASIS ACCIDENTS

OPTION

Design Basis Accidents	Percent of Core Reaching Cladding Temperature of 2,700°F	Peak Nuclear System Pressure	Maximum Total Off-Site Exposure (Rems)[1][4]	
			Whole Body	Thyroid
Rod drop accident[4]	N/A (0>3370°F)	<1,375 psig	0.19	6.4
Loss-of-coolant accident[4]	0	Not applicable[2]	0.66	15
Refueling accident[4]	0	Not applicable[3]	0.093	2.3
Main steam line break accident[4]	0	Not applicable[2]	0.023	88

[1] At the exclusion boundary
[2] This accident results in a depressurization.
[3] This accident occurs with the reactor vessel head off.
[4] See Section 14.9 for offsite doses.

14.5 ANALYSES OF ABNORMAL OPERATIONAL TRANSIENTS

14.5.1 Events Resulting in a Nuclear System Pressure Increase

Events that result directly in significant nuclear system pressure increases are those that result in a sudden reduction of steam flow while the reactor is operating at power. A survey of the plant systems has been made to identify events within each system that could result in the rapid reduction of steam flow. The following events were identified:

1. Electrical load rejection (turbine control valve fast closure).

2. Turbine trip (turbine stop valve closure).

3. Closure of the main steam line isolation valves.

4. Failure of the turbine bypass valves to open when required.

5. Loss of main condenser vacuum.

A consideration of the last two varieties of events shows that turbine bypass valve failure and loss of condenser vacuum are specific cases of the first three event types. A failure of the turbine bypass valves to open when required is analyzed as the most severe form of a turbine trip or generator load rejection. The most conservative transient, instantaneous loss of condenser vacuum, is a nearly identical transient, with essentially simultaneous scrams from the low vacuum signal and the turbine stop valve position indicating signals. For the loss of vacuum, the feedwater turbines would also be tripped. However, the parameters of main concern, fuel thermal margin and margin to vessel overpressure, are not significantly different from the analysis performed for the turbine trip (no bypass) transient (paragraph 14.5.1.2.2). Thus, all of the effects of these events are included in the effects described for the generator and turbine trip.

14.5.1.1 Electrical Load Rejection (Turbine Control Valve Fast Closure) with Bypass Failure

A loss of generator electrical load from high power conditions results in a significant reactor vessel pressurization due to a rapid (in approximately 0.08 sec) closure of all turbine control valves (TCVs) initiated by the action of the Power Load Unbalance (PLU) device with the turbine bypass valves assumed to fail shut. For this event, the reactor is assumed to be operating at rated

power. The event is evaluated at various flow conditions to ensure that the results bound the entire operating map. The event is normally analyzed with the EOC-RPT system both operable and inoperable (see 14.5.1.2.4). The event is also analyzed with the Power Load Unbalance device out-of-service (PLUOOS).

The system response for this event from an initial condition at the Increased Core Flow (ICF) point on the power/flow map (100% power and 110% core flow) with no equipment out of service flexibility options is shown in Figure 14.5.1a. Consequently, this figure does not reflect the most limiting combination(s) of initial operating point (ICF or MELLLA) and equipment flexibility options for each of the parameters discussed in this paragraph. A reactor scram is initiated by the TCV pressure switches. As the reactor vessel pressure increases, positive void reactivity due to void collapse in the core results in a power increase. The reactor scram quickly overcomes the positive void reactivity and the reactor is shut down. The calculated neutron flux reaches a peak value of 435% of its initial value and the core average surface heat flux reaches a peak value of 125% of its original value. This results in a calculated ΔCPR of 0.36. The relief valves open to limit the pressure rise, then reclose as pressure decreases. The calculated peak reactor vessel pressure (at the bottom of the vessel) is 1264 psig. Thus, unacceptable safety result 3 is avoided. Unacceptable safety results 1 and 2 are avoided by establishing core thermal operating limits based on the results of the analysis of the event such that the MCPR Safety Limit is not exceeded.

When the power load unbalance device is out-of-service, it is assumed in the analysis that the TCVs will not rapidly close, but instead close in their servo mode in response to the increase in turbine speed from the load rejection (Note: This is a conservative assumption; a backup turbine trip would be expected to occur during a load rejection event with the PLUOOS, resulting in an event similar to the normal load rejection). The closure of the TCVs causes an increase in reactor pressure and the event is terminated with a high pressure or high flux scram.

Additionally, as part of the normal load rejection event analysis, the PLU device is assumed not to generate a TCV fast closure signal below 55% power (Reference 14.0.28)

The Electrical Load Rejection Without Bypass event is a limiting event, and as such is reanalyzed each cycle to evaluate the required core thermal operating limits (i.e., MCPR). The Electrical Load Rejection Without Bypass event with PLUOOS is analyzed at full range of power levels. Peach Bottom 3 Cycle 21 was the first cycle to include PLUOOS in the reload analysis for

full range of power levels, and the results of the analysis are documented in Reference 14.0.32.

14.5.1.2 Turbine Trip (Turbine Stop Valve Closure)

A variety of turbine or nuclear system malfunctions can initiate a turbine trip. Once initiated, all of the turbine stop valves achieve full closure within about 0.10 sec. This event represents one of the fastest possible steam flow shutoff mechanisms and one of the most severe nuclear system pressure increases. Several variations in the turbine trip transient are possible according to the assumptions made concerning the initial power level and the turbine bypass system. These cases are discussed individually.

14.5.1.2.1 Turbine Trip from High Power with Bypass

The sequence of events for this case is similar to that described for the turbine trip from the high power without bypass - (Section 14.5.1.2.2) with the exception that the turbine bypass valves are assumed to function. Position switches on the turbine stop valves sense the valve closure and initiate a scram and bypass valve opening on demand from the EHC pressure regulator. The nuclear system relief valves open for a short time to relieve the pressure increase caused by the transient. No fuel damage or unacceptable system stress results from the transient. This transient is a non-limiting event bounded by the following case of a turbine trip in which the bypass valves are assumed to fail closed.

14.5.1.2.2 Turbine Trip from High Power without Bypass

The transient response of this event is very similar to an Electrical Load Rejection Without Bypass. However, due to the longer turbine stop valve closure time (relative to the turbine control valves) and a slightly quicker scram response, the Turbine Trip is somewhat less severe than an Electrical Load Rejection. The reactor core is assumed to be operating at rated power when the incident occurs. All other initial reactor parameters are consistent with those for an Electrical Load Rejection Without Bypass.

Figure 14.5.1B shows the predicted plant response to the trip from an initial condition at the Increased Core Flow (ICF) point on the power/flow map (100% power and 110% core flow) with no equipment out of service flexibility options. Consequently, this figure does not reflect the most limiting combination(s) of initial operating point (ICF or MELLLA) and equipment flexibility options for each of the parameters discussed in this paragraph. The scram is initiated by the position switches on the turbine stop valves. The calculated neutron flux reaches a peak value of 388% of its initial value, and the core average surface heat flux

reaches a peak value of 123% of its original. This results in a calculated ΔCPR of less than 0.36 for Unit 2 and 0.30 for Unit 3. The nuclear system relief valves open to limit the pressure rise, then sequentially reclose as pressure decreases. The calculated peak reactor vessel pressure (at the bottom of the vessel) is 1261 psig. Thus, unacceptable safety result 3 is avoided. The turbine trip without bypass event is reanalyzed each cycle to evaluate the required core thermal operating limits (i.e., MCPR).

14.5.1.2.3 Turbine Trip from Lower Power without Bypass

This abnormal operational transient is of interest because the turbine trip scram function is automatically bypassed when the reactor power level is below a pre-established value. Turbine first stage pressure is used to initiate the scram bypass. The highest power level for which the turbine stop valve closure scram remains bypassed is 26.27 percent of rated reactor thermal power. For this event, a reactor scram occurs in about 3.0 sec from high vessel pressure or at approximately 2 sec if a flow biased high neutron flux scram is considered. The turbine bypass valves are assumed not to open, and the relief valves act to limit the nuclear system pressure rise. Peak nuclear system pressure remains well below 1,375 psig. Thus, unacceptable safety result 3 is avoided. For off-rated conditions, the power and flow biased thermal limits implemented by the ARTS Improvement Program (References 14.0.4, 14.0.20, 14.0.21) assure that the consequences of this event remain bounded by the limiting event evaluated at rated conditions. Thus, unacceptable safety result 1 and 2 are avoided.

14.5.1.2.4 End-of-Cycle Recirculation Pump Trip (EOC-RPT) System

The EOC-RPT system consists of two high-speed circuit breakers in series with the conductors from the output of each recirculation Adjustable Speed Drive to the recirculation pump motor. The system receives turbine stop valve closure and turbine control valve fast closure signals from the RPS relays, and trips the EOC-RPT breakers when a turbine trip scram or a load reject scram is initiated (see Section 7.9.4.4.3). This trip rapidly reduces the core flow increasing the void content in the core. This addition of negative reactivity limits the power excursion due to pressurization. This is especially important when the reactor core is near end of cycle (EOC). At EOC the control rod density is low and the initial control rod motion during a scram has little reactivity effect.

As documented in Reference 14.0.15 the EOC-RPT feature provides a thermal limits benefit for pressurization events (i.e., Load Rejection, Turbine Trip, and Feedwater Controller Failure), particularly when the bypass valves are assumed to be inoperable.

Thermal operating limits are cycle specific and the analysis results for these events with and without RPT available are documented in the cycle specific SRLR and COLR. The response time for the EOC-RPT system used in the reload analysis is also documented in the COLR.

14.5.1.3 Main Steam Line Isolation Valve Closure

Automatic circuitry or operator action can initiate closure of the main steam line isolation valves. Position switches on the valves initiate a scram if the valves are less than 90 percent open and the mode switch is in RUN. However, RPS logic does permit the test closure of one valve without initiating a scram due to valve closure. These cases are investigated separately.

14.5.1.3.1 Closure of All Main Steam Line Isolation Valves

This is the design basis event to demonstrate compliance with the ASME nuclear system peak upset pressure limit (1375 psig). Position switches on the valves normally initiate a scram if the valves are less than 85% open and the mode switch is in RUN. However, this analysis conservatively assumes that the valve position scram fails and reactor scram is initiated by a high neutron flux signal. The analysis of this event accounts for uncertainty in reactor power as described in Section 14.0.

Figure 14.5.3 shows the predicted changes in important nuclear system variables during this event. The closure of all MSIVs causes a rapid pressure increase in the reactor vessel. The system pressure increase is mitigated by the actuation of the safety/relief valves in their safety mode and by the operation of the spring safety valves. The predicted peak reactor vessel pressure is 1352 psig, which is below the allowable value of 1375 psig **and the predicted reactor vessel dome pressure is 1324 psig, which is below the Technical Specification limit of 1325 psig.** Thus, unacceptable safety result 3 is avoided. Reactor core neutron flux and surface heat flux reach approximately 390% and 130% of their initial values. This event is not evaluated for unacceptable safety results 1 and 2.

14.5.1.3.2 Closure of One Main Steam Line Isolation Valve

Closure of one main steam isolation valve is desirable for testing purposes. Therefore the protection system logic from the valve position switches permits full shutoff of any steam line without initiating a scram. The MSIV's are partial closed to verify the operability of the RPS limit switches. Since the valves are slow closed to actuate the limit switches no power reduction is necessary for this test to avoid a high flux scram. MSIV's are also fast closed to verify closure times at reduced power.

Procedures for such a test require a power reduction to avoid a high neutron flux or a high pressure scram.

During a simulated 3-sec closure of one main steam line isolation valve from design power conditions, the steam flow disturbance raises reactor vessel pressure and the reactor power, causing a high neutron flux scram. The peak surface heat flux increases slightly. The fuel thermal-mechanical response is bounded by the results of an Electrical Load Rejection Without Bypass. Peak pressures remain below the lowest setting of the relief valves. Thus, all unacceptable safety results are avoided. This is a non-limiting event.

14.5.2 Events Resulting in a Reactor Vessel Water Temperature Decrease

Events that result directly in a reactor vessel water temperature decrease are those that either increase the flow of cold water to the vessel or reduce the temperature of water being delivered to the vessel. The events that result in the most severe transients in this category are the following:

1. Inadvertent pump start.

2. Feedwater controller failure - maximum demand.

3. Loss of feedwater heating.

4. Shutdown cooling (RHRS) malfunction - decreasing temperature.

14.5.2.1 Inadvertent Pump Start (HPCIS)

Several systems are available for providing high pressure supplies of cold water to the vessel for normal or emergency functions. The CRDS and the makeup water system, normally in operation, can be postulated to fail in the high-flow direction, introducing the possibility of increased power due to higher core inlet subcooling. The same type of transient would be produced by inadvertent startup of either the RCICS or the HPCIS. In all of these cases, the normal feedwater flow would be correspondingly reduced by the water level controls. The net result is simply a replacement of a portion of the feedwater flow (at rated power operation) by the colder HPCI flow.

The severity of the resulting transient is highest for the largest of these abnormal events, the inadvertent startup of the large (5,000 gpm) HPCIS.

In the original licensing basis of this event, for an inadvertent HPCIS startup, the additional fluid injection to the reactor vessel results in a water level increase. The colder HPCI water, assumed to be at 40°F, also results in a power increase and subsequent pressure increase. For the conditions analyzed the reactor is assumed to be a rated power. The analysis of this event accounts for uncertainty in reactor power as described in Section 14.0. Neither the high water level setpoint (Level 8) nor the high steam dome pressure scram setpoint are reached. The core negative reactivity void coefficient limits the power and pressure increase and the reactor settles into a new equilibrium state. The peak core average surface heat flux reaches a value of 115.0%. Based on a comparison of peak heat flux and reactor pressure, it is concluded that within this event category, the inadvertent HPCIS startup event is bounded by the Feedwater Controller Failure, Maximum Demand event (Section 14.5.2.2). The system will stabilize at the higher power level with neither fuel damage nor nuclear system process barrier damage. Thus, unacceptable safety results 1 through 3 are avoided.

Subsequent to the analysis presented above, changes were made to the reactor water level control system (i.e., digital feedwater control). The standard reload analysis transient models do not adequately represent the new reactor water level control system. Therefore, it cannot be confidently demonstrated that the inadvertent HPCIS startup will not result in reaching the high water level setpoint. Additional analysis of the inadvertent HPCIS startup event have been performed assuming the high water level setpoint is reached. The results of these analyses are more severe and are similar to the results for the Feedwater Controller Failure – Maximum Demand (Section 14.5.2.2).

Consequently, the inadvertent HPCIS startup event, assuming the high water level setpoint is reached, is a limiting event, and as such, is reanalyzed each cycle to evaluate the required core thermal operating limits (e.g., MCPR).

14.5.2.2 Feedwater Controller Failure - Maximum Demand

Failure of the feedwater controller in the direction of increased feedwater flow results in a moderator temperature decrease causing a reactor power increase through the effect of the negative reactivity void coefficient. This effect is the same as if a pump was inadvertently started, adding cold water to the core.

For this event, the reactor is assumed to be operating at rated power. The event is evaluated at various flow conditions to ensure that the results bound the entire operating map. This produces more severe results than assuming normal feedwater temperature. (Ref. 14.0.12)

The analysis is normally performed with the turbine bypass system both operable and inoperable. The initial reactor water level is assumed to be at the low level alarm setpoint (Level 4) which increases the severity of the event. The feedwater pumps are assumed to attain their maximum capability as established by the feedwater turbine controller limits (i.e., high speed stops). To ensure that the actual plant maximum feedwater capability is bounded, an additional 5% of rated feedwater flow is added to the maximum capability in the analysis. The maximum feedwater capability is confirmed for each cycle specific analysis.

The transient response of the plant to a failure of the feedwater controller resulting in a demand for maximum feedwater flow from an initial condition at the Increased Core Flow (ICF) point on the power/flow map (100% power and 110% core flow) with no equipment out of service flexibility options is shown in Figure 14.5.5. Consequently, this figure does not reflect the most limiting combination(s) of initial operating point (ICF or MELLLA) and equipment flexibility options for each of the parameters discussed in the following paragraph.

The water level increases during the initial part of the transient at about 2.0 in/sec. The high water level (Level 8) turbine trip is initiated at about 12 sec after the failure, when sensed level has increased about 30 in. The turbine trip prevents excessive moisture carryover to the turbine. Scram occurs essentially simultaneously with the turbine trip. After this point in the transient, the reactor vessel and core response is very similar to that described for the Electrical Load Rejection with Bypass Failure event. Neutron flux reaches a peak value of 328% of its initial value and the core average surface heat flux reaches a peak value of 126% of its original value. This results in a ΔCPR of 0.37. The peak reactor vessel pressure (at the bottom of the vessel) is 1264 psig. Thus, unacceptable safety result 3 is avoided. Unacceptable safety results 1 and 2 are avoided by establishing core thermal operating limits based on the results of the analysis of the event such that the MCPR Safety Limit is not exceeded.

The Feedwater Controller Failure, Maximum Demand event is a limiting event, and as such is reanalyzed each cycle to evaluate the required core thermal operating limits (e.g., MCPR). The event is normally analyzed with the Turbine Bypass system both operable and inoperable. This approach provides less restrictive thermal operating limits when the Turbine Bypass system is operable and allows for continued operation of the reactor if the system is inoperable. Similarly, the feedwater controller failure, maximum demand event is the most limiting pressurization event that can occur during continuous operation with one Turbine

Control Valve (TCV) or Turbine Stop Valve (TSV) closed. Restrictions for operation with one TCV or TSV closed have been established and are administratively controlled in operating procedures. The Feedwater Controller Failure, Maximum Demand event is the most limiting pressurization event that can occur during continuous operation with a TCV or TSV closed. (Ref. 14.0.14).

14.5.2.3 Loss of Feedwater Heating

Feedwater heating can be lost if one or more steam extraction lines are shut because of a loss of power to the extraction line valves, producing a gradual cooling of the feedwater. The reactor vessel receives cooler feedwater which produces an increase in core inlet subcooling. Due to the negative void reactivity coefficient, an increase in core power results.

For analytical purposes, the reactor is assumed to be at rated power and minimum licensed core flow conditions when feedwater heating is lost. Due to its slow progression, this event is treated as a quasi-steady state transient and is analyzed with a steady-state core simulator. The loss of heating event assumes a feedwater temperature reduction of 100°F (design basis for feedwater system) from 381°F to 281°F. Nuclear system pressure remains essentially constant during this event; therefore, unacceptable safety result 3 is avoided. The uncorrected ΔCPR for this event is approximately 0.12, which is significantly less than the uncorrected ΔCPR for the Feedwater Controller Failure event. Thus, unacceptable safety results 1 and 2 are avoided. For off-rated flow conditions, the severity of this event can increase. However, the power and flow biased thermal limits implemented by the ARTS Improvement Program (Reference 14.0.4) assure that the consequences of the event remain bounded and all unacceptable safety results are avoided.

As discussed in Section 3.7.5.4, the feedwater piping at PBAPS can lead to unbalanced or asymmetric feedwater temperature in the feedwater lines leading to the reactor, when some of the feedwater heaters are not in service. A special Loss of Feedwater Heating (LFWH) event can occur when the temperature decrease is asymmetric. Specifically, when operating with two reactor feedwater pumps and two feedwater heater strings, a total loss of the 3^{rd}, 4^{th}, and 5^{th} stage heating in one of the operating heater strings can lead to a condition that is more severe than the LFWH event analyzed above. For this reason, a thermal limit penalty on LHGR is imposed when operating in this configuration of feedwater pumps and feedwater heater strings.

14.5.2.4 Shutdown Cooling (RHRS) Malfunction – Decreasing Temperature

A shutdown cooling malfunction leading to a moderator temperature decrease could result from misoperation of the cooling water controls for the RHRS heat exchangers. The resulting temperature decrease causes a slow insertion of positive reactivity into the core. If the reactor were critical or near critical, a very slow reactor power increase could result. If no operator action were taken to control the power level, a high neutron flux reactor scram would terminate the transient without fuel damage and without any measurable nuclear system pressure increase.

14.5.3 Events Resulting in a Positive Reactivity Insertion

Events that result directly in positive reactivity insertions are the results of rod withdrawal errors and errors during refueling operations. The following events result in a positive reactivity insertion:

1. Continuous rod withdrawal during power range operation.
2. Continuous rod withdrawal during reactor startup.
3. Control rod removal error during refueling.
4. Fuel assembly insertion error during refueling.
5. Misoriented Bundle.

14.5.3.1 Continuous Rod Withdrawal During Power Range Operation

Control rod withdrawal errors are considered over the entire power range from any normally expected rod pattern. The continuous withdrawal, from any normal rod pattern, of the maximum worth rod results in a moderate core transient mitigated by the action of the Rod Block Monitor. MCPR remains greater than the MCPR Safety Limit during the transient. The increase in nuclear system pressure is small and is not considered for this event. The system will stabilize at the higher power level with neither fuel damage nor nuclear system process barrier damage. Thus, unacceptable safety results 1 through 3 are avoided. A control rod withdrawal error is not considered a limiting event. However, a cycle specific analysis is performed and the more limiting of the cycle specific and the ARTS generic values are used as the limits.

As part of the Rod Block Monitor (RBM) system modification included in the ARTS Improvement Program (Reference 14.0.4), the consequences of the rod withdrawal error transient are based on the ARTS generic (statistical) rod withdrawal error analysis. The ARTS-based generic maximum ΔCPR for the rod withdrawal error event is 0.13 for the 108% RBM setpoint. The rod withdrawal error event is initiated by an operator erroneously selecting and continuously withdrawing the maximum worth control rod. The core nuclear dynamic reactivity parameters are based on the cycle peak hot excess reactivity state point and the control rod patterns used to

simulate the event are assumed to be at nominal (i.e., expected) conditions. During this event, the core average power increases until the event is terminated by the Rod Block Monitor which does not allow further control rod withdrawal.

The calculated ΔCPR for this event at rated power and flow is the more limiting of the cycle specific result and the ARTS generic limits. These limits are established for the 108%, 111%, 114% and 117% analytical setpoints. The cycle specific calculation confirms the RBM bypass OLMCPR and that the applicable fuel thermal mechanical limits are met.

Since this event cannot result from a single operator error or component malfunction, it is not considered an abnormal operational transient. It is presented to show the use of the RBM.

14.5.3.2 Continuous Rod Withdrawal During Reactor Startup

The evaluation of the Continuous Rod Withdrawal During Reactor Startup described below was based on pre-EPU conditions. The evaluation of the Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition event for current conditions, like that for pre-EPU described below, is a comparison of the expected maximum increase in peak fuel enthalpy with the acceptance criterion of 170 cal/gram. The pre-EPU analysis is based on Reference 14.0.35. **The current analysis is based on a conservative generic methodology described in reference 14.0.36. Although no changes in peak fuel enthalpy were expected due to EPU or MUR uprates because an RWE is a localized low-power event, the reported enthalpy from this generic methodology was scaled based on rated thermal power**. The RWM installed provides the same level of protection for GE (GNF) fuel, and **Banked Position Withdrawal System** BPWS is used at power levels below the lower **Low Power Setpoint Analytical Limit (**LPSP AL). The evaluation of this event for current conditions considering these features and GE (GNF) fuel demonstrates the acceptance criterion is met. If the peak fuel rod enthalpy is conservatively assumed to increase by a factor **proportional to the increases in reactor power**, the RWE peak fuel enthalpy at current conditions will be 74 cal/gram. This enthalpy is well below the acceptance criterion of 170 cal/gram.

14.5.3.2.1 Identification of Causes and Frequency Classification

It is postulated that during a reactor startup, a single control rod is inadvertently withdrawn continuously due to a procedural error by the operator and operator failure to acknowledge continuous alarm annunciations prior to safety system actuation. The probability of initial causes or errors of this event alone is considered low enough to be categorized as an infrequent incident. The probability of further development of this event is low, because it is contingent upon the failure of the Rod Worth Minimizer (RWM) system, together with a high worth out-of-sequence rod selection contrary to procedures.

The WRNMS has period-based functions that will stop a continuous rod withdrawal by initiating a rod block if the flux excursion, caused by rod withdrawal, generates a period shorter than 20 seconds. The period-based functions also initiate a scram trip if the flux excursion generates a period shorter than 10 seconds. Any single WRNMS rod block trip initiates a rod block. One out of two taken twice logic is used to initiate a scram.

For this transient to happen, a large reactivity addition must be introduced. The reactor must be critical, with control rod density greater than 50%.

14.5.3.2.2 Sequence of Events and System Operation

14.5.3.2.2.1 Sequence of Events

Control rod withdrawal errors are not considered credible in the startup and low power ranges. The RWM system prevents the operator from selecting and withdrawing an out-of-sequence control rod. However, the sequence of events of a postulated continuous control rod withdrawal error during reactor startup is shown in Table 14.5.2.

14.5.3.2.3 Core and System Performance

14.5.3.2.3.1 Analysis Method and Assumptions

The analysis uses the reactivity insertion analysis code that is a two-dimensional adiabatic code assuming no heat transfer to the coolant. The analysis consists of four steps. In Step 1, with the error rod being continuously withdrawn from full-in, the model is used to calculate the average power and period change as a function of time with a continuous reactivity insertion simulating the rod withdrawal error (RWE). In Step 2, the power versus time data are used as input to a calculation of the WRNMS rod block and scram trip times. Both the rod block trip and scram trip times

are then determined. In Step 3, the reactivity insertion input to the adiabatic model is adjusted such that after the period reaches the rod block setpoint (20 sec), there is no further reactivity insertion. The RWE transient is than recalculated by the model with the adjusted reactivity input. The reactor scram time is also adjusted based on the time determined in Step 2. The calculated fuel enthalpy does not consider local peaking effect. In Step 4, the peak fuel enthalpy that includes the local peaking effect is calculated. Other assumptions used in the analysis are:

1. The standard BWR data of the adiabatic model is used.

2. The scram reactivity shape is derived from the design core, assuming no failing rods and same scram speed for all rods.

3. Six delayed neutron groups are assumed.

14.5.3.2.3.2 Analysis Conditions and Results

1. Analysis Conditions

    a. The reactor is assumed to be in the critical condition before the control rod withdrawal, with an initial power of 0.1% rated, and a core average temperature of 286°F.

    b. The worth of the withdrawn rod is 2.5% Δk from full-in to full-out.

    c. The control rod withdrawal speed is 3.11 in/s, the nominal withdrawal speed.

2. Analysis Results

    With this 2.5%Δk reactivity insertion, the flux excursion generates a period of approximately 0.5 seconds. The rod block trip is initiated at 7 seconds after the start of the transient. The scram is initiated at about 8 seconds. The event is terminated by the scram. The peak fuel enthalpy reached is approximately 19.9 cal/g, which is 2.9 cal/g higher than the initial fuel enthalpy.

14.5.3.2.3.3 Evaluation Based on Criteria

Due to the effective protection function of the period-based trip function, the fuel enthalpy increase is small. The criterion of 170 cal/gm for fuel enthalpy increase under RWE event is satisfied. It is concluded that a RWE at low power is not a limiting event, and thus, need not be reanalyzed for future fuel reloads.

14.5.3.2.3.4 Barrier Performance

An evaluation of the barrier performance is not made for this event, because there is no fuel damage in this event and only with mild change in gross core characteristics.

14.5.3.2.3.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, because no radioactive material is released from the fuel.

14.5.3.3 Control Rod Removal Error During Refueling

The nuclear characteristics of the core assure that the reactor is subcritical even in its most reactive condition with the most reactive control rod fully withdrawn during refueling. When the mode switch is in REFUEL, only one control rod can be withdrawn. Selection of a second rod initiates a rod block, thereby preventing the withdrawal of more than one rod at a time.

Therefore, the refueling interlocks prevent any condition which could lead to inadvertent criticality due to a control rod withdrawal error during refueling.

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, thus eliminating any hazardous condition.

14.5.3.4 Fuel Assembly Insertion Error During Refueling

The core is designed such that it can be made subcritical under the most reactive conditions with the strongest control rod fully withdrawn. Therefore, any single fuel bundle can be positioned in any available location without violating the shutdown criteria, providing all the control rods are fully inserted. The refueling interlocks require that all control rods must be fully inserted before a fuel bundle may be inserted into the core.

14.5.3.5 Misoriented Bundle Event

The core is designed such that the fuel bundles are required to be placed in each core location in a precise orientation. A misoriented fuel bundle during power operation potentially could lead to unmonitored violations of core thermal limits. Proper orientation of fuel bundles in the reactor core is easily verified by multiple independent visual observation; therefore, misorientation of a fuel bundle is considered to have a very low

probability. However, this event is considered to have a similar probability as other abnormal operational transients.

The potential impact (i.e., ΔCPR) of a misoriented bundle is dependent on the particular nuclear design of the fuel bundle. Therefore, this event is evaluated on a cycle specific basis to determine if it is a limiting event. Unacceptable safety results 1 and 2 are avoided by establishing MCPR operating limits such that the MCPR Safety Limit would not be exceeded in the event a misoriented bundle event occurred.

14.5.4 Events Resulting in a Reactor Vessel Coolant Inventory Decrease

Events that result directly in a decrease of reactor vessel coolant inventory are those that either restrict the normal flow of fluid into the vessel or increase the removal of fluid from the vessel. Four events are identified as causing the most severe transients in this category:

1. Pressure regulator failure.
2. Inadvertent opening of a relief or safety valve.
3. Loss of feedwater flow.
4. Loss of auxiliary power.

14.5.4.1 Pressure Regulator Failure

Potential system pressure increases or decreases can be produced by pressure regulator failure. If the controlling regulator fails in a closed direction, the backup regulator takes over control of the turbine admission valves, preventing a serious transient. The backup regulator takes over control bumplessly, resulting in no disturbance. However, if a pressure regulator fails downscale while the second regulator is out of service the resulting transient could be limiting. Operation with only one pressure regulator is allowed at all power levels Reference 14.0.32).

If the controlling regulator fails in an open direction, the backup regulator would take over control with no disturbance. However, if both of the regulators fail in an open direction, the turbine admission valves can be fully opened, and the turbine bypass valves can be partially opened. This action initially results in decreasing coolant inventory in the reactor vessel as the mass flow of steam leaving the vessel exceeds the mass flow of water entering the vessel. The total steam flow rate resulting from a pressure regulator malfunction is limited by the turbine controls to about 115 percent of rated flow.

Figure 14.5.8 graphically shows the transient resulting from a pressure regulator malfunction in which a 150-percent steam flow demand is assumed as a most severe case. This results in the full opening of the turbine control valves and turbine bypass valves. For analytical purposes, the reactor is assumed to be a rated power. The analysis of this event accounts for uncertainty in reactor power as described in Section 14.0. The depressurization results in the formation of voids in the reactor coolant causing a rapid rise in sensed reactor vessel water level. At about 2.5 seconds, the reactor high water level setpoint (Level 8) is reached and a main turbine trip and feedwater turbine trip are initiated. A reactor scram is initiated on turbine stop valve position. After the turbine trip and reactor scram, the reactor pressure begins to increase because the steam generation rate in the reactor core is greater than the turbine bypass system capacity. Pressure increases until a high pressure recirculation pump trip is initiated and the SRVs open to relieve pressure. After the SRVs reclose at about 10 seconds, the reactor pressure begins to decrease as the steam generation rate in the core becomes less than the turbine bypass system capacity. The pressure continues to decrease until the main steam line low pressure isolation setpoint (a conservatively bounding value of 750 psig is assumed for this analysis) is reached and an MSIV closure is initiated. Following MSIV closure, the system depressurization is terminated and the vessel water level begins to fall as the system pressure increases. The reactor, being scrammed and isolated, can now be brought to a controlled shutdown. The reactor vessel isolation limits the duration and severity of the depressurization so that no significant thermal stresses are imposed on the nuclear system process barrier (Reference 14.0.10). After the rapid portion of the transient is complete and isolation is effective, the nuclear system relief valves operate intermittently to relieve the pressure rise resulting from decay heat generation. No reduction in fuel thermal margins occur; thus, unacceptable safety results 1 and 2 are avoided. Because the rapid portion of the transient results in depressurization of the nuclear system and because the relief valves need operate only to relieve the pressure increase due to decay heat, the nuclear system process barrier is not threatened by high internal pressure for this pressure regulator malfunction. Unacceptable safety result 3 is, therefore, avoided.

Reference 14.0.10 describes the results of a "slow" Pressure Regulator Failure in which the Level 8 turbine trip and scram are not reached. Reactor scram is initiated by the MSIV isolation. The consequences of this event are no more severe than the "fast" event discussed previously. The Pressure Regulator Failure events are non-limiting events and are bounded by other events for operation at rated power.

14.5.4.2 Inadvertent Opening of a Relief Valve or Safety Valve

The opening of a relief or safety valve allows steam to be discharged into the primary containment. The sudden increase in the rate of steam flow leaving the reactor vessel causes the reactor vessel coolant (mass) inventory to decrease. The result is a mild depressurization transient.

The pressure regulator senses the nuclear system pressure decrease and closes the turbine control valves far enough to maintain constant reactor vessel pressure. Reactor power settles out at nearly the initial power level. Because pressure decreases throughout the transient, the nuclear system process barrier maximum allowable internal pressure (unacceptable safety result 3) is not approached. There is no significant change in ΔCPR for this event; therefore, unacceptable safety results 1 and 2 are avoided. This is a non-limiting event. The small amounts of radioactivity discharged with the steam are contained inside the primary containment; the situation is not significantly different, from a radiological viewpoint, than that normally encountered in cooling the plant using the relief valves to remove decay heat.

14.5.4.3 Loss of Feedwater Flow

A loss of feedwater flow results in a situation where the mass of steam leaving the reactor vessel exceeds the mass of water entering the vessel resulting in a net decrease in the coolant inventory available to cool the core.

The Loss of Feedwater Flow event represents the design basis for the performance of the reactor core isolation cooling (RCIC) system on all BWR/4, 5 and 6 plants. The following criteria are applied to this event:

1) The RCIC system shall maintain sufficient water level inside the core shroud to assure that the top of active fuel remains covered throughout the event.

2) The RCIC system shall maintain wide range sensed reactor water level high enough that the very low level instrument trip setpoint (Level 1) for low pressure emergency core cooling system initiation and MSIV closure is not activated.

This transient event does not pose any direct challenge to the reactor vessel or core in terms of a power or pressure increase. All unacceptable safety results are avoided. However, it is included in the evaluation to provide assurance that sufficient

makeup water capability is available to keep the core covered when all normal feedwater flow is lost.

The Loss of Feedwater Flow event was addressed on a generic basis for each of the BWR/4,5 and 6 class of plants in Reference 14.0.2. For BWR/4s, the generic analyses are bounding for plants with vessel diameters ranging from 183 inches to 251 inches (PBAPS class of plant). The system parameters used in the generic analyses for the BWR/4-251 class of plant are similar to the PBAPS plant-specific values.

14.5.4.3.1 Peach Bottom Specific Analysis

A Peach Bottom specific analysis was performed **in support of the EPU project, which continues to bound MUR conditions.** This analysis assumed failure of the HPCI system and used only the RCIC system to restore the reactor water level. It is also assumed a decay heat level of ANS 5.1-1979 with a two-sigma uncertainty was specified in CLTR. The assumed decay heat level for the EPU analysis was NAS 5.1-1979 decay heat +10 percent, which bound ANS 5.1-1979 +two sigma.

The results of the LOFW analysis show that the minimum water level inside the shroud is 129 inches above the top of active fuel (TAF). After the water level is restored, the operator manually controls the water level, reduces reactor pressure, and initiates RHR shutdown cooling.

The following is the general sequence of events in the analysis. The reactor is assumed to be at 4030 MWt when the LOFW occurs. The initial level in the reactor is conservatively set at the low-level scram setpoint and reactor FW is instantaneously isolated at event initiation. Scram is initiated at the start of the event. When the level decreases to the low-low level setpoint, the RCIC system is initiated. Only RCIC flow is credited to recover the reactor water level. There are no additional failures assumed beyond the failure of the HPCI system.

This LOFW analysis is performed to demonstrate acceptable RCIC system performance. The design basis criterion for the RCIC system is confirmed by demonstrating that it is capable of maintaining the water level inside the shroud above the TAF during the LOFW transient. RCIC flow to the vessel begins at 68 seconds and the minimum level is reached at 1273 seconds. The minimum level (see Figure 14.5.10) is maintained at least 129 inches above the TAF, thereby demonstrating acceptable RCIC system performance. There are no applicable equipment out of service assumptions for this transient.

An operational requirement is that the RCIC system restores the reactor water level while avoiding ADS timer initiation and MSIV closure activation functions associated with the low-low-low reactor water level setpoint (Level 1). This requirement is intended to avoid unnecessary initiations of safety systems. This requirement is not a safety-related function. The results of the LOFW analyses show that the nominal Level 1 setpoint trip is avoided.

14.5.4.3.2 Deleted

14.5.4.4 Loss of Auxiliary Power

NOTE: The description of this transient that follows is based on pre-EPU conditons. It was not re-analyzed at EPU because it is non-limiting and bounded by other events. Similarly, this was not re-analyzed at MUR because it is non-limiting and bounded by other events.

Loss of auxiliary power is defined as an event which de-energizes all buses that supply power to the unit auxiliary equipment such as recirculation pumps, condensate pumps, and circulating water pumps. This transient is classified as an event resulting in reactor vessel coolant inventory decrease because this is one of the dynamic results of the event. Two methods of experiencing this event are postulated:

1. A trip(s) or fault(s) occurring within the auxiliary power distribution system itself without transfer to outside power sources.

2. Complete loss of all external connections to the grid.

The following assumptions are used in evaluating the first transient:

1. All electric pumps are tripped at time = 0. Coastdown times are determined from momentum computations.

2. The RPS M-G sets are assumed to coast down in 5 sec to the point where scram and main steam lines isolation occur.

3. Loss of the main condenser circulating water pumps causes condenser vacuum to drop to the turbine trip setting in about 6 sec. The short period (about 2 sec) during which bypass flow would be permitted is neglected.

The initial portion of the transient is very similar to the loss of feedwater flow. Initiation of scram, isolation valve closure, and turbine trip all occur between about 5 and 6 sec; after the initial period, the characteristics of the transient change to that of a sudden stoppage of steam flow. The relief valves open for a short time, then sequentially reclose as the remainder of the stored thermal energy in the fuel is dissipated. The peak pressure is limited by the opening of the SRVs and the nuclear system pressure safety limit is not reached. The pressure rise and pressure relief cycle are continued with slower frequency and shorter relief discharges as the decay heat rate decreases up to the time that the RHRS heat exchangers can dissipate the heat. Wide range sensed reactor vessel water level drops to the HPCIS and RCICS initiation set point (Level 2). The HPCIS and RCICS maintain reactor water level to the required value.

For the transient which results from loss of all grid connections, the same sequence of events occurs except that the unit also experiences a generator load rejection and its associated fast control valve closure scram at the beginning of the transient.

No increase in neutron flux occurs due to the trip scrams and recirculation pumps trip. No increase in fuel surface heat flux occurs and the thermal behavior of the reactor is similar to that which results from a recirculation pump trip. Reactor pressure peaks at about 100 psi above initial values but at a slightly faster rate than for the previous case. Wide range sensed level initiates RCICS and HPCIS at 30 sec after the transient has started.

The RCICS reaches full flow within 50 sec after actuation, and by itself can raise reactor water level in time to prevent initiation of the low pressure portion of the CSCSs. The RCICS can maintain adequate reactor vessel water level during the shutdown period when the vessel is otherwise isolated. The fuel transient is almost the same as that experienced during the trip of both recirculation pump drive motors. MCPR remains above the Safety Limit MCPR in either case, and no fuel damage occurs. The long-term water level transient was conservatively evaluated (based on the initial core) considering RCIC operation only, beginning at 50 sec and reaching full RCIC flow (600 gpm) at 80 sec. The minimum calculated water level is 90 in. above the top of the active fuel, providing ample margin. Although the original analysis used a full flow start time for RCIC of 30 seconds, further analysis has shown that a full flow start time for RCIC of 50 seconds is acceptable. (Reference 14.6.20) Each of these events is non-limiting and bounded by the Electrical Load Rejection Without Bypass event. This remains a non-limiting event at MUR conditions (4016 MWt).

14.5.5 Events Resulting in a Core Coolant Flow Decrease

Events that result directly in a core coolant flow decrease are those that affect the reactor recirculation system. The following events result in the most significant transients in this category:

1. Recirculation flow control failure - decreasing flow.

2. Trip of one recirculation pump.

3. Trip of two recirculation ASDs.

4. Recirculation pump seizure.

5. Inadvertent EOC-RPT Initiation

With a loss of forced reactor coolant flow, additional core void will form and cause a decrease in reactor power through void feedback. The thermal inertia of the fuel will cause thermal power to lag behind the neutron flux and core flow decay. Critical power will reduce due to core flow reduction but the operating power will be sustained for a short period of time. This combination causes the calculated MCPR to decrease to a lower value, but not to SLMCPR. The fuel thermal margin is influenced by the rotating inertia of the motor-generator sets since it determines coast-down speed.

14.5.5.1 Recirculation Flow Control Failure - Decreasing Flow

The malfunction is one in which the flow controller for one ASD fails in such a way that the output signal changes in the direction of zero speed. This transient is similar to but less severe than the trip of one recirculation pump. A trip of one recirculation pump is evaluated in paragraph 14.5.5.2.

14.5.5.2 Trip of One Recirculation Pump

This transient is the result of opening the 13.8 kV breaker for one ASD. This transient is similar in nature to the Two Recirculation Pumps Trip transient, and has been evaluated by GE Hitachi in calculation PEAE-EPU-16. Calculation PEAE-EPU-16 determined that the One Recirculation Pump Trip transient is less severe than the Two Recirculation Pump Trip transient. A trip of two recirculation pump is evaluated in paragraph 14.5.5.3.

14.5.5.3 Trip of Two Adjustable Speed Drives

The results of a two ASD trip are identical to that of a two recirculation pump trip; specifically, an EOC-RPT. This transient

results if the power supply to the ASDs is lost. When the two ASDs trip, the recirculation pumps also trip and come to a rapid flow coastdown (no ASD inertia exists). The trip of two recirculation pumps can result in a core flow decrease of up to 75 percent; however, it results in a non-limiting transient for core thermal limits. (Reference 30)

14.5.5.4 Recirculation Pump Seizure

Seizure of a recirc pump is considered an accident due to the low probability of such an event, therefore, the criteria placed on evaluating a pump shaft seizure are the same for other accidents, such as a LOCA. However the consequences of this event are very mild compared to accident acceptance criteria. The seizure of a pump shaft causes an instantaneous cessation of pump rotational speed and a rapid decrease in drive and subsequently core flow. The sudden decrease in core coolant flow rate while operating at rated power results in degradation of core heat transfer. While critical power ratio (CPR) will decrease, the change will be small and remain above the MCPR criterion applied to moderate frequency events.

The rapid core flow decrease causes a water level swell in the annulus outside the core shroud. Under some conditions the rise in water level for this event will reach the high level trip set point and cause a feedwater pump trip, a main turbine trip, and subsequently a reactor scram. This turbine trip would be initiated from less than rated power. The peak neutron flux and average cladding surface heat flux would not increase significantly above their initial values. A pump seizure event that results in a turbine trip is depicted in Figure 14.5.13.

The pump seizure event has also been analyzed for single loop operation (SLO) and determined to provide a greater challenge to MCPR Safety Limit than the pump seizure from two loop operation (Ref. 14.0.16). For some fuel types, adjustments to the OLMCPR may be necessary for SLO to ensure that a postulated pump seizure event does not cause the MCPR Safety Limit to be exceeded. This event is not limiting for operation at rated **power.** SLO pump seizure MCPR limits remain unchanged for operation at rated power (4016 MWt) **because SLO operation limits are unchanged for the MUR power level.**

14.5.5.5 Inadvertent Initiation EOC-RPT

An inadvertent initiation of the EOC-RPT System is identical to the trip of the two ASDs (no drive inertia credited). Only the inertia of the pump motors and moving fluid is credited. The EOC-RPT transient is bounded by the pump seizure event and other transients. Like the trip of the recirculation ASD and the

seizure of recirculation pump shaft, there are no adverse effects on the fuel or on plant systems, although a high water level trip may occur.

14.5.6 Events Resulting in a Core Coolant Flow Increase

Events that result directly in a core coolant flow increase are those that affect the reactor recirculation system. The following events result in the most significant transients in this category:

1. Recirculation flow control failure - increasing flow.
2. Startup of idle recirculation pump.

14.5.6.1 Recirculation Manual Control Station Failure - Increasing Flow

A possibility exists for an unplanned increase in core coolant flow resulting from a recirculation flow control station malfunction. Failure of the manual controller can result in a speed demand increase for one recirculation pump. The simultaneous runout of the recirculation loops is not credible because all combined modes of recirculation flow control have been removed, leaving each loop controlled only by the individual manual flow control station.

Slow Flow Increase

Analyses demonstrate that a slow increase in recirculation flow results in a greater challenge to the fuel than a fast flow increase. Although a fast flow increase results in a higher neutron flux, a high flux scram is typically initiated, terminating the power transient before the cladding heat flux reaches its maximum achievable value. On the other hand, a slow run up of recirculation flow results in higher heat fluxes, and thus, is the limiting event for establishing flow dependent multipliers for operating limits(Reference 14.0.9).

In the case of a slow run up of recirculation flow, the neutron flux does not spike and stays below the scram setpoint. In this situation, the heat flux obtains its maximum value and causes the greatest change in MCPR. If one conservatively assumes the flow runup event begins with the core at its fuel thermal operating limits, it is possible the change in the MCPR during the flow increase could cause the fuel safety limits to be exceeded. For this reason flow dependent multipliers are used to adjust the operating limits when operating at less than rated flow conditions.

The Peach Bottom flow dependent operating limits multipliers are based on generic analyses developed for a slow run up of two

recirculation pumps, where the maximum core flow is clamped at 100%. However, as discussed above, the recirculation pumps flow controllers at Peach Bottom are arranged so that runout of only one pump is credible. Analyses have been performed which demonstrate that the results of a single pump runout to the maximum system capability (assuming no flow clamping) are less severe than a two pump runout clamped at 100% core flow. These analyses also conservatively assume the Turbine Bypass Valves are out-of-service (TBVOOS), bounding that contingency mode of operation (Reference 14.0.19). Therefore, based on this approach, the Peach Bottom flow dependent multipliers yield conservative operating limits. Calculation PEAE-EPU-16 further demonstrated that with the ASD, this event is bounded by the Two Recirculation Pump Trip event. A trip of two recirculation pumps is evaluated in paragraph 14.5.5.3.

<u>Fast Flow Increase</u>

Analyses performed at various rates of flow increase show that a reactor scram on high neutron flux is expected for recirculation pump speed increase rates greater than about 5% speed per second. The severity of the transient depends on the initial power and flow as well as the rate of increase. The reactor dome pressure and water level do not change appreciably during the transient. The effect of greatest interest is the decrease in MCPR. The most limiting flow increase transient is one that starts from relatively low flow, and thus, permits the greatest change in MCPR. For a flow run up from 68% power and 43% core flow at a rate of 5% speed per second, the neutron flux reaches a peak value of 121%. This results in a high flux scram, limiting the cladding heat flux to 104% of rated (Reference 14.0.9). The change in MCPR is calculated to be 0.23, however, the flow biased MCPR multiplier provides significant additional MCPR margin and makes this a non-limiting event.

Calculation PEAE-EPU-16 further simulated this event by applying the maximum (Breakdown) torque that the pump can develop. In that analysis, the maximum speed occurs at approximately 1.8 sec and the pump speed is conservatively maintained at or above this level for the duration of the transient, which makes this a non-limiting transient. Figure 14.5.14 provides plots for this transient.

This event remains non-limiting for operation at rated power (**4016** MWt).

14.5.6.2 <u>Startup of Idle Recirculation Pump</u>

The startup of an idle recirculation pump is also a core coolant flow increase event. The increase in core coolant flow results in

a neutron flux increase and subsequently a fuel surface heat flux increase.

As discussed in Ref. 14.0.17, with the implementation of ARTS/MELLLA, the startup of an idle recirculation pump (idle loop startup - ILS) is analyzed with the idle loop temperature 50°F below the active recirculation loop. As part of the implementation of ARTS/MELLLA, generic analyses were performed at several core power and flow operating points with the idle loop temperature 50°F below the active recirculation loop. These generic analyses show that, when a 50°F loop ΔT is assumed, the ARTS flow and power dependent thermal limits ensure the ILS event is bounded by the consequences of other events. Since this event is not limiting and is protected by flow and power dependent thermal limits, unacceptable safety results 1 and 2 are avoided.

ASME Upset category stress analyses have also been performed for the ILS event. These analyses also assume that the idle loop temperature is 50°F below the active recirculation loop. These analyses were performed for the reactor vessel nozzle and the reactor recirculation system and were shown to be consistent with the bases of ASME Upset category evaluations of thermal stresses. Since these analyses are consistent with the bases of ASME Upset category evaluations, unacceptable safety result 3 is avoided. This event remains non-limiting for operation at rated power (4016 MWt).

Table 14.5.1

(Deleted)

TABLE 14.5.2

CONTINUOUS ROD WITHDRAWAL
SEQUENCE OF EVENTS

Time (Sec)	Events
---	The reactor is critical and operating in the startup range.
>0	The operator selects and withdraws an out-of-sequence control rod at the maximum normal drive speed of 3.11 in/sec.
	The RWM fail to block the selection (selection error) and continuous withdrawal (withdraw error) of the out-of-sequence rod.
	Neutron flux increases rapidly (due to the continuous reactivity addition) with a very short period.
7	The WRNMS Period-Based Rod Block Trip initiates rod block due to short period (less than 20 seconds).
8	The WRNMS Period-Based Scram Trip initiates reactor scram due to short period (less than 10 seconds).
9	Reactor is scrammed and event is terminated.

14.6 ANALYSIS OF DESIGN BASIS ACCIDENTS

14.6.1 Introduction

The methods for identifying and evaluating accidents (paragraph 14.4.3) have resulted in the establishment of design basis accidents for the various accident categories as follows:

	Accident Category	Design Basis Accident
a.	Accidents that result in radioactive material release from the fuel with the nuclear system process barrier, primary containment, and secondary containment intact.	Rod drop accident (single control rod)
b.	Accidents that result in radioactive material release directly to the primary containment.	LOCA (rupture of one recirculation loop)
c.	Accidents that result in radioactive material release directly to the secondary containment with the primary containment initially intact.	Accidents in this category are less severe than those in Categories d and e below
d.	Accidents that result in radioactive material releases directly to the secondary containment with the primary containment not intact.	Refueling accident (fuel assembly drops on core during refueling)
e.	Accidents that result in radioactive material releases outside the secondary containment.	Steam line break accident (main steam line breaks outside secondary containment)

An investigation of accident possibilities reveals that accidents in Category c are less severe than those in Categories d and e. Category c includes two varieties of accidents: failures of the nuclear system process barrier inside the secondary containment and failures involving fuel that is located outside the primary containment but inside the secondary containment. Under the accident selection rules described in paragraph 14.4.3, a main steam line break inside the reactor building is the most severe accident of the first variety, but this accident results in a radioactive release to the environs no greater than that resulting from the main steam line break outside the secondary containment.

Similarly, the most severe accident of the second variety is the dropping of a fuel assembly into the fuel pool, but this results in a smaller radioactivity release to the environs than that resulting from dropping a fuel assembly on the fuel in the reactor vessel during refueling. Because the consequences of accidents in Category c are less severe than those resulting from similar accidents in other categories, the accidents in Category c are not described.

14.6.2 Control Rod Drop Accident

Note: Other than Section 14.6.2.8, the material presented in 14.6.2 is historical and describes the analysis for the original plant design. Current event description and analysis bases for the Control Rod Drop Accident are provided in Reference 14.0.6. The current licensing basis radiological consequences using the AST analysis are provided in Section 14.9

The accidents that result in releases of radioactive material from the fuel with the nuclear system process barrier, primary containment, and secondary containment initially intact are the results of various failures of the CRDS. Examples of such failures are collet finger failures in one CRD mechanism, a control drive system pressure regulator malfunction, and a CRD mechanism ball check valve failure. None of the single failures associated with the control rods or the control rod system results in a greater release of radioactive material from the fuel than the release that results when a single control rod drops out of the core after being disconnected from its drive and after the drive has been retracted to the fully withdrawn position. Thus, this control rod drop accident is established as the design basis accident for the category of accidents resulting in radioactive material release from the fuel with all other barriers initially intact. A highly improbable combination of actual events would be required for the design basis control rod drop accident to occur. The following actual events are required:

1. Failure of the rod-to-drive coupling. The design of the coupling itself reduces the probability of separation. Tests conducted under both simulated reactor conditions and the conditions more extreme than those expected in reactor service have shown that the coupling does not separate, even after thousands of scram cycles. Tests also show that the coupling does not separate when subjected to forces 30 times greater than those which can be achieved with a CRD.

2. Sticking of the control rod in its fully inserted position as the drive is withdrawn. The control rods are

designed to minimize the probability of sticking in the core. The control rod blades, which are equipped with rollers that make contact with the channel walls, travel in gaps between the fuel assemblies with approximately 1/2-in total clearance. Control rods of similar design, now in use in operating reactors, have shown no tendency to stick in the core due to distortion or swelling of the blade.

3. Full withdrawal of the CRD.

4. Failure of the operator to notice the lack of response of neutron monitoring channels as the rod drive is withdrawn.

5. Failure of the operator to verify rod coupling. The control rod bottoms on a seal, preventing the CRD from being withdrawn to the overtravel position. Attempting to withdraw a CRD to the overtravel position provides a method for verifying rod coupling; this verification is required whenever neutron monitoring equipment response does not indicate that the rod is following the drive.

The accident is analyzed over the full spectrum of power conditions. Nuclear excursion results are presented for three points in this range: the cold (68°F) critical condition for moderator and fuel; a hot (547°F) critical condition; and the 10 percent of rated power condition. The results of the rod drop accident initiated from higher than 10 percent power are less severe than the 10 percent power case due to the faster Doppler response. Only the radiological results of the most severe case are presented.

14.6.2.1 Initial Conditions (See note at the beginning of Section 14.6.2)

The following initial conditions are assumed for the three cases presented:

Case A (cold):	Reactor critical Moderator and fuel at 68°F Power level 10^{-8} x rated Rod worth (for dropped rod) 0.025 Δk.
Case B (hot):	Reactor critical Moderator and fuel at 547°F Power level 10^{-6} x rated Rod worth (for dropped rod) 0.025 Δk.

Case C (power): Reactor critical
Moderator and fuel at 547°F
Power level 10^{-1} x rated
Rod worth (for dropped rod) 0.038 Δk.

In considering the possibilities of a control rod drop accident, only the rod worths of the lower curve of Figure 14.6.1 are pertinent at less than 10 percent power. These are the rods which are normally allowed to be moved by operating procedures and the rod worth minimizer. The non-scheduled rods, those described by the central envelope, do not have a withdrawal permissive during the time their worths are greater than the lower curve, so they are held full in by the CRD and cannot drop from the core. If a non-scheduled rod were selected, the rod worth minimizer blocks rod movement. Therefore, the worth of the strongest rod which could be stuck is limited to about 0.01 Δk, and the 0.025 Δk worth assumed for cases A and B is considerably above the rod worth values available for stuck rods under the assumed reactor conditions. In the greater than 10 percent power range, the maximum rod worth is determined by the FLARE(1) and WANDA(2) computer codes and is shown in Figure 14.6.2. Thus, in case C, the rod worth is assumed to be 0.038 Δk.

14.6.2.2 Excursion Analysis Assumptions (See note at the beginning of Section 14.6.2.)

The following assumptions are used in the analysis of the nuclear excursion for each case:

1. The velocity at which the control rod falls out of the core is assumed to be 5 ft/sec. The control rod velocity limiter(3), an engineered safeguard, limits the rod drop velocity to less than this value.

2. Control rod scram motion is assumed to start at about 200 msec after the neutron flux has attained 122 percent of rated flux. This assumption allows the power transient to be terminated initially by the Doppler reactivity effect of the fuel. This assumption is particularly conservative for cases A and B because a high neutron flux scram would be initiated earlier by the WRNM channels.

3. No credit is taken for the negative reactivity effect resulting from the increased temperature of, or void formation in, the moderator. Since the time constant for heat transfer between the fuel and the moderator is long compared with the time required for control rod motion, this effect would be small.

4. No credit is taken for the prompt negative reactivity effect of heating in the moderator due to gamma heating and neutron thermalization.

14.6.2.3 Fuel Damage (See note at the beginning of Section 14.6.2.)

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, as far as fuel failure thresholds are concerned, come from the PULSTAR[5] and ANL-TREAT[6,7] tests.

The Pulstar tests, which used UO_2 pellets of 6 percent enrichment with Zr-2 cladding, achieved maximum fuel enthalpies of about 200 cal/g with a minimum period of 2.83 msec. 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 450 cal/g.

	Test 1	Test 2
Input energy (cal/g)	280	450
Final mean particle diameter (mils)	60	30
Pressure rise rate (psi/sec)	30	600

The ultimate degree of fuel fragmentation and dispersal of the two cases is not significantly different; however, the pressure rise rate in the higher energy test is increased by a factor of 20. This strongly implies that the dispersion rate in the higher energy test was significantly higher than that of the lower energy test. This leads to the logical conclusion that, although a high degree of fragmentation occurs for fuel in the range of 200 to 300 cal/g the breakup and dispersal into the water is gradual and pressure rise rates are very modest. On the other hand, for fuel above the range of 400 cal/g, 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/g is the threshold for eventual fuel cladding perforation. Fuel melting is estimated to occur in the 220 to 280 cal/g range and a minimum of 425 cal/g is required to cause immediate rupture of the fuel rods due to UO_2 vapor pressures.

A parametric analysis was made of the rod drop accident for various starting conditions and rod worths. The results are shown in Figures 14.6.3, 14.6.4, and 14.6.5, and the reduction in final peak fuel enthalpy with increasing initial power level is clearly shown. The cold critical case (case A) is shown as point A on Figure 14.6.3, and the hot standby critical case (case B) is shown as point B on Figure 14.6.4. Figure 14.6.4 is a conservative description of the consequences when the core is at rated temperature and the coolant is boiling. On Figure 14.6.5, the 10 percent of power case (case C) is represented by point C. In these cases, the maximum initial enthalpy generally is not in the fuel which experiences the greatest enthalpy addition during the excursion. If a rod were dropped from a high initial enthalpy region, the results would not be as great as with one dropped from a lower enthalpy region. However, for conservatism, it is assumed that the peak enthalpy increment is added to the maximum fuel enthalpy that existed in the vicinity of the excursion center prior to the accident.

In the hot standby critical case (case B), the power transient is calculated to have a total energy generation of 4,000 MW-seconds. The excursion energy is calculated to be distributed in the fuel such that about 330 fuel rods have enthalpies greater than 170 cal/g. The maximum UO_2 enthalpy is calculated to be 220 cal/g. Essentially no fuel will melt because fuel melting occurs in the range of 220 to 280 cal/g.

The power transient in the 10 percent of power rod drop accident (case C) is less severe than the one at hot standby (case B). The peak enthalpy is about 200 cal/g and only about 50 fuel rods have enthalpies in excess of 170 cal/g.

The power transient in the cold condition rod drop accident (case A) is calculated to be distributed in the fuel such that about 200 fuel rods have enthalpies greater than 170 cal/g. The maximum UO_2 enthalpy is calculated to be 250 cal/g. Approximately 50 lb of UO_2 have enthalpies in excess of 220 cal/g. Because fewer fuel rods are perforated and because the shutdown cooling system would be operating, allowing no radioactivity release to the main condenser, the radiological results of the cold rod drop accident are insignificant when compared to the hot standby critical case.

All of these peak enthalpies are far below 425 cal/g, which is estimated to be the threshold for immediate rupture of fuel rods due to UO_2 vapor pressure. Furthermore, the above peak enthalpies are well below the design limit of 280 cal/g. Thus, there are no damaging pressure pulses as a result of the rod drop accident; the only damage expected would be the failed fuel rods.

14.6.2.4 Fission Product Release From Fuel (See note at the beginning of Section 14.6.2)

The following assumptions are used in the calculation of fission product activity release from the fuel:

1. Three hundred thirty fuel rods fail (case B). This is the largest number of failed fuel rods resulting from the analysis of the rod drop accident over the full spectrum of power conditions.

2. The reactor has been operating at design power until 30 min before accident initiation. When translated into actual plant operations, this assumption means that the reactor was shut down from design power, taken critical, and brought to the initial temperature conditions within 30 min of the departure from design power. The 30-min time represents a conservative estimate of the shortest period in which the required plant changes could be accomplished and defines the decay time to be applied to the fission product inventory for the calculation.

3. The reactor has been operating at design power for 1,000 days prior to the accident. This assumption results in equilibrium concentration of fission products in the fuel. Longer operating histories do not increase significantly the concentration of longer lived fission products.

4. An average of 1.8 percent of the noble gas activity and 0.32 percent of the halogen activity in a perforated fuel rod are assumed to be released. These release percentages are consistent with actual measurements made on defective fuel experiments. The basis for these values is presented in APED-5756[8].

5. The following fission product concentration in the fuel rod plenums are applicable for the core at the time the accident occurs:

Noble gases	1.3×10^5 Ci/MWt
Halogens	2.2×10^5 Ci/MWt

   These concentrations are the result of a nuclear analysis of the fuel assuming operation at design power for 1,000 days followed by a 30-min decay period.

6. No solid fission products are released from the fuel. Because the fraction of solid fission product activity

available for release from the fuel is negligible, this assumption is reasonable.

7. The fission products produced during the nuclear excursion are neglected. The excursion is of such short duration that the fission products generated are negligible compared with the concentration of fission products already assumed present in the fuel.

Using the above assumptions the following amounts of fission product activity are released from the failed fuel rods to the reactor coolant:

Noble gases (Xe, Kr)	6.6×10^4 Ci
Halogens (Br, I)	2.2×10^4 Ci

14.6.2.5 Fission Product Transport (See note at the beginning of Section 14.6.2.)

The following assumptions are used in calculating the amounts of fission product activity transported from the reactor vessel to the main condenser:

1. The recirculation flow rate is 25 percent of rated and the stem flow to the condenser is 5 percent of rated. The 25 percent recirculation flow and 5 percent steam flow are the maximum flow rates expected when the reactor is being taken to power and the main condenser is still being evacuated by the mechanical vacuum pump. The recirculation flow rate is used in determining the volume of coolant in which the activity released from the fuel is deposited. The 5 percent steam flow rate is greater than that which would be in effect at the reactor power level assumed in the initial conditions for the accident. This assumption is conservative because it results in the transport of more fission products through the steam lines than would actually be expected. Because of the relatively long fuel-to-coolant heat transfer time constant, steam flow is not significantly affected by the increased core heat generation within the time required for the MSIV's to achieve full closure.

2. The MSIV's are assumed to receive an automatic closure signal 0.5 sec after the released activity reaches the main steam line radiation monitors, and to be fully closed 10 sec from the receipt of the closure signal. The 10-sec closure time of the MSIV's is the maximum closing time permitted by valve setting. The total time required to isolate the main steam lines (10.5

sec), combined with the assumptions in 1, allows calculation of the total amount of fission product activity transported to the condenser before the steam lines are isolated.

3. All of the noble gas activity is assumed to be released to the steam space of the reactor vessel. None is retained in the liquid reactor coolant.

5. The ratio of the halogen concentration in steam to that of water is assumed to be $3x10^{-5}$. Measurements of reactor coolant water and condensate at Dresden Nuclear Power Station Unit No. 1 indicate that the steam-to-water halogen concentration ratio is in the range of 1×10^{-5} to 3×10^{-5}.

5. Water carryover in the main steam lines is assumed to be 0.1 percent of the total mass of steam transferred to the condenser. Measurements of the steam separation effected by the same types of separators used in this reactor vessel show that water carryover is less than 0.1 percent even at rated steam flow. The carryover fraction permits computation of the halogen activity carried to the main condenser in the water entrained in the steam.

6. None of the fission products released from the fuel are assumed to plate out. Using the listed assumptions, the following amounts of fission product activities are carried to the condenser:

Noble gases	6.0×10^{3} Ci
Iodine 131	3.0×10^{-1} Ci
Iodine 132	4.6×10^{-2} Ci
Iodine 133	1.6×10^{-1} Ci
Iodine 134	3.7×10^{-2} Ci
Iodine 135	8.8×10^{-2} Ci

14.6.2.6 Fission Product Release to Environs (See note at the beginning of Section 14.6.2.)

The following assumptions and initial conditions were used in the calculation of fission product activity released to the environs:

1. All of the noble gas activity released to the condenser is assumed to be airborne in the condenser and available for release to the environs at a constant leak rate from the condenser of 0.5 percent/day.

2. The halogen activity airborne in the condenser is a function of the partition factor, air volume, and liquid volume in the condenser for which the respective values apply, PF = 100, VA = 2.08×10^5 cu ft, VW = 1.2×10^4 cu ft.

3. A high radiation signal from the main steam line radiation monitors results in isolation of the mechanical vacuum pump and vacuum pump suction line.

4. The fission products are released from the condenser to the turbine building below the operating floor and were assumed to be removed from the turbine building via the reactor building ventilation stack at a rate of 3 air changes/hr.

Based upon the above conditions, the fission product release rate to the environment is shown in Table 14.6.1.

14.6.2.7 Radiological Effects (See note at the beginning of Section 14.6.2.)

The radiological exposures resulting from the activity discharged to the environment have been determined for six meteorological conditions. These conditions range from very stable to unstable and consider wind speeds of 1 m/sec and 5 m/sec. Table 14.6.2 off-site dose occurs in the southwest sector at 7,300 m. At this location the 24-hr thyroid and cloud gamma exposures are 1.5×10^{-6} Rem and 2.3×10^{-5} Rem, respectively. In any case the above dose as well as those in Table 14.6.2 are well below the guidelines set forth in 10CFR100.

If consideration is given to a 30-day dose, the 24-hr cloud dose values presented in Table 14.6.2 would increase by a factor of about 1.7 and the thyroid dose would increase by a factor of about 1.4.

14.6.2.8 Elimination of Main Steamline Scram and Primary Containment Isolation High Radiation

The justification for eliminating the MSLRM high radiation trip and isolation function from initiating automatic reactor scram and automatic closure of the MSIVs is based on NEDO-31400A (Reference 26)and the applicability of this report to PBAPS, Units 2 and 3. NEDO-31400A provides a safety assessment that demonstrates the MSIV isolation and scram functions of the MSLRMs are not required to ensure compliance with the accident dose guidelines.

The changes defeat portions of MSLRM high radiation trip function logic circuitry in the RPS and PCIS. Although the PCIS system will no longer close the MSIVs; MSIV drain valves; or RHR, MSL, and reactor recirculation sample drain valves when an MSL high radiation signal is received, all other PCIS trip logic will remain unaffected by this change and will function as designed to perform their intended safety functions.

The MSLRM system high radiation Main Control Room (MCR) alarms and trip function for isolating the MVP will be retained. This will ensure that any radioactive material released from a fuel failure will be contained in the main condenser and processed through the offgas system, which continuously removes non-condensable gases from the main condenser by the SJAEs during plant operation. The offgas system reduces offgas radioactivity levels to permissible levels for release under all site atmospheric conditions. The system uses catalytic recombination for volume reduction and control of hydrogen concentration and activated charcoal filters to adsorb fission product and activation gases prior to release to the environment. Instrumentation permits system operation and monitoring from the MCR. All of the postulated radioactive material is assumed to be released to the condenser and turbine before the isolation occurs.

The automatic closure of the MSL drain valves was eliminated because the flow ultimately discharges into the main condenser, just as the flow from the MSIVs. Therefore, any radioactive material passing through the MSL drain valves to the main condenser and through the offgas treatment system is treated identically to any radioactive material that would pass through the MSIVs. The analysis in NED0-31400A evaluated removing the MSLRM system high radiation trip function for closing the MSIVs. This same analysis applies for closure of the MSL drain valves. The elimination of the automatic closure of the MSL sample line valves, RHR system sample line valves, and reactor recirculation loop sample line valves is based on the fact that the effects are negligible, since these lines are small in comparison to the size of the lines associated with the MSIVs and MSL drains. The sample lines are routed to a sample sink where inlet valves on the sample lines are normally closed. These inlet valves must be opened periodically to allow for chemistry sampling. The sample sinks are located in the reactor building and are under equipment ventilation hoods. The equipment ventilation hood exhaust is filtered prior to release to the environment. In the unlikely event that chemistry sampling is in progress and the valve is open, a minimal amount of radioactive material would be released to the environment from the sample flow path. Appropriate actions will be implemented at PBAPS to ensure that significant increases in MSL radiation levels are adequately controlled to

limit occupational exposures and environmental releases. In the event of a MSLRM system high radiation alarm, MSLRM and Offgas System radiation level trending data from radiation monitor recorders will be reviewed, and if necessary, reactor coolant samples will be obtained and analyzed. If high radiation levels are confirmed, as measured by the Offgas system radiation monitors, reactor power will be reduced to maintain offgas release rates within TS requirements. If these release rates cannot be maintained within required TS limits, an orderly plant shutdown will be initiated. Plant procedures will be in place to implement the appropriate mitigative measures in response to a MSLRM high radiation alarm signal.

Operators must isolate the reactor water (RW) sample lines (RHR and reactor loop sample lines) within 40 minutes of a CRDA, concurrent with the RW sample lines open. The Operators must close the Primary Containment Isolation Valves (PCIVs) from the room because the local controls are assumed to be unavailable due to radiation levels exceeding allowable levels. Local alarms and personal dosimetry will alert Operators of the need to evacuate the area around RW sample lines.

Since the automatic MSIV trip will no longer be available, operators will need to isolate the MSIVs from the control room within 24 hours of a CRDA. Plant procedures will direct Operators to initiate a manual shutdown in the case that adverse conditions continue to persist.

14.6.3 Loss of Coolant Accident

Note: The material presented in 14.6.3 is historical and describes the analysis for the original plant design. The current licensing basis radiological consequences using the AST analysis are provided in Section 14.9.

Accidents that could result in release of radioactive material directly into the primary containment are the result of postulated nuclear system pipe breaks inside the drywell. All possibilities for pipe break sizes and locations have been investigated including the severance of small lines, the main steam lines upstream and downstream of the flow restrictors, and the recirculation loop lines. The most severe nuclear system effects and the greatest release of radioactive material to the primary containment results from a complete circumferential break of one of the recirculation loop lines. This accident is established as the design basis LOCA.

14.6.3.1 Initial Conditions and Assumptions

The analysis of this accident is performed using the following assumptions:

1. At the time of the accident the reactor is operating at the condition which maximizes the parameter of interest: primary containment response, fission product release, or CSCS requirements.

2. A complete loss of normal AC power occurs simultaneously with the pipe break. This additional condition results in the longest delay time for the CSCS's become operational.

3. The recirculation loop line is considered to be instantly severed. This results in the most rapid coolant loss and depressurization, with coolant discharged from both ends of the break.

14.6.3.2 Nuclear System Depressurization and Core Heatup

In Section 6.0, "Core Standby Cooling Systems," the initial phases of the LOCA are described and evaluated. Included in that description are the rapid depressurization of the nuclear system, the operating sequences of the CSCS's, the heatup of the fuel, and the perforation of fuel rods.

14.6.3.3 Primary Containment Response

Note: The information presented in 14.6.3.3 is historical and does not represent the methodology used to analyze the containment response following a LOCA for current plant conditions. The methods and results for current plant conditions and operation along the Maximum Extended Load Line, and the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating domain are described in Section 14.10 and in Reference 25. Figures 14.6.10, 14.6.11, and 14.6.12 do not reflect the current containment analysis, but reasonably represent the general characteristics and relative differences between the three cases presented. The valid results for current plant conditions are presented in Section 14.10.

14.6.3.3.1 Initial Conditions and Assumptions

The following assumptions and initial conditions were used to calculate the effects of a LOCA described in paragraph 14.6.3.1:

a. The reactor is assumed to be shut down at the time of accident initiation due to void formation in the core

region. Actually, scram will occur in less than 1 sec from receipt of the high drywell pressure signal, but the difference in shutdown time between zero and 1 sec is negligible.

b. The sensible heat released in cooling the fuel to 547° (the normal primary system operating temperature) and the core decay heat were included in the reactor vessel depressurization calculation. The rate of energy release was calculated using a conservatively high heat transfer coefficient throughout the depressurization. Because of this assumed high energy release rate, the vessel is maintained at near rated pressure for almost 12 sec. The high vessel pressure increases the calculated depressurization flow rates; this is conservative for purposes of containment analysis. With the vessel fluid temperature remaining near 547°F, however, the release of sensible energy stored below 547°F is negligible during the first 6 sec. The later release of this sensible energy does not affect the peak drywell pressure. The small effect of this energy on the end-of-transient suppression pool temperature is included in the calculations.

c. The MSIV's were assumed to start closing at 0.5 sec after the accident, and the valves were assumed to be fully closed in the shortest possible time of 3 sec following closure initiation. Actually, the closures of the MSIV's are expected to be the result of low water level, so these valves may not receive a signal to close for more than 4 sec, and the closing time could be as high as 10 sec. By assuming rapid closure of these valves, the reactor vessel is maintained at a high pressure, which maximizes the discharge of high energy steam and water into the primary containment.

d. The feedwater flow was assumed to stop instantaneously at time zero. This is justified because the relatively cold feedwater flow, if considered to continue, tends to depressurize the reactor vessel, thereby reducing the discharge of steam and water into the primary containment.

e. The vessel depressurization flow rates were calculated using Moody's critical flow model[9], assuming "liquid only" outflow because this maximizes the energy release to the containment. "Liquid only" outflow means that all vapor formed in the vessel due to bulk flashing rises to the surface rather than being entrained in the

exiting flow. Some entrainment of the vapor would occur and would significantly reduce the reactor vessel discharge flow rates. Moody's critical flow model, which assumes annular, isentropic flow, thermodynamic phase equilibrium and maximized slip ratio, accurately predicts vessel outflows through small diameter orifices. However, actual flow rates through larger flow areas are less than the model indicated due to the effects of a near homogeneous, two-phase flow pattern and phase nonequilibrium. These effects are in addition to the reduction due to vapor entrainment discussed above.

f. The pressure response calculated for the containment is based on these assumptions:

1) Thermodynamic equilibrium in the drywell and suppression chamber. This assumption is justified because a gas mixing analysis has shown that the calculated containment pressure is decreased by 2 psi if complete separation of the air and steam is assumed. Because complete mixing is nearly achieved, the error introduced by assuming complete mixing is negligible and in the conservative direction.

2) The constituents of the fluid flowing in the drywell to suppression chamber vents are based on a homogeneous mixture of the fluid in the drywell. The consequences of this assumption result in complete liquid carryover into the drywell vents. Actually, some liquid will remain behind in a pool on the drywell floor so the calculated drywell pressure is conservatively high.

3) The flow in the drywell suppression pool vents is compressible, except for the liquid phase. In the development of the drywell flow model, it was noted that the mass fraction of liquid in the drywell was on the order of 0.60, while the volumetric fraction was only about 0.005. This fact resulted in the following interpretation of the flow pattern. The liquid is in the form of a fine mist that is carried along by the air flow (predominantly steam) and does not affect the flow except to add inertia to the flowing fluid. Except for corrections to account for the liquid inertia, flow was treated as compressible ideal gas in a duct with friction.

The accuracy of this interpretation of the effects of liquid carryover is supported primarily by a series of tests performed as part of the Humboldt Bay series of pressure suppression tests[10]. In this series of tests, changes in the drywell geometry resulted in variation in the amount of liquid carryover achieved. The liquid remaining in the drywell at the end of the test was measured and recorded. These tests were performed with a relatively small diameter orifice in the reactor vessel so that the reactor vessel outflow could be calculated accurately using Moody's critical flow model[16]. Figure 14.6.6 shows the calculated and measured pressure responses for these tests. Note that with 100 percent carryover the agreement is excellent. In this test, the drywell was preheated to 184°F before the steam-water mixture was introduced to the drywell, the preheating preventing any condensation on the drywell walls. A calculated response, assuming condensation but no carryover, is also shown in Figure 14.6.6, and agreement with the measured response with no carryover is excellent.

The model is compared with the Bodega Bay test data, for two of the smaller orifices tested, in Figures 14.6.7 and 14.6.8. The figures show that the reactor vessel depressurization model accurately predicted the results of these tests. However, the predicted drywell pressure response is slightly higher than the test results. The overprediction is believed to be due to a combination of the following:

a. No condensation assumed in calculated response.

b. Slight overprediction of reactor vessel discharge flow rates.

c. Incomplete liquid carryover into the drywell vents.

As the chosen size of the vessel orifice increases, the predicted vessel depressurization rate exceeds norm and the overprediction of drywell pressure increases. This trend is illustrated in Figure 14.6.6, which compares calculated and measured drywell peak pressures. In no case did the model under predict the test data.

4) No heat loss from the gases inside the primary containment is assumed. This adds extra conservatism to the analysis, i.e., the analysis will tend to predict higher containment pressures than would actually result.

14.6.3.3.2 Containment Response (See note at the beginning of Section 14.6.3.3.)

The calculated pressure and temperature responses of the containment are shown in Figures 14.6.10 and 14.6.11. Figure 14.6.10 shows that the calculated drywell peak pressure is 44.1 psig, which is well below the design pressure of 56 psig. After the discharge of the primary coolant from the reactor vessel into the drywell, the temperature of the suppression chamber water approaches 130°F, and the primary containment pressure stabilizes at about 27 psig, as shown on Figure 14.6.10. Most of the noncondensible gases are forced into the suppression chamber during the vessel depressurization phase. However, the noncondensables soon redistribute between the drywell and the suppression chamber via the vacuum-breaker system as the drywell pressure decreases due to steam condensation.

The core spray system removes decay heat and stored heat from the core, thereby controlling core heatup and limiting metal-water reaction to 0.1 percent or less. The core spray water transports the core heat out of the reactor vessel through the broken recirculation line in the form of hot water. This hot water flows into the suppression chamber via the drywell-to-suppression chamber vent pipes. Steam flow is negligible. The energy transported to the suppression chamber water is then removed from the primary containment system by the RHRS heat exchangers.

Prior to activation of the RHRS containment cooling mode (arbitrarily assumed to be 600 sec after the accident), the RHRS pumps (LPCI mode) have been adding liquid to the reactor vessel. After the reactor vessel is flooded to the height of the jet pump nozzles, the excess flow discharges through the recirculation line break into the drywell. This flow, in addition to heat losses to the drywell walls, offers considerable cooling to the drywell and causes a depressurization of the containment as the steam in the drywell is condensed. At 600 sec, the RHRS pumps are assumed to be switched from the LPCI mode to the containment cooling mode. The containment spray need never be activated and the changeover to the containment cooling mode need not be made for several hours.

To assess the primary containment long-term response after the accident, the effects of various containment spray and containment cooling combinations were analyzed. For all cases, at least one

of the core spray loops is assumed to be in operation. The long-term pressure and temperature response of the primary containment was analyzed for the following RHRS containment cooling mode conditions:

a. Case A Operation of both RHRS cooling loops with four RHRS pumps, four high-pressure service water pumps, and four RHRS heat exchangers with containment spray in the containment cooling mode.

b. Case B Operation of one RHRS cooling loop with one RHRS pump, one high-pressure service water pump, and one RHRS heat exchanger, with containment spray.

c. Case C Operation of one RHRS cooling loop with one RHRS pump, one high-pressure service water pump, and one RHRS heat exchanger, no containment spray.

The initial pressure response of the containment (the first 30 sec after break) is the same for each of the above conditions. During the long-term containment response (after depressurization of the reactor vessel is complete), the suppression pool is assumed to be the only heat sink in the containment system. The effects of decay energy, stored energy, and energy from the metal-water reaction on the suppression pool temperature are considered.

Case A

This case assumes that both RHRS loops are operating in the containment cooling mode. This includes four RHRS heat exchangers, four RHRS pumps, and four high-pressure service water pumps. The RHRS pumps draw suction from the suppression pool and pump water through the RHRS heat exchangers and back to the pool. This forms a closed cooling loop with the suppression pool, with half the flow being sprayed into the drywell. This suppression pool cooling condition is arbitrarily assumed to start at 600 sec after the accident. Prior to this time, the RHRS pumps are used to flood the core (LPCI mode). In addition, both core spray loops are assumed to be operating.

The containment pressure response to this set of conditions is shown as curve (a) in Figure 14.6.10. The corresponding drywell and suppression pool temperature responses are shown as curve (a) in Figures 14.6.11 and 14.6.12. After the initial rapid depressurization by the HPCIS and pipe break, and subsequent further depressurization due to core spray and LPCI core flooding, energy addition due to core decay heat results in a gradual pressure and temperature rise in the containment. When the energy removal rate of the RHRS exceeds the energy addition rate from the decay heat, the containment pressure and temperature decrease to

their pre-accident values. Table 14.6.3 summarizes the cooling equipment operation, the peak containment pressure following the initial blowdown peak, and the peak suppression pool temperature.

Case B

This case assumes that no off-site power is available following the accident, and during the first 600 sec following the accident, only one core spray loop and two RHRS (LPCI) pumps are used to cool the core. During this time an interlock prohibits the activation of containment spray.

After 600 sec, one RHRS heat exchanger is activated to remove energy from the containment. During this mode of operation, one of the two RHRS (LPCI) pumps is shut down and the high-pressure service water pump to the RHRS heat exchanger is activated. The flow is now cooled before being discharged into the containment as spray. The core spray continues to provide water to flood the core.

The containment pressure response to this set of conditions is shown as curve (b) in Figure 14.6.10. The corresponding drywell and suppression pool temperature responses are shown as curve (b) in Figures 14.6.11 and 14.6.12. Table 14.6.3 summarizes this case.

The conditions analyzed for this case are conservative because the loss of a single diesel results in the loss of only one RHRS pump.

Case C

This case is exactly the same as the one preceding except that the containment spray is not operating. During this mode of operation, the RHRS pump draws suction from the suppression pool and discharges flow through the RHRS heat exchanger, where it is cooled and then injected into the reactor vessel as LPCI flow. The flow removes energy from the fuel, spills into the drywell, and flows by gravity through the vents into the suppression pool.

The containment pressure response to this set of conditions is shown as curve (c) in Figure 14.6.10. The corresponding drywell and suppression pool temperature responses are shown as curve (c) in Figures 14.6.11 and 14.6.12. Table 14.6.3 summarizes this case.

When the "spray" case is compared with the "no spray" case, it can be concluded that the suppression pool temperature response is the same. This is because the same amount of energy is removed from the pool whether the exit flow from the RHRS heat exchanger is returned to the pool or injected into the drywell as spray. However, the peak containment pressure is higher for the "no

spray" case. This is of no consequence because the pressure is still much less than the containment design pressure. Figure 14.6.13 illustrates the containment leakage rate, which is calculated on the "no spray" basis since it results in the maximum containment pressure.

Case C' (Unit 3 only)

This case is the same as C above performed for rerated conditions, including operation on the Maximum Extended Load Line. The peak Drywell pressure is calculated to be 45.4 psig, which is well below the design pressure of 56 psig. The peak suppression chamber pressure is analyzed to be 31.1 psig. Assuming the same equipment in service as in case C above, the peak suppression pool temperature is 206°F.

Additional analyses have been performed to evaluate the Primary Containment response as a result of an initial drywell pressure of 2.5 psig. These analyses predict a peak drywell temperature of 294°F and a peak drywell pressure of 47.2 psig. For operation with 90°F Final Feedwater Temperature Reduction (FFWTR) the peak drywell pressure is 47.8 psig, the peak suppression chamber pressure is 33.5 psig, and the peak drywell airspace temperature is 295°F. Operation with 90°F FFWTR is restricted to the 100% load line or below to ensure these values for containment pressure and temperature remain bounding (Reference 22). The peak drywell temperature exceeds the drywell design temperature only for a brief period (approximately 20 seconds) and is considered acceptable. The increase in the initial drywell pressure has minimal affect on the suppression chamber parameters and the long term containment response.

Additional analyses have been performed to evaluate the Primary Containment response as a result of a Small Steam Line Break (SSLB). These analyses predict a peak drywell temperature of 337.9°F (see Figure 14.6.11C) and a peak drywell pressure of 34.6 psig (see Figure 14.6.10C). The peak drywell temperature exceeds the drywell design temperature; however, since the drywell design limit of 281°F is due to the drywell shell temperature limit, further analysis was performed under reference 14.0.26 and the peak drywell shell temperature is 281°F under SSLB conditions.

14.6.3.3.3 Metal-Water Reaction Effects on the Primary Containment (See note at the beginning of 14.6.3.3.)

If Zircaloy in the reactor core is heated to temperatures above about 2,000°F in the presence of steam, a chemical reaction occurs in which zirconium oxide and hydrogen are formed. This is accompanied with an energy release of about 2,800 Btu per pound of

zirconium reacted. The energy produced is accommodated in the suppression chamber pool. The hydrogen formed, however, will result in an increased drywell pressure due simply to the volume of gas added to the fixed containment volume. The metal-water reaction during core heatup is calculated by the core heatup method described in paragraph 14.8.3. The extent of the metal-water reaction thus calculated is 0.1 percent or less of all the zirconium surrounding the fuel.

As an index of the containment's ability to tolerate an arbitrarily large metal-water reaction, the concept of "containment capability" is used. Containment capability is defined as the maximum percent of fuel channels and fuel cladding material which can enter into a metal-water reaction for a specified duration without exceeding the maximum allowable pressure of the containment. To evaluate the containment capability, various percentages of metal-water reaction are assumed to take place over certain time periods. This analysis presents a method of measuring system capability without requiring prediction of the detailed events in a particular accident condition.

The basic approach to evaluating the capability of a containment system with a given containment spray design is to assume that the energy and gas are liberated from the reactor vessel over some time period. The rate of energy release over the entire duration of the release is arbitrarily taken as uniform, since the capability curve serves as a capability index only and is not based on any given set of accident conditions as an accident performance evaluation might be.

The energy release rate to the containment is calculated as follows:

$$\dot{q}_{in} = \frac{Q_O + Q_{MW} + Q_S}{T_D}$$

where:

$\dot{q}_{in}$ = Arbitrary energy release rate to the containment, Btu/sec.

Q_O = Integral of decay power over selected duration of energy gas release, Btu.

Q_{MW} = Total chemical energy released exothermically from selected metal-water reaction, Btu.

Q_S = Initial internal sensible energy of core fuel and cladding, Btu.

T_D = Selected duration of energy and gas release, sec.

The total chemical energy released from the metal-water reaction is proportional to the percent metal-water reaction. The initial internal sensible energy of the core is taken as the difference between the energy in the core after the blowdown and the energy in the core at a datum temperature of 250°F.

It is conservatively assumed that the suppression pool is the only body in the system which is capable of storing energy. The considerable amount of energy storage which takes place in the various structures of the containment is neglected. Hence, as energy is released from the core region, it is absorbed by the suppression pool. Energy is removed from the pool by heat exchangers which reject heat to the high-pressure service water. Because the energy release is taken as uniform and the high-pressure service water temperature and exchanger flow rate are constant, the temperature response of the pool can be determined. It is assumed that the suppression chamber gases are at the suppression chamber water temperature.

The temperature of the drywell gas is found by considering an energy balance on the flows through the drywell, as described in paragraph 14.8.4.

Based upon the drywell gas temperature, suppression chamber gas temperature, and the total number of moles in the system, the containment pressure is determined. The containment capability curves in Figure 14.6.14 present the results of the parametric investigation.

It should be noted that the curves are actually derived from separate calculations of two conditions, the "steaming" and the "non-steaming" situations. The minimum amount of metal-water reaction which the containment can tolerate for a given duration is given by the condition where all of the non-condensible gases are stored in the suppression chamber. This condition assumes that "steaming" from the drywell to the suppression chamber results in washing all the non-condensible gases into the suppression chamber. This is shown as the initial flat portion of the containment capability characteristic curve. Activation of containment sprays and/or CSCS flow condenses the drywell steam so that no steaming occurs, thus allowing storage of non-condensibles also in the drywell. This is denoted by the rising curve. The intersection between the two curves represents the duration and metal-water reaction energy release which just raises all the

incoming water to the saturation temperature at the maximum allowable containment pressure. All points below the curves represent a given metal-water reaction and a given duration, resulting in a containment peak pressure which is below the maximum allowable pressure. The calculations are made at the end of the energy release duration because the number of moles of gases in the system is then at a maximum, and the suppression pool temperature is higher at this time than at any other time during the energy release.

14.6.3.4 Fission Products Released to Primary Containment

Note: The radiological methodology and results described in Sections 14.6.3.4 through 14.6.3.7 are historical. See Section 14.9 for the methods and results for current plant conditions.

The following assumptions and initial conditions were used in calculating the amounts of fission products released from the nuclear system to the drywell:

1. The reactor has been operating at design power for 1,000 days prior to the recirculation line break. This assumption results in equilibrium concentrations of fission products in the fuel. Longer operating histories would not increase the concentrations of the longer lived fission products of significance.

2. Twenty-five percent of the fuel rods in the core are conservatively assumed to be perforated. This is about a factor of 3 above the predicted extent of fuel damage.

3. An average of 1.8 percent of the noble gas activity and 0.32 percent of the halogen activity contained in a fuel rod is in the plenum and available for release if the cladding is perforated. These assumptions are consistent with measurements made on defective fuel experiments[8].

4. Due to negligible particulate activity available for release from non-molten fuel, none of the radioactive solids is released from the reactor vessel during the accident.

5. All of the noble gases and halogens released from the perforated fuel rods are assumed to be transported to the drywell.

6. The equilibrium fission product activity is:

Noble gases	8.4×10^8 Ci
Iodine	9.5×10^8 Ci

The above fission product inventory reflects an assumed 1,000 days at design power followed by a decay period of 1 min. The 1-min assumption results in the decay of the very short lived fission products which contribute significantly to the fission products in the fuel but are insignificant as far as plenum activity is concerned.

7. The only mechanisms which will reduce the noble gas concentration will be radioactive decay and leakage to the secondary containment.

8. The halogen activity released to the primary containment will experience such removal effects as washout, fallout, plateout, and removal by the pressure suppression pool. The effects of washout and plateout have been shown by various investigators to vary between 10 and 1,000[8]. A value of 2 has been chosen as representing these removal effects for this plant.

9. An iodine partition factor has been conservatively assumed to be 10^2. Numerous experiments have also been conducted to investigate the iodine partition factor between water and air. The results of these experiments show that a partition factor of 10^3 to 10^8 [8] is appropriate for conditions existing in the primary containment as a result of a LOCA.

10. As a consequence of releasing elemental iodine into the primary containment, the possibility exists that some of this activity may be converted to an organic form which is generally removed to a lesser extent by the various removal mechanisms previously discussed than is elemental iodine. Various experiments[8] have been conducted which show that for the conditions representative of a LOCA the conversion ratio of elemental iodine to methyl iodide varies between 0.001 percent and 1.0 percent. The conversion ratio chosen for this analysis is 1.0 percent. In addition to this conversion ratio, it is also conservatively assumed that the only removal mechanism affecting the organic iodines is radioactive decay.

Using the previous assumptions, the calculations result in the following amounts of fission products released to the drywell through the pipe break:

Noble gases	1.9×10^6 Ci
Iodine 131	3.0×10^5 Ci
Iodine 132	4.6×10^4 Ci
Iodine 133	1.7×10^5 Ci
Iodine 134	4.3×10^4 Ci
Iodine 135	9.3×10^4 Ci
Other Halogens	1.3×10^5 Ci

The noble gas activity and the sum of the elemental and organic iodines airborne in the primary containment are shown in Table 14.6.4.

14.6.3.5 Fission Product Release to Secondary Containment (See note at the beginning of Section 14.6.3.4)

The fission product activity in the secondary containment is a function of the leakage rate from the primary containment and the volumetric discharge rate from the secondary containment. The primary containment leak rate is assumed constant at 0.5 percent per day of the contained free volume while the standby gas treatment system provides 1 air change/day for the secondary containment at its design leak rate. Any fission product removal effects in the secondary containment such as plateout, fallout, etc, are neglected. However, the effects of decay are considered. Based upon these values, the reactor building fission product inventory is calculated to reach maximum concentrations as noted in Table 14.6.5.

14.6.3.6 Fission Product Release to Environs (See note at the beginning of Section 14.6.3.4.)

The fission product activity released to the environs is dependent upon the fission product inventory airborne in the secondary containment, the volumetric flow from the secondary containment, and the efficiency of the various components of the standby gas treatment system.

The following assumptions and initial conditions were used in calculating the fission products released to the environs.

1. The activity airborne in the secondary containment is as presented in Table 14.6.5.

2. The standby gas treatment system, in addition to iodine and particulate filters, also contains a demister for the removal of entrained water droplets and electric heaters for heating the incoming air, upstream of the particulate and iodine filters, to a @T of 10°F above the temperature of the mixture entering the standby gas

treatment system. A @T of 10°F will reduce the relative humidity of the incoming mixture to approximately 70 percent.

3. A filter efficiency of 99 percent is used for elemental and organic iodine and 0 percent for the noble gases. Tests[8] on iodine filter systems have shown removal efficiencies of greater than 99.9 percent even in environments approaching 100 percent relative humidity. The initial filter acceptance test is based on filter efficiencies of 99.97 percent for the particulate filter and 99.9 percent for the iodine removal filter. Annual tests will demonstrate that the filter systems will achieve filter efficiencies of at least 99.9 percent and 99.0 percent for the particulates and iodines respectively.

4. There is 1 air change per day through the standby gas treatment system.

Based upon these conditions, the fission product activity being released to the environs is shown in Table 14.6.6.

14.6.3.7 <u>Radiological Effects</u> (See note at the beginning of Section 14.6.3.4.)

The radiological exposures resulting from the activity released to the environment as a consequence of the LOCA have been determined for 6 meteorological conditions. These conditions represent the type of meteorology which may occur at the site during a LOCA. While more favorable conditions could occur, only those conditions which would tend to maximize the off-site doses have been investigated.

As noted in Table 14.6.7, the maximum 24-hr exposure occurs at the site boundary (0.5 mi from the release point) and under unstable meteorological conditions. The cloud gamma dose and thyroid inhalation dose at the site boundary are 1.9×10^{-3} Rem and 1.8×10^{-4} Rem, respectively. These values are well below the guideline doses set forth in 10CFR100.

The values presented in Table 14.6.7 neglect the effects of terrain features and are therefore applicable to those locations which are at the same elevation as the base of the stack. If consideration is given to the effects of terrain, the maximum 24 hr off-site exposure occurs at 7,300 m in the southwest sector, for which the respective 24-hr thyroid inhalation and cloud gamma doses are 6.6×10^{-4} Rem and 1.7×10^{-3} Rem.

If consideration is given to a 30-day dose, the 24-hr cloud dose values presented in Table 14.6.7 would increase by a factor of 2.2 and the thyroid dose would increase by a factor of about 1.5.

14.6.4 Refueling Accident

Note: The material presented in Section 14.6.4 is historical. Current event description and analysis bases for the Fuel Handling Accident are provided in Reference 14.0.6. The current licensing basis radiological consequences using the AST analysis are provided in Section 14.9.

Accidents that result in the release of radioactive materials directly to the secondary containment are events that can occur when the primary containment is open. A survey of the various plant conditions that could exist when the primary containment is open reveals that the greatest potential for the release of radioactive material exists when the primary containment head and reactor vessel head have been removed. With the primary containment open and the reactor vessel head off, radioactive material released as a result of fuel failure is available for transport directly to the reactor building.

Various mechanisms for fuel failure under this condition have been investigated. The refueling interlocks, which impose restrictions on the movements of refueling equipment and control rods, prevent an inadvertent criticality during refueling operations. The RPS is capable of initiating a reactor scram in time to prevent fuel damage for errors or malfunctions occurring during deliberate criticality tests with the reactor vessel head off. The possibility of mechanically damaging the fuel has been investigated.

The design basis for this case is the drop of a spent fuel assembly and fuel grapple mast assembly onto the reactor core. The fuel grapple mast assembly consists of the telescoping mast (4 sections) and the grapple head. Analyses were performed to evaluate the consequences of a variety of drop scenarios. Analytical assumptions were selected to provide a viable worst-case scenario that would produce the greatest number of failed fuel rods.

The postulated accident scenario begins with a spent fuel assembly raised to the highest position (overhoist position) with the grapple mast by the refueling platform (bottom of the fuel bundle is 32.9 feet above the top of the core). At this time, the main hoist cables and the mast mounts fail, allowing the fuel assembly and grapple mast to fall onto the top of the core impacting a group of fuel assemblies. The grapple head and mast are

conservatively assumed to have remained fixed (vertically) to the dropped fuel assembly such that all kinetic energy is transferred through the dropped assembly to the impacted assemblies. The dropped fuel assembly impacts the core at a slight angle, and the fuel rods in this assembly are subjected to bending failure. After impact with the core, the fuel assembly falls onto the core horizontally without contacting the side of the pressure vessel. During the tipping of the fuel assembly, the grapple mast remains attached to the fuel bundle bail handle and remains in a vertical position. The mast vertically impacts core fuel, as the dropped fuel assembly comes to rest. With the fuel assembly laying on its side, the grapple mast then falls onto the core horizontally without contacting the side of the pressure vessel.

The selected scenario addresses three impacts. The first impact occurs when the assembly (fuel bundle and attached grapple mast) is dropped above the core. The second impact occurs when the fuel bundle (with attached grapple mast modeled as a point load on the bail handle) falls from a vertical to horizontal position while in contact with the core. The third impact occurs when the grapple mast falls from the vertical to the horizontal position while the grapple head remains in contact with the core.

14.6.4.1 <u>Accident Scenario Assumptions</u> (See note at beginning of Section 14.6.4)

1. The fuel assembly and grapple mast assembly are dropped from the maximum height allowed by the fuel handling equipment. This will require the complete detachment of both assemblies from the platform which will involve multiple failures. Both cables of the dual hoist cable system and the mast-to-platform securing hardware are postulated to fail.

2. The entire amount of potential energy, referenced to the top of the reactor core, is available for application to the fuel assemblies involved in the accident. This assumption conservatively neglects the dissipation of any mechanical energy (drag effects) from the fuel assembly falling in water above the core.

3. The energy of the dropped assembly will be absorbed by the fuel, cladding, and other core structures. However, in this analysis no energy was considered to be absorbed by the fuel pellets, i.e., the energy was absorbed entirely by nonfuel components of the assemblies. The energy available for cladding deformation was considered to be proportional to the mass ratio:

$$\frac{\text{mass of cladding}}{\text{mass of assembly - mass of fuel pellets}}$$

and is equal to a maximum of .519 for the fuel designs presently in use.

4. The energy is considered to be absorbed equally by the falling assembly and the impacted fuel bundles.

5. To estimate the expected number of failed fuel rods, an energy approach has been used. Since the total energy available to cause damage to the cladding is independent of the number of impacts, multiple impacts need not be considered.

6. For the fuel designs currently in use, there is no propensity for preferential tie rod failure due to bending as a result of impact while in the core. Therefore, impacted core bundle fuel rods that fail are expected to do so by gross compression distortion (based on 1 percent uniform plastic deformation of the rod) after absorption of approximately 250 ft-lb of energy.

7. The fuel grapple mast assembly and the fuel assembly fall together. Detachment of the fuel assembly from the grapple mast may occur after the bundle has reached a horizontal position by either bail handle failure or bail release from grapple hooks opening.

8. The dropped fuel assembly impacts the core slightly off the vertical.

9. The grapple mast assembly impacts the core in a fully retracted condition.

10. Fuel assembly associated values used in the analysis are:

length	14.68 ft
weight	700 lbs
drop distance	32.9 ft

11. Grapple mast assembly associated values used in the analysis are:

length	21.87 ft
weight	600 lbs

14.6.4.2 <u>Fuel Damage</u> (See note at beginning of Section 14.6.4)

Dropping a fuel assembly and grapple mast assembly onto the reactor core from the maximum height allowed by the refueling equipment results in an (air equivalent) impact velocity of 46.03 ft/sec.

The fuel assembly is expected to impact on the reactor core at a small angle from the vertical, inducing a bending mode of failure on the fuel rods of the dropped assembly. Fuel rods are expected to absorb little energy prior to failure due to bending if it is assumed that each fuel rod resists the imposed bending load by a couple consisting of two equal, opposite concentrated forces. Actual bending tests with concentrated point loads show that each fuel rod absorbs about 1 ft-lb prior to cladding failure. For rods which fail due to gross compression distortion, each rod is expected to absorb about 250 ft-lb before cladding failure (this is based on 1 percent uniform plastic deformation of the rods).

The initial impact dissipates 32.9 ft x (700 lbs + 600 lbs) or 42770 ft-lb of energy. It is assumed that 50 percent of this energy is absorbed by the dropped fuel assembly, and that the remaining 50 percent is absorbed by the struck fuel assemblies in the core. Because the fuel rods of the dropped fuel assembly are susceptible to the bending mode of failure, and because 1 ft-lb of energy is sufficient to cause cladding failure due to bending, all 62 fuel rods of the dropped fuel assembly are assumed to fail.

If the dropped fuel assembly strikes only one or two fuel assemblies, the energy absorption by the core support structure results in about the same energy dissipation as in the case where four fuel assemblies are struck.

The fuel rods of the struck assemblies are held rigidly in place in the core, so they are susceptible only to the compression mode of failure. To cause cladding failure of one fuel rod due to compression, 250 ft-lb of energy is required.

The number of fuel rod failures due to compression occurring during the first impact is computed as follows:

$$\frac{.5 \text{ x } 42770 \text{ ft-lb x } .519}{250 \text{ ft-lb/fuel rod}} = 45 \text{ failed fuel rods}$$

Thus, during the initial impact, the fuel rod failures are as follows:

Dropped assembly	62 fuel rods	(bending)
Struck assembly	45 fuel rods	(compression)
	107 failed fuel rods	

For the second impact, the broad side of the dropped assembly was assumed to tip over and impact horizontally on the top of the core. The available energy was calculated by assuming a linear weight distribution in the assembly with a point load at the top of the assembly to represent the fuel grapple weight. This produces conservative energy production and damage estimates since again the drag effects of a water environment are not considered, and the fuel bundle weight is assumed to be concentrated at a point rather than distributed over its entire length. This impact dissipates:

14.68 ft x 600 lbs + 1/2 x 14.68 ft x 700 lbs = 13946 ft-lb

of energy. As before, the energy was considered to be absorbed equally by the falling assembly and the impacted assemblies, and the energy fraction available for cladding deformation was .519. The number of fuel rod failures due to compression occurring during the second impact is computed as follows:

$$\frac{.5 \times 13946 \text{ ft-lb} \times .519}{250 \text{ ft-lb/fuel rod}} = 15 \text{ failed fuel rods}$$

During the second impact, the grapple mast assembly was assumed to have remained in the vertical position as the fuel bundle tipped onto its side. So, for the third and final impact, the grapple mast assembly tipped over and impacted horizontally on the top of the core. Again the available energy was calculated by assuming a linear weight distribution in the mast assembly. Conservative estimates for energy production and damage are obtained since the drag effects of a water environment are not considered, the mast is assumed to be vertical and the mast weight is assumed to be concentrated at a single point rather than distributed over its entire length. This impact dissipates:

1/2 x 21.87 ft x 600 lbs = 6561 ft-lb

of energy. Again, the energy was considered to be absorbed equally by the falling and impacted assemblies, and the energy fraction available for cladding deformation was .519. The number of fuel rod failures due to compression occurring during the third impact is computed as follows:

$$\frac{.5 \times 6561 \text{ ft-lb} \times .519}{250 \text{ ft-lb/fuel rod}} = 7 \text{ failed fuel rods}$$

The total number of failed fuel rods resulting from the accident is as follows:

First impact	107 fuel rods
Second impact	15 fuel rods

Third impact	7 fuel rods
	129 failed fuel rods

14.6.4.3 Fission Product Release From Fuel (See note at beginning of Section 14.6.4)

Fission product release estimates for the accident are based on the following assumptions:

1. The reactor fuel has an irradiation time of 1,000 days at design power up to 24 hr prior to the accident. This assumption results in an equilibrium fission product concentration at the time the reactor is shut down. Longer operating histories do not significantly increase the concentration of the fission products of concern. The 24-hr decay time allows time for the reactor to be shut down, the nuclear system depressurized, the reactor vessel head removed, and the reactor vessel upper internals removed. It is not expected that these evolutions could be accomplished in less than 24 hr.

2. An average of 1.8 percent of the noble gas activity and 0.32 percent of the halogen activity is in the fuel rod plenums and available for release. This assumption is based on fission product release data from defective fuel experiments[8].

3. Due to the negligible particulate activity available for release in the fuel plenums or from the unmelted fuel, none of the solid fission products is assumed to be released from the fuel.

4. 140 GE 8x8 fuel rods (111 GE 7x7 equivalent) are assumed to fail for the fission product release analysis. The conclusion from the analysis of fuel bundle mechanical damage during the postulated fuel handling accident indicates that 129 GE 8x8 fuel rods could fail, therefore, the radiological release analysis that follows is conservative. GE14 (10x10) fuel was evaluated and determined to be bound by the 7x7 bundle (Reference 23).

5. The fission product activity contents of the core at the time of the accident are as follows:

Noble gases	2.5×10^{8} Ci
Halogens	3.1×10^{8} Ci

These activity contents are the result of an analysis of the fission product inventories in the core assuming equilibrium conditions at design power followed by a 24-hr decay period.

Using these assumptions, the following amounts of fission product activity are released from the fuel to the water in the reactor vessel as a result of the dropped fuel assembly:

Noble gases	1.9×10^{4} Ci
Iodine 131	3.3×10^{3} Ci
Iodine 132	4.6×10^{2} Ci
Iodine 133	8.9×10^{2} Ci
Iodine 134	1.6×10^{-5} Ci
Iodine 135	1.1×10^{2} Ci

14.6.4.4 <u>Fission Product Release To Secondary Containment</u> (See note at the beginning of Section 14.6.4.)

The following assumptions and initial conditions are used to calculate the fission product release to the secondary containment:

1. The fission product activity released to the secondary containment will be in proportion to the removal efficiency of the water in the refueling pool. Since water has a poor affinity for the noble gases they are assumed to be instantaneously released from the pool to the secondary containment.

2. As noted in paragraph 14.6.3.4, the removal efficiency of the water for halogens can be defined in terms of the partition factor, for which values between 10^{3} and 10^{5} have been experimentally determined to be applicable for the conditions under investigation. A partition factor of 10^{2} for the halogens has been conservatively assumed for this accident. Thus the computed inhalation exposures will be overestimated by a factor of 10 to 10^{3}.

3. The conservative assumption is also made that instantaneous equilibrium is attained between the refueling pool and secondary containment. In reality, if a true equilibrium is maintained, the effects of plateout or fallout would be compensated for by the evolutions of activity from the refueling pool.

4. The effects of plateout and fallout are neglected. Fission product plateout and/or fallout will occur in the secondary containment; however, for the assumption

that a true equilibrium is maintained, the effects of plateout or fallout would be compensated for by the evolutions of activity from the refueling pool.

5. The refueling cavity liquid volume is 3.1×10^4 cu ft and the effective air volume in the secondary containment is 1.1×10^6 cu ft.

6. The standby gas treatment systems removes 1 secondary containment air volume per day.

Based upon these assumptions, the airborne activity is as shown in Table 14.6.8.

14.6.4.5 Fission Product Release To Environs (See note at the beginning of Section 14.6.4.)

The following assumptions and initial conditions are used to calculate the fission product release to the environs:

1. High radiation levels in the reactor building exhaust plenum will isolate the normal ventilation system and actuate the standby gas treatment system.

2. The relative humidity in the secondary containment is 70 percent. Since the refueling accident does not result in the release of any liquid or vapor to the secondary containment, the normal environmental condition existing prior to the accident will also exist after the accident, except for the addition of the released fission products. The relative humidity in the secondary containment will therefore be considerably below any levels which may be detrimental to the filter media in the standby gas treatment system. However, as mentioned previously, the charcoal beds and absolute filter media, as well as the air flowing through the filter system, are heated 10°F above the mixture entering the system, reducing the relative humidity to 70 percent or less.

3. The filter efficiency is assumed to be 99 percent for iodines and 0 percent for the noble gases (see paragraph 14.6.3.6c).

4. There is 1 secondary containment air change per day through the standby gas treatment system. The activity in the secondary containment is shown on Table 14.6.8.

Based upon these conditions, the fission product activity release rate to the environs is as shown in Table 14.6.9.

14.6.4.6 Radiological Effects (See note at the beginning of Section 14.6.4.)

The radiological exposures to the general population have been evaluated for 6 meteorological conditions encompassing very stable to unstable meteorology occurring with 1-m/sec and 5-m/sec winds. These conditions represent the spectrum of conditions which could exist at the site and which would result in maximizing the off-site exposures. Two exposure periods have been evaluated, a 2-hr exposure period and a 24-hr exposure period, commonly referred to as the total dose. It should be emphasized that the radiological exposures presented in Table 14.6.10 are based upon the assumption that the stated meteorological conditions exist for the duration under consideration and that the wind blows in one direction during the entire release period.

As noted in Table 14.6.10, the maximum off-site exposure occurs at the site boundary, ~0.5 mi from the release point. The maximum 24-hr cloud gamma dose and thyroid inhalation dose at the site boundary are 1.1×10^{-2} Rem and 1.4×10^{-2} Rem, respectively. These values are well below the guidelines set forth in 10CFR100.

If consideration is given to a 30-day dose, the 24-hr cloud gamma values would be increased by a factor of 1.1 while the thyroid inhalation dose would be increased by a factor of 1.1.

The values presented in Table 14.6.10 are based on the assumption of a stack release with no terrain effects considered. If terrain effects are considered, the maximum off-site dose occurs in the southwest sector at 7,300 m. The 24-hr radiological exposures at this location are 5.0×10^{-2} Rem thyroid inhalation and 1.0×10^{-2} Rem cloud gamma.

It can therefore be concluded that the radiological exposures for the refueling accident are well below the guidelines set forth in 10CFR100.

14.6.5 Main Steam Line Break Accident

Note: The methodology and results described in Section 14.6.5 are historical. See Section 14.9 for the current methods and results for current plant conditions using the AST analysis.

Accidents that result in the release of radioactive materials outside the secondary containment are the results of postulated breaches in the nuclear system process barrier. The design basis accident is a complete severance of one main steam line outside the secondary containment. Figure 14.6.15 shows the break

location. The analysis of the accident is described in three parts as follows:

1. Nuclear System Transient Effects

   This includes analysis of the changes in nuclear system parameters pertinent to fuel performance and the determination of fuel damage.

2. Radioactive Material Release

   This includes determination of the quantity and type of radioactive material released through the pipe break and to the environs.

3. Radiological Effects

   This portion determines the dose effects of the accident to off-site persons.

14.6.5.1 Nuclear System Transient Effects (See note at the beginning of Section 14.6.5.)

14.6.5.1.1 Assumptions

The following assumptions are used to evaluate response of nuclear system parameters to the steam line break accident outside the secondary containment:

1. The reactor is operating at design power.

2. Reactor vessel water level is normal for initial power level assumed at the time the break occurs.

3. Nuclear system pressure is normal for the initial power level.

4. The steam line is assumed to be instantly severed by a circumferential break. The break is physically arranged so that the coolant discharge through the break is unobstructed. These assumptions result in the most severe depressurization rate of the nuclear system.

5. The MSIV's are assumed to be closed 10.5 sec after the break. This assumption is based on the 0.5-sec time required to develop the automatic isolation signal (high differential pressure across the main steam line flow restrictor) and the 10-sec closure time for the

valves, which is the maximum setting for valve closure time.

Faster MSIV closure could reduce the mass loss until finally some other process line break would become controlling. However, the resulting radiological dose for this break would be less than the main steam line break with a 10-sec valve closure. Thus the postulated main steam line break outside the primary containment with a 10-sec isolation valve closure results in maximum calculated radiological dose and is therefore the design basis accident.

6. The mass flow rate through the upstream side of the break is assumed not to be affected by isolation valve closure until the isolation valves are closed far enough to establish limiting critical flow at the valve location. After limiting critical flow is established at the isolation valve, the mass flow rate is assumed to decrease linearly as the valve is closed. This assumption results in an almost constant mass flow out of the break until the last 3 to 4 sec of a 10-sec valve closure.

7. The mass flow rate through the downstream side of the break is assumed not to be affected by the closure of the isolation valves in the unbroken steam lines until those valves are far enough closed to establish limiting critical flow at the valves. After limiting critical flow is established at the isolation valve positions the mass flow is assumed to decrease linearly as the valves close. This assumption results in an almost constant mass flow through the break until the last 3 to 4 sec of a 10-sec valve closure.

8. In calculating the rate of water level rise inside the vessel, it is assumed that the steam bubbles formed during depressurization rise at an average velocity of about 1 ft/sec relative to the liquid. This assumption is predicted by analysis [11,12] and confirmed experimentally[13].

9. After the level of the mixture inside the reactor vessel rises to the top of the steam dryers, the quality of the two-phase mixture discharged through the break is assumed constant at its minimum value. This assumption maximizes the total mass of coolant discharged through the break because most of the mixture flow will actually be at a higher quality.

10. Feedwater flow is assumed to decrease linearly to zero over the first 4 sec.

11. A loss of auxiliary ac power is assumed to occur simultaneous with the break. This results in the immediate loss of power to the recirculation pumps. Recirculation flow is assumed to coast down according to momentum computations for the recirculation system.

12. Recirculation system drive pump head is assumed to be zero when the coolant at the pump suction reaches 1 percent quality. This assumption accounts for the effects of cavitation on recirculation drive pump capacity as the pumps coast down.

13. Following closure of the MSIV's, no high pressure injection occurs and ADS is manually initiated after 10 minutes.

14.6.5.1.2 Sequence Events

The sequence of events following the postulated main steam line break is as follows.

The steam flow through both ends of the break increases to the value limited by critical flow considerations. The flow from the upstream side of the break is limited initially by the main steam line flow restrictor. The flow from the downstream side of the break is limited initially by the downstream break area. The decrease in steam pressure at the turbine inlet initiates closure of the MSIV's within about 200 msec after the break occurs (see subsection 7.3, "Primary Containment and Reactor Vessel Isolation Control System"). Also, MSIV closure signals are generated as the differential pressures across the main steam line flow restrictors increase above isolation set points. The instruments sensing flow restrictor differential pressures generate isolation signals within about 200 msec after the break occurs; however, 500 msec is conservatively assumed in the analysis.

A reactor scram is initiated as the MSIV's begin to close (see subsection 7.2, "Reactor Protection System"). In addition to the scram initiated from MSIV closure, voids generated in the moderator during depressurization contribute significant negative reactivity to the core even before the scram is complete. Because the main steam line flow restrictors are sized for the main steam line break accident, reactor vessel water level remains above the top of the fuel throughout the accident.

14.6.5.1.3 Coolant Loss and Reactor Vessel Water Level

The steam flow rate through the upstream side of the break increases from the initial value of 1,000 lb/sec in the line to 2,000 lb/sec (about 200 percent of rated flow for one steam line) with critical flow initially occurring at the flow restrictor. The steam flow rate was calculated using an ideal nozzle model. That the flow model predicts the behavior of the flow limiter has been substantiated by tests conducted on a scale model over a variety of pressure, temperature, and moisture conditions.

The steam flow rate through the downstream side of the break consists of equal flow components from each of the unbroken lines.

The steam flow rate in each of the unbroken lines increases from an initial value of 1,000 lb/sec to 2,000 lb/sec.

The total steam flow rate leaving the vessel is approximately 8,000 lb/sec, which is in excess of the steam generation rate of 4,000 lb/sec. The steam flow-steam generation mismatch causes an initial depressurization of the reactor vessel at a rate of 50 psi/sec. The formation of bubbles in the reactor vessel water causes a rapid rise in the water level. The analytical model used to calculate level rise predicts a rate of rise of about 6 ft/sec. Thus, the water level reaches the vessel steam nozzles at 2 to 3 sec after the break, as shown in Figure 14.6.16. From that time on, a two-phase mixture is discharged from the break. The two-phase flow rates are determined by vessel pressure and mixture enthalpy[14]. The vessel depressurization is calculated using a digital computer model in which the reactor vessel is divided into five major nodes. The model includes the flow resistance between nodes, as well as heat addition from the core.

As shown in Figure 14.6.16, two-phase flow is discharged through the break at an almost constant rate until late in the transient. This is the result of not taking credit for the effect of valve closure on flow rate until isolation valves are closed far enough to establish critical flow at the valve locations. The slight decrease in discharge flow rate is caused by depressurization inside the reactor vessel. The linear decrease in discharge flow rate at the end of the transient is the result of the assumption regarding the effect of valve closure on flow rate after critical flow is established at the valve location.

The following total masses of steam and liquid are discharged through the break prior to isolation valve closure.

Steam	25,000 lb
Liquid	160,000 lb

Analysis of fuel conditions reveals that no fuel rod perforations due to high temperature occur during the depressurization, even

with the conservative assumptions regarding the operation of the recirculation and feedwater systems. MCHFR remains above 1.0 at all times during the transient. No fuel rod failures due to mechanical loading during the depressurization occur because the differential pressures resulting from the transient do not exceed the designed mechanical strength of the core assembly.

After the MSIV's close (10.5 sec), depressurization stops and natural convection is established through the reactor core. No fuel cladding perforation occurs even if the stored thermal energy in the fuel were simply redistributed while natural convection is being established; cladding temperature would be about 1,000°F, well below the temperatures at which cladding can fail. Thus, it is concluded that even for a 10.5-sec MSIV closure, fuel rod perforations due to high temperature do not occur. For shorter valve closure times, the accident is less severe. After the MSIV's are closed, the reactor can be cooled by operation of any of the normal or standby cooling systems. Reference 14.6.18 shows that adequate core cooling can be accomplished with no high pressure injection, manual initiation of ADS 10 minutes after MSIV closure, and either LPCI or Core Spray injection. The core flow and MCHFR during the first 10.5 sec of the accident are shown in Figures 14.6.17 and 14.6.18. Since the MCHFR never drops below 1.0, the core is always cooled by very effective nucleate boiling.

14.6.5.2 Radioactive Material Release (See note at the beginning of Section 14.6.5.)

14.6.5.2.1 Assumptions

The following assumptions are used in the calculation of the quantity and types of radioactive material released from the nuclear system process barriers outside the secondary containment.

1. The amounts of steam and liquid discharged are as calculated from the analysis of the nuclear system transient.

2. The concentration of biologically significant radionuclides contained in the coolant are as follows and are based on an off-gas release rate of 0.35 Ci/sec:

Iodine 131	0.17 μCi/cc
Iodine 132	1.02 μCi/cc
Iodine 133	1.04 μCi/cc
Iodine 134	1.47 μCi/cc
Iodine 135	1.30 μCi/cc

Because the steam-to-water halogen concentration ratio is on the order of 3 x 10^{-5}[8], only the halogens carried out of the reactor vessel by the liquid phase during the discharge of the steam-water mixture are significant. Because the coolant activity contents are based on data derived from reactor operation with an unusually large number of cladding failures, and because high normal stack gas discharge rate is assumed, considerable conservatism is inserted into the analysis.

3. The noble gas discharge rate, after 30-min holdup, is assumed to be 0.35 Ci/sec. This assumption permits direct computation of the amount of noble gas activity leaving the reactor vessel at the time of the accident. The result is that 1.9 Ci of noble gas activity leaves the reactor vessel each second that the isolation valve is open.

4. It is assumed that the MSIV's are fully closed at 10.5 sec after the pipe break occurs. This allows 500 msec for the generation of the automatic isolation signal and 10 sec for the valves to close. The valves and valve control circuitry are designed to provide main steam line isolation in no more than 10.5 sec. The actual closure time setting for the isolation valves is less than 10 sec.

5. Due to the short half-life of nitrogen-16, the radiological effects from this isotope are of no major concern and are not considered in the analysis.

14.6.5.2.2 <u>Fission Product Release from Break</u>

Using the above assumptions, the following amounts of radioactive materials are released from the nuclear system process barrier:

Noble gases	2.0 x 10^1 Ci
Iodine 131	1.8 x 10^1 Ci
Iodine 132	1.0 x 10^2 Ci
Iodine 133	1.1 x 10^2 Ci
Iodine 134	1.5 x 10^2 Ci
Iodine 135	1.4 x 10^2 Ci

The above releases take into account the total amount of liquid released as well as the liquid converted to steam during the accident.

14.6.5.2.3 <u>Steam Cloud Movement</u>

The following initial conditions and assumptions are used in calculating the movement of the steam cloud:

1. Additional flashing to steam of the liquid exiting from the steam line break will occur due to its superheated condition in the atmosphere.

2. The pressure buildup inside the turbine building will result in release of the steam cloud in a matter of seconds.

3. Steam cloud rise as predicted by the following equation could vary between 100 and 600 m depending upon the assumptions made regarding wind speed.

$$h = \frac{11\ Q^{1/3}}{u}$$

where:

h = Height of cloud rise (ft)

u = Wind speed (ft/sec)

Q = Heat output of cloud (cal/sec)

While the effect of cloud rise is a physical reality, this effect has been neglected for this accident and the assumption is made that the steam cloud does not attain an elevation greater than 30 m.

14.6.5.3 Radiological Effects (See note at the beginning of Section 14.6.5.)

The following assumptions and initial conditions are used in calculating the radiological effects of the steam line break accident:

1. Height of release of 30 m.

2. All of the activity released from the reactor vessel to the turbine building is conservatively assumed to escape to the environment.

The resulting radiological exposures are shown in Table 14.6.11 with the maximum occurring at the site boundary. The cloud gamma dose and thyroid inhalation dose at the site boundary is 6.0×10^{-3} Rem and 3.1×10^{0} Rem, respectively. These values are well below the guideline doses set forth in 10CFR100.

Since all of the activity is released to the environment in the form of a puff, the doses indicated are maximum values regardless of what dose period is being evaluated.

The exposures presented in Table 14.6.11 neglect any effects due to terrain. If these effects are taken into consideration, the maximum site boundary dose occurs at 915 m in the northwest sector. The exposures at this location are 1.2×10^{-2} Rem cloud gamma and 2.7×10^{1} Rem thyroid inhalation.

14.6 ANALYSIS OF DESIGN BASIS ACCIDENTS

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14.6 ANALYSIS OF DESIGN BASIS ACCIDENTS (CONTINUED)

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TABLE 14.6.1*

CONTROL ROD DROP ACCIDENT

FISSION PRODUCT RELEASE RATE TO ENVIRONS

Time After Accident	Fission Product Activity Being Released to Environment: Noble Gases (Ci/sec)	Iodines (Ci/sec)
1 min	1.96×10^{-5}	2.65×10^{-12}
30 min	3.03×10^{-4}	4.05×10^{-11}
1 hr	3.61×10^{-4}	4.78×10^{-11}
2 hr	3.63×10^{-4}	4.74×10^{-11}
8 hr	3.13×10^{-4}	3.93×10^{-11}
12 hr	2.95×10^{-4}	3.63×10^{-11}
1 day	2.60×10^{-4}	3.05×10^{-11}
2 days	2.21×10^{-4}	2.46×10^{-11}
4 days	1.71×10^{-4}	1.88×10^{-11}
30 days	1.89×10^{-5}	1.92×10^{-12}

* Historical Information – not applicable to Current Plant Design (see Section 14.9.2)

TABLE 14.6.2**

CONTROL ROD DROP ACCIDENT

Distance (m)	2-Hr dose VS-1	2-Hr dose MS-1	2-Hr dose N-1	2-Hr dose N-5	2-Hr dose U-1	2-Hr dose U-5	24-Hr Dose VS-1	24-Hr Dose MS-1	24-Hr Dose N-1	24-Hr Dose N-5	24-Hr Dose U-1	24-Hr Dose U-5
	Radiological effects (REM)											
Passing Cloud Whole-Body Dose												
730*	$1.8x10^{-6}$	$1.8x10^{-6}$	$1.9x10^{-6}$	$2.9x10^{-7}$	$2.9x10^{-6}$	$4.1x10^{-7}$	$1.5x10^{-5}$	$1.5x10^{-5}$	$1.7x10^{-5}$	$2.5x10^{-6}$	$2.5x10^{-5}$	$3.6x10^{-6}$
805	$1.7x10^{-6}$	$1.7x10^{-6}$	$1.9x10^{-6}$	$2.8x10^{-7}$	$2.9x10^{-6}$	$4.1x10^{-7}$	$1.5x10^{-5}$	$1.5x10^{-5}$	$1.7x10^{-5}$	$2.5x10^{-6}$	$2.6x10^{-5}$	$3.6x10^{-6}$
1,609	$1.2x10^{-6}$	$1.2x10^{-6}$	$1.7x10^{-6}$	$1.7x10^{-7}$	$2.0x10^{-6}$	$4.5x10^{-7}$	$1.1x10^{-5}$	$1.1x10^{-5}$	$1.5x10^{-5}$	$1.5x10^{-6}$	$1.8x10^{-5}$	$2.2x10^{-6}$
8,045	$3.7x10^{-7}$	$4.1x10^{-7}$	$4.0x10^{-7}$	$9.8x10^{-8}$	$2.0x10^{-7}$	$4.7x10^{-8}$	$3.2x10^{-6}$	$3.6x10^{-6}$	$3.5x10^{-6}$	$8.5x10^{-7}$	$1.7x10^{-6}$	$4.1x10^{-7}$
16,090	$1.9x10^{-7}$	$2.3x10^{-7}$	$1.2x10^{-7}$	$4.2x10^{-8}$	$5.1x10^{-8}$	$1.6x10^{-8}$	$1.7x10^{-6}$	$2.0x10^{-6}$	$1.1x10^{-6}$	$3.6x10^{-7}$	$4.5x10^{-7}$	$1.4x10^{-7}$
Lifetime Thyroid Dose												
730*	0	$1.6x10^{-21}$	$2.6x10^{-11}$	$2.4x10^{-14}$	$4.1x10^{-8}$	$5.4x10^{-9}$	0	$1.4x10^{-21}$	$2.3x10^{-10}$	$2.1x10^{-13}$	$3.6x10^{-7}$	$4.7x10^{-8}$
805	0	$1.4x10^{-20}$	$1.1x10^{-10}$	$2.4x10^{-13}$	$4.6x10^{-8}$	$6.6x10^{-9}$	0	$1.8x10^{-19}$	$9.4x10^{-10}$	$2.1x10^{-12}$	$4.0x10^{-7}$	$5.8x10^{-8}$
1,609	0	$4.1x10^{-15}$	$1.2x10^{-8}$	$7.6x10^{-10}$	$3.8x10^{-8}$	$8.1x10^{-9}$	0	$3.6x10^{-14}$	$1.0x10^{-7}$	$6.6x10^{-9}$	$3.4x10^{-7}$	$7.1x10^{-8}$
8,045	$9.0x10^{-29}$	$8.6x10^{-10}$	$9.0x10^{-9}$	$2.3x10^{-9}$	$4.0x10^{-9}$	$1.0x10^{-9}$	$7.9x10^{-28}$	$7.5x10^{-9}$	$7.9x10^{-8}$	$2.0x10^{-8}$	$3.5x10^{-8}$	$9.0x10^{-9}$
16,090	$3.0x10^{-19}$	$3.2x10^{-9}$	$3.5x10^{-9}$	$9.8x10^{-10}$	$1.4x10^{-9}$	$3.7x10^{-10}$	$2.7x10^{-18}$	$2.8x10^{-8}$	$3.1x10^{-8}$	$6.6x10^{-9}$	$1.2x10^{-8}$	$3.2x10^{-9}$

*Site Boundary
**Historical information - not accurate for current plant conditions. (See Section 14.9)

KEY:	Meteorology	Wind speed (m/s)
VS-1	Very stable	1
MS-1	Moderately stable	1
N-1	Neutral	1
N-5	Neutral	5
U-1	Unstable	1
U-5	Unstable	5

TABLE 14.6.3

PRIMARY CONTAINMENT RESPONSE SUMMARY

Case	RHR Loops	RHR Pumps	High-Pressure Service Water Pumps	Containment Spray (gpm)	Core Spray (gpm)	LPCI (gpm)	Peak Pool Temperature (°F)	Secondary Peak Pressure (psig)
A*	2	4	4	20,000	12,500	20,000	145	None
B*	1	1	1	10,000	6,250	None	195	10.2
C*	1	1	1	None	6,250	10,000	195	12.6
C'	1	1	1	None	6,250	10,000	206	15.0

* Historical information - not accurate for current plant conditions. (See Section 14.9)

TABLE 14.6.4*

LOSS-OF-COOLANT ACCIDENT

PRIMARY CONTAINMENT AIRBORNE FISSION PRODUCT INVENTORY

	Fission Product Activity Airborne Primary Containment	
Time After Accident	Noble Gases (Ci)	Iodines (Ci)
1 min	1.67×10^6	1.36×10^4
30 min	1.57×10^6	1.30×10^4
1 hr	1.52×10^6	1.26×10^4
2 hr	1.46×10^6	1.18×10^4
8 hr	1.30×10^6	9.77×10^3
12 hr	1.26×10^6	8.99×10^3
1 day	1.16×10^6	7.54×10^3
2 days	1.02×10^6	6.05×10^3
4 days	7.89×10^5	4.63×10^3
30 days	8.75×10^4	4.51×10^3

* Historical information - not applicable to current plant conditions. (See Section 14.9.2)

TABLE 14.6.5*

LOSS-OF-COOLANT ACCIDENT

SECONDARY CONTAINMENT AIRBORNE FISSION PRODUCT INVENTORY

Time After Accident	Fission Product Activity Airborne Secondary Containment: Noble Gases (Ci)	Fission Product Activity Airborne Secondary Containment: Iodines (Ci)
1 min	5.81×10^{0}	4.75×10^{-2}
30 min	1.62×10^{2}	1.35×10^{0}
1 hr	3.12×10^{2}	2.57×10^{0}
2 hrs	5.86×10^{2}	4.75×10^{0}
8 hrs	1.85×10^{3}	1.39×10^{-1}
12 hr	2.48×10^{3}	1.77×10^{-1}
1 day	3.69×10^{3}	2.39×10^{-1}
2 days	4.42×10^{3}	2.63×10^{-1}
4 days	3.89×10^{3}	2.28×10^{-1}
30 days	4.40×10^{2}	2.26×10^{0}

* Historical information - not applicable to current plant conditions. (See Section 14.9.2)

TABLE 14.6.6*

LOSS-OF-COOLANT ACCIDENT

FISSION PRODUCT RELEASE RATE TO ENVIRONS

	Fission Product Activity Being Released to Environment	
Time After Accident	Noble Gases (Ci/sec)	Iodines (Ci/sec)
1 min	6.74×10^{-5}	5.40×10^{-9}
30 min	1.88×10^{-3}	1.56×10^{-7}
1 hr	3.62×10^{-3}	2.98×10^{-7}
2 hr	6.80×10^{-3}	5.51×10^{-7}
8 hr	2.15×10^{-2}	1.61×10^{-6}
12 hr	2.88×10^{-2}	2.05×10^{-6}
1 day	4.28×10^{-2}	2.77×10^{-6}
2 days	5.12×10^{-2}	3.05×10^{-6}
4 days	4.52×10^{-2}	2.64×10^{-6}
30 days	5.10×10^{-3}	2.62×10^{-7}

* Historical information - not applicable to current plant conditions. (See Section 14.9.2)

TABLE 14.6.7**

LOSS-OF-COOLANT ACCIDENT

Distance (m)	2-Hr Dose VS-1	2-Hr Dose MS-1	2-Hr Dose N-1	2-Hr Dose N-5	2-Hr Dose U-1	2-Hr Dose U-5	24-Hr Dose VS-1	24-Hr Dose MS-1	24-Hr Dose N-1	24-Hr Dose N-5	24-Hr Dose U-1	24-Hr Dose U-5
	Radiological Effects (Rem)											
Passing Cloud Whole-Body Dose												
730*	$2.0x10^{-5}$	$2.0x10^{-5}$	$2.1x10^{-5}$	$3.3x10^{-6}$	$3.3x10^{-5}$	$4.6x10^{-6}$	$1.2x10^{-3}$	$1.2x10^{-3}$	$1.3x10^{-3}$	$1.9x10^{-4}$	$1.9x10^{-3}$	$2.7x10^{-4}$
805	$1.9x10^{-5}$	$1.9x10^{-5}$	$2.1x10^{-5}$	$3.2x10^{-6}$	$3.3x10^{-5}$	$4.6x10^{-6}$	$1.1x10^{-3}$	$1.1x10^{-3}$	$1.2x10^{-3}$	$1.9x10^{-4}$	$1.9x10^{-3}$	$2.7x10^{-4}$
1,609	$1.4x10^{-5}$	$1.4x10^{-5}$	$1.9x10^{-5}$	$1.9x10^{-6}$	$2.3x10^{-5}$	$2.8x10^{-6}$	$8.0x10^{-4}$	$8.0x10^{-4}$	$1.1x10^{-3}$	$1.1x10^{-4}$	$1.3x10^{-3}$	$1.6x10^{-4}$
8,045	$4.1x10^{-6}$	$4.6x10^{-6}$	$4.5x10^{-6}$	$1.1x10^{-6}$	$2.2x10^{-6}$	$5.2x10^{-7}$	$2.4x10^{-4}$	$2.7x10^{-4}$	$2.6x10^{-4}$	$6.4x10^{-5}$	$1.3x10^{-4}$	$3.1x10^{-5}$
16,090	$2.2x10^{-6}$	$2.6x10^{-6}$	$1.4x10^{-6}$	$4.7x10^{-7}$	$5.8x10^{-7}$	$1.8x10^{-7}$	$1.3x10^{-4}$	$1.5x10^{-4}$	$8.2x10^{-5}$	$2.7x10^{-5}$	$3.4x10^{-5}$	$1.1x10^{-5}$
Lifetime Thyroid Dose												
730*	0	$1.1x10^{-19}$	$1.8x10^{-9}$	$1.7x10^{-12}$	$2.9x10^{-6}$	$3.8x10^{-7}$	0	$6.1x10^{-18}$	$1.0x10^{-7}$	$9.5x10^{-11}$	$1.6x10^{-4}$	$2.1x10^{-5}$
805	0	$9.9x10^{-19}$	$7.5x10^{-9}$	$1.7x10^{-11}$	$3.2x10^{-6}$	$4.6x10^{-7}$	0	$5.5x10^{-17}$	$4.2x10^{-7}$	$9.4x10^{-10}$	$1.8x10^{-4}$	$2.6x10^{-5}$
1,609	0	$2.8x10^{-13}$	$8.2x10^{-7}$	$5.3x10^{-8}$	$2.7x10^{-6}$	$5.7x10^{-7}$	0	$1.6x10^{-11}$	$4.6x10^{-5}$	$3.0x10^{-6}$	$1.5x10^{-4}$	$3.2x10^{-5}$
8,045	$6.3x10^{-27}$	$6.0x10^{-8}$	$6.3x10^{-7}$	$1.6x10^{-7}$	$2.8x10^{-7}$	$7.2x10^{-8}$	$3.5x10^{-25}$	$3.4x10^{-6}$	$3.5x10^{-5}$	$9.1x10^{-6}$	$1.6x10^{-5}$	$4.0x10^{-6}$
16,090	$2.1x10^{-17}$	$2.2x10^{-7}$	$2.5x10^{-7}$	$6.8x10^{-8}$	$9.8x10^{-8}$	$2.6x10^{-8}$	$1.2x10^{-15}$	$1.2x10^{-5}$	$1.4x10^{-5}$	$3.8x10^{-6}$	$5.5x10^{-6}$	$1.5x10^{-6}$

*Site Boundary
**Historical information - not accurate for current plant conditions. (See Section 14.9)

KEY:		Meteorology	Wind speed (m/s)
	VS-1	Very stable	1
	MS-1	Moderately stable	1
	N-1	Neutral	1
	N-5	Neutral	5
	U-1	Unstable	1
	U-5	Unstable	5

TABLE 14.6.8*

REFUELING ACCIDENT

SECONDARY CONTATINMENT AIRBORNE FISSION PRODUCT INVENTORY

Time After Accident	Fission Product Activity Airborne Secondary Containment: Noble Gases (Ci)	Fission Product Activity Airborne Secondary Containment: Iodines (Ci)
1 min	1.88×10^{4}	1.24×10^{3}
30 min	1.83×10^{4}	1.21×10^{3}
1 hr	1.77×10^{4}	1.19×10^{3}
2 hr	1.67×10^{4}	1.14×10^{3}
8 hr	1.17×10^{4}	9.54×10^{2}
12 hr	9.41×10^{3}	8.73×10^{2}
1 day	5.04×10^{3}	6.93×10^{2}
2 days	1.57×10^{3}	4.60×10^{2}
4 days	1.65×10^{2}	2.19×10^{2}
30 days	1.06×10^{-10}	2.60×10^{-2}

* Historical information - not applicable to current plant conditions. (See Section 14.9.2)

TABLE 14.6.9*

REFUELING ACCIDENT

FISSION PRODUCT RELEASE RATE TO ENVIRONS

Time After Accident	Fission Product Activity Being Released to Environment: Noble Gases (Ci/sec)	Iodines (Ci/sec)
1 min	2.18×10^{-1}	1.44×10^{-4}
30 min	2.12×10^{-1}	1.41×10^{-4}
1 hr	2.05×10^{-1}	1.37×10^{-4}
2 hr	1.93×10^{-1}	1.32×10^{-4}
8 hr	1.36×10^{-1}	1.11×10^{-4}
12 hr	1.09×10^{-1}	1.01×10^{-4}
1 day	5.84×10^{-2}	8.04×10^{-5}
2 days	1.82×10^{-2}	5.33×10^{-5}
4 days	1.91×10^{-3}	2.54×10^{-5}
30 days	1.23×10^{-15}	3.02×10^{-9}

* Historical information - not applicable to current plant conditions. (See Section 14.9.2)

TABLE 14.6.10**

REFUELING ACCIDENT

	Radiological Effects (Rem											
	2-Hr Dose						24-Hr Dos					
Distance (m)	VS-1	MS-1	N-1	N-5	U-1	U-5	VS-1	MS-1	N-1	N-5	U-1	U-5
Passing Cloud Whole-Body Dose												
730*	$1.2x10^{-3}$	$1.1x10^{-3}$	$1.2x10^{-3}$	$1.9x10^{-4}$	$1.9x10^{-3}$	$2.7x10^{-4}$	$6.6x10^{-3}$	$6.6x10^{-3}$	$7.1x10^{-3}$	$1.1x10^{-3}$	$1.1x10^{-2}$	$1.5x10^{-3}$
805	$1.1x10^{-3}$	$1.1x10^{-3}$	$1.2x10^{-3}$	$1.8x10^{-4}$	$1.9x10^{-3}$	$2.7x10^{-4}$	$6.4x10^{-3}$	$6.4x10^{-3}$	$7.1x10^{-3}$	$1.1x10^{-3}$	$1.1x10^{-2}$	$1.5x10^{-3}$
1,609	$7.9x10^{-4}$	$8.0x10^{-4}$	$1.1x10^{-3}$	$1.1x10^{-4}$	$1.3x10^{-3}$	$1.6x10^{-4}$	$4.6x10^{-3}$	$4.6x10^{-3}$	$6.3x10^{-3}$	$6.4x10^{-4}$	$7.6x10^{-3}$	$9.3x10^{-4}$
8,045	$2.4x10^{-4}$	$2.7x10^{-4}$	$2.6x10^{-4}$	$6.4x10^{-5}$	$1.3x10^{-4}$	$3.0x10^{-5}$	$1.4x10^{-3}$	$1.5x10^{-3}$	$1.5x10^{-3}$	$3.7x10^{-4}$	$7.5x10^{-4}$	$1.7x10^{-4}$
16,090	$1.3x10^{-4}$	$1.5x10^{-4}$	$8.1x10^{-5}$	$2.7x10^{-5}$	$3.4x10^{-5}$	$1.1x10^{-5}$	$7.2x10^{-4}$	$8.6x10^{-4}$	$4.6x10^{-4}$	$1.6x10^{-4}$	$1.9x10^{-4}$	$6.1x10^{-5}$
Lifetime Thyroid Dose												
730*	0	$6.6x10^{-17}$	$1.1x10^{-6}$	$1.0x10^{-9}$	$1.7x10^{-3}$	$2.3x10^{-4}$	0	$4.6x10^{-16}$	$7.8x10^{-6}$	$7.2x10^{-9}$	$1.2x10^{-2}$	$1.6x10^{-3}$
805	0	$6.0x10^{-16}$	$4.6x10^{-6}$	$1.0x10^{-8}$	$2.0x10^{-3}$	$2.8x10^{-4}$	0	$4.2x10^{-15}$	$3.2x10^{-5}$	$7.1x10^{-8}$	$1.4x10^{-2}$	$2.0x10^{-3}$
1,609	0	$1.7x10^{-10}$	$5.0x10^{-4}$	$3.2x10^{-5}$	$1.6x10^{-3}$	$3.4x10^{-4}$	0	$1.2x10^{-9}$	$3.5x10^{-3}$	$2.2x10^{-4}$	$1.1x10^{-2}$	$2.4x10^{-3}$
8,045	$3.8x10^{-24}$	$3.6x10^{-5}$	$3.8x10^{-4}$	$9.9x10^{-5}$	$1.7x10^{-4}$	$4.4x10^{-5}$	$2.7x10^{-23}$	$2.5x10^{-4}$	$2.6x10^{-3}$	$6.9x10^{-4}$	$1.2x10^{-3}$	$3.0x10^{-4}$
16,090	$1.3x10^{-14}$	$1.3x10^{-4}$	$1.5x10^{-4}$	$4.2x10^{-5}$	$6.0x10^{-5}$	$1.6x10^{-5}$	$9.0x10^{-14}$	$9.4x10^{-4}$	$1.0x10^{-3}$	$2.9x10^{-4}$	$4.1x10^{-4}$	$1.1x10^{-4}$

*Site Boundary
** Historical information - not accurate for current plant conditions. (See Section 14.9)

KEY:		Meteorology	Wind speed (m/s)
	VS-1	Very stable	1
	MS-1	Moderately stable	1
	N-1	Neutral	1
	N-5	Neutral	5
	U-1	Unstable	1
	U-5	Unstable	5

TABLE 14.6.11**

STEAM LINE BREAK ACCIDENT

Distance (m)	VS-1	MS-1	N-1	N-5	U-1	U-5
	Radiological effects (REM)					
Passing Cloud Whole-Body Dose						
730*	$6.0x10^{-3}$	$6.0x10^{-3}$	$4.8x10^{-3}$	$1.2x10^{-3}$	$2.0x10^{-3}$	$6.0x10^{-4}$
805	$6.0x10^{-3}$	$6.0x10^{-3}$	$5.2x10^{-3}$	$1.2x10^{-3}$	$2.2x10^{-3}$	$5.2x10^{-4}$
1,609	$4.8x10^{-3}$	$4.4x10^{-3}$	$2.8x10^{-3}$	$7.2x10^{-4}$	$8.8x10^{-4}$	$2.3x10^{-4}$
8,045	$1.5x10^{-3}$	$1.2x10^{-3}$	$2.8x10^{-4}$	$1.0x10^{-4}$	$4.8x10^{-5}$	$1.6x10^{-5}$
16,090	$7.8x10^{-4}$	$4.4x10^{-4}$	$6.4x10^{-5}$	$3.2x10^{-5}$	$1.0x10^{-5}$	$4.4x10^{-6}$
Lifetime Thyroid Dose						
730*	$6.4x10^{-3}$	$3.1x10^{0}$	$3.0x10^{0}$	$8.0x10^{-1}$	$8.0x10^{-1}$	$2.0x10^{-1}$
805	$6.8x10^{-3}$	$3.0x10^{0}$	$2.8x10^{0}$	$7.2x10^{-1}$	$6.4x10^{-1}$	$1.7x10^{-1}$
1,609	$3.2x10^{-2}$	$2.2x10^{0}$	$1.0x10^{0}$	$2.9x10^{-1}$	$1.8x10^{-1}$	$4.8x10^{-2}$
8,045	$2.9x10^{-1}$	$4.4x10^{-1}$	$6.4x10^{-2}$	$1.8x10^{-2}$	$1.0x10^{-2}$	$2.6x10^{-3}$
16,090	$2.6x10^{-1}$	$1.6x10^{-1}$	$1.8x10^{-2}$	$6.4x10^{-3}$	$2.8x10^{-3}$	$7.2x10^{-4}$

*Site Boundary
** Historical information - not accurate for current plant conditions. (See Section 14.9)

KEY:	Meteorology	Wind speed (m/s)	
VS-1	Very stable	1	
MS-1	Moderately stable		1
N-1	Neutral	1	
N-5	Neutral	5	
U-1	Unstable	1	
U-5	Unstable	5	

14.7 CONCLUSIONS

Tables 14.4.1 and 14.4.2 provide summaries of the results of the plant safety analyses. Because the spectrum of abnormal operational transients has been approached and analyzed by a method that included the various combinations of plant problems and operating conditions, general conclusions regarding the plant's behavior in response to operational problems can be made. Because none of the abnormal operational transients results in an MCPR of less than the Safety Limit MCPR, it can be concluded that unacceptable safety results 1 and 2 for abnormal operational transients are avoided. Because peak nuclear system pressure does not exceed 1,375 psig as a result of any abnormal operational transient, it can be concluded that unacceptable safety result 3 for abnormal operational transients is avoided.

The broad approach to and methodical categorization of accidents leading to unplanned releases of radioactive material from the fuel barrier and the nuclear system process barrier also justify general conclusions. A comparison of each of the design basis accident analyses with the unacceptable safety results for accidents shows that items 1, 2, 3, 4 and 5 are satisfied.

14.8 ANALYTICAL METHODS

Note: The material presented in Section 14.8 is largely historical. Descriptions of current analysis methods for the topics discussed in Sections 14.8.2, 14.8.3 and 14.8.4 can be found in References 14.8.9 through 14.8-20. The current licensing basis radiological consequences using the AST analysis are provided in Section 14.9.

14.8.1 Nuclear Excursion Analysis

14.8.1.1 Introduction

Although extensive preventative measures in the forms of equipment design and procedural controls are taken to avoid nuclear excursions, such an event is assumed as a design basis accident. Continued effort is made in the area of analytical methods to assure that nuclear excursion calculations reflect the state of the art in the field. This section outlines only the broader aspects of the subject. Greater detail is available in technical literature[1].

14.8.1.2 Description

There are many ways of inserting reactivity into a large-core BWR; however, most result in a relatively slow rate of reactivity insertion and therefore pose no threat to the system. The one category of reactivity additions that must be considered in evaluating large nuclear excursions is that associated with the control rod system. The rapid removal of a high worth control rod is the only way to obtain a rate of reactivity insertion high enough to result in a potentially significant excursion.

The rapid removal of a high-worth rod results in a high local $k\infty$ in a small region of the core. For large, loosely-coupled cores, this would result in a highly-peaked power distribution and subsequent shutdown mechanisms. Significant shifts in the spatial power generation would occur during the course of the excursion; therefore, the method of analysis must be capable of properly accounting for any possible effects of the power distribution shifts. This is an effect which is not significant in small cores.

With this background in mind, it is now possible to categorize nuclear excursions in water-moderated, oxide cores. The categorization criterion that seems most definitive is one based on the principal shutdown mechanisms that come into play. This method is particularly useful here because for fuel such as that in the current General Electric product line reactors, the principal shutdown mechanisms have a direct relationship to both

the consequences of the excursion and the applicable method of analysis. With respect to the energy densities presented, the following reference points are used:

Enthalpy = 0 cal/g at ambient temperature.

Enthalpy = 220 cal/g at incipient melting of UO_2

Enthalpy = 280 cal/g at fully-molten UO_2

Enthalpy = 425 cal/g when UO_2 vapor pressure is 1000 psi.

The following describes the three categories of nuclear excursions, assuming a very low initial power level. As shown in the table, there is some overlap in the three ranges of excursions. The indicated numbers for reactivity insertion rate, minimum period and peak energy density are nominal values and will vary somewhat from one reactor to another.

CHARACTERISTICS OF NUCLEAR EXCURSIONS

WATER-MODERATED OXIDE CORES

Range	Reactivity Insertion Rate ($/sec)	Minimum Period (ms)	Peak Energy Density (cal/g)	Principal Shutdown Mechanisms
Low	<2.5	>4	<120	Doppler Effect Moderator Effects
Medium	2-25	7-2	100-425	Doppler Effect
High	>20	<3	>380	Doppler Effect Core Disassembly

In the low reactivity insertion rate range, the reactor is barely prompt critical, and the energy that is stored in the fuel as a result of the nuclear burst is built up at a relatively slow rate. As a result, there may be a significant amount of heat transfer out of the fuel during the burst, and the negative moderator coefficient as well as the U-238 Doppler effect contributes to the shutdown mechanisms. In the medium range, the period is much shorter, and there is very little heat transfer out of the fuel during the burst. In this case, the principal shutdown mechanism is the Doppler effect. Finally, in the high range, there exists the possibility of core disassembly during the burst, due to high internal pressure causing prompt failure of fuel rods. This

results in a significant contribution toward shutdown of the excursion.

In terms of consequences, the low range is limited to no fuel cladding damage or at worst, a small amount of burnout. This poses no threat to nuclear system integrity; therefore, from a safety viewpoint, only the medium and high ranges are considered. The design basis rod drop accident is in the medium range, well below the range where core disassembly is possible.

14.8.2 Reactor Vessel Depressurization Analysis (See note at the beginning of Section 14.8.)

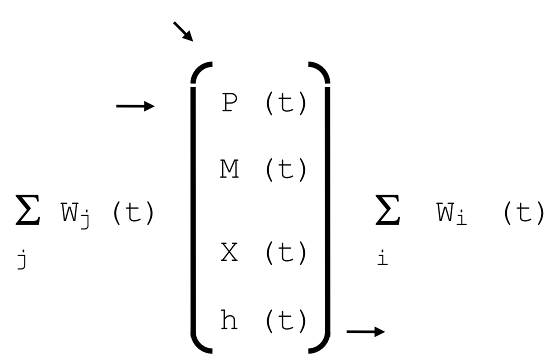
14.8.2.1 Introduction

The analytical methods used to calculate the energy and mass release rates issuing from a reactor vessel during rapid depressurization are described in this section. Conservation of mass and energy equations are written for a constant-volume system containing saturated steam and liquid in thermodynamic equilibrium to determine the thermodynamic state in the vessel. Mass flow rates into and out of the vessel are then used to find the rate of change of system pressure and mass inventory.

14.8.2.2 Theoretical Development

The mathematical formulation for the depressurization of the reactor vessel can be derived by considering the conservation of mass and energy in the constant-volume system during rapid depressurization, as shown in the control volume sketch below. If the mass flow rates are known it is possible to develop expressions of the rate of change of mass, energy, and pressure within the system.

CONTROL VOLUME, V



14.8.2.2.1 Mass Balance

The volume of the control system is comprised of saturated liquid and saturated vapor in equilibrium:

$$V = M_f v_f + M_g v_g = \text{constant} \qquad (14.1)$$

where:

V = Total volume of the system (reactor vessel)

v = Specific volume

M = Mass.

(The subscripts f and g refer to the liquid and vapor phases, respectively.)

Since the total mass in the system is simply

$$M = M_f + M_g \qquad (14.2)$$

then the steam quality by weight is given as:

$$X = \frac{M_g}{M} \qquad (14.3)$$

14.8.2.2.2 Mass Rate of Change in Vessel

From continuity, the rate of change of vapor mass in the system is equal to the net in-flow of vapor plus the rate at which liquid is flashed to vapor due to depressurization. Hence:

$$\frac{dM_g}{dt} = \sum_j W_{g_j} - \sum_i W_{g_i} + W_{fg} \qquad (14.4)$$

where:

w = Mass flow rate

W_{fg} = Net flashing rate

(The subscript j corresponds to inflow, while i refers to the outflow from the vessel evaluated at the thermodynamic conditions within the system.) Similarly, the rate of change of liquid mass in the vessel is:

$$\frac{dM_f}{dt} = \sum_j W_{f_j} - \sum_i W_{f_i} - W_{fg} \qquad (14.5)$$

14.8.2.2.3 Rate of Change of Energy in Vessel

The rate of change of energy in the system can be expressed from the First Law of Thermodynamics:

(Net energy inflow) - (Net energy outflow) = (Rate of change of internal energy)

$$\left(\dot{q} + \sum_j w_f h_f + \sum_j w_g h_g\right) - \left(\sum_i w_f h_f + \sum_i w_g h_g\right)$$

$$= \frac{d}{dt}\left(M_f h_f + M_g h_g - VP,\right) \qquad (14.6)$$

where:

h = Enthalpy

P = Saturated pressure in the system

$\dot{q}$ = Heat transfer rate to the fluid from the surroundings (solids)

The right-hand side of Equation (14.6) can be expanded, using the chain rule, to yield:

(Rate of change of internal energy)

$$= \left[M_g \frac{dh_g}{dP} + M_f \frac{dh_f}{dP}\right] \frac{dP}{dt} + h_g \frac{dM_g}{dt} + h_f \frac{dM_f}{dt} - V\frac{dP}{dt}. \qquad (14.6a)$$

14.8.2.2.4 Flashing Rate in Vessel

After substituting Equations (14.4), (14.5), and (14.6a) into Equation (14.6), the expression for the net flashing rate is:

$$W_{fg} = \frac{1}{h_{fg}} \Bigg\{ \dot{q} + \sum_j w_g h_g - \left(\sum_j w_g\right) h_g$$

$$+ \sum_j w_f h_f - \left(\sum_j w_f \right) h_f - \left[M_g \frac{dh_g}{dP} + M_f \frac{dh_f}{dP} - V \right] \frac{dP}{dt} \Bigg\} . \qquad (14.7)$$

14.8.2.2.5 <u>Vessel Depressurization Rate</u>

To arrive at an expression for depressurization rate, start by differentiating Equation (14.1) realizing that for a fixed total system volume $\frac{dv}{dt} = 0$ then:

$$M_g \frac{dv_g}{dt} + v_g \frac{dM_g}{dt} + M_v \frac{dv_f}{dt} + v_f \frac{dM_f}{dt} = 0 . \qquad (14.8)$$

Now expand this by means of the chain rule to obtain:

$$v_g \frac{dM_g}{dt} + v_f \frac{dM_f}{dt} + \left(M_f \frac{dv_f}{dP} + M_g \frac{dv_g}{dP} \right) \frac{dP}{dt} = 0 . \qquad (14.9)$$

With expressions for dM_g/dt and dM_f/dt as given in Equations (14.4) and (14.5), Equation (14.9) can be written:

$$v_g \left[\sum_j w_g - \sum_i w_g + w_{fg} \right] + v_f \left[\sum_j w_f - \sum_i w_f - w_{fg} \right] + \left[M_f \frac{dv_f}{dP} + M_g \frac{dv_g}{dt} \right] \frac{dP}{dt} = 0 . \qquad (14.10)$$

After substituting Equation (14.7) into Equation (14.10) and rearranging, the following expression for depressurization rate is obtained:

$$\frac{dP}{dt} = - \left[\frac{f_1 \ (P) + f_2(P)}{f_3(P)} \right] \qquad (14.11)$$

Where:

$$f_1(P) = v_f\left(\sum_j w_f - \sum_i w_f\right) + v_g\left(\sum_j w_g - \sum_i w_g\right)$$

$$f_2(P) = \frac{v_{fg}}{h_{fg}}\left[q + \sum_j w_f h_f - \left(\sum_j w_f\right) h_f + \sum_j w_g h_g - \left(\sum_j w_g\right) h_g\right]$$

$$f_3(P) = M_g\left[\frac{dv_g}{dP} - \left(\frac{v_{fg}}{h_{fg}}\right)\frac{dh_g}{dP}\right] + M_f\left[\frac{dv_f}{dP} - \left(\frac{v_{fg}}{h_{fg}}\right)\frac{dh_f}{dP}\right] + \left(\frac{v_{fg}}{h_{fg}}\right)\frac{v}{J}\ .$$

J = 778 ft-lb/Btu.

14.8.2.2.6 Mass Flow Rates

The mass flow rates entering the reactor vessel during the blowdown are treated as functions of time and are independent of the internal thermodynamic conditions in the vessel. These flow rates can be liquid or vapor, or some combination of the two. The outlet flow rate can be calculated from one of two flow models: critical flow as a function of the control volume stagnation properties P_o and h_o, or supercritical flow as a function of the pressure difference $P_o - P_{sink}$ (sink refers to the pressure outside the vessel).

Critical flow is flow which is "choked" at some point where the Mach number is unity in the line through which depressurization is taking place. Critical or maximum flow (both single-phase and two-phase) persists when the ratio of driving pressure (vessel pressure) to sink pressure (drywell) is greater than approximately two. The critical flow analysis of F. J. Moody[2] is used to determine the flow rate for critical flow conditions.

For the instantaneous values of pressure, P, enthalpy, h, and friction coefficient, $\bar{f}L/d$, a three-variable interpolation is performed using Moody's results to find the critical mass velocity:

$$G_c = G(P, h, \bar{f}L/d)\ . \qquad (14.12)$$

The mass flow rate is now calculated from:

$$w_c = AG_c, \qquad (14.13)$$

where:

A = Minimum flow area in the line.

Supercritical flow will exist prior to the formation of bubbles in a liquid flow and establishment of two-phase critical flow, or when the source pressure is low so that the ratio of $P_o/P_{sink} < 2$.

Supercritical mass velocity is calculated from:

$$G_{sc} = \frac{2g\left(P_o - P_{sink}\right)}{v_f\,(1.4 + \bar{f}\,L/d)\,\phi^2}$$

where: (14.14)

ϕ = Martinelli-Nelson two-phase multiplier.[3]

The mass flow rate is:

$$w_{sc} = AG_{sc} \quad (14.15)$$

14.8.2.3 <u>Numerical Solution</u>

If a function of time and its time derivatives are known at time t_1, the value of the function at time $t_1 + \Delta t$ obtained from a Taylor series expansion. The first three terms of the series are:

$$f\left(t_1 + \Delta t\right) = f\left(t_1\right) + \frac{\Delta t}{1!} f'\left(t_1\right) + \frac{\Delta t^2}{2!} f''\left(t_1\right) + \ldots \quad (14.16)$$

where:

$f'\left(t_1\right) = \frac{df}{dt}$ at $t = t_1$,

$f''\left(t_1\right) = \frac{d^2f}{dt^2}$ at $t = t_1$, and

Δt = Size of time step.

<u>Integration.</u> If the term involving the second derivative is negligible, the Euler forward integration method is obtained:

$$f(t_1 + \Delta t) = f(t_1) + \Delta t f'(t_1) \quad (14.17)$$

Time Step. A variable time step based on an accuracy criterion has been used in the integration method. The error made in one extrapolation of the Euler method can be approximated by the third term of Taylor's series, given by Equation (14.16), i.e.,

$$e \approx \frac{\Delta t^2}{2!} f''\left(t_1\right) \tag{14.18}$$

An exact equation for the second time derivative can be approximated by the rate of change of the first derivative, i.e.:

$$f''\left(t_1\right) = \frac{f'\left(t_1 + \Delta t\right) - f'\left(t_1\right)}{\Delta t} \tag{14.19}$$

After substituting Equation (14.10) into (14.18), an approximation of the error made in one time step can be calculated:

$$e \approx \frac{\Delta T}{2}\left[f'\left(t_1 + \Delta t\right) - f'\left(t_1\right)\right] \tag{14.20}$$

If the magnitude of this error is within the error criterion, ε, then the time step is doubled for the next calculation. If $|e| > \varepsilon$, then the time step is halved and the previous calculations are repeated.

Equations (14.4), (14.5), and (14.11) are programmed for machine calculation using the numerical methods described above.

14.8.3 Reactor Core Heatup Analysis (See note at the beginning of Section 14.8.)

14.8.3.1 Introduction

The analytical method used to calculate the reactor core thermal transient following a LOCA is described in this section. The fuel temperature, cladding temperature, channel temperature, and amount of metal-water reaction are calculated as functions of time from the start of the accident. In this analysis the power of decaying fission products, the chemical energy released by metal-water reactions, and the stored heat in the fuel, cladding, and other metal in the core are included as heat sources.

The fuel rods are classified such that those with similar power levels and fuel bundle locations are analyzed as a group. A one-dimensional heat balance is then written for each type of fuel rod. Heat is transferred from the surface of the fuel rods by convection to the water, steam, or hydrogen formed in the metal-water reaction. In addition, thermal radiation between fuel rods

and from the rods to the channel is accounted for in the overall heat balance.

14.8.3.2 Theoretical Development

A typical fuel rod consists of uranium dioxide fuel with a Zircaloy cladding. A fuel bundle consists of 49 fuel rods, grouped together to form a square array which is surrounded by a metal channel. The fuel rods are divided into four radial temperature zones for the numerical calculations, as shown in Figure 14.8.1. The cladding, on the other hand, is described by the average cladding temperature, with an outer surface temperature computed from the average temperature. The channel (Figure 14.8.1) is considered to be at a uniform temperature radially. The fuel rods within the channel are divided into four representative zones to describe the spatial variation of power generation. The entire reactor core consists of several hundred fuel bundles and channels. To describe the radial variations of power generation, the core is divided into five radial zones. The fuel rods and channels are divided into five axial regions. Axial conduction between regions is neglected. Each channel is considered to be isolated from the rest of the core so that ineractions between adjacent channels are neglected.

14.8.3.2.1 Heat Sources

The energy generated by delayed neutrons and decaying fission products is assumed to be uniform within a fuel rod and to have the same radial and axial variation within the core as the steady state power distribution. The chemical energy released by the metal-water reaction is described by the parabolic rate law given by Baker[4], where the rate of change of the metal oxide thickness is written as:

$$\frac{d\delta}{dt} = \frac{K}{\delta} \exp\left(-D / T_c\right) \qquad (14.21)$$

where:

K = Rate coefficient

T_c = Cladding temperature

D = Activation coefficient

δ = Oxide thickness

The heat generation rate and hydrogen release rate are proportional to the rate of change of oxide generated. The chemical heat liberated is given as follows:

$$\frac{dQ_c}{dt} = \frac{d\delta}{dt} \Delta H \rho_c A_s \qquad (14.22)$$

where:

H = Heat of reaction

ρ_c = Density of metal

A_s = Exposed surface area of oxide

The mass rate of hydrogen generated is:

$$\frac{dw_H}{dt} = 2 \frac{d\delta}{dt} \rho_c A_s \frac{N_{H_2}}{N_{METAL}} \qquad (14.23)$$

where:

W_H = Mass of hydrogen generated

N = Molecular weight

The above reaction rate considers that there is an unlimited source of saturated steam available for the reaction. The empirical reaction constants, K and D, are based upon experimental data obtained under conditions where the metal and water are at the same temperature. Therefore, for Equation (14.21) to be correct the water must be heated to the cladding temperature. The energy required to heat this water is deducted from the total chemical energy added to the system.

14.8.3.2.2 Conduction Heat Transfer

The heatup analysis considers only radial conduction of heat from the fuel to the cladding surface. Axial conduction along the fuel rods or to support structures is neglected. Resistance to heat flow through the fuel-cladding gap is taken into account.

14.8.3.2.3 Convection Heat Transfer

Heat is transferred from the cladding and channel to the surrounding fluid by thermal radiation and convection. During the blowdown, a convection heat transfer coefficient must be calculated. The water level is calculated from the mass inventory in the reactor vessel during the blowdown. If an axial node is

covered with water or steam water mixtures, the heat transfer coefficient for that node is obtained from the Jens-Lottes correlation for boiling heat transfer:

$$h_B = \frac{e^{P/900}}{1.9} \left(Q_s\right)^{0.75} \tag{14.24}$$

where:

P = Reactor pressure

Q_s = Surface heat flux

Equation (14.24) is used to describe the heat transfer coefficient if the calculated water level is above the center of the node. When water level drops below the center of the node, it is treated as being completely uncovered and the convective heat transfer rate diminishes to zero.

14.8.3.2.4 Radiation

Thermal radiation between fuel rods and the fuel channel box is permitted if they are not covered with water. To simplify calculations, the fuel rods are grouped into four groups. Figure 14.8.1 shows the channel configuration. Group 1 rods exchange radiation with groups 2, 3, and 4 rods and the channel. Group 2 rods exchange radiation with groups 1, 3, and 4 rods and the channel. Group 3 rods exchange radiation with groups 1, 2, and 4 rods and the channels. Finally, group 4 rods exchange radiation only with groups 1, 2, and 3 rods. Radiation view factors are also calculated for each group of rods. The view factors together with the emissivity and relative areas are converted to radiation coefficients used in the Stephan-Boltzman equation for obtaining the radiant heat transfer.

14.8.3.3 Method of Solution

The fuel, cladding, and channel temperature are calculated at each time step by considering the aforementioned energy consideration. All temperatures are integrated using a simple Euler forward difference method:

$$\phi \left(t + \Delta t\right) = \phi\left(t\right) + \frac{d\phi\left(t\right)}{dt} \Delta t \tag{14.25}$$

All physical properties are considered constant with temperature and time. The model utilizes the calculated histories of pressure, water level, and heat transfer coefficients. The sink

temperature for all convective heat transfer calculations is determined by the saturation temperature at the given pressure.

14.8.4 <u>Containment Response Analysis</u> (See note at the beginning of Section 14.8.)

14.8.4.1 <u>Short-Term Containment Response</u>

14.8.4.1.1 <u>Introduction</u>

This section describes the analytical model used to study the pressure suppression part of the LOCA. The system consists of a drywell which has mass and energy flowing into it from the reactor vessel, and a suppression chamber which receives flow from the drywell via the drywell vents.

The chain of events occurring during the pressure suppression phase of an accident is as follows:

1. Mass and energy enter the drywell from the reactor vessel.

2. Drywell temperature and pressure increase.

3. The increased pressure acts to clear water from the vents.

4. When the vents are cleared, air, steam, and water flow through the vents and into the suppression pool. The steam is condensed by the pool, increasing the pool temperature. The air passes through the pool and enters the suppression chamber air space, causing the pressure in the suppression chamber to rise.

5. As the transient proceeds, energy in-flow from the reactor vessel decreases, because of reduced reactor vessel pressure, while vent flow continues. The result is reduced drywell pressure.

6. Vent flow decreases with decreasing drywell pressure and the drywell and suppression chamber come to an equilibrium pressure.

14.8.4.1.2 <u>Theoretical Development</u>

<u>Drywell Conditions.</u> The total energy, E_D, air mass, M_{aD}, and water mass, M_{wD}(liquid vapor), in the drywell at any time are known by numerical integration of the appropriate rate equations, which are developed by taking mass and energy balances on the drywell.

An independent expression for total drywell energy is:

$$E_D = M_{f_D} u_f + M_{g_D} u_g + M_{a_D} C_v T_D \qquad (14.26)$$

where:

$$M_{f_D} = \left(1 - x_D\right) M_{w_D} \qquad (14.27)$$

$$M_{g_D} = x_D M_{w_D} \qquad (14.28)$$

$$x_D = \frac{v - v_f}{v_g - v_f} \qquad (14.29)$$

$$v = \frac{V_D}{M_{w_D}} \qquad (14.30)$$

and all thermodynamic properties are evaluated at T_D, the equilibrium saturation temperature of the air-steam-liquid mixture within the drywell. Numerical solution of Equation (14.26) gives T_D.

With T_D known, all thermodynamic properties are known, and the total drywell pressure is:

$$P_D = P_{g_D} + P_{a_D} \qquad (14.31)$$

where:

$$P_{a_D} = \frac{M_{a_D} R_a T_D}{V_{a_D}} \qquad (14.32)$$

$$V_{a_D} = V_D - v_f M_{f_D} \qquad (14.33)$$

Flow Rate Through Vents. The flow through vents is treated as compressible flow of an ideal gas with friction, except modifications are made to account for the effects of the liquid flowing in the vents. The total mass flow rate is then given as a function of friction factor, vent area, drywell/suppression chamber pressure ratio, and drywell fluid properties. Specifically:

$$\dot{w} = f\left(fL / D, A_V, \frac{P_D}{P_B}, V_M, x\right) \qquad (14.34)$$

where:

$\frac{fL}{D}$ = Vent friction factor,

A_V = Vent flow area,

$\frac{P_D}{P_B}$ = Pressure ratio (P_B = back pressure at vent exit)

v_M = Specific volume of gaseous phase (air and steam)

x = Quality in drywell

The specific volume of the gaseous phase is given by:

$$v_M = \frac{V_{a_D}}{M_{a_D} + M_{g_D}} \qquad (14.35)$$

The following expressions are used to determine the vent flow rate of each constituent:

$$\dot{w}_{g_2} = \frac{M_{g_D}}{M_{a_D} + M_{g_D} + M_{f_D}} \dot{w}_T \qquad (14.36)$$

$$w_{a_2} = \frac{M_{a_D}}{M_{a_D} + M_{g_D} + M_{f_D}} \dot{w}_T \qquad (14.37)$$

$$w_{f_2} = \frac{M_{f_D}}{M_{a_D} + M_{g_D} + M_{f_D}} \dot{w}_T \qquad (14.38)$$

The subscript 2 represents flow from drywell to suppression chamber.

Drywell Time Derivatives. To compute the rate of change of mass in the drywell, a mass balance gives:

$$\dot{M}_{w_D} = \dot{w}_{g_1} + \dot{w}_{f_1} - \dot{w}_{g_2} - \dot{w}_{f_2} \tag{14.39}$$

$$\dot{M}_{a_D} = \dot{w}_{a_2} \tag{14.40}$$

where the subscript 1 represents the source flow into the drywell. The rate of change of energy in the drywell, assuming no heat loss, is:

$$\dot{E}_D = h_{g_1}\ \dot{w}_{g_1} + h_{f_1}\ \dot{w}_{f_1} - h_{g_2}\ \dot{w}_{g_2} - h_{f_2}\ \dot{w}_{f_2}$$

$$- C_p T_D\ \dot{w}_{a_2} \tag{14.41}$$

An energy balance on the lumped suppression pool, assuming complete condensation of the steam entering the pool and neglecting heat losses and evaporation, gives:

$$h_{g_2}\ \dot{w}_{g_2} + C_p\left(T_D - T'_a\right)\ \dot{w}_{a_2}$$

$$= \frac{d}{dt}\left[M_{w_s}\ h\right] \tag{14.42}$$

where M_w is the pool mass and h is the enthalpy of the pool. After expanding the right-hand side, Equation (14.42) becomes:

$$\frac{d}{dt}\left[M_{w_s}\ h\right] = h\dot{M}_{w_s} + M_{w_s}\left(\frac{h}{T}\right)_p \dot{T}_S$$

$$+ \quad M_{w_s}\left(\frac{h}{P}\right)_T \dot{P}_S \tag{14.43}$$

For liquid water, $(h/T)_p = 1.0$, $(h/P)_T = 0.0$, thus:

$$\dot{T}_S = \frac{h_{f_2} w_{f_2} + h_{g_2} w_{g_2} + C_p\left(T_D - T'_a\right) w_{a_2} - h\dot{M}_{w_s}}{M_{w_s}} \tag{14.44}$$

A mass balance gives:

$$\dot{M}_{w_s} = \dot{w}_{f_2} + \dot{w}_{g_2} \tag{14.45}$$

It is assumed that in bubbling through the pool the air reaches the pool temperature, thus:

$$T'_a = T_s \tag{14.46}$$

Also, knowing T_s from the integral of Equation (14.44), P_{g_s}, the partial pressure of vapor in the suppression chamber can be determined:

$$P_s = P_{g_s} + \frac{M_{a_s} R_a T_{a_s}}{V_{a_s}} \tag{14.47}$$

where:

$$V_{a_s} = V_s - vM_{w_s}, \tag{14.48}$$

and

v = The specific volume of the pool water.

14.8.4.1.3 Solution

The equations described have been programmed for computer solution. Time derivatives are integrated using numerical methods. Thermodynamic properties in the drywell are found using tabular interpolations.

14.8.4.2 Long-Term Containment Pressure Response

The previously developed equations are used to calculate the containment transient during the reactor vessel depressurization and during the containment depressurization which follows the vessel transient. Once the depressurization transient is over (about 600 sec after the accident) a considerably simplified model can be used. The key assumptions employed in the simplified model are:

1. Drywell and suppression chamber, both are saturated and at the same total pressure.

2. An energy balance is performed to determine the temperature of the emergency core cooling flow as it drains by gravity back into the suppression chamber. The drywell is conservatively assumed to be 5 F hotter than the water draining back into the suppression pool.

3. The suppression chamber air temperature is taken equal to the pool temperature, which is determined from an energy balance on the pool mass.

4. No credit is taken for heat losses from the primary containment.

Since no mass is being added to the suppression pool, the pool temperature can be calculated based on the following energy balance:

$$\dot{T}_S = \frac{h_D \dot{m}_{D_o} - h_S \dot{m}_{S_o} - \dot{q}_{H_x}}{M_{W_s}} \tag{14.49}$$

where:

h_D = Enthalpy of water leaving drywell

$\dot{m}_{D_o}$ = Flow rate out of drywell

h_s = Enthalpy of water in suppression chamber

$\dot{m}_{s_o}$ = Flow rate out of suppression chamber

$\dot{q}_{H_x}$ = Heat removal rate of heat exchanger

M_{w_s} = Mass of water in suppression chamber

Assuming no storage in drywell:

$$\dot{m}_{D_o} = \dot{m}_{S_o} = \dot{m}_{CSCS}$$

And since the only heat source is the core decay heat, we have:

$$(h_D - h_S)\,\dot{m}_{CSCS} = \dot{q}_D \tag{14.50}$$

Therefore:

$$\dot{T}_S = \frac{q_D - q_{H_x}}{M_{W_S}} \qquad (14.51)$$

which can be integrated to give T_s as a function of time. At any point in time the drywell temperature is given by:

$$T_D = T_S + \frac{\dot{q}_D}{\dot{m}_{CSCS}} + 5\ F \qquad (14.52)$$

With the suppression chamber and drywell temperatures known and their total pressures assumed equal, it is now possible to solve for the total pressure

$$P_D = P_s$$

$$P_{a_D} + P_{v_D} = P_{a_s} + P_{v_s} \qquad (14.53)$$

$$\frac{M_{a_D} R_{T_D}}{V_D} + P_{v_D} = \frac{M_{a_s} R_{T_s}}{V_{a_s}} + P_{v_s}$$

The total mass, M_T, can be determined from a mass balance on the primary containment:

$$\dot{M}_T = \dot{M}_{a_D} + \dot{M}_{a_s} = \dot{m}_i - \dot{m}_{LEAK} \qquad (14.54)$$

where:

m_i = All non-condensible flow into containment, e.g., hydrogen from metal-water reaction

m_{LEAK} = leakage from primary containment

Therefore, at any time, M_T is known, and

$$M_T = M_{a_s} + M_{a_D} \qquad (14.55)$$

The two equations (14.53) and (14.55) can be solved for the two unknowns (M_{a_s} and M_{a_D}), and the pressure can be determined.

The leakage rate from the primary containment is determined from the following relationship:

$$\dot{m}_{LEAK} = L_T \left[\frac{1 - \left(\frac{1}{P}\right)^2}{1 - \left(\frac{1}{P_T}\right)^2} \right]^{1/2} \qquad (14.56)$$

where:

L_T = Leak rate at test pressure

P_T = Test pressure in absolute atmospheres

P = Containment pressure in absolute atmospheres

The above equations are solved simultaneously on a step-by-step basis to obtain the long-term pressure transient of the primary containment.

14.8.5 Analytical Methods for Calculating Radiological Effects

14.8.5.1 Introduction

This section describes the analytical techniques used to calculate the radiological exposures for each design basis accident. The external exposures are from airborne fission products and the internal exposures are from inhalation of airborne radioactive material. The first portion of the analysis concerns the meteorological considerations that describe the diffusion of the radioactive material as it emanates from the source and spreads through the atmosphere. The second portion of the analysis describes the radiological effects as a result of the dispersed materials.

The radiological effects of the design basis accidents are evaluated at various distances from the plant. The nearest distance is approximately the site boundary, with other distances included to illustrate the decrease of the radiological dose with distance. Since airborne materials are released via the elevated release point, the effects at short distance for any diffusion condition are much less for all modes of exposure except the passing cloud. At these short distances, the plume has not yet reached ground level so that exposure from inhalation is small. The passing cloud effect, however, remains nearly constant due to essentially line-source geometry of the elevated plume.

14.8.5.2 Meteorological Diffusion Evaluation Methods

14.8.5.2.1 General

Six points in the atmospheric diffusion spectrum are used to evaluate the radiological effects of secondary containment leakage via the elevated release point. These points represent the meteorological conditions which could exist at the site.

The atmospheric diffusion methods are the same as those reported in the Journal of Applied Meteorology(5).

14.8.5.2.2 Height of Release

Discharge from the secondary containment to the atmosphere emanates from the elevated release point. The effective height of release is the sum of the release heights plus an effluent rise due to momentum or buoyancy. For most of the design basis accidents, the additional effects of momentum and buoyancy are negligible, so that the effective release height is equal to the elevated release point heights. For the off-gas stack there is an effective height of 765 ft C.D. For the building ventilation stack the effective stack height equals 305 ft MSL. While buoyancy effects are significant for the steam line break accident, the conservative assumption is made that the release height is equal to the top of the turbine building.

14.8.5.2.3 Diffusion Conditions

An important parameter used in the atmospheric diffusion calculation is the measure of wind direction persistence and variability of direction. This parameter is the product of the standard deviation of the horizontal wind direction fluctuations, σ_θ, and the average wind velocity, $\acute{u}$. Combined with the assumed stability condition, specification of $\sigma_\theta\acute{u}$ permits calculation of air concentrations at various distances from the source.

A conservative value of 0.11 radian-m/sec is used for this parameter to describe the horizontal spreading of the plume for 1-m/sec wind speed conditions. A value of 0.55 radian-m/sec is typical for a 5-m/sec condition. These values are typical for a 1-hr period. A choice of wind direction persistence (the number of continuous hours) of 24 hr is used for poor diffusion conditions. This period is conservative when used with $\sigma_\theta\acute{u}$ of 0.1 radian-m/sec and 1-m/sec wind speed, based on the U. S. Weather Bureau data shown below(6).

DOSE COMPUTATIONAL METHODS
WIND DIRECTION PERSISTENCE

Station	Direction* (One Sector - 22 1/2°)	50%	10%	1%	0.1%	Longest No.Hr	Longest No. Hr** in any Direction	
		Frequency of Duration in Hours						
Augusta, Georgia	W	2	3	8	13	18	W	18
Birmingham, Alabama	S	2	4	9	16	16	SSE	20
Chicago, Illinois	SSW	2	5	12	21	22	NNE	25
Little Rock, Arkansas	SSW	2	4	9	17	28	SSE	28
Phoenix, Arizona	E	2	3	6	9	12	E	12
Rochester, New York	WSW	2	6	13	23	28	WSW	28
Salt Lake City, Utah	SSE	2	4	7	13	15	S	17
San Diego, California	NW	2	6	12	16	17	WNW	33
Tampa, Florida	ENE	2	3	7	13	14	SSW	18
Yakima, Washington	W	2	5	8	14	17	WNW	19

These data show that wind persistency of periods as long as 24 hr occurs only about 0.1 percent of the time, or less, at the sites listed. The sites include flat terrain, coastal and lake shore sites, and some valley locations. For a wind speed of 5 m/sec, a value of 0.55 radian-m/sec corresponds to a σ_θ of 0.11 radians, which is identical to the value of 0.11 radians for the 1 m/sec case. Thus, the same amount of wind variability is considered and the conservative 24-hr persistence assumption is applicable to both cases.

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*Direction examined is the one showing greatest frequency of persistent winds.

**Longest number of hours observed may not be same direction as direction showing most frequency of persistent winds.

14.8.5.2.4 Applied Meteorology

The diffusion and wind direction persistence conditions and breathing rates used for the design basis accident calculations are given below.

METEOROLOGY APPLICABLE TO DESIGN BASIS ACCIDENTS (H_{eff}>0)

Time After Accident	Diffusion Conditions Investigated*	Wind Variance During Indicated Time Period (% Occurrence)	Breathing Rate (cu m/sec)
0-8 hr	VS-1, MS-1, N-1, N-5, U-1, U-5	100%/C_L	$3.47x10^{-4}$
8-24 hr	VS-1, MS-1, N-1, N-5, U-1, U-5	100%/C_L	$1.75x10^{-4}$
1-4 days	VS-2	25%/22.5°	$2.32x10^{-4}$
	U-2	25%/22.5°	$2.32x10^{-4}$
4-30 days	VS-2	16.5%/22.5°	$2.32x10^{-4}$
	U-3	16.5%/22.5°	$2.32x10^{-4}$

METEOROLOGY APPLICABLE TO DESIGN BASIS ACCIDENTS (H_{eff}=0)

Time After Accident	Diffusion Conditions Investigated*	Wind Variance During Indicated Time Period (% Occurrence)	Breathing Rate (cu m/sec)
0-8 hr	VS-1, MS-1, N-1, N-5, U-1, U-5	100%/C_L	$3.47x10^{-4}$
8-24 hr	VS-1, MS-1, N-1, N-5, U-1, U-5	100%/C_L	$1.75x10^{-4}$
1-4 days	VS-2	25%/22.5°	$2.32x10^{-4}$
	N-2	25%/22.5°	$2.32x10^{-4}$
4-30 days	VS-2	16.5%/22.5°	$2.32x10^{-4}$
	N-3	16.5%/22.5°	$2.32x10^{-4}$

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*VS denotes very stable meteorological conditions;

MS - moderately stable, N - neutral, and U - unstable meteorological conditions. 1, 2, 3, and 5 denotes wind speed in m/sec.

14.8.5.2.5 Cloud Dispersion Calculations

The dispersion of the released effluent is described by the Gaussian Diffusion Equation given below.

$$X/Q_o = \frac{f_d}{\pi\kappa\sigma_y\sigma_z\,\bar{u}}\; e{-1/2}\left(\frac{y^2}{\sigma_y^2} + \frac{z^2}{\sigma_z^2}\right) \qquad (14.57)$$

where:

X/Q_o = Integrated air concentration per unit activity release

y = Distance from centerline crosswind (since plume centerline is used, $y = 0$)

z = Height of plume above ground

f_d = Cloud depletion factor (halogens only) (see paragraph 14.8.5.2.6)

σ_y = Horizontal diffusion coefficient

σ_z = Vertical diffusion coefficient

ú = wind speed at point of release (m/sec)

κ = Ground reflection factor

σ_y and σ_z are defined as follows:

$$\sigma_y^2 = At - A\alpha + A\alpha e^{-t/\alpha}\ ^{(5)} \qquad (14.58)$$

where:

$A = 13 + 232.5\,(\sigma_\theta\,\bar{u})$

$\alpha = \dfrac{A}{2(\sigma_\theta\,\bar{u})^2}$

t = Time after release and is = $\dfrac{x}{\bar{u}}$

where x is downwind dis tan ce.

The vertical cloud growth, as defined by the standard deviation of width σ_z is given by:

$$\sigma_z^2 = a\left[1 - e^{-k^2 t^2}\right] + bt \quad \text{stable case}^{(5)} \qquad (14.59)$$

$$\sigma_z^2 = \frac{C_z^2 x^{2-n}}{2} \quad \text{neutral and unstable case}^{(7)} \qquad (14.60)$$

The values of the constants in Equations (14.59) and (14.60) are given below.

Stability	Wind Speed:	a	b	k^2	c_z	n
	(M/sec)	(M^2)	(M^2/sec)	(Sec^{-2})	($M^{N/2}$)	-
VS,MS	1	3.4×10^{1}	2.5×10^{-2}	8.8×10^{-4}	-	-
U	1	-	-	-	3.0×10^{-1}	2.0×10^{-1}
U	5	-	-	-	2.6×10^{-1}	2.0×10^{-1}
N	1	-	-	-	1.5×10^{-1}	2.5×10^{-1}
N	5	-	-	-	1.2×10^{-1}	2.5×10^{-1}

The conventional "reflection" factor (K) of 2 usually applied for ground level releases is not included for $H_{eff}>0$, but is included for $H_{eff}=0$. For the passing cloud dose, which is primarily a gamma dose, the entire cloud volume is integrated as an "infinite" number of point sources to plus and minus infinity in the z-direction ignoring interception by the ground, so that the entire cloud volume is included. Inhalation doses are a function of concentration at the ground and subject to "reflection" effects if they exist. Since the materials of interest in inhalation effects deposit on the ground, "perfect" reflection will not occur, but rather the cloud will expand distorting the gaussian mass distribution resulting in at most a small increase in concentration. In addition, no account is taken of the better diffusion near the ground compared to the stack exit elevation used. In any event, an increase by a factor of less than 2 but perhaps more than 1 may be a result of this "reflection" effect. A factor of 1.0 is used in this analysis for elevated release points where terrain features are less than the height of release. However, for terrain features equal to the height of release (i.e., $H_{eff}=0$) a reflection factor of 2 is considered appropriate.

No distinction in the choice of the diffusion parameter is made between the first 2-hr period and the total accident dose calculations. This is inconsistent because larger values of this parameter are obviously appropriate for the longer time period.

That is, the values used, as discussed in paragraph 14.8.5.2.3, are for 1-hr periods, and thus are somewhat conservative when applied to the 2-hour period dose calculation and are markedly conservative for the total accident calculation.

14.8.5.2.6 Cloud Depletion and Ground Deposition

The fallout concentrations of radioactive materials are determined on the basis of particle settling by eddy diffusion only, since settling by gravity is expected to be negligible in this case.

The extent of halogen and solid fission product deposition on the ground is a function of the apparent deposition velocity which, in turn, is considered to be a function of the diffusion condition and wind speed. Deposition velocities used in this evaluation are given below[7].

	Wind Velocity	Deposition Velocity (cm/sec)	
Meteorology	(M/sec)	Noble Gases	Halogens
Very stable	1	0	0.24
Moderately stable	1	0	0.34
Unstable	1	0	0.80
Unstable	5	0	4.00
Neutral	1	0	0.46
Neutral	5	0	2.30

These values of deposition velocity are used in the calculation of the cloud depletion term, f_d.

$$f_d = \exp\left[\left(\frac{-v_g}{u_h}\right)\left(\sqrt{\frac{2}{\pi}}\right)\left\{\int_0^t \frac{u_h \exp\left(\frac{-z^2}{2\sigma_z^{\,2}}\right) dt}{\sigma_z}\right\}\right]$$

where:

f_d = Cloud depletion factor due to fallout

V_g = Deposition velocity of isotope in question (cm/sec)

u_h = Wind speed at height of release (cm/sec)

σ_z = Vertical diffusion coefficient (cm)

14.8.5.2.7 Air Concentration Calculation

Using the equations developed above, the integrated air concentration from a release of 1 curie of activity is calculated in Ci-sec/cu m. These data are given in Tables 14.8.1, 14.8.2, and 14.8.3 for the specified release heights and meteorological conditions.

14.8.5.3 Radiological Effects Calculation

The radiological doses of primary consideration are inhalation and cloud gamma. While the deposition gamma dose may be important from a decontamination viewpoint, it is of minor importance in evaluating the radiological consequences of a design basis accident and is, therefore, insignificant in this analysis.

The downwind radiological effects, such as cloud gamma and inhalation exposure, are a function principally of the integrated air concentration at any point. Calculation of this integrated concentration has been described in the preceding subsections. This subsection describes the conversion of air concentration to radiation dose.

14.8.5.3.1 Passing Cloud Dose

The ground level whole body cloud gamma dose which is received from airborne radioactive materials is determined by summing the dose contribution from each incremental volume of air containing fission product activity. The dose from a point in space to a receptor located at coordinates x, y and z is determined as follows:

$$D_g = \sum_{K=1}^{m\bullet} C_1 C_K f_K \int_{-\infty}^{\infty}\int_{-\infty}^{\infty}\int_{-\infty}^{\infty} \chi G_K \; dx dy\, dz \qquad (14.61)$$

where:

D_g = Gamma dose at the receptor point (Rem)

C_1 = Conversion factor (3.7 x 10^{10} disintegrations/sec-Ci)

χ = Integrated air concentration (Ci-sec/cu m)

f_K = The number of photons of the K^{th} isotope released per disintegration (photons/dis)

C_K = Flux to dose conversion factor, $\frac{\text{Rem - sq m}}{\text{photon}}$

G_K = Dose attenuation kernel, which is defined as follows:

where:

$$G_K = Be^{-ut}/4\pi T^2$$

where:

B = Buildup factor = 1 + kuT

k = $\dfrac{u - u_a}{u_a}$ where u is the total absorption coefficient and u_a is the energy absorption coefficient (m^{-1})

T = Distance from the source to the detector position and is equal to

$$\sqrt{x_1^2 + y_1^2 + z_1^2}$$

14.8.5.3.2 Inhalation Dose

The inhalation dose is an internal exposure which is received as a consequence of inhaling airborne radioactive fission products. Depending upon the isotopes inhaled there may be one or more organs which are affected.

The total activity inhaled during the inhalation period is

$$Q_{dep.} = \chi_i B_r \qquad (14.62)$$

where:

χ = Time integral of the air concentration previously defined in subsection 14.8.5.2.5 (Ci-sec/cu m)

B_r = Breathing rate (cu m/sec).

When the above equation is multiplied by an appropriate conversion factor, C_i (Rem/sec-Ci inhaled), a dose rate in the organ is obtained. The total dose resulting from inhalation of a mixture of fission products is

$$D_I = \sum_{i=1}^{N} \int_{o}^{t} \chi_i B_r C_i e^{-\lambda_i t} dt \qquad (14.63)$$

where:

D_I = Total inhalation dose (rad)

λ_i = Effective decay constant of the i^{th} isotope in the organ of reference (sec^{-1})

The summation sign indicates that all isotopes contributing to the organ dose are added to obtain the total inhalation dose.

The conversion factor, C_i, which is applicable to the isotope of the organ of interest, is calculated using the following mathematical model:

$$C_i = \frac{C_1 f_a E C_3}{M C_2} \qquad (14.64)$$

where:

C_i = Activity to dose conversion factor (rad/sec-curie inhaled)

C_1 = 1.6×10^{-6} ergs/Mev

f_a = Fraction of inhaled material reaching the organ of reference

E = The effective energy absorbed per disintegration (Mev/dis)

C_3 = 3.7×10^{10} dis/sec-Ci

M = Mass of the organ (g)

C_2 = 100 ergs/g-rad

Therefore,

$$C_i = \frac{(1.6 \times 10^{-6})\,(f_a)\,(E)\,(3.7 \times 10^{10})}{(M)\ (100)} = \frac{5.92 \times 10^{2}\ f_a E}{M} \quad \text{(rad/sec Ci inhaled)}$$

Upon integration of Equation (14.63), the total inhalation dose is:

$$D_T = \sum_{i=1}^{N} \frac{\chi_i \; B_r \; C_i \,(1 - e^{-\lambda_i t})}{\lambda_i} \qquad (14.65)$$

If t is large compared to λ, Equation (14.65) can be simplified to:

$$D_T = \sum_{i=1}^{N} \frac{\chi_i \; B_r \; C_i}{\lambda_i} = \sum_{i=1}^{N} = \frac{\chi_i \; B_r \; C_i \; T_i}{0.693} \qquad (14.66)$$

where:

T_i = Effective half life of the i^{th} isotope and is equal to,

$$T_i = \frac{T_b \; T_r}{T_b + T_r}$$

T_b = Biological half life (sec)

T_r = Radioactive half life (sec)

If the effective half life is defined in terms of days and is combined with the conversion factor, C_i, Equation (14.66) can be expressed as follows:

$$D_T = \sum_{i=1}^{N} \chi_i \; B_r \; C_i' \qquad (14.67)$$

where:

$$C_i' = \frac{8.64 \times 10^4}{6.93 \times 10^{-1}} \; T_i \; C_i$$

$$= 1.25 \times 10^5 \; T_i \; C_i \; (\text{rad / Ci inhaled})$$

For the thyroid gland the dose conversion factor is:

$$C_i' = \frac{(1.25 \times 10^5)(5.92 \times 10^2)(f_a) E T_i}{M}$$

$$= \frac{7.40 \times 10^7 f_a E T_i}{20} = 3.7 \times 10^6 f_a E T_i$$

The numerical values which are used in Equation (14.64) and the dose conversion factor, C_i, are given below:

THYROID DOSE CONVERSION FACTORS

Isotope	Effective Half-Life (Days)	f_a	E (Mev/dis)	C_i rad/Ci inhaled
I-131	7.6×10^0	2.3×10^{-1}	2.3×10^{-1}	1.49×10^6
I-132	9.7×10^{-2}	2.3×10^{-1}	6.5×10^{-1}	5.65×10^4
I-133	8.7×10^{-1}	2.3×10^{-1}	5.4×10^{-1}	4.21×10^5
I-134	3.6×10^{-2}	2.3×10^{-1}	8.2×10^{-1}	2.64×10^4
I-135	2.8×10^{-1}	2.3×10^{-1}	5.2×10^{-1}	1.30×10^5

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TABLE 14.8.1*

CALCULATED AIR CONCENTRATION FOR 0-METER RELEASE HEIGHT

(Ci-sec/cu m/Ci released)

Distance (m)	Activity of Interest	VS-1	MS-1	N-1	N-5	U-1	U-5
730	Noble Gases	6.4×10^{-4}	2.5×10^{-4}	1.2×10^{-4}	3.0×10^{-5}	5.0×10^{-5}	1.9×10^{-5}
	Halogens	4.9×10^{-4}	2.1×10^{-4}	1.1×10^{-4}	2.8×10^{-5}	4.7×10^{-5}	1.1×10^{-5}
1,609	Noble Gases	2.6×10^{-4}	9.1×10^{-5}	3.0×10^{-4}	7.5×10^{-6}	1.2×10^{-5}	2.9×10^{-6}
	Halogens	1.6×10^{-4}	6.8×10^{-5}	2.7×10^{-4}	6.9×10^{-6}	1.1×10^{-5}	2.7×10^{-6}
8,045	Noble Gases	4.4×10^{-5}	1.3×10^{-5}	2.1×10^{-6}	5.9×10^{-7}	8.3×10^{-7}	2.2×10^{-7}
	Halogens	9.7×10^{-6}	6.1×10^{-6}	1.9×10^{-6}	5.3×10^{-7}	7.7×10^{-7}	2.0×10^{-7}
16,090	Noble Gases	2.1×10^{-5}	6.0×10^{-6}	7.7×10^{-7}	2.2×10^{-7}	2.9×10^{-7}	7.9×10^{-8}
	Halogens	2.0×10^{-6}	2.0×10^{-6}	6.9×10^{-7}	1.9×10^{-7}	2.7×10^{-7}	7.3×10^{-7}

NOTE: The diffusion parameter $\sigma\theta\bar{u}$ assumed is 0.1 radian-m/sec for the 1-m/sec cases and 1.0 radian-m/sec for the 5-m/sec cases.

KEY:

	Meteorology	Wind speed (m/s)
VS-1	Very stable	1
MS-1	Moderately stable	1
N-1	Neutral	1
N-5	Neutral	5
U-1	Unstable	1
U-5	Unstable	5

* Historical information – not applicable to current plant design. (See Section 14.9.2)

TABLE 14.8.2*

CALCULATED AIR CONCENTRATION FOR 30-METER RELEASE HEIGHT

(Ci-sec/cu m/Ci released)

Distance (m)	Activity of Interest	VS-1	MS-1	N-1	N-5	U-1	U-5
730	Halogens	1.62×10^{-7}	9.10×10^{-5}	9.01×10^{-5}	2.26×10^{-5}	2.21×10^{-5}	5.73×10^{-6}
	Noble Gases	1.62×10^{-7}	9.25×10^{-5}	9.34×10^{-5}	2.34×10^{-5}	2.31×10^{-5}	6.02×10^{-6}
805	Halogens	1.98×10^{-7}	8.88×10^{-5}	8.04×10^{-5}	2.09×10^{-5}	1.87×10^{-5}	4.85×10^{-6}
	Noble Gases	1.98×10^{-7}	9.05×10^{-5}	8.37×10^{-5}	2.17×10^{-5}	1.96×10^{-5}	5.12×10^{-6}
1,609	Halogens	9.25×10^{-7}	6.39×10^{-5}	2.94×10^{-5}	8.64×10^{-6}	5.47×10^{-6}	1.43×10^{-6}
	Noble Gases	9.25×10^{-7}	6.67×10^{-5}	3.14×10^{-5}	9.28×10^{-6}	5.88×10^{-6}	1.56×10^{-6}
8,045	Halogens	8.60×10^{-6}	1.24×10^{-5}	1.82×10^{-6}	5.55×10^{-7}	2.92×10^{-7}	7.65×10^{-8}
	Noble Gases	9.09×10^{-6}	1.53×10^{-5}	2.06×10^{-6}	6.41×10^{-7}	3.30×10^{-7}	8.78×10^{-8}
16,090	Halogens	7.75×10^{-6}	4.71×10^{-6}	5.22×10^{-7}	1.58×10^{-7}	8.14×10^{-8}	2.12×10^{-8}
	Noble Gases	9.26×10^{-6}	6.79×10^{-6}	6.14×10^{-7}	1.92×10^{-7}	9.48×10^{-8}	2.52×10^{-8}

NOTE: $\sigma\theta\bar{u}$ for the above stabilities and wind speeds is 0.11, 0.11, 0.11, 0.55, 0.11, and 0.55.

KEY:		Meteorology	Wind speed (m/s)
	VS-1	Very stable	1
	MS-1	Moderately stable	1
	N-1	Neutral	1
	N-5	Neutral	5
	U-1	Unstable	1
	U-5	Unstable	5

* Historical information - not applicable to current plant design. (See Section 14.9.2)

TABLE 14.8.3*

CALCULATED AIR CONCENTRATION FOR 152-METER RELEASE HEIGHT

(Ci-sec/cu m/Ci released)

Distance (m)	Activity of Interest	VS-1	MS-1	N-1	N-5	U-1	U-5
730	Halogens	0	$1.65x10^{-19}$	$2.81x10^{-9}$	$2.57x10^{-12}$	$4.37x10^{-6}$	$5.71x10^{-7}$
	Noble Gases	0	$1.65x10^{-19}$	$2.81x10^{-9}$	$2.57x10^{-12}$	$4.40x10^{-6}$	$5.74x10^{-7}$
805	Halogens	0	$1.50x10^{-18}$	$1.14x10^{-8}$	$2.54x10^{-11}$	$4.88x10^{-6}$	$7.05x10^{-7}$
	Noble Gases	0	$1.50x10^{-18}$	$1.14x10^{-8}$	$2.54x10^{-11}$	$4.92x10^{-6}$	$7.08x10^{-7}$
1,609	Halogens	0	$4.29x10^{-13}$	$1.25x10^{-6}$	$8.03x10^{-8}$	$4.08x10^{-6}$	$8.60x10^{-7}$
	Noble Gases	0	$4.29x10^{-13}$	$1.26x10^{-6}$	$8.04x10^{-8}$	$4.17x10^{-6}$	$8.77x10^{-7}$
8,045	Halogens	$9.56x10^{-27}$	$9.11x10^{-8}$	$9.50x10^{-7}$	$2.46x10^{-7}$	$4.21x10^{-7}$	$1.09x10^{-7}$
	Noble Gases	$9.56x10^{-27}$	$9.13x10^{-8}$	$9.89x10^{-7}$	$2.58x10^{-7}$	$4.48x10^{-7}$	$1.16x10^{-7}$
16,090	Halogens	$3,22x10^{-17}$	$3.37x10^{-7}$	$3.73x10^{-7}$	$1.04x10^{-7}$	$1.49x10^{-7}$	$3.92x10^{-8}$
	Noble Gases	$3.22x10^{-17}$	$3.42x10^{-7}$	$4.00x10^{-7}$	$1.13x10^{-7}$	$1.63x10^{-7}$	$4.33x10^{-8}$

NOTE: $\sigma\theta\overline{u}$ for the above stabilities and wind speeds is 0.11, 0.11, 0.11, 0.55, 0.11, and 0.55.

KEY:	Meteorology	Wind speed (m/s)
VS-1	Very stable	1
MS-1	Moderately stable	1
N-1	Neutral	1
N-5	Neutral	5
U-1	Unstable	1
U-5	Unstable	5

* Historical information – not applicable to current plant design. (See Section 14.9.2)

14.9 EVALUATIONS USING AEC METHOD

14.9.1 Evaluation of Plant Systems Using TID-14844 Source Terms

Note: The material presented in Section 14.9.1 is historical and describes the analysis for the original plant design. The current licensing basis radiological consequences using the AST analysis are provided in Section 14.9.2.

In addition to the analyses presented in the preceding subsections, an evaluation was made of the adequacy of the containment and engineered safeguards using the assumptions of TID-14844 with regard to the fission product source terms as a basis.

14.9.1.1 Source Term Assumptions

For the purpose of calculating the "worst" dose to critical materials and heat loading, as well as the "worst" airborne or waterborne activity, the following assumptions were made:

1. Activity in suppression pool - 50 percent of the core halogen inventory and 1 percent of the core particulate inventory are instantaneously released to the suppression pool.

2. Activity airborne in primary containment - 100 percent of the core noble gas activity, 25 percent of the halogen activity, and 1 percent of the core particulate activity are airborne in the primary containment.

3. Activity airborne in secondary containment - The airborne primary containment activity noted in item 2 is released at a constant leak rate of 0.635 percent per day to the secondary containment, uniformly mixed in the secondary containment, and released to the standby gas treatment system at the rate of 1 air change/day.

4. Filter Activity and Heat Loading - The airborne primary containment activity noted in item 2 is released at a constant leak rate of 0.635 percent per day directly to the standby gas treatment system where the filter efficiency is assumed to be 100 percent.

Use of the above assumptions results in the activities and heat loadings presented in Table 14.9.1.

The effect of increasing the primary containment leak rate from 0.635 percent per day to 2.0 percent per day is to in-

crease activity or heat loading in the secondary containment or on standby gas treatment system filters. While the increase is actually time dependent, a conservative approach is to increase the activities and heat loadings determined for 0.635 percent per day by about a factor of three. A leak rate greater than 0.635 percent per day would reduce the activity and heat load in the primary containment at a rate inversely proportional to the leak rate.

14.9.1.2 Standby Gas Treatment System

The data presented and the discussion that follows show that the standby gas treatment system, as designed, will tolerate service conditions that arise from utilizing TID-14844 fission product source terms for a LOCA. Data are included for both 0.635 percent per day and 2.0 percent per day primary containment leak rates. A comparison of design basis capability and "as-built" capability is shown in Table 14.9.2.

The standby gas treatment system has been designed with all active components redundant to prevent a single component failure from preventing operation of the system. It is designed to process a maximum of 10,500 scfm of air from the secondary containment. The temperatures shown in Table 14.9.2 represent filter media (HEPA filter) or carbon (charcoal filter) temperature at the indicated air flows.

High Efficiency Particulate Air Filter

These filters are rated at a temperature of 200°F for continuous service. As shown in Table 14.9.2, with 2,000 scfm air flow, based on a filter efficiency of 100 percent and 1.0 percent of the core particulate activity available for release at a 2.0-percent primary containment leakage rate, after 30 days of continuous operation the filter temperature will reach 112°F. If at this time it is postulated that the air flow would decrease to 250 scfm, the temperature of the filter media would rise to 182°F.

The design incorporates two such filters, one upstream of the charcoal filter and one downstream. The analysis above applies to the upstream filter since all fission products are assumed to be contained on it. However, the downstream filter acts as a backup filter and would retain particulates which escape the upstream filter in the event that leakage or bypassing of the first filter occurs.

Charcoal Filter

Each of the charcoal filter trains contains a minimum of 1,320 lb of charcoal.

As shown in Table 14.9.2, the maximum temperature of the charcoal filter is 127°F, based on a 2,000 scfm air flow, 25 percent of the core halogen inventory retained on the filter with 100 percent filter efficiency and a 2.0 percent per day primary containment leakage rate, 12 days following a postulated LOCA. If the air flow over the filter is assumed to be reduced to 250 scfm at this time, the maximum charcoal temperature is calculated to be 209°F, well below the ignition temperature of 640°F. Water sprays are provided for the charcoal filter to prevent overheating if the air flow should be totally lost while a large radioactive iodine inventory is present.

14.9.1.3 Core Standby Cooling System Components

There are no CSCS components located within the primary containment that would be required to function following the postulated LOCA that could suffer significant radiation damage other than electrical penetrations that are discussed separately. Valves in the RHR and core spray systems are fluid-actuated check valves and consequently not subject to loss of operability because of radiation damage.

Components that are within the secondary containment that could be affected by radiation doses are pump seals and motor operator valves as discussed below:

1. Pumps - The RHR and core spray pumps that would be expected to operate following a LOCA have seals consisting of Buna "N" O-rings and carbon seal rings. Irradiation tests(1) on the Buna "N" material indicate that an integrated dose of 4×10^6 rads results in a 25 percent decrease in compression set properties. Carbon materials have a dose threshold on the order of 3×10^{11} rads(2). Source terms presented in Table 14.9.1 were converted to the equivalent integrated doses as shown in Table 14.9.3. For the outside surface of a schedule 80, 24-in diameter pipe containing water from the suppression chamber, it is shown that the integrated dose for 180 days is about 8.1×10^5 rads. It is expected that this dose would represent the maximum dose received by the seals and therefore failure due to radiation damage is not expected.

2. Valves - No valve operation is expected to be required after the accident is contained and the RHR system is placed in the shutdown cooling mode. This event could occur about 8 hr

after the accident so that the total integrated dose given in Table 14.9.3 at 12 hr of 7.7 x 10^4 rads adequately represents the equipment dose. No loss of valve operability would be expected at this dose.

14.9.1.4 Electrical Penetrations

As shown in Table 14.9.3, the total integrated dose at the interior surface of the drywell would be 3.4 x 10^7 rads at 180 days. The electrical penetrations are sealed with materials with radiation breakdown threshold of at least 1 X 10^8 rads.

Each penetration has two such seals, one inside and one outside the primary containment and both seals will maintain the structural integrity for specified integrated radiation dose.

14.9.1.5 Control Room

1. Ventilation - Fission products from a LOCA will be released from the top of the stack. Should these fission products reach ground level, the control room ventilation system outside air may be processed through a gas treatment system employing a HEPA filter and a charcoal filter to remove radioactive particulate and halogen fission products.

2. Shielding - Biological dose rates inside the control room have been evaluated based on fission product source terms specified in TID-14844.

3. A separate secondary containment bypass leakage component of primary containment leakage is also evaluated. This is treated as a ground level release, with credit for holdup and plateout (elemental and particulate iodine only) in steam line piping and the condenser.

The fission product activity in the secondary containment as a function of time after a LOCA assuming a 0.635 percent primary containment leak rate and a secondary containment ventilation rate of 100 percent per day is given in Table 14.9.1. These activity levels result from primary containment airborne activity levels based on a release of 100 percent of the noble gases, 50 percent of the halogens, and 1 percent of the solids contained in the core at the time of the accident. A plateout factor of 2 is assumed appropriate for the halogens. Therefore, the airborne halogen activity is 25 percent of the total core activity.

The gamma ray energy spectrums of the fission product activity in the secondary containment 1 hr and 1 day after the accident are shown in Table 14.9.4. The average gamma ray activity decreases

in energy as a function of time. The average energies may be grouped as a function of time after the accident as follows:

Time After Accident	Average Photon Energy
0-12 hr	0.6 Mev
12-48 hr	0.3 Mev
2 days-30 days	0.08 Mev
30 days-6 months	0.02 Mev

Dose rate calculations were performed using activity levels based on a 0.635 percent per day leak rate. Approximate variations in dose rate attributable to variations in leak rate may be obtained by multiplying by a direct ratio of the leak rates. These results will be conservative for times after the LOCA greater than 2 days.

Dose rate calculations were performed by using a point kernel type shielding computer code. The reactor building was represented by a cylindrical drywell of solid concrete with 1-ft-thick concrete floors at the 135, 165, 195, and 234-ft levels, and the 2.5-ft-thick concrete reactor building wall separating the reactor building and the control room. The control room was modeled to include the 2.5-ft-thick control room wall, the 2.5-ft-thick ceiling in the control room, and the 1-ft-thick control room floor. The source was distributed throughout the reactor building based on total mixing. Each floor was treated as a separate source region. Six gamma ray energy groups were used (Table 14.9.4). The results of these calculations are shown in Table 14.9.5. The integrated dose over the first 30 days following a LOCA is given in Table 14.9.6.

The secondary containment bypass is assumed to be a separate 0.145 percent per day primary containment leak rate. This is based on the assumption that all four steam lines are leaking at their Technical Specification allowable isolation valve leak rate of 11.5 scfh. The condenser is assumed to leak at a rate of 1 percent of its volume per day. This leakage path dominates the control room LOCA dose because of the lower dispersion compared with stack releases.

The shielding design and ventilation system of the main control room were also evaluated under design basis conditions for the main steam line break, the control rod drop, and the refueling accidents. In evaluating each accident, the sources of radioactivity were determined from the assumptions of paragraph 14.9.2. Thyroid inhalation doses and whole-body doses were calculated for each of the design basis accidents considered.

The main steam line break accident analysis is based on a release of 165,120 lb of reactor water and 25,800 lb of steam. All associated activity was assumed to be released to the turbine

building and hence to the building wake via the reactor building vent. Radioisotope concentrations in the reactor water and steam were based on the values given in Table 9.4.2 and Table 9.4.4 of the FSAR evaluated at an off-gas rate of 350,000 μCi/sec at 30-min decay. It was conservatively assumed that all activity is released from the turbine building for the purpose of computing the whole-body dose and the thyroid inhalation dose.
The cavity dilution effects were calculated by the method described in Section 5-5 of "Meteorology and Atomic Energy-1968" (TID-24190) using a shape factor of 0.5. A constant breathing rate of 347 cc/sec was assumed for the 2-hr duration of the accident. Charcoal filters in the control room ventilation intake were assumed to be 90 percent efficient for iodines. Using these assumptions a thyroid inhalation dose of 1.6 Rem and a whole-body dose of 36 mRem were calculated. Doses were also calculated to account for iodine spiking concentrations in the reactor coolant. The resultant doses for this case were 18.0 Rem to the thyroid and 36 mRem to the whole body.

In evaluating the control rod drop accident, the radioactive source term was computed from the assumptions made in paragraph 14.9.2 of the FSAR. Rupture of 330 rods from the highest burnup region of the core was assumed, releasing 50 percent of the rod inventory of halogens and 100 percent of the rod inventory of noble gases to the reactor water. These assumptions are not the same as found in Reference 14.0.6 or in current standard methodology for calculating radiological consequences; however, the net effect of these assumptions provides conservative and bounding results. Ten percent of the halogens in the water and 100 percent of the noble gases were released to the condenser. Halogen plateout in the condenser was assumed to be 50 percent. The condenser was assumed to leak at a rate of 0.5 percent per day for 24 hr. This leakage was assumed to pass unfiltered and unmixed through the reactor building vent. Dilution in the building wake was assumed to occur prior to intake at the control room vent. Charcoal filters in the control room vent were assumed to be 90 percent efficient for iodines. The resultant doses were 0.8 Rem to the whole body and 8.0 Rem to the thyroid, based on continuous control room occupancy for the 24-hr duration of the accident.

For the refueling accident (described in FSAR paragraph 14.9.2.2), one bundle of 49 rods was assumed to be damaged, releasing 30 percent of the inventory of noble gases and 15 percent of the inventory of halogens. Ninety percent of the halogens were assumed to be retained by the refueling water after release. These assumptions are not the same as found in Reference 14.0.6 or in current standard methodology for calculating radiological consequences, however, the net effect of these assumptions provides conservative and bounding results. The inventory of the

bundle was determined by assuming that the reactor has been operating at 3,528 MWt for 1,000 days. A power peaking factor of 1.5 was applied to the damaged rods and 24 hr of decay were assumed to occur. Charcoal filters in the standby gas treatment system, which exhausts the refueling area to the main stack, and charcoal filters in the control room ventilation system were both assumed to be 90 percent efficient for iodine removal.

The release was assumed to pass directly to the environment via the main stack over a 2-hr period with no mixing in any building volume. The meteorological conditions on which the accident was evaluated were Pasquill stability class F with a 1-m/sec wind speed and fumigation conditions. The resultant whole-body dose and inhalation thyroid dose are 0.5 Rem and 2.2 Rem, respectively.

14.9.1.6 Conclusion

All the necessary safety-related systems and equipment are capable of performing their intended functions in the presence of either the design basis radiation sources or postulated TID-14844 sources.

14.9.2 Current Licensing Basis Dose Evaluations Using Alternative Source Term (RG 1.183)

14.9.2.1 Loss-of-Coolant Accident (LOCA)

The radiological dose consequences are analyzed using the AST, guidance in RG 1.183, Appendix A, and TEDE dose criteria for the post-LOCA containment, ESF, and MSIV leakage release paths.

Containment Leakage:

1. The reactor has operated for an extended period at 4,030 MWt.

2. The isotopic core inventory (source term) accounts for operation at 4,030 MWt and equilibrium End of Cycle (EOC) core average exposure with nuclear fuel using various fuel enrichments. Fuel enrichments that bound the expected core designs are evaluated in the source term calculation. The LOCA calculation uses the bounding isotopic distribution that is based on the most limiting source term from these bounding enrichments.

   The core inventory release fractions, by radionuclide compositions, for the gap release and early in-vessel release phases for a design basis LOCA are obtained from Table 1 in Regulatory Guide 1.183 and listed in Table 14.9.9. These fractions are applied to the equilibrium average core fission products, which are assumed to instantaneously and

homogeneously become airborne in the primary containment (drywell) air volume.

3. Tthe primary containment leak rate is assumed to be 0.70 percent per day for 720 hours. No credit is taken for reduced containment leakage due to a reduction in containment pressure.

4. The airborne fission products immediately flow through the standby gas treatment system (SGTS) and the stack without mixing in the secondary containment building.

5. The SGTS charcoal and HEPA filters are not credited for removing the radioiodine and aerosols prior to release to the environment.

6. The chemical form of radioiodine released into the containment is assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide.

ESF Leakage:

7. The ESF systems that recirculate torus water outside of the primary containment are assumed to leak during their intended operation.

8. With the exception of noble gases, all the fission products released from the core to the containment (as defined in Sections 14.9.2.1.2 above) are assumed to instantaneously and homogeneously mix in the torus water at the time of release from the core.

9. The total ESF leakage from all components in the ESF systems is assumed to be 10 gpm, which is equal to 2 times the expected leakage of 5 gpm and assumed to start immediately after the onset of a LOCA. With the exception of iodine, all remaining fission products in the recirculating liquid are assumed to be retained in the liquid phase.

10. The post-LOCA temperature of torus water recirculated through the ESF system is expected to be less than 212° F; therefore, 10% of the iodine activity in the ESF leaked liquid is assumed to become airborne.

11. The radioiodine that is postulated to be available for release to the environment due to ESF leakage is assumed to be 97% elemental and 3% organic.

MSIV Leakage:

12. The MSIV leakage is postulated to release to the environment through the MSIV failed steam line and one of the three remaining intact steam lines. Each release path consists of two well-mixed volume nodes consistent with the AEB 98-03 two segment nodalization - piping between the Reactor Pressure vessel (RPV) nozzle to outboard MSIV and that between the outboard MSIV to Turbine Stop valve (TSV).

13. The main steam isolation valves (MSIVs) are postulated to leak at a total design leak rate of 300 scfh for at 49.1 psig and the activity available for release via MSIV leakage is assumed to be that time dependent activity released into the drywell.

14. A total of 300 scfh MSIV leakage is assumed to occur as follows:

    - 150 scfh through the shortest steam line. This line is modeled as having a failed inboard MSIV. Conservatively, the deposition of aerosol and removal of elemental iodine activities are not credited in the steam line between the RPV nozzle and the outboard MSIV. The deposition of aerosols and removal of elemental iodine are conservatively credited only in the horizontal pipe between the outboard MSIV and Turbine Stop Valve (TSV) for 0-96 hours.

    - 150 scfh through the shortest of the three remaining (intact) steam lines. The deposition of aerosol and removal of elemental iodine activities are conservatively credited only in the horizontal pipe segments between the RPV nozzle and TSV for 0-96 hours.

15. The aerosol deposition removal efficiencies for the main steam lines are determined based on the methodology in Appendix A of AEB-98-03 using only the horizontal pipe projected area (Diameter x Length). The natural removal efficiency for elemental iodine in each steam line volume is calculated based on the steam line wall temperature profile using the J.E. Cline methodology.

<u>Offsite Dose Consequences:</u>

16. The offsite dose is determined as a TEDE, which is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure from all radionuclides that are significant with regard to dose consequences and the released radioactivity.

17. The maximum exclusion area boundary (EAB) TEDE for any two-hour period following the start of the radioactivity release is determined and used in determining compliance with the dose acceptance criteria in 10CFR50.67.

18. TEDE is determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and is used in determining compliance with the dose criteria in 10CFR50.67.

19. No correction is made for depletion of the effluent plume by deposition on the ground. The breathing rates specified in RG 1.183, Section 4.1.3 and 4.1.5 are used.

20. The offsite dose analysis uses the Committed Effective Dose Equivalent (CEDE) Dose Conversion Factors (DCFs) for inhalation exposure per RG 1.183, Section 4.1.2 and Table 2.1 of Federal Guidance Report 11.

21. The dose calculation calculates Deep Dose Equivalent (DDE) using whole body submergence in semi-infinite cloud with appropriate credit for attention by body tissue. As a result, the DDE can be assumed nominally equivalent to the Effective Dose Equivalent (EDE) from external exposure. Therefore, the dose analysis uses EDE, taken from Table III.1 of US Environmental Protection Agency Report EPA-402-R-93-081, "Federal Guidance Report 12, External Exposure to Radionuclides in Air, Water and Soil" (Sept. 1993), in lieu of DDE Dose Conversion Factors in determining external exposure.

<u>Control Room Dose Consequences:</u>

22. The radioactivity from the containment, ESF, and MSIV leakage sources is assumed to be released into the environment and transported to the CR air intake, where it may leak into the CR envelope or be filtered by the CR intake filtration system prior to being distributed in the CR envelope. The following four major radioactive source are analyzed to calculate the CR TEDE dose:

    - Post-LOCA airborne activity inside the CR
    - Post-LOCA airborne cloud external to CR
    - Post-LOCA containment shine to CR
    - Post-LOCA Main Control Room Emergency Ventilation (MCREV) filter shine

23. The radioactive material releases and radiation levels used in the control room dose analysis are determined using the

same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values.

24. The models used to transport radioactive material into and through the control room, and the shielding models used to determine radiation dose rates from external sources, are determined to provide suitably conservative estimates of the exposure to control room personnel.

Dose Acceptance Criteria:

25. The following NRC regulatory requirement and guidance documents are applicable to this PBAPS AST LOCA Calculation:

- Regulatory Guide 1.183, Table 6
- 10CFR50.67
- Standard Review Plan section 15.0.1

Dose Acceptance Criteria are:

Regulatory Dose Limits

Dose Type	Control room (rem)	EAB and LPZ (rem)
TEDE Dose	5	25

Atmospheric Dispersion Factors (χ/Qs):

26. The control room atmospheric dispersion factors (χ/Qs) for the onsite release points such as the Main stack for containment and ESF leakage release paths and the TB/RB exhaust vent for the MSIV leakage release path are developed using PBAPS plant specific meteorology and the NRC sponsored computer code ARCON96. The EAB and LPZ χ/Qs are developed using the PBAPS plant specific meteorology and the NRC sponsored computer code PAVAN. The control room, EAB, and LPZ χ/Qs are shown in Table 14.9.10.

    The remaining design inputs and assumptions for the LOCA analysis are listed in Table 14.9.10.

Radiological Dose Consequences:

27. The post-LOCA control room dose consequences are shown in Table 14.9.6. The corresponding EAB and LPZ dose consequences are shown in Table 14.9.7.

14.9.2.2 Refueling Accident

1. There are 172 fuel pins that are expected to be damaged in the limiting case. The fuel burnup and linear heat generation rate limits are bounded by those specified in RG 1.183, footnote 11, which allows application of the gap activity fractions per RG 1.183, Table 3.

2. All of the damaged fuel pins are assumed to have a limiting Peaking Factor (PF) of 1.7.

3. 5 percent of the noble gases (excluding Kr-85), 10 percent Kr-85, 5 percent of the iodine inventory (excluding I-131), 8% of the I-131, and 12% of the Alkali metal inventory contained within the damaged rods are released to the pool water.

4. For a drop over the spent fuel pool, coverage over the dropped assemblies slightly less than 23 feet and the plenum in struck assemblies is 23 feet. Coverage is sufficient to maintain a 200 DF based on the Reg. Guide 1.183 Appendix B recommended 500 DF for inorganic iodines, and the recommended inorganic/organic iodine ratios.

5. The refuel floor exhaust rate is set artificially high at 6 air changes per hour. This results in 99.9999% of the contained radioactivity being exhausted within two hours.

Various post-FHA release cases are analyzed with the different fuel decay times as follows:

Release Pathway	Decay Time
All releases except hatches (Note 2)	24 hours
Hatch H17, H18, H33 (Note 1)	24 hours
Hatch H1/H19 (Note 1)	24 hours
Hatch H2/H20 (Note 1)	288 hours
Hatch H21, H22, H34 (Note 1)	312 hours
Hatch H23, H24, H15, H16 (Note 2)	24 hours

Note 1: MCREV initiation is credited with CR unfiltered inleakage of 500 cfm.
Note 2: MCREV initiation is not credited with CR in normal mode of operation.

Offsite Dose Consequences:

7. The offsite dose assumptions are same as those in Section 14.9.2.1 for the LOCA analysis.

Control Room Dose Consequences:

8. The radioactive material releases and radiation levels used in the control room dose analysis are determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values. The MCREV parameters for two different licensing basis cases analyzed are given in Table 14.9.11. The source terms of FHA are bounded by those of LOCA based on comparison of source terms in Tables 1 & 3 of RG 1.183, the CR operator direct doses from the external sources are not analyzed.

Dose Acceptance Criteria:

9. The following NRC regulatory requirement and guidance documents are applicable to FHA analysis:

   - Regulatory Guide 1.183, Table 6
   - 10CFR50.67
   - Standard Review Plan section 15.0.1

   Dose Acceptance Criteria are:

Regulatory Dose Limits

Dose Type	Control Room (rem)	EAB and LPZ (rem)
TEDE Dose	5	6.3

Atmospheric Dispersion Factors (χ/Qs):

10. The χ/Qs are based on RG 1.194 methodology as implemented by ARCON96 for onsite locations (Control Room) and on Reg. Guide 1.145 methodology as implemented by PAVAN for offsite locations (EAB & LPZ). The control room, EAB, and LPZ χ/Qs are shown in Table 14.9.11.

Radiological Dose Consequences:

11. The post-FHA control room dose consequences are shown in Table 14.9.6. The corresponding EAB and LPZ dose consequences are shown in Table 14.9.7.

14.9.2.3 Main Steam Line Break Accident (MSLB)

The radiological dose consequences are analyzed using the guidance in RG 1.183, Appendix D and TEDE dose criteria for a Main Steam Line Break (MSLB) accident.

1. No fuel damage is expected to result from a MSLB. Therefore, the activity available for release from the break is that in

the reactor coolant and present in steam lines prior to the break.

2. Two cases analyzed corresponding to the Reactor Coolant System Specific Activity limits in Technical Specification 3.4.6 and its Basis. Case 1 is for continued full power operation with a maximum equilibrium coolant concentration of 0.2 uCi/gm dose equivalent 1-131. Case 2 is for a maximum coolant concentration of 4.0 uCi/gm dose equivalent I-131, based on a pre-accident iodine spike caused by power excursion.

3. The release model is identical to that historically used. The previously determined mass of reactor coolant release and mass of steam release, before the break is isolated by MSIV closure, are used.

4. 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 masses of steam and water released are 25,800 lb and 165,120 lb respectively.

5. The steam cloud is assumed to consist of the initial steam blowdown and that 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, not just the portion that flashes to steam. This assumption is conservative because it considers the maximum release of fission products including iodine and noble gas.

<u>Offsite Dose Consequences:</u>

6. Dose models for both onsite and offsite are simplified and meet RG 1.183 requirements, providing results in units of Total Effective Dose Equivalent (TEDE). Dose conversion factors are based on Federal Guidance Reports 11 and 12.

<u>Control Room Dose Consequences:</u>

7. CR operator doses are determined somewhat differently, because steam cloud concentrations are used, rather than χ/Q times a curie release rate. No MCREV system or filtration is credited during the MSLB and inhalation doses are determined based on the isotopic concentrations at the intake for the duration of plume traverse. External exposure doses are determined based on concentrations at the intake for the duration of the plume traverse. A geometry factor is used to credit the reduced plume size seen in the control room per RG 1.183, Section 4.2.7.

Dose Acceptance Criteria:

8. The following NRC regulatory requirement and guidance documents are applicable to MSLB accident analysis:

   - Regulatory Guide 1.183, Table 6
   - 10CFR50.67
   - Standard Review Plan section 15.0.1

   Dose Acceptance Criteria are:

Regulatory Dose Limits

Accident Case	CR (30 days)	EAB (2 hours)	LPZ (30 days)
Maximum Equilibrium Iodine	5	2.5	2.5
Pre-accident Iodine Spike Case	5	25	25

Atmospheric Dispersion Factors (χ/Qs):

9. The activity concentrations for AST were originally those for Pasquill F Stability and 1-m/sec wind speed for the duration of the accident. However, the site specific χ/Qs were determined to be more conservative and are used for the MSLB dose calculations that are performed at 4,030 MWt. The χ/Qs are based on Reg. Guide 1.145 methodology as implemented by PAVAN for offsite locations (EAB and LPZ). For the MSLB, the site specific χ/Qs at 0 - 0.5 hours are used for EAB, and LPZ that are 9.11E-04 and 1.38E-04 sec/m3, respectively.

Radiological Dose Consequences:

10. The post-MSLB accident control room dose consequences are shown Table 14.9.6. The corresponding EAB and LPZ dose consequences are shown Table 14.9.7.

14.9.2.4 Control Rod Drop Accident (CRDA)

1. The number of Failed Fuel Rods is assumed to be 1,200 rods for bounding case of 10x10 bundle for GNF2 fuel. An average power peaking factor of 1.7 per pin was assumed. Ten percent (10%) of the core Inventory of noble gases and iodine, and 12% of the core Inventory of alkali metals, are released from the breached fuel gap.

2. Five percent (5.0%) of the breached fuel is conservatively assumed to melt during the CRDA. One hundred percent (100%)

of noble gases and 50% of the iodines contained in the melted fuel fraction are assumed to be released to the reactor coolant.

3. The activity released from the fuel from either the gap or from fuel pellets is assumed to be instantaneously mixed with the reactor coolant within the pressure vessel. One hundred (100%) of all noble gases, 10% of the iodines, and 1% of alkali metal nuclides are transported to the Turbine/Condenser.

4. Of the activity that reaches the turbine and condenser, 100% of the noble gases, 10% of the iodine, and 1% of the alkali metal nuclides are available for release to the environment.

5. Upon detection of high radiation levels during a CRDA by the MSLRM, no credit is taken for MSIV closure, nor SJAE shutdown. Credit is taken for MVP cessation.

6. It is assumed that the purging of the Reactor Water (RW) chemistry sample line continues for 40 minutes until the sample valves are closed by a MCR operator. These sample line valves are classified as Primary Containment Isolation Valves (PCIVs). The post-CRDA release through the RW sample line path contributes additional doses, which are added to corresponding total post-CRDA doses at EAB, LPZ, and CR locations.

7. The following potential release paths were reviewed and analyzed to determine the most limiting combination of the credible release paths as a result of the MSIVs remaining open during the CRDA:

    a) An isolated Main Condenser is assumed to exhaust the post-CRDA activity through the reactor building stack as a ground level release at a rate of 1% per day. No credit is taken for dilution or holdup within the turbine building. Radioactive decay during holdup in the turbine is not assumed (i.e., the activity is instantaneously transported to the condenser). Radioactive decay during holdup in the condenser is assumed.

    b) For a release without the automatic MSIV trip, the methods of analysis are the same as that used for the first case except for those that pertain to the release path. For this case, the evaluation assumes that the MSIVs do not close and that steam flow continues for approximately 24 hours before this path is Isolated. If the event occurs at low

power and the steam jet air ejector does not operate, then the offsite dose is equivalent to that of the first case because the total activity is assumed to be transferred to the condenser instantaneously. If sufficient power is available for SJAE operation, then some of the available activity in the Main Condenser is transported to the Offgas System and thus provides a different release path to the environment. The charcoal beds in the Offgas System ensure that the iodine is retained in the charcoal beds and not released to the environment. The noble gases are held for significant decay times (refer to Table 14.9.12) before release from the stacks and to the environment.

c) During the tow power operating conditions there are forced flow paths from the Turbine/Main Condenser. For instance, the CRDA can occur during MVP operation, which can exhaust an unprocessed release from the condenser at a significantly larger rate. Since the MSLRM trip function of the MVP is retained, this release path would be promptly terminated with the resulting doses considerably less than rounding error.

d) A release path not automatically Isolated during this event is via the Turbine Gland Seal Condenser. For the turbine gland seal release, the reactor steam containing the CRDA source is assumed to be directly released to the environment without any holdup, dilution, and partitioning of the radioiodine and particulates in the gland seal condenser.

e) Another release path not automatically isolated during this event is the consideration of a concurrent RW chemistry sample being taken when the CRDA occurs. For this case, it is assumed that an MCR Operator closes the PCIVs by remote-manual means within 40 minutes.

As discussed in the preceding discussion, the four credible release paths that exist during a CRDA are through the isolated Main Condenser, SJAE, gland seal condenser, and RW sample line path. The SJAE release path exists when there is enough steam pressure available to sustain its operation. Only the SJAE or Isolated Main Condenser release path exists at a given time. The operation of the SJAE maintains the Main Condenser at a vacuum, and thereby eliminates the

potential for a release through the Main Condenser. The CRDA occurring at a low power level secures the operation of the SJAE, which pressurizes the Main Condenser and establishes a post-CRDA release path. The Turbine Gland Seal Condenser is operational as long as there is pressurized steam available during the CRDA event. Since the MSIV is postulated to remain open for 24 hours, the reactor decay heat continues to produce steam and thereby establishes the Turbine Gland Seal Condenser release path irrespective of Main Condenser or SJAE operation. Consequently, in addition to the assumed 40 minute chemistry RW sample release path, the two possible combinations of release paths that may exist at any time during a CRDA event are:

a) The isolated Main Condenser, gland seal condenser, and RW sample release paths.

b) The SJAE, gland seal condenser, and RW sample release paths.

8. All leakage from the main steam turbine condenser leaks to the atmosphere from the Turbine Building/Reactor Building Ventilation Exhaust Slack at a rate of 1% per day, for a period of 24 hours.

9. The accident release duration is 24 hours except for the RW sample line which is 40 minutes.

Control Room Dose Consequences:

10. The radioactive material releases and radiation levels used in the control room dose analysis are determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values. The MCREV parameters are given in Table 14.9.12. The MCR external cloud is negligible because less than 2% of the core fuel is damaged, and 60 fuel rods are melted. Also there is no release into containment, so there is no containment shine dose and the MCR emergency filtration is not modeled, so there is no MCR filter shine dose. Therefore, the MCR Operator doses from the external cloud, containment shine, and MCR filter shine doses are not analyzed for the CRDA.

Dose Acceptance Criteria:

11. The following NRC regulatory requirement and guidance documents are applicable to CRDA analysis:

- Regulatory Guide 1.183, Table 6
- 10CFR50.67

- Standard Review Plan section 15.0.1

Dose Acceptance Criteria are:

Regulatory Dose Limits

Dose Type	Control Room (rem)	EAB and LPZ (rem)
TEDE Dose	5	6.3

<u>Atmospheric Dispersion Factors (χ/Qs):</u>

11. The χ/Qs are based on RG 1.194 methodology as implemented by ARCON96 for onsite locations (Control Room) and on Reg. Guide 1.145 methodology as implemented by PAVAN for offsite locations (EAB & LPZ). The control room, EAB, and LPZ χ/Qs are shown in Table 14.9.12.

<u>Radiological Dose Consequences:</u>

12. The post-CRDA control room dose consequences are shown in Table 14.9.6. The corresponding EAB and LPZ dose consequences are shown in Table 14.9.7.

14.9.2.5 <u>Conclusion</u>

The EAB, LPZ, and control room doses following the design basis accidents including the loss of coolant accident, fuel handling accident, main steam line break accident, and control rod drop accident using the AST, guidance in RG 1.183, and TEDE dose criteria are within the allowable regulatory dose limits.

14.9 EVALUATIONS USING AEC METHOD

REFERENCES

1. Boalt, R.O. and Cornal, J.G., "Radiation Effects on Organic Materials," Academic Press, 1963.

2. Rockwell, "Reactor Shielding Design Manual," TID-7044.

3. Standard Review Plan.

TABLE 14.9.1*

ACTIVITY, MASS LOADING, AND HEAT LOADING AT VARIOUS LOCATIONS

FOR TID RELEASE ASSUMPTIONS

0.635%/day Primary Containment Leak Rate

Time and Parametric Value

	1 Hr	10 Hr	1 Day	10 Days	30 Days	Peak Value and Time
Act. In SP (Ci)	4.6×10^{8}	2.0×10^{8}	1.2×10^{8}	3.5×10^{7}	1.7×10^{7}	6.1×10^{8} @ 0 hr
Heat Load SP (kW)	5.1×10^{3}	1.3×10^{3}	6.4×10^{2}	1.6×10^{2}	9.4×10^{1}	7.9×10^{8} @ 0 hr
Act. Airborne PC (Ci)	6.6×10^{8}	3.7×10^{8}	2.8×10^{8}	9.1×10^{7}	2.5×10^{7}	10.0×10^{8} @ 0 hr
Heat Load PC (kW)	3.5×10^{3}	1.0×10^{3}	5.2×10^{2}	1.4×10^{2}	5.1×10^{1}	8.0×10^{3} @ 0 hr
Act. Airborne SC (Ci)	1.4×10^{5}	6.8×10^{5}	1.1×10^{6}	4.9×10^{5}	7.7×10^{4}	1.2×10^{6} @ 50 hr
Heat Load SC (W)	10.0×10^{2}	2.1×10^{3}	2.3×10^{3}	8.1×10^{2}	1.7×10^{2}	2.3×10^{3} @ 24 hr
Act. on HEPA (Ci)	10.0×10^{3}	6.6×10^{4}	1.3×10^{5}	6.9×10^{5}	1.4×10^{6}	1.4×10^{6} @ 30 days
Heat Load on HEPA (W)	3.3×10^{1}	1.5×10^{2}	2.9×10^{2}	1.2×10^{2}	2.3×10^{3}	2.3×10^{3} @ 30 days
Act. on CF (Ci)	4.3×10^{4}	2.0×10^{5}	3.3×10^{5}	6.9×10^{5}	3.5×10^{5}	$6. \times 10^{5}$ @ 10 days
Heat Load on CF (W)	2.1×10^{2}	6.8×10^{2}	8.7×10^{2}	1.3×10^{3}	6.9×10^{2}	1.3×10^{3} @ 10 days

KEY: SP = Suppression pool
PC = Primary containment
SC = Secondary containment
HEPA = High-efficiency particulate absorber filter
CF = Charcoal filter

* Historical information - not applicable to current plant design.

Historical Information – not applicable to current plant design

TABLE 14.9.2

STANDBY GAS TREATMENT SYSTEM PERFORMANCE

	Design Temperature Capability	Assuming TID-14844 Sources: 0.635%/day leakage	Assuming TID-14844 Sources: 2.0%/day leakage
Iodine Accumulated (g)		1,344	3,488
HEPA Filter	250°F		
Watts		2,257[1]	5,848
Temp. at 250 scfm		135°F	182°F
Temp. at 2,000 scfm		105°F	112°F
Charcoal Filter	640°F		
Watts		1,334	3,488[2]
Temp. at 250 scfm		145°F	209°F
Temp. at 2,000 scfm		110°F	127°F

(1) Peak heat load evaluated at ~ 30 days.
(2) Peak load occurs at ~ 10 days.

TABLE 14.9.3*

DOSES FOR VARIOUS EQUIPMENT OR LOCATIONS BASED ON

TID-14844 FISSION PRODUCT RELEASE ASSUMPTIONS

Location or Equipment	Max. Dose Rate (R/hr)	Integrated Dose (Rad) For			
		12 Hr	2 Days	30 Days	180 Days
Surface 24 in - Sch 80-pipe	1.4×10^4	7.7×10^4	2.6×10^5	5.8×10^5	8.1×10^5
Interior surface drywell	1.0×10^6	4.9×10^6	1.2×10^7	2.4×10^7	3.4×10^7
Floor of corner comp. containing core spray pump seals	3.4×10^1	1.3×10^2	1.3×10^3	3.9×10^3	9.3×10^3
Pumps seals	$\sim 1.4 \times 10^4$	7.7×10^4	2.6×10^5	5.8×10^5	8.1×10^5
Refueling floor	5.4×10^2	2.3×10^3	2.1×10^4	6.0×10^4	1.4×10^5

* Historical information – not applicable to current plant design.

TABLE 14.9.4*

GAMMA RAY ENERGY SPECTRUM OF FISSION PRODUCTS IN THE SECONDARY CONTAINMENT

1 Hr After LOCA Total Activity = 41.3 Ci/MWt
Gamma Ray Energy Mev/sec-MWt x 10^{-10}

Nuclide	Ci/MWt	0.0-0.5	0.5-1.0	1.0-1.5	1.5-2.0	2.0-2.5	>2.5
Kr 83	0.053	0.0224					
85	1.4	1.090					
85	0.078	0.0604					
87	1.8	2.660	0.106			0.809	7.10
88	3.4	3.480	7.010		7.86	4.94	
Xe 131	0.047	0.0285					
133	0.48	0.4160					
133	14.00	41.90					
135	0.29	0.568					
135	2.4	2.230	0.014				
138	0.99	1.760	0.149	4.31			
I 131	1.9	4.070	0.103				
132	2.2	0.501	14.30	6.314			
133	3.1	7.030	1.03				
134	1.8	0.247	11.83	2.88	1.070		
135	3.1	0.316	1.07	12.70	5.92		
	37.4	66.40	36.6	20.20	14.90	5.75	7.10

$= \frac{\text{Total Energy}}{\text{Total Photons}} = 0.64$ Mev

0-0.5	6.64×10^{11} Mev/sec-MWt
0.5-1.0	3.66×10^{11}
1.0-1.5	2.02×10^{11}
1.5-2.0	1.49×10^{11}
2.0-2.5	5.75×10^{10}
>2.5	7.10×10^{10}

Nuclide	Ci/MWt	0.0-0.5	0.5-1.0	1.0-1.5	1.5-2.0	2.0-2.5	>2.5
Kr 83	0.0015	0.00127					
85	0.60	0.465					
85	1.21	0.940					
87	1.45					0.00613	0.00056
88	0.18	0.180	0.363		0.408	2.56	
Xe 131	0.69	0.415					
133	5.6	4.82					
133	191.0	57.20					
135	0.0						
135	6.4	5.88	0.038				
138	0.0						
I 131	28.0	59.70	1.51				
132	0.32	0.00635	0.210	0.0051			
133	22.0	49.70	7.30				
134	0.0						
135	4.3	0.462	1.51	17.98	8.38		
	259.0	179.8	10.92	17.98	8.78	2.56	0.00056

$\bar{E} = 0.197$ Mev

0.0-0.5	1.798×10^{12} Mev/sec-MWt
0.5-1.0	1.092×10^{11}
1.0-1.5	1.798×10^{11}
1.5-2.0	8.780×10^{10}
2.0-2.5	2.560×10^{10}
>2.5	5.600×10^{6}

* Historical information - not applicable to current plant design.

TABLE 14.9.5*

BIOLOGICAL DOSE RATE AT THE CENTER OF THE CONTROL ROOM

FLOOR FOLLOWING A LOSS-OF-COOLANT ACCIDENT

1 Hr After the Accident Assuming a 0.635%/day Leak Rate

Contributing Volume of Reactor Building

Energy	El 134'-163'	El 165'-195'	El 195'-234'	El 234'-290'	Total mRem/hr
0.0-0.5	0.000	0.000	0.000	0.0078	0.0078
0.5-1.0	0.197	0.000	0.263	0.380	0.840
1.0-1.5	1.260	0.000	0.360	1.244	2.86
1.5-2.0	3.52	0.000	0.843	2.483	6.85
2.0-2.5	2.73	0.0078	0.516	1.51	4.76
>2.5	6.22	0.0404	1.031	2.91	10.20
Total	13.930	0.048	3.010	8.53	25.52

0.635%/day Leak Rate Total Mixing = 25.5 mRem/hr

24 Hr After the Accident Assuming a 0.635%/day Leak Rate

Contributing Volume of Reactor Building

Energy	El 134'-163'	El 165'-195'	El 195'-234'	El 234'-290'	Total mRem/hr
0.0-0.5	0.000	0.000	0.000	0.0278	0.028
0.5-1.0	0.0156	0.000	0.033	0.114	0.163
1.0-1.5	0.301	0.000	0.349	1.108	1.758
1.5-2.0	0.052	0.000	0.0508	1.488	1.592
2.0-2.5	0.0300	0.0026	0.231	0.671	0.935
>2.5	0.000	0.000	0.000	0.000	0.000
Total	0.400	0.0026	0.664	3.41	4.48

0.635%/day Leak Rate Total Mixing = 4.5 mRem/hr

* Historical information – not applicable to current plant design.

TABLE 14.9.6

INTEGRATED DOSE IN THE CONTROL ROOM
(30 Days Continuous Occupancy)

Post-LOCA Control Room Dose

Radioactive Sources	Post-LOCA CR Dose (Rem TEDE)
Containment Leakage	2.63E-01
ESF Leakage	5.81E-02
MSIV Leakage	4.43E+00
Containment Shine	3.31E-02
External Cloud	1.30E-02
CR Filter Shine	3.09E-03
Total	4.80E+00
Allowable TEDE Limit	5.00E+00

Design Basis Accident Radiological Doses

Control Room Doses

Design Basis Accidents	Control Room Dose (Rem TEDE)
Loss-of-Coolant-Accident	4.80E+00
Fuel Handling Accident	4.30E+00
Roof Scuttles, Personnel Access Doors, and Railroad Bay Doors	4.30E+00
Ground Hatches H17, H18, H33	3.17E+00
Ground Hatches H1, H19	3.75E+00
Ground Hatches H2, H20	4.52E+00
Ground Hatches H21, H22, H34	4.59E+00
Ground Hatches H23, H24, H15, H16	3.58E+00
Control Rod Drop Accident	2.06E+00*
Main Steam 4.0 μCi/g I-131 DE:	2.10E+00
Line Break 0.2 μCi/g I-131 DE:	1.10E-01

* Assumes Reactor Water Sample Lines release.

TABLE 14.9.7

DESIGN BASIS ACCIDENT RADIOLOGICAL DOSES

EXCLUSION AREA BOUNDARY DOSES

Design Basis Accidents		Exclusion Area Boundary Dose (Rem TEDE)
Loss-of-Coolant-Accident		9.04E+00
Fuel Handling Accident		2.99E+00
Roof Scuttles, Personnel Access Doors, and Railroad Bay Doors		2.99E+00
Ground Hatches H17, H18, H33		2.99E+00
Ground Hatches H1, H19		8.04E-01
Ground Hatches H2, H20		7.35E-01
Ground Hatches H21, H22, H34		2.99E+00
Control Rod Drop Accident		2.33E+00*
Main Steam Line Break	4.0 μCi/g I-131 DE:	5.43E+00
	0.2 μCi/g I-131 DE:	2.70E-01

LOW POPULATION ZONE DOSES

Design Basis Accidents		Low Population Zone Dose (Rem TEDE)
Loss-of-Coolant-Accident		9.59E+00
Fuel Handling Accident		4.50E-01
Roof Scuttles, Personnel Access Doors, and Railroad Bay Doors		4.53E-01
Ground Hatches H17, H18, H33		4.53E-01
Ground Hatches H1, H19		4.53E-01
Ground Hatches H2, H20		1.22E-01
Ground Hatches H21, H22, H34		1.11E-01
Ground Hatches H23, H24, H15, H16		4.53E-01
Control Rod Drop Accident		3.90E-01*
Main Steam Line Break	4.0 μCi/g I-131 DE:	8.20E-01
	0.2 μCi/g I-131 DE:	4.00E-02

* Assumes Reactor Water Sample Lines release.

TABLE 14.9.8*

SENSITIVITY OF DOSES TO VARIATION OF ASSUMPTIONS - LOSS-OF-COOLANT ACCIDENT

Assumptions	Section 14.6 Analysis	Section 14.9 Analysis	Factor Affecting Thyroid Dose	Factor Affecting Whole-Body Dose
Fission products released to drywell	1.8% noble gases[1] 0.32% iodines released from 25% of the fuel rods which are assumed to be perforated. 1% of total iodines in organic form. Negligible solids	100% noble gases 50% iodines and 1% solids in total core inventory. 5% of total iodines in organic form	625	220
Iodine retained in water	Based on partition factor of 100 between the volumes of air and water in suppression chamber and drywell	None	12.5[3]	1.0
Elemental iodine plateout in drywell	50%	50%	1.0	1.0
Leakage rate from primary containment	Function of drywell pressure; peaks close to 0.5% volume per day	0.635% per day constant throughout the accident	1.3(2-hr)[2] ~1.0(30-day)	1.3(2-hr)[2] ~1.0(30-day)
Uniform mixing in reactor building	Yes	No	22(2-hr)[2] 3(30-day)	28(2-hr)[2] 3(30-day)
Iodine filter efficiency	99%	90%	10	1.0
Effectiveness of stack	Yes	Yes	1.0	1.0

(1) 1% of iodines released in organic form, which is not reduced by fallout or plateout in the drywell and reactor building. Elemental iodines are carried into suppression pool during blowdown, and a fraction retained according to the assumed equilibrium partition factor of 100. Iodines become airborne in the suppression chamber and drywell before leaking out to the secondary containment.

(2) 2-hr dose is evaluated at 1,040 m; 30-day dose is evaluated at low population zone of 7,300 m.

(3) Takes into account the organic iodine fraction.

* Historical Information – not applicable to current plant design.

TABLE 14.9.8 (Continued)*

SENSITIVITY OF DOSES TO VARIATION OF ASSUMPTIONS - LOSS-OF-COOLANT ACCIDENT

Assumptions	Section 14.6 Analysis	Section 14.9 Analysis	Factor Affecting Thyroid Dose	Factor Affecting Whole-Body Dose
Fission product released to reactor water[(1)]	1.8% noble gases, 0.32% iodines from 111 perforated fuel rods, solids negligible	30% noble gases, 15% iodines from 49 perforated fuel rods[(2)]	13.8	4.9
Iodines retained in water	Equilibrium partition[(3)] factor of 100 for iodines and water	90%[(2)]	0.4	1.0
Plateout of iodines in reactor building	None	50%	0.5	1.0
Uniform mixing in refueling chamber	Yes	No		
	Fission products exponentially released from water to reactor building until exhausted	Fission products exponentially released from reactor building in 2 hr	14(2-hr)[(4)] ~1.3(30-day)	18(2-hr)[(4)] ~1.1(30-day)
Iodine filter efficiency	99%	90%	10	1.0
Effectiveness of stack	Yes	Yes	1.0	1.0

(1) Accident occurs 24 hr after shutdown.
(2) These assumptions are not the same as found in Reference 14.0.6 or in current standard methodology for calculating radiological consequences, however, the net effect of these assumptions provides conservative and bounding results.
(3) Amount of retention depends on the ratio of air space to water space. In this case, the equivalent value of 75% is obtained.
(4) 2-hr dose is evaluated at 1,040 m. 30-day is evaluated at low population zone of 7,300 m.

* Historical information – not applicable to current plant design.

TABLE 14.9.8 (Continued)*

SENSITIVITY OF DOSES TO VARIATION OF ASSUMPTIONS - LOSS-OF-COOLANT ACCIDENT

Assumptions	Section 14.6 Analysis	Section 14.9 Analysis	Factor Affecting Thyroid Dose	Factor Affecting Whole-Body Dose
Fission products released to water	1.8% noble gases, 0.32% iodines from 330 perforated fuel rods Solids negligible	100% noble gases, 50% iodines from 330 perforated fuel rods (2)	156	55
Noble gases carry-over to condenser hotwell	Uniformly mixed with steam, carried over at 5.0% steam flow rate, isolation valve closure at 10.5 sec	100%	1.0	10
Iodine carryover to condenser hotwell	Retention in water[1], uniform mixing in steam dome, carryover at 5.0% steam flow, and isolation at 10.5 sec	10%	2,700	1.0
Iodine plateout in condenser hotwell	None	50%	0.5	1.0

[1] Amount of retention in condenser hotwell water depends on relative ratio of steam space to water space. The base case uses an equilibrium partition factor of 100 and a steam-water space ratio of about 12.

[2] These assumptions are not the same as found in Reference 14.0.6 or in current standard methodology for calculating radiological consequences, however, the net effect of these assumptions provides conservative and bounding results.

* Historical information – not applicable to current plant design.

TABLE 14.9.8 (Continued)*

SENSITIVITY OF DOSES TO VARIATION OF ASSUMPTIONS - LOSS-OF-COOLANT ACCIDENT

			Factor Affecting	
Assumptions	Section 14.6 Analysis	Section 14.9 Analysis	Thyroid Dose	Whole-Body Dose
Steam and Water mass lost in blowdown (10.5-sec closure)	25,800 lb steam 165,120 lb water	190,920 lb	1.0	1.0
Total fission gases released	518 Ci iodines and 20 Ci noble gases[1]	Proportional to operating limit, assumed 2.86 times the base case value	2.86	2.86
Concentration in water and steam	Equilibrium separation	Equilibrium separation	1.0	1.0
Steam cloud rise	No	No	1.0	1.0
Release height	30 m	30 m (with downwash)	10	2.0

[1] Based on fission product concentration in coolant consistent with an off-gas release rate at the stack of 350,000 μCi/sec.

* Historical information – not applicable to current plant design.

Table 14.9.9

AST Radionuclide and Magnitude

Source Term Composition

Radionuclide Composition

Group	Elements
Noble Gases	Xe, Kr
Halogens	I, Br
Alkali Metals	Cs, Rb
Tellurium Group	Te, Sb, Se, Ba, Sr
Noble Metals	Ru, Rh, Pd, Mo, Tc, Co
Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am
Cerium	Ce, Pu, Np

Source Term Magnitude

BWR Core Inventory Fraction
Released Into Containment

Group	Gap Release Phase	Early In-vessel Phase	Total
Noble Gases	0.05	0.95	1.0
Halogens	0.05	0.25	0.3
Alkali Metals	0.05	0.2	0.25
Tellurium Group	0.00	0.05	0.05
Barium, Strontium	0.00	0.02	0.02
Noble Metals	0.00	0.0025	0.0025
Cerium Group	0.00	0.0005	0.0005
Lanthanides	0.00	0.0002	0.0002

TABLE 14.9.10

Parameters And Assumptions Used In
Post-LOCA Radiological Consequences Analysis

Parameter	Value
Reactor power level	4,030 MWt
Equilibrium EOC Core Avg. Exposure	36,471 MWd/MTU
Bounding Fuel Enrichment	3.8 & 4.2 wt% U-235
Drywell Pressure	49.1 psig
Drywell Temperature	305°F
Minimum Drywell air volume	1.59E+5 ft^3
Minimum Suppression Chamber Air Volume	1.277E+5 ft^3
Drywell Plus Suppression Chamber air volume	2.867E+5 ft^3
Drywell Surface Area	3.32E+4 ft^2
Containment analytical leak rate to environment:	
0 - 720 hours	0.70% per day
Reactor building pressure drawdown time	3 minutes
Reactor Building Volume	2.50E+6 ft^3
SGTS Flow Rate	10,500 cfm
Minimum Suppression Pool Water Volume	1.229E5 ft^3
ESF Leakage Rate (2 x allowable ESF Leak Rate)	10 gpm
ESF Leakage Flashing Fraction	0.10
ESF Leakage Maximum Temperature	< 212°F
Long-Term Suppression Pool Water pH	>7.0
Chemical Form of Iodine in ESF Leakage Release	
Elemental Iodine	97%
Organic Iodine	3%
MSIV Leak Rate, 0-720 hrs, 49.1 psig	
All Four Lines	300 scfh
MSIV Failed Line	150 scfh
First Intact Line	150 scfh

TABLE 14.9.10 (Cont.)

Parameter	Value
Aerosol Settling velocity On Main Steam Lines	8.1E-4 meters/sec
Control Room Volume	1.76E+5 ft^3
MCREV System Outside Air Intake Flow (+/-10%)	2,700 cfm
Control Room Isolation Time (assumed)	30 minutes
Elemental Iodine Removal Efficiency for Failed Line, Outboard MSIV to TSV	
0 min. - 2 min.	0
2 min. - 8 hrs.	11.27%
8 - 24 hrs.	14.89%
24 - 48 hrs.	25.00%
48 - 72 hrs.	47.07%
72 - 96 hrs.	71.01%
Elemental Iodine Removal Efficiency for Intact Line, RPV to Outboard MSIV	
0 min. - 2 min.	0
2 min. - 8 hrs.	4.45%
8 - 24 hrs.	5.96%
24 - 48 hrs.	10.40%
48 - 72 hrs.	21.67%
72 - 96 hrs.	38.24%
Elemental Iodine Removal Efficiency for Intact Line, Outboard MSIV to TSV	
0 min. - 2 min.	0
2 min. - 8 hrs.	11.40%
8 - 24 hrs.	15.05%
24 - 48 hrs.	25.25%
48 - 72 hrs.	47.45%
72 - 96 hrs.	71.41%
Unfiltered Air Inleakage Rate Into Control Room (includes ingress/egress leakage of 10 cfm):	
0 to 30 minutes	18,500 cfm
30 minutes to 30 days	500 cfm

TABLE 14.9.10 (Cont.)

Parameter	Value
MCREV System Charcoal and HEPA Filter Efficiencies (includes 1% system bypass) (used in the analysis):	
Elemental Iodine	89%
Organic Iodide	89%
Aerosol (particulate)	98%
Control Room Occupancy Factors:	
0 - 24 hrs	100%
24 - 96 hrs	60%
96 - 720 hrs	40%
Control Room Breathing Rate	3.5E-4 m^3/sec
CR χ/Qs For Containment & ESF Leakage Releases:	
0 – 0.05 hrs	1.18E-3 sec/m^3 Note 1
0.05 - 2 hrs	3.31E-6 sec/m^3
2 - 8 hrs	1.00E-15 sec/m^3
8 - 24 hrs	1.00E-15 sec/m^3
24 - 96 hrs	1.64E-8 sec/m^3
96 - 720 hrs	4.54E-9 sec/m^3
CR χ/Qs For MSIV Leakage Releases:	
0 - 2 hrs	1.18E-3 sec/m^3
2 - 8 hrs	9.08E-4 sec/m^3
8 - 24 hrs	4.14E-4 sec/m^3
24 - 96 hrs	2.90E-4 sec/m^3
96 - 720 hrs	2.26E-4 sec/m^3
EAB χ/Qs For Containment & ESF Leakage Releases:	
0 – 0.05 hrs	9.11E-4 sec/m^3
0.05 – 0.5 hrs	5.30E-5 sec/m^3
0.5 - 720 hrs	9.17E-6 sec/m^3
EAB χ/Qs For MSIV Leakage Releases:	
0 - 720 hrs	9.11E-4 sec/m^3
EAB Breathing Rate	3.5E-4 m^3/sec
LPZ χ/Qs For Containment & ESF Leakage Releases:	
0 – 0.05 hrs	1.38E-4 sec/m^3
0.05 – 0.5 hrs	1.75E-5 sec/m^3

Note 1: Conservative ground-level release used during 3-minute secondary containment drawdown.

TABLE 14.9.10 (Cont.)

Parameter	Value
0.5 - 2 hrs	9.05E-6 sec/m^3
2 - 8 hrs	4.01E-6 sec/m^3
8 - 24 hrs	2.67E-6 sec/m^3
24 - 96 hrs	1.10E-6 sec/m^3
96 - 720 hrs	3.10E-7 sec/m^3
LPZ χ/Qs For MSIV Leakage Releases:	
0 - 2 hrs	1.38E-4 sec/m^3
2 - 8 hrs	5.81E-5 sec/m^3
8 - 24 hrs	3.77E-5 sec/m^3
24 - 96 hrs	1.48E-5 sec/m^3
96 - 720 hrs	4.15E-6 sec/m^3
LPZ Breathing Rates:	
0 - 8 hrs	3.5E-4 m^3/sec
8 - 24 hrs	1.8E-4 m^3/sec
24 - 720 hrs	2.3E-4 m^3/sec

TABLE 14.9.11

Parameters And Assumptions Used In
Fuel Handling Accident Radiological Consequence Analysis

Parameters	Value
Reactor Power level	4,030 MWt
Radial Peaking Factor	1.7
Number of Fuel Pins Damaged (GNF2 with a 85.6 fuel pin bundle)	172
Pool Decontamination Factor (DF):	
Iodine	200
Noble Gas	1
Particulate	Infinity
Release Of Activity From Damaged Fuel By Nuclide:	Percent (%)
Noble Gas (excluding Kr-85)	5%
Kr-85	10%
I-131	8%
Other Halogens	5%
Alkali Metals	12%
Fractions Released to the Environment:	Percent (%)
Organic Iodine	43%
Elemental Iodine	57%
Particulate Iodine	0%
Noble Gases	100%
Alkali Metals	0%
SGTS Filtration	Not credited
Release Duration	2 hours
Fuel Decay Times	
Roof Scuttles, Personnel Access Doors, and Railroad Bay Doors Releases	24 hours
Ground Hatches H15, H16, H17, H18, H1/H19, H23, H24 & H33 Releases	24 hours
Ground Hatch H2/H20 Release	288 hours
Ground Hatches H21, H22 & H34 Releases	312 hours
Control Room Volume	1.76E+5 ft^3
Control Room Ventilation Mode	
Roof Scuttles, Personnel Access Doors, and Railroad Bay Doors Releases	Normal
Ground Hatches H1/H19, H2/H20, H17, H18, H21, H22, H33, H34	CREV
Ground Hatches H23, H24, H15, H16	Normal

TABLE 14.9.11 (continued)	
Parameters	Value
Control Room Ventilation Operation	
MCREV Intake Flow Rate	2,700 cfm
MCREV Unfiltered Inleakage Rate	500 cfm
Normal Ventilation Flow Rate	20,600 cfm
Normal Ventilation Inleakage Rate	1,600 cfm
MCREV Filter Efficiencies:	
Elemental Iodine	89%
Organic Iodine	89%
Particulate	98%
Control Room Occupancy Factors:	
0 - 24 hrs	100%
24 - 96 hrs	60%
96 - 720 hrs	40%
Control Room Breathing Rate	3.5E-4 m^3/sec
CR Atmospheric Dispersion Factors (χ/Qs):	
All releases except ground hatches (sec/m^3)	
0 – 2 hrs	1.90E-03 sec/m^3
2 – 8 hrs	1.33E-03 sec/m^3
8 – 24 hrs	5.96E-04 sec/m^3
24 – 96 hrs	4.18E-04 sec/m^3
96 – 720 hrs	3.27E-04 sec/m^3
Unit 2 Ground Hatch H18 Release (sec/m^3)	
0 – 2 hrs	1.48E-03 sec/m^3
2 – 8 hrs	6.87E-04 sec/m^3
8 – 24 hrs	2.45E-04 sec/m^3
24 – 96 hrs	2.10E-04 sec/m^3
96 – 720 hrs	1.65E-04 sec/m^3
Unit 2 Ground Hatch H1/H19 Release (sec/m^3)	
0 – 2 hrs	1.75E-03 sec/m^3
2 – 8 hrs	9.51E-04 sec/m^3
8 – 24 hrs	3.05E-04 sec/m^3
24 – 96 hrs	2.82E-04 sec/m^3
96 – 720 hrs	2.44E-04 sec/m^3
Unit 3 Ground Hatch H2/H20 Release (sec/m^3)	
0 – 2 hrs	5.59E-03 sec/m^3
2 – 8 hrs	4.61E-03 sec/m^3
8 – 24 hrs	1.63E-03 sec/m^3
24 – 96 hrs	1.55E-03 sec/m^3
96 – 720 hrs	1.34E-03 sec/m^3

TABLE 14.9.11 (continued)	
Parameters	Value
Unit 3 Ground Hatch H21 Release (sec/m^3)	
0 – 2 hrs	6.20E-03 sec/m^3
2 – 8 hrs	5.35E-03 sec/m^3
8 – 24 hrs	2.14E-03 sec/m^3
24 – 96 hrs	1.84E-03 sec/m^3
96 – 720 hrs	1.61E-03 sec/m^3
Unit 3 Ground Hatch H23 Release (sec/m^3)	
0 – 2 hrs	1.58E-03 sec/m^3
2 – 8 hrs	1.42E-03 sec/m^3
8 – 24 hrs	6.25E-04 sec/m^3
24 – 96 hrs	4.41E-04 sec/m^3
96 – 720 hrs	3.86E-04 sec/m^3
EAB Atmospheric Dispersion Factors (χ/Qs):	
0 – 720 hrs	9.11E-04 sec/m^3
LPZ Atmospheric Dispersion Factors (χ/Qs):	
0 – 2 hrs	1.38E-04 sec/m^3
2 – 8 hrs	5.81E-05 sec/m^3
8 – 24 hrs	3.77E-05 sec/m^3
24 – 96 hrs	1.48E-05 sec/m^3
96 – 720 hrs	4.15E-06 sec/m^3
Dose Conversion Factors	FGR 11 & 12

TABLE 14.9.12

Parameters And Assumptions Used In
Control Rod Drop Accident Radiological Consequence Analysis

Parameters	Value
Reactor Power level	4,030 MWt
Radial Peaking Factor	1.7
Number Failed Fuel Rods (Bounding case for 10x10 bundle type)	1200
Melted Fuel Rods (% of Failed Fuel Rods) Unit 2: Unit 3:	5.0% 0.77%
Fission Product Release Fractions:	
Failed Fuel Rods:	
Noble Gas	10%
Iodine	10%
Alkali Metals	12%
Melted Fuel Rods:	
Noble Gas	100%
Iodine	50%
Alkali Metals	25%
Remaining Fission Products	RG 1.183 Table 1, Early-In-Vessel Fraction
Fission Product Fraction Transfer To Condenser:	
Noble Gas	100%
Iodine	10%
Remaining Fission Products	1%
Fission Product Fraction Available for Condenser Release:	
Noble Gas	100%
Iodine	10%
Remaining Fission Products	1%
Condenser Release Rate (volume%/day)	1
Gland Seal Condenser (Extraction Steam) Flow Rate	0.15% of Main Steam Flow Rate (18,920 lbs/hr)
Offgas System Charcoal Delay Beds Hold-up Time:	
Krypton	34 hours
Xenon	410 hours
Duration of Release	24 hours
Control Room Parameters:	
Control Room Volume	1.76E+5 ft^3
MCREV Operation	Not credited

TABLE 14.9.12 (Cont)

Parameters	Value
Control Room Normal Intake Flow Rate	20,600 cfm
Assumed Unfiltered Inleakage Rate	500 cfm
Control Room Occupancy Factors:	
0 - 24 hrs	100%
24 -96 hrs	60%
96 - 720 hrs	40%
Control Room Breathing Rate	3.5E-4 m^3/sec
CR χ/Qs for Condenser Release	
0 - 2 hrs	1.18E-3 sec/m^3
2 – 8 hrs	9.08E-4 sec/m^3
8 – 24 hrs	4.14E-4 sec/m^3
24 – 96 hrs	2.90E-4 sec/m^3
96 – 720 hrs	2.26E-4 sec/m^3
EAB χ/Qs for Condenser Release	
0 - 720 hrs	9.11E-4 sec/m^3
LPZ χ/Qs for Condenser Release	
0 - 2 hrs	1.38E-4 sec/m^3
2 – 8 hrs	5.81E-5 sec/m^3
8 – 24 hrs	3.77E-5 sec/m^3
24 – 96 hrs	1.48E-5 sec/m^3
96 – 720 hrs	4.15E-6 sec/m^3
Dose Conversion Factors	FGR 11 & 12

14.10 ANALYIS OF THE CONTAINMENT RESPONSE

14.10.1 Methodology

The analyses of containment pressure and temperature responses for design basis accident were performed at a power level of 102% of EPU rated thermal power (RTP) (4030 MWt) in accordance with NEDC-32424P-A, Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate (ELTR1)(Reference 1) using GEH codes and models. **Analyses performed at 102% of EPU RTP satisfy the requirements for a MUR uprate of 100.35% MUR RTP and therefore were not reperformed for the MUR Uprate.**.

The LAMB code models the recirculation loop as a separate pressure node. It also allows for inclusion of flashing in the pipe and vessel during the blowdown and flow choking at the jet pump nozzles when the conditions warrant. The use of the LAMB blowdown flow in M3CPT was identified in ELTR1 by reference to the LAMB code qualification in Reference 4.

The SHEX code was used to model the long-term containment pressure and temperature response. The key models in SHEX are based on models described in Reference 3. The GEH containment analysis methodologies have been applied to all BWR power uprate projects performed by GEH and accepted by the NRC.

Original long-term containment analyses did not credit passive heat sinks in the DW, Torus airspace, and suppression pool. This conservative assumption was identified to the NRC as Assumption 6 of Attachment 1 to the March 12, 1993, GE letter referenced in Reference 5. Long-term containment analyses performed for PBAPS now includes credit for these passive heat sinks. This was identified as a change in methodology in Reference 6. These long-term containment analyses continue to conservatively neglect any heat loss from the containment through the containment walls to the reactor building or environs (Assumption 8 of the same GE letter, Reference 5).

The metal-water reaction energy versus time relationship is calculated using the method described in NRC Regulatory Guide 1.7 (Reference 7) as a normalized value (fraction of reactor thermal power). All of the energy from the metal-water reaction is assumed transferred to the reactor coolant in the first 120 seconds into the LOCA. The metal-water reaction energy represents a very small fraction of the total shutdown energy transferred to the coolant. A summary of the codes, version and NRC approval is provided below:

Computer Code	Version	NRC Approved	Comments
SHEX	06	Yes	Note 2

M3CPT	05	Yes	NEDO-10320, 04/71 (NUREG-0661)
LAMB	08	Note 1	NEDE-20566-P-A, 09/86

Note 1: The LAMB code is approved for use in ECCS-LOCA applications (NEDE-20566-P-A), but no approving SER exists for the use of LAMB in the evaluation of RIPDs or containment system response. The use of LAMB for these applications is consistent with the model description of NEDE-20566-P-A.

Note 2: The application of the methodology in the SHEX code to the containment response is approved by the NRC in the letter to G. L. Sozzi (GE) from A. Thadani (NRC), "Use of the SHEX Computer Program and ANSI/ANS 5.1-1979 Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," July 13, 1993 (Reference 8).

14.10.2 Short-Term Containment Pressure Response

The short-term containment response analysis was performed for the limiting DBA LOCA that assumes a double-ended guillotine break of a recirculation suction line to demonstrate that operation at RTP does not result in exceeding the containment design limits. The short-term analysis covers the blowdown period during which the maximum DW pressure and Torus pressure occur. The analysis was performed at 102% of EPU RTP (4030 MWt) **which satisfies the requirements for the MUR Uprate**. The time-dependent results of the limiting short-term analysis are presented in Figures 14.10.1 through 14.10.6 and are summarized in Table 14.10.2. The maximum calculated containment pressure remains within the design value. The short-term analysis was performed for three different initial containment conditions. The Design case (D) considers the most limiting initial containment conditions of 70°F in the DW and 2.5 psig in the DW and Torus. The Bounding case (B) considers initial containment conditions of 125°F (two standard deviations below the PBAPS five year average temperature) in the DW and 2.0 psig (two standard deviations above the PBAPS five year average pressure) in the DW and Torus, bounding normal operation. It is conservative to use 125°F as the initial drywell gas temperature instead of 145°F because it leads to a higher peak drywell temperature in the analysis used for equipment qualification evaluation. A lower initial drywell gas temperature in the analysis results in more initial non-condensable gas in the drywell leading to a higher pressure and accordingly higher peak temperature for the small steam line break analysis.

The PBAPS short-term RSLB containment temperature and pressure responses are affected by the change in enthalpy as a result of MELLLA+ operating domain expansion. Short term RSLB containment analyses were performed at the power / flow points which define the MELLA+ operating domain (102% RTP/83% rated core flow and

78.8% RTP/55% rated core flow). Note flow is the more limiting point of the MELLLA+ domain. The results demonstrated that peak drywell pressures and temperatures from the short-term RSLB for the MELLLA+ operating domain are bounded by the results presented in Table 14.10.2 (Reference 15).

A Reference Case (R) is also evaluated that assumes initial conditions of 145°F in the DW and 0.75 psig in the DW and Torus (initial conditions used in the PBAPS power rerate analysis, Reference 8). The Design case (D) was also performed at 3514 MWt (rerate) conditions to provide comparison for evaluating the impact of operation at EPU 4030 MWt. The key parameters used to model and analyze the plant response at RTP are provided in Table 14.10.1

14.10.3 <u>Drywell and Suppression Pool Temperature</u>

The bounding DW temperature occurs during a Small Steam Line Break (SSLB). A spectrum of steam line break sizes have been evaluated to ensure a bounding DW EQ temperature profile is established (Figure 14.10.7). The analysis has been performed in accordance with NUREG-0588 (Reference 9), and the most limiting DW temperature from this analysis is shown in Table 14.10.2. Although the DW environment may see temperatures as high as 340°F, the most limiting temperature for the DW shell has been analyzed to be within the design temperature of 281°F.

This analysis includes the impact of a LOCA signal from both the accident unit and the non-accident unit as discussed in Section 14.10.4 below. In order to encompass the timing of LOCA signal generation for the spectrum of SSLBs, Suppression pool cooling (SPC) is interrupted three separate times. For the larger SSLB break sizes, a LOCA signal is assumed to occur at 10 minutes just as operators are initiating Drywell Sprays. This delays Drywell Sprays for an additional 10 minutes which provides the worst case assumption for the DW air space and shell temperatures. To account for intermediate SSLB break sizes, the analysis assumes a LOCA signal is received after initiation of drywell sprays as reactor pressure reaches 450 psig either due to the break or due to operators depressurizing the reactor at a cooldown rate of 100°F/hr. This assumption results in the worst case reheat of the drywell air space and shell temperatures. Finally, to account for the smallest SSLB break sizes, the LOCA signal is assumed to occur 10 minutes prior to the peak Suppression Pool temperature. This results in the worst case for Suppression Pool temperature.

The torus gas space peak temperature response was calculated assuming a heat and mass transfer model between the pool and torus gas space that is calculated mechanistically. Table

14.10.2 shows the calculated peak torus gas space temperature for the DBLOCA of 184°F with consideration of dual unit interaction. The torus gas temperatures are bounded by the torus design temperature of 281°F.

14.10.4 Long-Term Bulk Suppression Pool Temperature Response-Design Basis Accidents

The long-term bulk pool temperature response at RTP is evaluated for the limiting DBLOCA originally analyzed in Section 14.6.3.3 (Case C). This DBLOCA is an instantaneous guillotine break of the recirculation loop suction line (RSLB). Small break LOCAs and special events were also analyzed at RTP conditions.

The GE Safety Communication SC 06-01 (Reference 10) identified the potential that a single failure that eliminated only the RHR heat exchanger could prove more limiting than the typically analyzed scenario of the single failure of an entire AC electrical power source. The PBAPS RHR system is configured with 2 loops of RHR, with each loop having its own separate injection point to the RPV, and with each loop having its own separate return to the suppression pool. Each loop is comprised of 2 RHR pumps, each pump having its own separate suction from the suppression pool, and with each pump having its own separate heat exchanger on its discharge. This configuration is such that GE Safety Communication SC 06-01 has been determined to not be applicable to PBAPS.

As stated in Appendix F, design consideration has been given to the possible effects of interaction of Units 2 and 3 on the diesel generators. For the postulated case of DBA conditions for one unit, including the total loss of offsite power (LOOP), it has been recognized that a loss of DW cooling to the non-accident unit could eventually result in High Drywell Pressure (HDWP) for that unit. The ECCS logic at PBAPS requires HDWP coincident with low reactor pressure (LRPVP) to initiate LPCI injection. Therefore, the HDWP will result in the generation of a LOCA signal only when the non-accident unit is depressurized below 450 psig. This design feature allows the operator to control the non-accident unit and take preemptive actions as described below so that automatic actuations do not result in overloading of the shared EDGs.

The non-accident unit LOCA signal will result in the start and alignment of the ECCS pumps to the LPCI mode of operation. As stated Appendix F, in order to prevent overloading of the EDGs the RHR pumps are electrically interlocked between units at the breaker level to preclude the operation of the RHR pumps of both units on one diesel generator. This logic could result in tripping RHR pumps on the accident unit; therefore operators are

instructed to secure RHR and HPSW pumps on the accident unit, that are not required for adequate core cooling, prior to depressurizing the non-accident unit below 500 psig in Abnormal Operating Procedures (AOPs). After the non-accident unit is depressurized below 450 psig and the ECCS pumps start, adequate core cooling is verified, unnecessary pumps are secured and containment cooling is restored on the accident unit.

The analysis of the above scenario is referred herein as the dual unit interaction, This analysis was performed as described for the single unit events except containment cooling is assumed interrupted due to dual unit interaction when the accident unit suppression pool temperature is 10°F below the peak suppression pool temperature that would be experienced by the accident unit if there was no containment cooling interruption due to dual unit interaction. The depressurization of the non-accident unit is controlled and the operators are instructed to verify a margin of 10°F exists, on the accident unit, to the NPSH curves in the EOPs prior to depressurizing the non-accident unit below 450 psig so the effect on the accident unit is acceptable and doesn't result in loss of NPSH for the ECCS pumps.

The accident unit containment cooling interruption is assumed for a period of 10-minutes due to the dual unit interaction. After the 10-minute interruption, containment cooling on the accident unit is restored with the same containment cooling configuration as existed prior to the dual unit interaction interruption.

Long term suppression pool temperature response is not affected by operation in the MELLLA+ operating domain (Reference 15).

14.10.4.1 Suppression Pool Temperature Response - DBLOCA

The analysis of the RSLB DBLOCA was performed at 102% EPU rated power (4030 MWt) **which satisfies the requirements for the MUR Uprate**. The calculated SP and torus (WW) temperature responses are presented in Figures14.10.8 and 14.10.8a, the DW and torus (WW) temperature responses are presented in Figures14.10.9 and 14.10.9a, and the peak values are provided in Table 14.10.2. The analysis was performed using a decay heat table based on ANS/ANSI 5.1-1979 with 2-sigma adders with additional actinides and activation products per GE SIL 636 (Reference 11). No modifications were made to this standard. The analysis assumed the worst-case single active failure and concurrent loss of off-site power. For the first 10 minutes of the DBLOCA, there is no operator action. As such, CS (one loop with two pumps) and RHR (one loop with two pumps and another loop with only one pump) automatically align to provide cooling to the fuel by either spray (CS) or flooding (LPCI). With a LOOP and a single active failure (SAF) of one EDG, a third CS pump in the other loop is

expected to initiate as well, however, operators are assumed to stop this pump when they take manual control. Ten minutes after the event initiation, the operators take manual control and align one loop of RHR (one RHR pump, one RHR heat exchanger and one HPSW pump providing cooling flow to the RHR heat exchanger) to provide containment cooling with a flow rate of 8600 gpm. All three modes of containment cooling were evaluated, suppression pool cooling (SPC), containment spray cooling (CSC) and coolant injection cooling (CIC). The other two RHR pumps are manually turned off, and CS flow maintains core cooling. At one hour after event initiation, the RHR heat exchanger cross-tie is placed in service which results in a containment cooling configuration of 1 RHR pump with flow split between each of the two RHR heat exchangers (total RHR flow of 8600 gpm). The RHR heat exchanger cross-tie results in a significant increase in the capacity of a single RHR pump to cool the containment and suppression pool. The resulting calculated peak bulk suppression pool temperature at 4030 MWt for the DBLOCA is 186°F.

The dual unit interaction analysis, for the RSLB DBLOCA was performed as described above for the single unit RSLB DBLOCA except for the interruption of containment cooling due to the non-accident unit LOCA signal. The resulting calculated peak bulk suppression pool temperature at 4030 MWt for the RSLB DBLOCA with dual unit interaction is 187.2°F.

14.10.4.2 Suppression Pool Temperature Response - Small Steam Break LOCA

For a large DBLOCA, sufficient flow from the break can be established to provide long-term core cooling to achieve cold shutdown. For a small break LOCA, such as the SSLB, the Alternate Shutdown Cooling (ASDC) method may be required to achieve cold shutdown within 36 hours following initiation of the event. Therefore, the SSLB analyses performed for EQ profiles in the DW are also used to evaluate suppression pool temperature for small LOCA events, and to confirm the ability to achieve cold shutdown within 36 hours following initiation of the event.

For this small LOCA analysis, a spectrum of SSLBs is evaluated. Initial reactor conditions are consistent with operation at 100.35% of RTP, and the same decay heat, relaxation and metal-water reaction energies are assumed as is used for the large DBLOCA analysis. Consistent with the large DBLOCA assumptions, a complete LOOP and a worst-case single failure are also assumed for this analysis. For smaller steam breaks, HPCI may be available for vessel makeup, but is not credited for 10 minutes to ensure a bounding DW temperature is evaluated. At 10 minutes, operators turn off two of the RHR pumps, and align the remaining RHR pump to provide containment cooling with a flow of 8600 gpm

through one RHR heat exchanger, and one HPSW pump providing cooling flow to the RHR heat exchanger. When DW pressure exceeds 2.0 psig, operators initiate torus spray, with the remainder of the RHR flow remaining in SPC mode. Prior to torus pressure exceeding 9.0 psig, operators initiate DW spray and stop all SPC flow. When bulk suppression pool temperature exceeds 110°F (but not earlier than 10 minutes following initiation of the event), operators initiate a controlled reactor vessel cooldown at 100°F per hour. At one hour from initiation of the event, operators establish the RHR heat exchanger cross-tie to the other RHR heat exchanger in the same loop, such that a total RHR flow rate of 8600 gpm is maintained to the DW and torus spray headers. When reactor pressure is decreased below the pressure permissive for Normal Shutdown Cooling (NSDC) (70 psig), operators maintain DW and Torus spray cooling with the RHR heat exchanger cross-tie in service, and maintain reactor vessel pressure as low as possible to limit steam flow from the break. When bulk suppression pool temperature is below a pre-determined value, based on NPSH requirements, operators open one or more ADS valves and increase vessel water level in the reactor vessel to the MSL nozzles using the one loop of CS until water flows from the open ADS valves back to the suppression pool, establishing ASDC. DW and torus sprays continue to be used to provide containment and SPC, bulk reactor water temperature decreases below 200°F and cold shutdown is achieved prior to 36 hours from initiation of the event, which is conservative for plants with cold shutdown defined at a higher temperature. The resulting time-dependent bulk suppression pool temperature response is presented in Figure 14.10.10 and the peak bulk suppression pool temperature at 4030 MWt is 187°F.

The SSLB analysis was also performed for dual unit interaction, as described above for the single unit analysis except as noted in the following paragraphs. Only the limiting (0.01 sqft break size), with respect to peak suppression pool temperature, SSLB LOCA was analyzed for dual unit interaction.

When bulk suppression pool temperature exceeds 110°F (but not earlier than 10 minutes following initiation of the event), operators initiate a controlled reactor vessel cooldown at 100°F per hour. At one hour from initiation of the event, operators establish the RHR heat exchanger cross-tie to the other RHR heat exchanger in the same loop, such that a total RHR flow rate of 8600 gpm is maintained to the DW and torus spray headers. At one-hour following the start of reactor depressurization it is assumed that a LOCA signal on the SSLB LOCA unit may occur on HDWP commensurate with low reactor pressure. This timing for the HDWP/low reactor pressure LOCA signal on the small break LOCA/accident unit is based on the cooldown rate of 100°F/hr mentioned above. This LOCA signal will stop all HPSW flow, start all available low-pressure ECCS pumps, and realign the RHR pumps

to LPCI mode in the SSLB LOCA/accident unit. The analysis assumes this interruption in SPC continues for 10 minutes, after which operators are assumed to stop the additional low-pressure ECCS pumps (two RHR and unnecessary CS pumps), restart the HPSW flow to both RHR heat exchangers in the remaining one RHR loop, and reestablish SPC or CSC as applicable.

A second interruption of containment cooling on the accident unit is assumed due to dual unit interaction when the accident unit suppression pool temperature is 10°F below the peak suppression pool temperature that would be experienced by the accident unit if there was no containment cooling interruption due to dual unit interaction. The accident unit cooling interruption is assumed for a period of ten minutes due to the dual unit interaction. After the ten minute interruption, containment cooling on the accident unit is restored with the same containment cooling configuration as existed prior to the dual unit interaction interruption. The remainder of the transient proceeds as analyzed for the single unit. The resulting time-dependent bulk suppression pool temperature response is presented in Figure 14.10.10a and the peak bulk suppression pool temperature at 100.35% of MUR RTP is 187.6°F.

14.10.4.3 <u>Suppression Pool Temperature Response - Non-Accident Unit</u>

Evaluation of the containment response for the non-accident unit was also evaluated based on the conservative assumption that the RSLB DBLOCA occurs concurrently with a LOOP, resulting in reactor isolation and scram on the non-accident unit. It is conservatively assumed that the SAF during the RSLB DBLOCA is loss of an EDG. This could result in insufficient electrical capacity to place in service two HPSW pumps and thereby initiate the RHR heat exchanger cross-tie in the non-accident unit. In addition, loss of a specific EDG will result in the inability to place in service the NSDC mode of the RHR system in order to achieve cold shutdown of the non-accident unit. The capability of the non-accident unit to achieve cold shutdown within 36 hours was analyzed at 4030 MWt and decay heat based on ANS/ANSI 5.1-1979 with 2-sigma adders with additional actinides and activation products per GE SIL 636 (Reference 11). HPCI is conservatively assumed available and provides reactor inventory makeup until SP temperature reaches 180°F. If SP temperature reaches 180°F, HPCI is secured because HPCI availability cannot be assured with SP temperature greater than 180°F. No credit is assumed for CST volume. DW cooling fans in the non-accident unit are assumed as unavailable. When non accident unit SP temperature reaches 110°F, but no sooner than ten minutes after reactor shutdown; the operators commence manual reactor depressurization and reactor cooldown at a rate of 100°F/hr. At one-hour after reactor

shutdown, the operators align one loop of RHR (one RHR pump, one RHR heat exchanger and one HPSW pump providing cooling flow to the RHR heat exchanger) in SPC mode with a flow rate of 8600 gpm.

At one-hour following the start of reactor depressurization, it is assumed that a LOCA signal (on the non-accident unit) may occur on HDWP commensurate with low reactor pressure. This assumed timing for the LOCA signal on the non-accident unit is based on a cooldown rate of 100°F/hr. This LOCA signal will stop all HPSW flow, start all available low-pressure ECCS pumps, and realign the RHR pumps to LPCI mode in the non-accident unit.

The analysis assumes this operation continues for 10 minutes, after which operators are assumed to stop all but one RHR pump (if SP temperature is below 180°F, HPCI will also be available), restart the HPSW flow to one RHR heat exchanger in one RHR loop, and reestablish SPC. A second interruption of containment cooling on the non-accident unit is assumed due to dual unit interaction, as described above, when the non-accident unit suppression pool temperature is 10°F below the peak suppression pool temperature that would be experienced by the non-accident unit if there was no containment cooling interruption due to dual unit interaction. Note that this assumption of a second containment cooling interruption is conservative for the large pipe break DBLOCA because, for the DBLOCA, the LOCA signal on the accident unit will occur within the first minute of the accident progression. This second containment cooling interruption for the non-accident unit was assumed in order to provide a bounding analysis of the non-accident unit for all events in which the RHR heat exchanger cross-tie modification is not available.

The non-accident unit containment cooling interruption is assumed for a period of 10-minutes due to the dual unit interaction. After the 10-minute interruption, containment cooling on the non-accident unit is restored with the same containment cooling configuration as existed prior to the dual unit interaction interruption. When RPV pressure reaches 150 psig, the analysis assumes that the operators will maintain the RPV at this pressure. Containment cooling is maintained using RHR SPC mode. The containment spray mode of RHR may be initiated to maintain the containment shell temperature less than the containment design temperature of 281°F. When bulk suppression pool temperature is below a pre-determined value, based on NPSH requirements, operators are assumed to flood the RPV to the level of the MSLs, open the ADS valves to completely depressurize the RPV, and initiate ASDC in the CIC mode. Cooldown of the RPV to cold shutdown conditions on the non-accident unit is accomplished with ASDC. Cold shutdown is achieved when bulk reactor liquid water temperature is below 200°F.

The peak bulk SPC temperature for this analysis is 203.8°F, and the time to achieve cold shutdown was 35.1 hours. The resulting time-dependent bulk suppression pool temperature response is presented in Figure 14.10.11. The results of this evaluation of the second PBAPS (non-accident) unit response is applicable and bounding for small break LOCAs and other accidents/events on the other PBAPS unit because the scenario includes a bounding dual unit interaction and uses the minimum equipment (no RHR Cross-tie) to reach cold shutdown.

14.10.4.4 Suppression Pool Temperature Response – Loss of RHR Normal Shutdown Cooling Function Event

Analyses were also performed to confirm the ability of PBAPS to reach cold shutdown conditions within 36 hours, considering the inability to cool down the reactor assuming loss of the Normal Shut Down Cooling (NSDC) mode of the RHR system. This method to achieve cold shutdown is termed Alternate Shut Down Cooling (ASDC). ASDC can be established in various ways. Because the method of CIC mode results in the greatest heat addition to the suppression pool, only CIC mode was evaluated in this analysis. This method establishes ASDC using a single RHR pump aligned in CIC mode to both flood the vessel and to provide containment cooling, which results in the most limiting peak bulk suppression pool temperature.

The suppression pool temperature response for the analysis of the loss of normal RHR shutdown cooling function event with cold shutdown achieved by ASDC also represents the bounding bulk suppression pool temperature response for a small liquid line break LOCA wherein SPC mode is used in lieu of CSC mode. The analysis results provide a peak bulk suppression pool temperature for the small liquid line break LOCA of 186°F when the accident unit was treated singularly and 186.7°F when additional containment cooling interruption is considered due to dual unit interaction. The capability of the ASDC method to achieve cold shutdown within 36 hours was analyzed at 4030 MWt and decay heat based on ANS/ANSI 5.1-1979 with 2-sigma adders with additional actinides and activation products per GE SIL 636 (Reference 11). Reactor shutdown is assumed initiated by a LOOP with concurrent loss of one division power. The loss of one division power prevents the use of the NSDC mode of the RHR system. HPCI is conservatively assumed available and provides reactor inventory makeup until reactor pressure decreases below the HPCI isolation pressure, after which low-pressure ECCS provides reactor inventory makeup. If HPCI is not available, ADS would be used to rapidly reduce reactor pressure to allow low-pressure ECCS to provide vessel makeup. Such use of ADS results in a faster heatup of the suppression pool. The total (integrated) decay heat to the pool at the time of peak pool temperature is less for

the fast pool heatup. In addition, the heat removed from the pool is greater for the faster pool heatup. Thus, a faster pool heatup results in a lower peak pool temperature. For this reason, the assumption of crediting the HPCI as available is conservative. No credit is assumed for CST volume. When suppression pool temperature reaches 110°F, but no sooner than ten minutes following initiation of the event, the operators commence manual reactor depressurization and reactor cooldown at a rate of 100°F/hr. At ten minutes after initiation of the event, the operators align one loop of RHR (one RHR pump, one RHR heat exchanger and one HPSW pump providing cooling flow to the RHR heat exchanger) in SPC mode. At one hour after event initiation, the RHR heat exchanger cross-tie is placed in service, wherein the RHR flow from the single RHR pump is split, with half of the pump discharge directed to a second RHR heat exchanger.

When RPV pressure reaches the NSDC RPV pressure permissive of 70 psig, the analysis assumes that the operators will maintain the RPV at this pressure. Containment cooling is maintained using SPC mode.

Just before suppression pool temperature peaks (ten minutes is assumed in the analysis), it is conservatively assumed that a LOCA signal may occur on HDWP commensurate with low reactor pressure. This LOCA signal will stop all HPSW flow, start all available low-pressure ECCS pumps, and realign the RHR pumps to LPCI mode. The analysis assumes this operation continues for 10 minutes, after which operators are assumed to stop the additional low-pressure ECCS pumps (two RHR and all CS pumps), restart the HPSW flow to both RHR heat exchangers in the remaining one RHR loop, and reestablish SPC.

When bulk Suppression Pool temperature is below a pre-determined value, based on NPSH requirements, operators are assumed to choose to attempt NSDC. The analysis assumes operators take 30 minutes to attempt NSDC, after which time they determine NSDC cannot be accomplished. Another five minutes is conservatively assumed to realign and establish containment cooling using either SPC or CIC mode. Because the CIC mode results in a more limiting Suppression Pool temperature response, this mode was assumed in the analysis.

Operators then initiate action to establish ASDC by opening the ADS valves and allowing reactor vessel water level to increase above the elevation of the MSL nozzles, flooding the MSLs, and flowing out the ADS valves (liquid flow) and returning to the suppression pool. Cold shutdown is achieved when bulk reactor liquid water temperature is below 200°F. The peak bulk SPC temperature for this analysis at 4030 MWt conditions is 186°F,

and the time to achieve cold shutdown is 16 hours. The resulting time dependent bulk suppression pool temperature response is presented in Figure 14.10.12.

Accomplishing ASDC using either SPC or CSC modes would be expected to take slightly longer to achieve cold shutdown. However, the time to achieve cold shutdown assuming CIC is significantly earlier than the acceptance limit time of 36 hours. Therefore additional analysis runs using either SPC or CSC modes were considered unnecessary.

The dual-unit interaction scenario was analyzed as described above for the single unit event, except as follows. At one-hour following the start of reactor depressurization, it is assumed that a LOCA signal (on the Loss of NSDC Event unit) may occur on HDWP commensurate with low reactor pressure. This assumed timing for the LOCA signal on the Loss of NSDC Event unit is based on a depressurization and reactor cooldown at a rate of 100°F/hr. This LOCA signal will stop all HPSW flow, start all available low-pressure ECCS pumps, and realign the RHR pumps to LPCI mode in the Loss of NSDC Event unit. The analysis assumes this operation continues for 10 minutes, after which operators are assumed to stop the additional low-pressure ECCS pumps (two RHR and all CS pumps), restart the HPSW flow to both RHR heat exchangers in the remaining one RHR loop, and reestablish SPC. A second interruption of containment cooling on the Loss of NSDC Event unit is assumed due to dual unit interaction when the Loss of NSDC Event unit suppression pool temperature is 10°F below the peak suppression pool temperature that would be experienced by the Loss of NSDC Event unit if there was no containment cooling interruption due to dual unit interaction. The depressurization of the Second PBAPS unit is controlled and the operators are instructed to verify a margin of 10°F exists, on the Loss of NSDC unit, to the NPSH curves in the EOPs prior to depressurizing the Second PBAPS unit below 450 psig so the effect on the Loss of NSDC unit is acceptable and doesn't result in loss of NPSH for the ECCS pumps.

The peak bulk SPC temperature for this analysis at 4030 MWt conditions is 186.7°F, and the time to achieve cold shutdown was 16 hours. The resulting time-dependent bulk suppression pool temperature response is presented in Figure 14.10.12a.

14.10.5 Long-Term Bulk Suppression Pool Temperature Response - Special Events

14.10.5.1 Station Blackout

The containment response to a Station Blackout (SBO) was evaluated at 100.05% MUR RTP (4018 MWt). The evaluation used the

NRC approved method of NUMARC 87-00 (Reference 12) and NRC Regulatory Guide 1.155 (Reference 13). Peach Bottom is evaluated as an "Alternate AC Approach" plant per Reference 12. This entails a short period of time (up to one hour) where the plant relies on available process steam, DC power and compressed air to maintain reactor level and pressure. Once alternate power is available the plant transitions to the Alternate AC state to provide decay heat removal until off-site or emergency power becomes available. The containment response was analyzed using the SHEX analysis code as was utilized for the DBA responses. Nominal values were used as inputs to the analysis rather than Technical Specification limits. The specific values are shown in Table 14.10.3.

The SBO scenario is based on a LOOP which causes turbine trip and reactor scram on both units. HPCI and RCIC are the only available sources of makeup to maintain reactor water level. HPCI and RCIC start on low reactor level and take suction from the Condensate Storage Tank (CST). The analysis assumes that at 30 minutes operators secure HPCI and continue RCIC operation to maintain reactor level. In addition, at 30 minutes operators begin vessel depressurization at the maximum allowed cooldown rate of 100°F/hr. At one hour, the alternate AC source (Conowingo line) is available and 30 minutes later operators place one RHR loop in SPC. The RHR configuration is one RHR pump, one RHR heat exchanger and one HPSW pump. The alternate source has the capacity to power this RHR/HPSW configuration plus battery chargers and other small essential loads on both units. Operators continue depressurization until reactor pressure reaches 150 psig. Reactor pressure is maintained at 150 psig for approximately 30 minutes, then RCIC is secured and operators commence entry into ASDC using the RHR pump which is in service. The containment analysis includes the effect of a LOCA signal initiated when each reactor is depressurized to 450 psig coincident with high drywell pressure. This results in two ten minute interruptions in SPC. The peak bulk SPC temperature for this analysis at 4018 MWt conditions is 200°F and the peak drywell pressure is 40.47 psig.

In addition to the containment response, CST inventory, battery capacity and compressed gas capacity were evaluated to verify the response to a SBO.

<u>Condensate Inventory for Decay Heat Removal</u>

Analyses have shown that the PBAPS Condensate inventory is adequate to meet the SBO coping requirement for RTP conditions. The current CST inventory reserve for RCIC and HPCI use ensures that adequate water volume is available to remove decay heat,

depressurize the reactor and maintain reactor vessel level above the Top of Active Fuel (TAF) during the coping period.

Class 1E Battery Capacity

Evaluation of the PBAPS Class 1E Battery Capacity has shown that PBAPS has adequate battery capacity to support decay heat removal during a SBO for the required coping duration. The battery capacity remains adequate to support required coping equipment operation at RTP.

Compressed Gas Capacity

PBAPS meets the requirement for compressed gas capacity. An evaluation has shown that the PBAPS air operated SRVs required for decay heat removal have sufficient compressed gas capacity for the required automatic and manual operation during the SBO event at RTP conditions. Sufficient capacity remains to perform emergency RPV depressurization in case it is required. Adequate compressed gas capacity exists to support the SRV actuations because the maximum number of SRV valve operations is less than the capacity of the pneumatic supply.

14.10.5.2 Appendix R Fire Safe Shutdown

The limiting Appendix R fire events were analyzed at RTP conditions. The fuel heatup analysis was performed using the SAFER/GESTR-LOCA analysis model. The containment analysis was performed using the SHEX model. This evaluation determined the effect on fuel cladding integrity, reactor vessel integrity, and containment integrity as a result of the Appendix R fire event.

Four shutdown methods defined in the PBAPS Fire Protection Program were reanalyzed at RTP conditions. These shutdown methods are described below:

1. Shutdown Method “A”: Utilizes RCIC, two SRVs and one RHR pump (in LPCI, pool cooling and ASDC modes) to achieve the plant shutdown.
2. Shutdown Method “B”: Utilizes HPCI, two SRVs and one RHR pump (in LPCI, pool cooling and ASDC modes) to achieve the plant shutdown.
3. Shutdown Method “C”: Utilizes manual control of three SRVs of the ADS for the depressurization of RPV, along with either, one CS pump and one RHR pump in pool cooling and ASDC modes, or one RHR pump in LPCI, pool cooling and ASDC modes.
4. Shutdown Method “D”: Utilizes HPCI, one SRV and one RHR pump (in LPCI, pool cooling and ASDC modes) to achieve the plant shutdown at alternative control station.

The bounding peak cladding temperature (PCT) for PBAPS occurs when utilizing shutdown Method "C" with one RHR train in LPCI mode. The resulting peak cladding Temperature is 1485°F and the peak RPV pressure is 1145.3 psig. The bounding peak suppression pool temperature for PBAPS occurs when utilizing shutdown Method A with RCIC (CST and Refueling Water Storage Tank (RWST) are credited), one RHR train in LPCI mode. The results of the Appendix R evaluation for this Method at 100.05% MUR RTP (4018 MWt) are provided in Table 14.10.2. Key inputs for the Appendix R evaluations are provided in Table 14.10.3.

14.10.5.3 Anticipated Transient Without Scram (ATWS)

The PBAPS ATWS evaluation considered the limiting cases for RPV overpressure and for suppression pool temperature / containment pressure. Previous evaluations considered four ATWS events. Based on experience and the generic analyses performed for Reference 14 (ELTR2), only two cases need to be further analyzed for PBAPS: (1) Main Steam Isolation Valve Closure (MSIVC) and (2) Pressure Regulator Failure Open (PRFO). For PBAPS, a LOOP does not result in a reduction in the RHR pool cooling capability relative to these cases. Thus, with the same RHR pool cooling capability, the containment responses for the MSIVC and PRFO cases bound the LOOP case. The RTP ATWS analysis is performed using the NRC-approved code ODYN. The key inputs to the ATWS analysis are provided in Table 14.10.3. The results of the analysis for the containment response are provided in Table 14.10.2. The limiting ATWS event for the containment response is PRFO at the MELLLA+ operating point 100% RTP and 83% rated core flow.

TABLE 14.10.1
DBA Containment Response Key Analysis Input Values

Parameter	Unit	Value
Reactor		
Initial Power Level (100.35% RTP)	MWt	4030
Normal FWT at 100.35% RTP	°F	384
RPV Dome Pressure	Psia	1,068
Decay Heat Model- Short Term DBA LOCA		ANS 5 1971+20%
Decay Heat Model-Long Term		1979 ANS 5.1+2σ
RPV Free Volume	Ft^3	20,682
RPV Liquid Volume	Ft^3	11,790
RPV Attached Piping Volume	Ft^3	4,142
RPV related masses for long term calculation		
Liquid mass in recirculation loops	Lbm	51,292
Steam mass to first MSIV (total for all lines)	Lbm	3,216
HPCI Liquid from RPV to first closed valve	Lbm	349
RCIC liquid from RPV to first closed valve	Lbm	67
RHR shutdown line liquid from RPV to first closed valve	Lbm	22,103
CS liquid from RPV to first closed valve	Lbm	8,684
MSIV closure initiation	Sec.	0.5
MSIV full closure	Sec.	3.5
Drywell Vent System		
Drywell free volume (including vent system)	Ft^3	175,800
Initial drywell pressure (D=design, B=bounding, R=reference)	Psig.	2.5 (D) 2.0 (B) 0.75 (R)
Initial drywell Temperature (D=design, B=bounding, R=reference)	°F	70 (D) 125 (B) 145 (R)
Initial relative humidity (D=design, R=reference)	%	20 (D) 100 (R)
Downcomer submergence- Low water level	Ft	4.0
Downcomer submergence- High water level	Ft	4.4
Loss coefficient for vent system	Real	5.17
Drywell holdup volume	Ft^3	4,416
Suppression Pool		
Suppression pool volume- low water level (LWL)	Ft^3	122,900
Suppression pool volume-high water level (HWL)	Ft^3	127,300

Table 14.10.1 (continued)

Parameter	Unit	Value
Initial pool temperature (N=minimum, B=bounding, X=maximum)	°F	70 (N) 86 (B) 95 (X)
Suppression Pool air volume-LWL	Ft^3	132,000
Suppression Pool air volume-HWL	Ft^3	127,700
Initial Suppression Pool air space pressure (N=minimum, B=bounding, X=maximum)	psig	0.0 (N) 2.0 (B) 2.5 (X)
Initial Suppression Pool air temperature (N=minimum, B=bounding, X=maximum)	°F	70 (N) 86 (B) 95 (X)
Initial Suppression Pool air space relative humidity (min. and max.)	%	100
RHR		
K value (single HX)	BTU/sec °F	305
K value (cross tie mode)	BTU/sec °F	500
HPSW temperature	°F	92
Drywell spray flow rate (1 RHR pump)	Gpm	7,867
Suppression Pool spray flow rate (1 RHR pump)	Gpm	733
RHR flow rate in SPC mode	Gpm	8,600
Suppression Pool to Drywell Vacuum Breakers		
Opening pressure difference	Psid	0.5
Number of valves		12
Flow area per valve	Ft^2	1.62
Loss coefficient per valve system	Real	2.78

Table 14.10.2

Containment Response Results Dual Unit Interaction

Parameter	ST DBA LOCA	LT DBA LOCA	LT SSLB	Non-Accident Unit	Loss of NSDC	SBO	Appendix R[1]	ATWS[3]
Peak DW Pressure (psig)	50.4 (D) 48.7 (B)	37.8	N/A	27.7	N/A	40.7	N/A	8.8
Peak DW Air Temperature (°F)	298 (D) 296 (B)	322	340	280.7	N/A	N/A	N/A	N/A
Peak Bulk Pool Temperature (°F)	N/A	187.2	187.6	203.8	186.7	200	205.0	171.7
Peak Torus Pressure (psig)	35 (D) 30.6 (B)	32.4	37.8	N/A	N/A	N/A	N/A	N/A
Peak Torus Air Temperature (°F)	N/A	184	N/A	N/A	N/A	N/A	N/A	N/A
Cold Shut Down Time (hrs)	N/A	N/A	13	35.1	16	N/A	63	20.4 min.[2]

Notes:

N/A - denotes a noncritical parameter for this analysis.
1 - Value for Peak Bulk Pool Temperature is for the limiting method, A1. The Cold Shut Down Time is for the limiting method, Modified B modified to use Core Spray instead of RHR.
2 - Time to hot shutdown in minutes is provided for ATWS.
3 – ATWS values are based on the MELLLA+ analysis since MELLLA+ is bounding for ATWS.

Table 14.10.3
Special Event Containment Response Key Analysis Input Values

Parameter	Units	SBO	Appendix R	ATWS	Comments
Initial Rx Power	MWt	4018	4018	4018	Initial power is 100% RTP
Initial Dome Pressure	psia	1050	1050	1050	Maximum normal operating pressure
Initial Rx water Level	In. above Vessel Zero (AVZ)	562	562	562	Normal operating water level
CST volume Required	Gal.	94,570	154,000	135,000	Minimum protected CST volume of 103,377 Gal. is supplemented with up to 51,500 Gal. from the RST
CST Water Temperature	°F	120	120	120	Normal operating upper bound
Initial Torus Water and air Temperature	°F	86	86	86	PBAPS 5 year operating average plus 2σ
Initial Torus Volume	Ft^3	125,100	125,100	125,100	Normal torus water volume
Initial DW air Temperature	°F	145	135	N/A	Tech. Spec upper operating limit for SBO. Nominal value for App. R
Initial DW relative humidity	%	20	20	N/A	Normal operating value
Initial DW pressure	Psig	2.5	0.35	N/A	Maximum operating limit for SBO. Nominal value for App. R
HPSW Temperature	°F	86	86	86	PBAPS 5 year operating average plus 2σ

Table 14.10.3 (continued)

Parameter	Units	SBO	Appendix R	ATWS	Comments
RPV cooldown rate	°F/hr	100	100	100	Tech Spec Limit
RHR K Factor	BTU/sec°F	305	305	305	Single RHR Hx K factor
Max. operating Temperature for HPCI and RCIC	°F	180	180	180	

14.10 ANALYSIS of CONTAINMENT RESPONSE

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