CHAPTER 15: ACCIDENT ANALYSES

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15.0 PLANT SAFETY ANALYSIS

15.0.1 ANALYTICAL OBJECTIVE

The objective of the 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 Chapters of this UFSAR describe and evaluate the reliability of major systems and components of the plant from a safety standpoint. This Chapter assumes that certain incidents occur notwithstanding precautions taken to prevent their happening. This Chapter then examines the potential consequences of each occurrence to determine the effect on the plant, to determine whether the plant design evaluated in earlier Sections is adequate to minimize the consequences of these occurrences, and finally, to ensure that the health and safety of the public are protected from the consequences of even the most severe of the hypothetical accidents analyzed.

15.0.2 ANALYTICAL CATEGORIES

Postulated plant events are categorized as to the expected frequency of occurrence – typically from normal operation up to and including highly improbable (hypothetical) events that represent the greatest challenge to the plant's overall design capability. Each category of event type is discussed briefly below.

<u>Normal Operation</u>

Normal Operation includes the startup of the plant from a cold, shutdown (sub-critical) condition, the approach to criticality of the reactor core, achieving reactor core critical conditions, plant heat up and pressurization to rated conditions and power ascension to rated thermal power. The reverse sequence of these events, a plant shutdown, when planned and manually executed is also considered Normal Operation. Normal Operation also includes the disassembly of the reactor pressure vessel and primary containment to allow reactor core refueling operations and their subsequent re-assembly.

No specific evaluations of Normal Operations are performed, other than to establish the initial plant conditions, e.g., reactor pressure, fuel thermal parameters, etc. that are used in the evaluations of the remaining event categorizes, which are initiated from the various states within Normal Operations. For example, a Turbine Trip transient from full-power, a Rod Withdrawal Error transient during Startup, or a Fuel Handling Accident during Refueling.

The plant's Technical Specifications contain the prescribed limitations on Normal Operation (i.e., the Limiting Conditions for Operation (LCOs)) for those process variables that have been shown to be sensitive to the outcome/results of the other category events or those systems, structures or components (SSCs) that are necessary to mitigate the consequences of these events to remain within the applicable acceptance criteria.

Transients

Transients are sub-divided into two categories: Anticipated Operational Occurrences (AOOs), events which are expected to occur at least once over the life of plant and are the result of single equipment failures or single operator errors that can be reasonably expected during any mode of normal plant operations; and, Abnormal Operating Transients (AOTs), which are generally AOOs with an additional equipment failure or operator error, in addition to that which initiated the AOO. These are generally the most limiting Transients for impact on the plant. AOTs are not postulated to occur over the life of the plant. The general method of identifying and evaluating Transients is shown in Figure 15.0-1.

The following types of operational 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 misoperation 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 standard operating practices. A single operator error is the set of actions that 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 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 withdrawal of a single control rod out of sequence.
   3. An incorrect calibration of an average power range monitor.

4. Manual isolation of the main steam lines because of 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.

Transient events contained in this Section are discussed in individual categories as required by Reference 1, based upon their effect on the various primary system process variables. Each event evaluated is assigned to one of the following applicable categories:

1. Decrease in Core Coolant Temperature: A reduction in reactor vessel water (moderator) temperature results in an increase in core reactivity. This could lead to fuel-cladding damage.

2. Increase in Reactor Pressure: Increases in nuclear system pressure threaten to rupture the reactor coolant pressure boundary (RCPB). Increasing pressure also collapses the voids in the core moderator, thereby increasing core reactivity and power level, which threatens fuel cladding because of overheating.

3. Decrease in Reactor Core Coolant Flow Rate: A reduction in the core coolant flow rate threatens to overheat the cladding as the coolant becomes unable to adequately remove the heat generated by the fuel.

4. Reactivity and Power Distribution Anomalies: Transient events included in this category are those which cause rapid increases in power resulting from increased core flow disturbance events. Increased core flow reduces the void content of the moderator, thereby increasing core reactivity and power level.

5. Increase in Reactor Coolant Inventory: Increasing coolant inventory could result in excessive moisture carryover to the main turbine.

6. Decrease in Reactor Coolant Inventory: Reductions in coolant inventory could threaten the fuel as the coolant becomes less able to remove the heat generated in the core.

Parameter variations (for events 1 through 6 above), if uncontrolled, could result in excessive damage to the reactor fuel or damage to the nuclear system process barrier or both.

- An increase in nuclear system pressure 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 to damage the fuel from overheating.
- A decrease in reactor vessel water (moderator) temperature results in an insertion of positive reactivity as density increases. This could lead to fuel overheating.

- Positive reactivity insertions are possible from causes other than nuclear system pressure or moderator temperature changes. Such reactivity insertions threaten fuel damage by overheating.
- Both a decrease in reactor vessel coolant inventory 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 analytical 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.

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.

Accidents

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

1. Mechanical failure of various components leading 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 the breakage of the coupling between a control rod drive (CRD) and the control rod, the failure of a crane cable, and the failure of a spring used to close an isolation valve.

2. Overheating of the fuel barrier. 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 direct overheating situation. For example, Primary Containment failure would require a combination of high temperature and high pressure.

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 a rupture is assumed only if the component is subjected to significant pressure.

The method of identifying and evaluating Accidents is shown in Figure 15.0-2.

For analytical 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 Accident resulting in potential radiation exposures greater than any other Accident considered, under the same general Accident assumptions, are designated a Design-Basis Accident (DBA) 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 misoperation of a device that 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 misoperation. 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.

Federal Regulations also specify that for certain Accidents (such as loss-of-coolant accidents (LOCAs), in addition to the single failure discussed above, the mitigation of the event must be demonstrated with the worst case scenario of either offsite AC power available or unavailable. In most cases it has been shown that the loss-of-offsite power (LOOP) is the bounding of the two scenarios.

In the analyses of the DBAs, analysis assumptions are made regarding the initial conditions and equipment responses 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 DBA.

<u>Special Events</u>

Events in this Category are those where the plant has demonstrated its ability to respond to specific events that were postulated after the plant was initially designed and licensed, usually in response to a change in regulations (e.g., 10 CFR 50.62). These events are sometimes referred to as "beyond design basis" events, because they were postulated after the plant was originally designed and constructed. These events have unique requirements for demonstrating the plant's ability to mitigate the Special Event; in particular, initial conditions, equipment response assumptions, methods used. These are described in the event description of the individual event in Section 15.3.

15.0.3 EVENT EVALUATION

Situations, causes, and their frequency were not presented in the original FSAR. Causes and frequencies listed in current docketed FSARs for boiling-water reactors (BWRS) are generally applicable to the Duane Arnold Energy Center (DAEC). Situations, causes, and frequency of limiting events are discussed in Reference 2.

15.0.4 UNACCEPTABLE RESULTS

The following are considered unacceptable results of Transients:

1. Safety Analysis Fuel Design Limits (SAFDLs) shall not be exceeded as a result of any Transient. (MCPR, LHGR (MOP & TOP), and MAPLHGR)

2. No damage to the nuclear system process barrier (RCPB) shall result from the forces associated with Transients. (ASME Code – Upset limits)

3. The radiological effects of Transients shall not exceed the applicable limits of 10 CFR Part 20 and 10 CFR Part 50, App. I (ODAM). Note: this is typically satisfied by meeting Criterion 1 above for the SAFDLs.

The following are considered unacceptable results of Accidents:

1. The radiological effects, both onsite and offsite, of the accident shall not exceed the guidelines of 10 CFR 50.67.

2. Fuel response to LOCAs shall not be in excess of 10 CFR 50.46 limits.

3. Fuel response to CRDA shall not be in excess of 280 cal/gm (peak fuel enthalpy).

4. No damage to the nuclear system process barrier (RCPB) shall result from the forces associated with Accidents. (ASME Code – Emergency & Faulted Conditions). For Accidents that assume an initial failure of the nuclear system

process barrier, the primary containment integrity protection criteria must be satisfied.

5. Proper initial response (10 minutes) to Accidents shall be automatic and require no decision or manipulation of controls by plant operations personnel.

The acceptance criteria for Special Events are generally unique to the event and are described in the individual event description in Section 15.3.

15.0.5 SEQUENCE OF EVENTS AND SYSTEMS OPERATIONS

Each transient or accident is discussed and evaluated in terms of an event description and the analyzed results.

15.0.6 Acceptance Criteria for Fission Product Barriers

2013-014 See Table 15.0-7.

15.0.6.1 Fuel Cladding

Section 4.4 describes the various fuel failure mechanisms and establishes fuel damage limits for various plant conditions. The satisfaction of safety design bases for Abnormal Operational Transients is determined by demonstrating that such transients do not result in established values [2,3] being exceeded.

The satisfaction of safety design bases for Accidents is shown by demonstrating that fuel overheating (e.g., peak clad temperature remains below 2200ºF or peak fuel enthalpy remains below 280 cal/gm) is not postulated to occur.

15.0.6.2 Reactor Coolant Pressure Boundary

The satisfaction of safety design bases for Abnormal Operational Transients is assessed by comparing peak Reactor Coolant Pressure Boundary (RCPB) pressure with the overpressure transient allowed by the applicable industry code. The only significant areas of interest for internal RCPB pressure damage are the high-pressure portions of the nuclear system primary barrier, the reactor vessel, and the high-pressure pipelines attached to the reactor vessel. The over-pressure below which no damage can occur is taken as the lowest pressure increase over the design pressure allowed by either: the ASME Boiler and Pressure Vessel Code, Section III, for the reactor vessel; or, ANSI B31.1 Code for the high-pressure nuclear system piping. The ASME Code, Section III, permits pressure transients up to 10% over design pressure (110% x 1250 psig = 1375 psig); ANSI B31.1 permits pressure transients up to 15% over the design pressure (115% x 1250 psig = 1438 psig). The overpressure protection analysis is contained in the SRLR for the current operating fuel cycle and is discussed further in Section 15.1.2.3.2.

An analysis 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 process barrier damage results from nuclear excursion accidents. The results of the analysis are discussed in References 2 and 3.

15.0.6.3 Containment

The satisfaction of safety design bases requires that the primary and secondary containments retain their integrities during plant events. Containment integrity is maintained as long as internal pressures and temperatures remain below the maximum allowable values. For added conservatism, the design pressure (56 psig) is used as the acceptance criterion for Accidents (Section 15.2). However, some special events (Section 15.3) use the maximum allowable pressure as the acceptance criterion. (Note: the containments also have negative pressure allowances as well. However, these are not challenged by any of the Abnormal Operating Transients or Accidents analyzed.) The maximum allowable internal pressures and temperatures are as follows:

Drywell (primary containment)	62 psig/281 °F
Suppression chamber (primary containment)	62 psig/281 °F
Secondary containment	7 in H_20

Accident-initiated fluid impingement and jet forces are considered in the design of the primary containment as described in Chapter 3. DBAs are used in determining 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.

15.0.7 ANALYSIS BASIS

The analysis basis for Chapter 15 is given in References 2 and 3. The capability to operate the plant in various Extended Operating Domains (such as Single Recirculation Loop Operation and Increased Core Flow) and with certain Equipment Out-of-Service configurations (such as Turbine Bypass Valves) has been evaluated and found to be acceptable, (References 5,
2012-020 39, 47, 60, and 61). See the Fuel Reload-Licensing Engineering Data (FRED) form for the current operating cycle for the complete listing. The continued applicability of the original analysis is validated as part of each cycle's SRLR.

Table 15.0-1 contains a listing of all the events in Chapter 15 and delineates which are part of the cycle-specific reload analysis; those that are the current evaluations of record, updated to Extended Power Uprate conditions; and those that are historical in nature and presented for
2013-003 completeness. The loss of stator cooling event identified in Reference 64 was evaluated specifically for DAEC (Reference 65) with the conclusion that the event is not limiting on a cycle-independent basis.

15.0.8 EVALUATION MODELS

The models used to analyze the plant for Abnormal Operating Transients, Accidents, and Special Events are given in Table 15.0-2.

15.0.9 INPUT PARAMETERS AND INITIAL CONDITIONS FOR ANALYZED EVENTS

The generic input parameters and initial conditions for Abnormal Operating Transients are specified in the Operating & Plant Licensing (OPL)-3 form, Table 15.0-3. The generic input parameters and initial conditions for Accidents are specified in the OPL-4, 4a and 5 forms,
2012-020
2017-001 Tables 15.0-4, 5 and 6, respectively.

For the evaluation of the radiological consequences of events that are analyzed, the input parameters and assumptions used in those evaluations are discussed in the individual event discussions in Section 15.2.

Analyses that assume data inputs and/or assumptions different than these values are specifically discussed in the appropriate event discussion. For example, most events assume that the plant is initially at rated power and flow. However, some events are more limiting from "off-rated" conditions. Also, most transients assume end-of-cycle (EOC) fuel conditions (such as exposure, power shape, all control rods withdrawn). Again, some events can be more limiting at other points in the fuel cycle, such as the beginning-of-cycle (BOC) or at the most reactive point in the middle-of-cycle (MOC).

15.0.10 INITIAL POWER/FLOW OPERATING CONSTRAINTS

The analyses basis for most of the safety analyses is 102% of the rated thermal power (1912 MWt), i.e., 1950.2 MWt, at rated core flow (49.0 Mlb_m/hr), per Reference 6 (RG 1.49). Some analyses have been re-evaluated at 105% of rated core flow (Ref. 60). Reactor Heat Balances for 100% and 102% power at rated (100%) core flow are provided as Figures 15.0-3 and 4, respectively. Reactor Heat Balances for 100% and 102% power at 105% core flow, are provided as Figures 15.0-5 and 6, respectively.

The operating power/flow map is the operating basis of the analysis, as shown in the current cycle's COLR. A "typical" power/flow map is provided as Figure 15.0-7.

Certain events are evaluated at other than the above-mentioned conditions. These conditions are discussed in the appropriate event descriptions.

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2017-001 67. Deleted.

TABLE 15.0 – 1
SAFETY ANALYSIS EVENT CLASSIFICATION

TRANSIENTS

2012-020

UFSAR Section #	Event Title	Original FSAR	Extended Power Uprate	Cycle-Specific (Reload)*
15.1.1.1	Feedwater Controller Failure - Maximum Demand			X
15.1.1.2	Loss of Feedwater Heating			X
15.1.1.3	Inadvertent HPCI Actuation			X
15.1.2.1.1	Main Generator Load Rejection With Bypass Valves – High Power		X	
15.1.2.1.2	Main Generator Load Rejection Without Bypass Valves – High Power			X
15.1.2.1.3	Main Generator Load Rejection Without Bypass Valves – Low Power		X	
15.1.2.2.1	Main Turbine Trip With Bypass – High Power		X	
15.1.2.2.2	Main Turbine Trip Without Bypass – High Power			X
15.1.2.2.3	Main Turbine Trip Without Bypass – Low Power		X	
15.1.2.3.1	Closure of All Main Steam Line Isolation Valves – High Power		X	
15.1.2.3.2	Closure of All Main Steam Line Isolation Valves with Direct Scram Failure – High Power			X
15.1.2.3.3	Closure of One Main Steam Line Isolation Valve – High Power		X	
15.1.3.1	Recirculation Flow Control Failure – Decreasing Flow	X		
15.1.3.2	Trip of One Recirculation Pump	X		
15.1.3.3	Trip of Two Recirculation Pumps	X		
15.1.4.1	Rod Withdrawal Error at Power			X
15.1.4.2	Rod Withdrawal Error at Startup		X	

*Beginning with Cycle 22, the Cycle-specific Reload includes Increased Core Flow (105% of Rated Core Flow) analysis.

TABLE 15.0 – 1
SAFETY ANALYSIS EVENT CLASSIFICATION

TRANSIENTS

UFSAR Section #	Event Title	Original FSAR	Extended Power Uprate	Cycle-Specific (Reload)*
15.1.4.3	Control Rod Removal Error during Refueling	X		
15.1.4.4.1	Fuel Assembly Insertion Error during Refueling – Inadvertent Criticality	X		
15.1.4.4.2	Fuel Loading Error – Mislocated Bundle			X
15.1.4.4.3	Fuel Loading Error – Rotated Bundle			X
15.1.5.1	Startup of an Idle Recirculation Pump	X		
15.1.5.2	Recirculation Flow Controller Failure – Increasing Flow		X	
15.1.5.3	Recirculation Flow Controller Failure – Slow Flow Runout		X	
15.1.6	No Events for BWRs			
15.1.7.1	Pressure Regulator Failure - Open	X		
15.1.7.2	Inadvertent Opening of a Safety/Relief Valve	X		
15.1.7.3	Loss of Feedwater Flow		X	
15.1.7.4	Trip of One Feedwater Pump		X	

2012-020 (margin, beside 15.1.4.4.2)

ACCIDENTS

UFSAR Section #	Event Title	Original FSAR	Extended Power Uprate	Cycle-Specific (Reload)
15.2.1	Loss-of-Coolant Accidents (LOCA)			
15.2.1.1	Reactor Recirculation Pipe Breaks		X**	
15.2.1.2	Core Spray Line Break	X		
15.2.1.3	Feedwater Line Break	X		
15.2.1.4	Main Steamline Break – Inside		X	

2012-020

** GE did an evaluation of the LOCA response of the reactor core and containment at Increased Core Flow (105% of Rated Core Flow) conditions (Reference 15.0-60). GE also did a LOCA analysis for the GNF2 new fuel introdu cction (Reference 15.0-62).

TABLE 15.0 – 1
SAFETY ANALYSIS EVENT CLASSIFICATION

ACCIDENTS

	UFSAR Section #	Event Title	Original FSAR	Extended Power Uprate	Cycle-Specific (Reload)
		Primary Containment			
	15.2.1.5	Main Steamline Break – Outside Primary Containment		X	
	15.2.2	Instrument Line Break	X		
	15.2.3	Recirculation Pump Seizure			X
	15.2.4	Control Rod Drop Accident		X	
2012-020	15.2.5	Fuel Handling Accident+		X	

SPECIAL EVENTS

	UFSAR Section #	Event Title	Original FSAR	Extended Power Uprate	Cycle-Specific (Reload)	Increased Core Flow
2012-020	15.3.1	Anticipated Transients Without Scram (ATWS)+				
	15.3.1.1	ATWS – Closure of All Main Steamline Isolation Valves		X		
	15.3.1.2	ATWS – Pressure Regulator Failure – Maximum Demand		X		
	15.3.1.3	ATWS – Loss-of-Offsite Power		X		
	15.3.1.4	ATWS – Inadvertent Opening of One Safety/Relief Valve		X		
	15.3.2.1	Station Blackout (SBO)		X		
2012-020 2013-013	15.3.3	NFPA 805 Fire – Safe Shutdown+				
2013-013	15.3.3.1	Fire – No Spurious Operations		X		
2013-013	15.3.3.2	Fire – Spurious Opening of One Safety/Relief Valve (20 minutes)		X		
2013-013	15.3.3.3	Fire – Spurious Opening of One Safety/Relief Valve (10 minutes)		X		
2014-003	15.3.3.4	Fire – Spurious Leakage from a One-inch Line		X		
	15.3.4.1	Thermal-Hydraulic Stability (Zones)			X	

2012-020 + GE performed evaluation as part of GNF new fuel introduction.

TABLE 15.0 – 1
SAFETY ANALYSIS EVENT CLASSIFICATION

SPECIAL EVENTS

	UFSAR Section #	Event Title	Original FSAR	Extended Power Uprate	Cycle-Specific (Reload)	Increased Core Flow
2012-020	15.3.4.2	Thermal-Hydraulic Stability (HCOM)			X	
2012-020	15.3.5	Reactor Internal Pressure Differentials (RIPD) [+]				
	15.3.5.1	RIPD – Normal				X
	15.3.5.2	RIPD – Upset				X
	15.3.5.3	RIPD – Emergency				X
	15.3.5.4	RIPD – Faulted				X
	15.3.5.5	RIPD – Flow Induced Loads		X		[***]
	15.3.5.6	RIPD – Acoustic Loads		X		[***]

[***] GE did an evaluation of the RIPD response at Increased Core Flow (105% of Rated Core Flow) conditions (Reference 15.0-60).

Table 15.0-2

Table of Computer Codes/Methods of Evaluation used in Chapter 15.0 Analyses

	Evaluation Subject	Computer Code	Version	NRC Approved	Reference
	Accident Radiological Consequence Analysis	PAVAN	2.0	Yes	NUREG/CR-2858
		ARCON96	NA	Yes	NUREG/CR-6331
2016-006		RADTRAD	3.02/3.03	Yes	NUREG/CR-6604 See Note 22
		MicroShield	5.03a	No	See Note 1
		ORIGEN2	N/A	No	See Note 2
	ATWS	ODYN	10	Yes	NEDE 24154P-A, Feb. 2000
		TASC	3	Yes	NEDC-32084P-A, Rev. 2, July 2002
		STEMP	4	Yes	NEDE-32868P, Dec. 1998
		ISCOR	9	Yes	NEDE-32868P, Dec. 1998
	Containment System Response	M3CPT	5	Yes	NEDO-10320, Apr. 1971
		SHEX	4	Yes	See Note 4
		LAMB	8	No	See Note 16
2012-020 2013-013 2017-001	ECCS-LOCA and NFPA 805 Fire Protection	SAFER	4	Yes	NEDC-32950P, Jan 2000; see Note 5
		LAMB	8	Yes	NEDE-20566P-A, Sept 1986
		ISCOR	9	Yes	NEDE-30130-P-A, Apr 1985; MFN-212-78, May 1978
		TASC	3	Yes	NEDEC-32084P-A, Rev. 2, July 2002
		SHEX	4	Yes	See Note 4
		PRIME	3.6	Yes	NEDC-33256P-A, Sep. 2010 NEDC-33257P-A, Sep. 2010 NEDC-33258P-A, Sep. 2010
	Flux Analysis	DORT	0	No	See Note 6
		TGBLA	6	Yes	NEDE-24011-P-A, Nov. 1999
	NUREG-0737 Assessment	MicroShield	5.03a	No	See Note 1
	Probabilistic Risk Assessment Evaluation	REBECA	3.0Q	No	See Note 8
		CAFTA	3.2b	No	
		PRAQuant	3.3bx	No	
		OMNICUT	1.00b	No	
		MAAP	3.0B	No	
	Reactor Core and Fuel Performance	TGBLA	6	Yes	NEDE-24011-P-A, Nov. 1999
		PANAC	11	Yes	NEDE-24011-P-A, Nov. 1999
		ISCOR	9	Yes	NEDE-30130-P-A, Apr. 1985; MFN-212-78, May 1978
		GESAM	2	Yes	NEDO-10958-A, Jan. 1977; NEDC-32601P-A, Aug. 1999; NEDC-32694-P-A, Aug. 1999
	Reactor Heat Balance	ISCOR	9	No	See Note 9
	Reactor Internals Structural Evaluation	ANSYS	5.3	No	See Note 10

Table 15.0-2

Table of Computer Codes/Methods of Evaluation used in Chapter 15.0 Analyses

	Evaluation Subject	Computer Code	Version	NRC Approved	Reference
	Reactor Internal Pressure Differences	LAMB	7	No	NEDE-23008, Apr 1978; NEDE-20566-P-A, Sept. 1986 See Note 16
		TRACG	2	Yes	NEDC-32176P, Feb. 1993; NEDC-32177P, Rev. 1, June 1993; NEDC-32192, Dec. 1993; See Note 11
		ISCOR	9	Yes	NEDE-32227, Oct. 1993; NEDC-32082P, Aug. 1992; See Note 9
	Refueling Shutdown Margin	PANAC	11	Yes	NEDE-24011-P-A, Nov. 1999
2012-020	Thermal Hydraulic Stability	PANAC	11	Yes	NEDE-24011-P-A, Nov. 1999; See Note 17
		ISCOR	9	Yes	NEDE-24011-P-A, Nov. 1999; See Note 18
		CRNC	6	Yes	NEDO-24154-A, 1986; See Note 19
		OPRM	4	Yes	NEDO-32465-A, Aug, 1996; See Note 20
		ODYSY	5	Yes	NEDE-33213P-A, April 2009
2017-001		TRACG	04	No	NEDE-32465 Supplement 1 P-A, Rev. 1, October 2014; See Note 21
	Transients	ISCOR	9	Yes	NEDE-24011-P-A, Nov. 1999
		PANAC	11	Yes	NEDE-24011-P-A, Nov. 1999
		ODYN	10	Yes	NEDO-24154-A, 1986
		TASC	3	Yes	NEDC-32084P-A, Rev. 2, July 2002
		CRNC	6	Yes	NEDO-24154-A, 1986
		SAFER	4	Yes	See Note 13
		GROMT	1	No	See Note 14
	Vibration Analysis - Main Steamline Flow Restrictor	SABOR-Free	N/A	N/A	See Note 15

NOTES:

1. MicroShield is used in safety related applications by many nuclear plants in the United States. The code has been used to support licensing submittals that have been accepted by the NRC.
2. The ORIGEN2 computer code is a widely recognized program by the BWR community and NRC as an appropriate methodology to obtain fission product inventories. In addition, ORIGEN2 output for fission product inventories has been used by NRC and in licensing submittals accepted by NRC.

3. Not used.
4. Letter from Ashok Thadani, Director Division of System Safety and Analysis, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, to Gary L. Sozzi, Manager Technical Services, General Electric Nuclear Energy, " Use of SHEX Computer Program and ANSI/ANS 5.1-1979 Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," July 13, 1993.

Table 15.0-2

Table of Computer Codes/Methods of Evaluation used in Chapter 15.0 Analyses

5. Letter, S.A. Richards (NRC) to J.F. Klapproth, "General Electric Nuclear Energy (GENE) Topical Reports GENE (NEDC)-32950P and GENE (NEDC)-32084P Acceptability Review," May 24, 2000.
6. DORT is distributed as part of the TORT code package by the Radiation Shielding Information Center at Oak Ridge National Laboratory and is the updated version of the DOT series of codes. Although NRC has not published an approval SER for DORT, standardization of the usage of the DOT code package has been endorsed in the recently published Draft Regulatory Guide DG-1053.
7. Not Used.
8. These code packages are standard industry-accepted codes for the development of Probabilistic Risk Assessment models and calculations, which have been used to support NRC submittals for Individual Plant Examination (IPE).
9. The heat balance application of ISCOR is not considered to be NRC approved. Simple reactor systems heat balance equations are used in ISCOR. The reactor core coolant hydraulics implemented in ISCOR was approved per Letter MFN-212-78, D.G. Eisenhut (NRC) to R.L. Gridley (GE), "Safety Evaluation for the GE LTR, Generic Reload Fuel Application, Original Document NEDE-24011," May 12, 1978. The use of ISCOR to provide core thermal-hydraulic information in Reactor Internal Pressure Differences, Transient, Anticipated Transients Without Scram (ATWS), Stability, and LOCA applications is consistent with the approved models and methods.
10. ANSYS is a general-purpose structural analysis code, verified in accordance with the approved GE Engineering Operating Procedures.
11. NRC has approved the TRACG application for the flow-induced loads on the core shroud in NRC SER TAC No. M90270, dated September 30, 1994.
12. Not Used.
13. SAFER02 and SAFER03 have been reviewed and approved by the NRC per NEDE-23785-1-PA, Rev. 1, Oct. 1984, and NEDE-30996P-A, Oct. 1987. Changes since NRC approval are documented in MFN (Master File Number) -040-88, MFN-023-90, MFN-025-91, and MFN-090-93.
14. GROMT is an RBM simulator, which is described in GESTAR II, NEDE-24011-P-A. This application has been used in previous uprate submittals. In addition, this methodology is used as part of the current DAEC reload licensing analysis and there are no changes for the EPU.
15. This code was used in the original FSAR vibration analysis of the Main Steamline flow venturi (Section 5.4.4.2). No other information about this code is available.

Table 15.0-2

Table of Computer Codes/Methods of Evaluation used in Chapter 15.0 Analyses

16. NEDE-23008 documents the LAMB07 model description. NRC approval has not been identified for the use of LAMB for the evaluation of reactor internal pressure differences or containment system response. NRC has approved use of the LAMB code (NEDE-20566P-A) for ECCS-LOCA application. However, the use of LAMB07 for RIPD and LAMB08 for Containment Response is consistent with the model description of NEDE-20566P-A.

2012-020

17. The physics code PANACEA provides inputs to the transient code ODYN. The improvements to PANACEA that were documented in NEDE-30130-P-A were incorporated into ODYN by way of Amendment 11 of GESTAR II (NEDE-24011-P-A). The use of TGBLA Version 06 and PANACEA Version 11 in this application was initiated following approval of Amendment 26 of GESTAR II by letter from S.A. Richards (NRC) to G.A. Watford (GE) Subject: "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, GESTAR II Implementing Improved GE Steady-State Methods," (TAC NO. MA6481), November 10, 1999. TGBLA06 with Error Correction 6 was used in the DAEC Core Design analysis and it meets the requirements established by the SER for LTR NEDC-33173P-A.
18. The ISCOR code is not approved by name. However, the SER supporting approval of NEDE-24011P Rev. 0 by the May 12, 1978 letter from D. G. Eisenhut (NRC) to R. Gridley (GE) finds the models and methods acceptable, and mentions the use of a digital computer code. The referenced digital computer code is ISCOR. The use of ISCOR to provide core thermal-hydraulic information in RIPDs, Transient, ATWS, Stability, Reactor Core and Fuel Performance and LOCA applications is consistent with the approved models and methods.
19. The CRNC code is not approved by name. However, NEDO-24154-A finds the models and methods acceptable. The use of CRNC to provide input for Exclusion/Buffer Region calculations is consistent with the approved models and methods.
20. The OPRM code is not approved by name. However, NEDO-32465-A finds the models and methods acceptable. The use of OPRM to provide Stability OPRM Amplitude Setpoints is consistent with the approved models and methods.

2017-001

21. Migrated to TRACG04/PANAC11 from TRACG02/PANAC10.

2016-006

22. RADTRAD version 3.03 is NRC approved for Control Rod Drop Accident only by Amendment 261. All other accidents were NRC approved using RADTRAD version 3.02.

Table 15.0-3

2019-001

OPL-3									
Transient Protection Parameters Verification For									
Reload Licensing Analyses	Plant:	Duane Arnold							
	Project:	CYCLE 27							
		CYCLE 26	GE	CUSTOMER	CYCLE 27				
		RESOLVED	PROPOSED	PROPOSED	RESOLVED	CUSTOMER	CUSTOMER	GE	GE
PARAMETER DESCRIPTION	UNITS	VALUE	VALUE	VALUE	VALUE	REFERENCE	COMMENTS	REFERENCE	COMMENTS
1. PARAMETERS USED IN RELOAD LICENSING ANALYSES (RLA)									
1.1 INITIAL OPERATING CONDITIONS									
A. CORE THERMAL POWER									
1. RATED POWER (LICENSED)	A. MWT	1912.00	1912.00		1912.00	1,43			
	B. % RATED	100.00	100.00		100.00	1,43			
2. RELOAD LICENSE ANALYSIS (RLA) POWER	A. MWT	1912.00	1912.00		1912.00	1,43			
	B. % RATED	100.00	100.00		100.00	1,43			
B. VESSEL DOME PRESSURE									
1. AT RATED POWER	PSIG	1025.3	1025.3		1025.3	1,43	39		
2. AT RLA POWER	PSIG	1025.3	1025.3		1025.3	1,43	39		
C. STEAMLINE PRESSURE DROP	PSID	57.0	57.0		57.0	1,43			
D. CORE INLET FLOW (RATED)	LB/HR	49.00E+06	49.00E+06		49.00E+06	1,43			
E. VESSEL STEAM FLOW									
1. AT RATED POWER	LB/HR	8.35E+06	8.35E+06		8.35E+06	1,43			
2. AT RLA POWER	LB/HR	8.35E+06	8.35E+06		8.35E+06	1,43			
F. FEEDWATER TEMPERATURE	DEG F	431.40	431.40		431.40	1,43			
(AT RLA POWER)									
G. CHANGE IN FEEDWATER TEMPERATURE (MAXIMUM)	DEG F	100.00	100.00		100.00	4			
(WORST SINGLE FAILURE OF FW HEATERS)									

Table 15.0-3

2019-001

OPL-3									
Transient Protection Parameters Verification For									
Reload Licensing Analyses	Plant:	Duane Arnold							
	Project:	CYCLE 27							
		CYCLE 26	GE	CUSTOMER	CYCLE 27				
		RESOLVED	PROPOSED	PROPOSED	RESOLVED	CUSTOMER	CUSTOMER	GE	GE
PARAMETER DESCRIPTION	UNITS	VALUE	VALUE	VALUE	VALUE	REFERENCE	COMMENTS	REFERENCE	COMMENTS
1. PARAMETERS USED IN RELOAD LICENSING ANALYSES (RLA)									
1.1 INITIAL OPERATING CONDITIONS AT RATED POWER PLUS UNCERTAINTY (1,2,3,4)									
A. CORE THERMAL POWER									
2. RELOAD LICENSE ANALYSIS RATED POWER PLUS UNCERTAINTY	A. MWT	1950.24	1950.24		1950.24	3,44	40		
	B. % RATED	102.00	102.00		102.00	3,44			
B. VESSEL DOME PRESSURE									
2. AT RATED POWER PLUS UNCERTAINTY	PSIG	1040.3	1040.3		1040.3	3,44	41		
C. STEAMLINE PRESSURE DROP									
2. AT RATED POWER PLUS UNCERTAINTY	PSID	59.8	59.8		59.8	1,3,43,44	1		
D. CORE INLET FLOW (RATED)	LB/HR	49.00E+06	49.00E+06		49.00E+06	3,44			
E. VESSEL STEAM FLOW									
2. AT RATED POWER PLUS UNCERTAINTY	LB/HR	8.55E+06	8.55E+06		8.55E+06	3,44			
F. FEEDWATER TEMPERATURE									
2. AT RATED POWER PLUS UNCERTAINTY	DEG F	433.80	433.80		433.80	3,44			
(1) The above parameter values apply to the MSIV-FSCRAM event.									
(2) All of these initial operating conditions, with the exception of CORE INLET FLOW,									
will differ from those at rated power to account for the change in operating state.									
(3) A Thermal Power Optimization uprate may not be at 102% of rated power									
(4) Initial operating conditions at rated power plus uncertainty are not applicable									
for TRACG plants.									

Table 15.0-3

2019-001

OPL-3									
Transient Protection Parameters Verification For									
Reload Licensing Analyses	Plant:	Duane Arnold							
	Project:	CYCLE 27							
		CYCLE 26	GE	CUSTOMER	CYCLE 27				
		RESOLVED	PROPOSED	PROPOSED	RESOLVED	CUSTOMER	CUSTOMER	GE	GE
PARAMETER DESCRIPTION	UNITS	VALUE	VALUE	VALUE	VALUE	REFERENCE	COMMENTS	REFERENCE	COMMENTS
1.1 INITIAL OPERATING CONDITIONS (CONT.)									
H. RECIRCULATION SYSTEM PUMP OPERATING CONDITIONS (1)									
(ALL PARAMETERS SHOULD BE MEASURED AT RATED CORE									
POWER/FLOW CONDITIONS AND SAME CYCLE EXPOSURE)									
POWER	% RATED	99.94	99.94		99.94		43		
FLOW	% RATED	100.94	100.94		100.94		43		
CYCLE	NUMBER	24	24		24		74		
EXPOSURE	MWD/ST	1455.5	1455.5		1455.5		43		
1. PUMP HEAD A. (LOOP A)	PSID	174.0	174.0		174.0		43		
B. (LOOP B)	PSID	166.0	166.0		166.0		43		
2. PUMP FLOW A. (LOOP A)	LB/HR	10.850E+06	10.850E+06		10.850E+06		43		
B. (LOOP B)	LB/HR	10.580E+06	10.580E+06		10.580E+06		43		
3. SPEED A. (LOOP A)	RPM	1694.0	1694.0		1694.0		43		
MEASUREMENT	MG/PUMP	PUMP	PUMP		PUMP		43		
B. (LOOP B)	RPM	1693.0	1693.0		1693.0		43		
MEASUREMENT	MG/PUMP	PUMP	PUMP		PUMP		43		
4. FLOW CONTROL VALVE POSITION A. (LOOP A)	% OPEN						44		
(VALVE FLOW CONTROL PLANTS) B. (LOOP B)	% OPEN						44		
5. CORE PLATE PRESSURE DROP	PSID	23.84	23.84		23.84		43		
I. NORMAL VESSEL WATER LEVEL	IN AVZ	535.5	535.5		535.5	54,55	45		
(NARROW RANGE AT RATED POWER)									
J. LICENSED POWER/FLOW OPERATING MAP		see FRED							
NOTE: AVZ = ABOVE VESSEL ZERO									
(1) For TRACG plants, recirculation tuning will not be performed on the ODYN basedeck unless there is a change that may impact the jet pump loss coefficients.									

Table 15.0-3

2019-001

OPL-3									
Transient Protection Parameters Verification For									
Reload Licensing Analyses	Plant:	Duane Arnold							
	Project:	CYCLE 27							
		CYCLE 26	GE	CUSTOMER	CYCLE 27				
		RESOLVED	PROPOSED	PROPOSED	RESOLVED	CUSTOMER	CUSTOMER	GE	GE
PARAMETER DESCRIPTION	UNITS	VALUE	VALUE	VALUE	VALUE	REFERENCE	COMMENTS	REFERENCE	COMMENTS
1.2 SCRAM PARAMETERS									
A. APRM NEUTRON FLUX SCRAM SETPOINT									
(AT RATED DRIVE FLOW)									
1. RELOAD LICENSING ANALYSIS (RLA) VALUE	% RATED	127.18	127.18		127.18	6			
2. NTSP	% RATED	120.36	120.36		120.36	6			
3. AFS RESPONSE TIME (MAXIMUM VALUE)	SEC	0.02	0.02		0.02	45			
B. APRM THERMAL POWER SCRAM (TPS) SETPOINT									
(AT RATED DRIVE FLOW)									
1. RLA VALUE	% RATED								
2. NTSP	% RATED								
C. TPS TIME CONSTANT (MAXIMUM VALUE)	SEC								
D. VESSEL DOME PRESSURE SCRAM SETPOINT									
1. RLA VALUE	PSIG	1075.90	1075.90		1075.90	7			
2. NTSP	PSIG	1067.00	1067.00		1067.00	7	3, 26		
E. RESPONSE TIME OF PRESSURE SCRAM SENSOR	SEC	0.50	0.50		0.50	28, 71			
F. MSIV POSITION SWITCH SETPOINT									
RLA VALUE	% OPEN	88.00	88.00		88.00	8,47			
G. TURBINE STOP VALVE (TSV) POSITION SWITCH SETPOINT									
RLA VALUE	% OPEN	87.50	87.50		87.50	4,9			
H. RESPONSE TIME OF TCV FAST CLOSURE SENSOR (MAXIMUM)	SEC	0.03	0.03		0.03	10			
NOTES:									
The utility is responsible for determine the setpoint methodology that applies for the RLA value.									
NTSP = NOMINAL TRIP SET POINT									

Table 15.0-3

2019-001

OPL-3									
Transient Protection Parameters Verification For									
Reload Licensing Analyses	Plant:	Duane Arnold							
	Project:	CYCLE 27							
		CYCLE 26	GE	CUSTOMER	CYCLE 27				
		RESOLVED	PROPOSED	PROPOSED	RESOLVED	CUSTOMER	CUSTOMER	GE	GE
PARAMETER DESCRIPTION	UNITS	VALUE	VALUE	VALUE	VALUE	REFERENCE	COMMENTS	REFERENCE	COMMENTS
1.2 SCRAM PARAMETERS (CONT.)									
I. ADDITIONAL RPS DELAY FOR BPV INQUIRY	SEC								
(TCV FAST CLOSURE/FULL BYPASS PLANTS ONLY)									
J. RESPONSE TIME (DELAY) OF RPS LOGIC (MAXIMUM)	SEC	0.05	0.05		0.05	28,46			
K. RESPONSE TIME OF CRD DURING SCRAM	TABLE	TSIP	TSIP		TSIP	4,69			

CRD CONTROL FRACTION	SCRAM TIME, SEC(1) 67A	67B	OPTB	------BWR6-------- SS(2)	AOPT	OPPT	TSIP
0	0.200	0.200	0.200	0.100	0.100	0.100	0.200
1				0.138	0.138	0.153	
5	0.375	0.375	0.324	0.218	0.219	0.252	0.490
10				0.317	0.321	0.376	
20	0.900	0.900	0.694	0.503	0.516	0.644	0.900
40				0.874	0.907	1.179	
50	2.000	2.000	1.459	1.087	1.175	1.540	2.000
75				1.620	1.844	2.444	
90	5.000	3.500	2.535	1.940	2.246	2.986	3.500
100	5.750	3.875	2.804	2.153	2.690	3.380	3.875

NOTES:
(1) TIME FROM DE-ENERGIZATION OF SCRAM SOLENOID
TO SPECIFIED CRD INSERTION
(2) TECH SPEC (SURVEILLANCE REQUIREMENTS) VALUES

SS = STEADY STATE OPERATION
AOPT = ABNORMAL OPERATING PRESSURIZATION TRANSIENT
OPPT = OVERPRESSURE PROTECTION TRANSIENTS

Table 15.0-3

2019-001

OPL-3									
Transient Protection Parameters Verification For									
Reload Licensing Analyses	Plant:	Duane Arnold							
	Project:	CYCLE 27							
		CYCLE 26	GE	CUSTOMER	CYCLE 27				
		RESOLVED	PROPOSED	PROPOSED	RESOLVED	CUSTOMER	CUSTOMER	GE	GE
PARAMETER DESCRIPTION	UNITS	VALUE	VALUE	VALUE	VALUE	REFERENCE	COMMENTS	REFERENCE	COMMENTS
1.3 OTHER TRANSIENT PROTECTION PARAMETERS									
A. RECIRCULATION PUMP TRIP (RPT)									
1. INSTALLED AND ENABLED?	YES/NO	YES	YES		YES		46		
2. RESPONSE TIME (DELAY) OF RPT	SEC	0.175	0.175		0.175	11,4,68	47		
B. ATWS RECIRCULATION SYSTEM TRIP (RST)									
1. INSTALLED AND ENABLED?	YES/NO	YES	YES		YES		48		
2. ATWS RST PRESSURE SETPOINT									
2.1 RLA VALUE	PSIG	1168.600	1168.600		1168.600	12	49		
2.2 TECH SPEC UPPER LIMIT	PSIG	1154.200	1154.200		1154.200	12	50		
3. ATWS RST TRIP LOCATION									
3.1 LOW FREQUENCY M-G SET (LFMG)	YES/NO	NO	NO		NO		51		
3.2 M-G SET DRIVE MOTOR BREAKERS	YES/NO	NO	NO		NO		52		
3.3 M-G SET FIELD BREAKERS	YES/NO	NO	NO		NO		53		
3.4 PUMP DRIVE MOTOR BREAKERS	YES/NO	YES	YES		YES	29	54		
4. ATWS RST FUNCTIONS									
4.1 START HPCI/HPCS/RCIC	YES/NO	NO	NO		NO		55		
4.2 LIMIT FEEDWATER FLOW	YES/NO	NO	NO		NO		56		
C. HIGH WATER LEVEL (L8) TRIP FUNCTIONS									
1. TSV CLOSURE	YES/NO	YES	YES		YES	65			
1.1 L8 TRIP LEVEL SENSOR TIME CONSTANT	SEC	1.00	1.00		1.00		68		
1.2 DELAY IN START OF TSV CLOSURE AFTER TURBINE TRIP	SEC	0.00	0.00		0.00		68		
1.3 DELAY IN START OF TCV CLOSURE AFTER TURBINE TRIP	SEC	0.10	0.10		0.10	4	24		
2. REACTOR FEEDWATER PUMP (RFP) TRIP	YES/NO	YES	YES		YES	66			
3. DIRECT REACTOR SCRAM	YES/NO	NO	NO		NO	4			
3.1 SCRAM DELAY	SEC								

Table 15.0-3

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Transient Protection Parameters Verification For

Reload Licensing Analyses

Plant: Duane Arnold

Project: CYCLE 27

PARAMETER DESCRIPTION	UNITS	CYCLE 26 RESOLVED VALUE	GE PROPOSED VALUE	CUSTOMER PROPOSED VALUE	CYCLE 27 RESOLVED VALUE	CUSTOMER REFERENCE	CUSTOMER COMMENTS	GE REFERENCE	GE COMMENTS
1.3 OTHER TRANSIENT PROTECTION PARAMETERS (CONT.)									
D. HIGH WATER LEVEL (L8) SETPOINT (NR)									
RLA VALUE	IN AVZ	571.600	571.600		571.600	13	73		
E. LOW WATER LEVEL ALARM (L4) SETPOINT									
RLA VALUE	IN AVZ	530.500	530.500		530.500	14			
F. PRESSURE MARGIN(1) RECOMMENDATION(2)	25/60 PSID	60	60		60	15	4		
(PLANTS WITH UNPIPED SPRING SAFETY VALVES (SSV))									
G. MSIV CLOSURE TIME (MINMUM)									
RLA VALUE	SEC	3.00	3.00		3.00	30			
H. DOME PRESSURE SAFETY LIMIT(3)	PSIG	1335.0	1335.0		1335.0	31			
I. THE FW LEVEL CONTROL SYSTEM WILL PREVENT A LEVEL 8 TURBINE TRIP FOR AN INADVERTENT HPCI?	YES/NO	YES	YES		YES	39			

NOTE:

(1) REFER TO THE DIFFERENCE BETWEEN THE LOWEST NOMINAL SSV SETPOINT AND THE PEAK CALCULATED TRANSIENT PRESSURE AT THE STEAMLINE

(2) THE 25 PSID RECOMMENDATION IS BASED UPON HAVING A 25 PSID PRESSURE MARGIN FOR TURBINE/GENERATOR TRIPS WITH ASSUMED BYPASS FAILURE.
THE 60 PSID RECOMMENDATION IS BASED UPON HAVING A 60 PSID PRESSURE MARGIN FOR MODERATE FREQUENCY EVENTS AND NO REQUIREMENT FOR INFREQUENT EVENTS
MODERATE FREQUENCY EVENTS ARE DEFINED AS THOSE WITH AN EXPECTED FREQUENCY OF GREATER THAN ONCE IN A PLANT LIFETIME.
INFREQUENT EVENTS ARE THOSE WITH AN EXPECTED FREQUENCY OF OCCURRENCE OF LESS THAN ONCE IN A PLANT LIFETIME (1/40 YRS) SUCH AS TURBINE AND GENERATOR TRIPS WITHOUT BYPASS.
THE 60 PSID RECOMMENDATION IS BASED UPON PROPER SSV MAINTENANCE PROCEDURES AND A RELIABLE TURBINE BYPASS SYSTEM.

(3) THIS TECH SPEC SAFETY LIMIT PROTECTS THE REACTOR COOLANT SYSTEM AGAINST OVERPRESSURIZATION.

Table 15.0-3

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OPL-3														
1.4 RV, S/RV AND SSV PARAMETERS														
					PREVIOUS CYCLE									
				RELIEF VALVE SETPOINT		SAFETY VALVE SETPOINT		CERTIFIED CAPACITY						
										CAPACITY				
			THROAT					REFERENCE		AT REF.				
	VALVE		DIA	NTSP	RLA VALUE	NTSP	RLA VALUE	PRESSURE	ACC	PRESSURE	CUSTOMER	CUSTOMER	GE	GE
#	TYPE	MFR	INCHES	PSIG	PSIG	PSIG	PSIG	PSIG	%	PPH	REFERENCE	COMMENTS	REFERENCE	COMMENTS
1	S/RV	TR	5.03	1110.0	1143.3			1080.0	3.0	829000.0	Cus Com 71	15,37,38,71		
2	S/RV	TR	5.03	1120.0	1153.6			1080.0	3.0	829000.0	Cus Com 71	16,37,38,71		
3	S/RV	TR	5.03	1130.0	1163.9			1080.0	3.0	829000.0	Cus Com 71	17,37,38,71		
4	S/RV	TR	5.03	1130.0	1163.9			1080.0	3.0	829000.0	Cus Com 71	18,37,38,71		
5	S/RV	TR	5.03	1140.0	1174.2			1080.0	3.0	829000.0	Cus Com 71	19,37,38,71		
6	S/RV	TR	5.03	1140.0	1174.2			1080.0	3.0	829000.0	Cus Com 71	20,37,38,71		
7	SSV	DS	4.27			1240.0	1277.2	1240.0	3.0	642100.0	Cus Com 72	21,37,38,71		
8	SSV	DS	4.27			1240.0	1277.2	1240.0	3.0	642100.0	Cus Com 72	22,37,38,71		
9														
10														
11														
12														
13														
14														
15														
16														
17														
18														
19														
20														
21														
22														
23														
24														
25														
	The TYPE should be labeled RV if opens electrically at the opening setpoint. Use the RELIEF VALVE SETPOINT columns.													
	The TYPE should be labeled S/RV if opens fully mechanically at the opening setpoint. Use the RELIEF VALVE SETPOINT columns.													
	The TYPE should be labeled SSV if opens mechanically with accumulation at the opening setpoint. Use the SAFETY VALVE SETPOINT columns.													
	The TYPE should be labeled DS/RV if its acts as both a RV and a SSV. Use RELIEF VALVE SETPOINT and SAFETY VALVE SETPOINT columns.													

Table 15.0-3

2019-001

OPL-3														
1.4 RV, S/RV AND SSV PARAMETERS														
						RESOLVED								
				RELIEF VALVE SETPOINT		SAFETY VALVE SETPOINT		CERTIFIED CAPACITY						
										CAPACITY				
			THROAT					REFERENCE		AT REF.				
	VALVE		DIA	NTSP	RLA VALUE	NTSP	RLA VALUE	PRESSURE	ACC	PRESSURE	CUSTOMER	CUSTOMER	GE	GE
#	TYPE	MFR	INCHES	PSIG	PSIG	PSIG	PSIG	PSIG	%	PPH	REFERENCE	COMMENTS	REFERENCE	COMMENTS
1	S/RV	TR	5.03	1110.0	1143.3			1080.0	3.0	829000.0	Cus Com 71	15,37,38,71		
2	S/RV	TR	5.03	1120.0	1153.6			1080.0	3.0	829000.0	Cus Com 71	16,37,38,71		
3	S/RV	TR	5.03	1130.0	1163.9			1080.0	3.0	829000.0	Cus Com 71	17,37,38,71		
4	S/RV	TR	5.03	1130.0	1163.9			1080.0	3.0	829000.0	Cus Com 71	18,37,38,71		
5	S/RV	TR	5.03	1140.0	1174.2			1080.0	3.0	829000.0	Cus Com 71	19,37,38,71		
6	S/RV	TR	5.03	1140.0	1174.2			1080.0	3.0	829000.0	Cus Com 71	20,37,38,71		
7	SSV	DS	4.27			1240.0	1277.2	1240.0	3.0	642100.0	Cus Com 72	21,37,38,71		
8	SSV	DS	4.27			1240.0	1277.2	1240.0	3.0	642100.0	Cus Com 72	22,37,38,71		
9														
10														
11														
12														
13														
14														
15														
16														
17														
18														
19														
20														
21														
22														
23														
24														
25														
	The TYPE should be labeled RV if opens electrically at the opening setpoint. Use the RELIEF VALVE SETPOINT columns.													
	The TYPE should be labeled S/RV if opens fully mechanically at the opening setpoint. Use the RELIEF VALVE SETPOINT columns.													
	The TYPE should be labeled SSV if opens mechanically with accumulation at the opening setpoint. Use the SAFETY VALVE SETPOINT columns.													
	The TYPE should be labeled DS/RV if its acts as both a RV and a SSV. Use RELIEF VALVE SETPOINT and SAFETY VALVE SETPOINT columns.													

Table 15.0-3

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OPL-3

1.4A ADDITIONAL RV, S/RV AND SSV PARAMETERS

PREVIOUS CYCLE

RELIEF VALVE DATA(2)

#	OOS FOR MCPR?	OOS FOR OPP?	ATTACHED TO THE MAX CAP(1) STEAMLINE?	QUALIFIED FOR OPP(3)?	PRESSURE SENSING LOCATION(4)	OPENING DELAY SEC	OPENING STROKE SEC	CLOSING NTSP PSIG	USED FOR ADS?	LLS CLOSING NTSP PSIG	LLS OPENING NTSP PSIG	CUSTOMER REFERENCE	CUSTOMER COMMENTS	GE REFERENCE	GE COMMENTS
1	NO	NO	YES	YES	LOCAL	0.20	0.20	1065.6	NO	920.9	1059.9	32, 33, 34			
2	NO	NO	NO	YES	LOCAL	0.20	0.20	1075.2	NO	915.9	1054.9	32, 33, 34			
3	NO	NO	NO	YES	LOCAL	0.20	0.20	1084.8	YES			32, 33, 34	17, 23		
4	NO	NO	NO	YES	LOCAL	0.20	0.20	1084.8	YES			32, 33, 34	18, 23		
5	NO	NO	NO	YES	LOCAL	0.20	0.20	1094.4	YES			32, 33, 34	19, 23		
6	NO	NO	YES	YES	LOCAL	0.20	0.20	1094.4	YES			32, 33, 34	20, 23		
7	NO	NO	NO									32, 33, 34	21		
8	NO	NO	NO									32, 33, 34	22		
9															
10															
11															
12															
13															
14															
15															
16															
17															
18															
19															
20															
21															
22															
23															
24															
25															

(1) VALVE IS ATTACHED TO THE STEAMLINE THAT HAS THE LARGEST TOTAL RV AND SV CAPACITY

(2) APPLIES ONLY TO RELIEF VALVES

(3) THE RELIEF VALVE IS QUALIFIED FOR APPLICATION TO ASME UPSET OVERPRESSURE EVENTS

(4) IF THE RELIEF VALVES OPENS ON LOCAL PRESSURE ENTER "LOCAL". IF OPENS ON DOME PRESSURE ENTER "DOME"

Table 15.0-3

2019-001

OPL-3															
1.4A ADDITIONAL RV, S/RV AND SSV PARAMETERS															
						RESOLVED									
					RELIEF VALVE DATA(2)										
#	OOS FOR MCPR?	OOS FOR OPP?	ATTACHED TO THE MAX CAP(1) STEAMLINE?	QUALIFIED FOR OPP(3)?	PRESSURE SENSING LOCATION(4)	OPENING DELAY SEC	OPENING STROKE SEC	CLOSING NTSP PSIG	USED FOR ADS?	LLS CLOSING NTSP PSIG	LLS OPENING NTSP PSIG	CUSTOMER REFERENCE	CUSTOMER COMMENTS	GE REFERENCE	GE COMMENTS
1	NO	NO	YES	YES	LOCAL	0.20	0.20	1065.6	NO	920.9	1059.9	32, 33, 34			
2	NO	NO	NO	YES	LOCAL	0.20	0.20	1075.2	NO	915.9	1054.9	32, 33, 34			
3	NO	NO	NO	YES	LOCAL	0.20	0.20	1084.8	YES			32, 33, 34	17, 23		
4	NO	NO	NO	YES	LOCAL	0.20	0.20	1084.8	YES			32, 33, 34	18, 23		
5	NO	NO	NO	YES	LOCAL	0.20	0.20	1094.4	YES			32, 33, 34	19, 23		
6	NO	NO	YES	YES	LOCAL	0.20	0.20	1094.4	YES			32, 33, 34	20, 23		
7	NO	NO	NO									32, 33, 34	21		
8	NO	NO	NO									32, 33, 34	22		
9															
10															
11															
12															
13															
14															
15															
16															
17															
18															
19															
20															
21															
22															
23															
24															
25															

(1) VALVE IS ATTACHED TO THE STEAMLINE THAT HAS THE LARGEST TOTAL RV AND SV CAPACITY

(2) APPLIES ONLY TO RELIEF VALVES

(3) THE RELIEF VALVE IS QUALIFIED FOR APPLICATION TO ASME UPSET OVERPRESSURE EVENTS

(4) IF THE RELIEF VALVES OPENS ON LOCAL PRESSURE ENTER "LOCAL". IF OPENS ON DOME PRESSURE ENTER "DOME"

Table 15.0-3

2019-001

OPL-3									
Transient Protection Parameters Verification For									
Reload Licensing Analyses	Plant:	Duane Arnold							
	Project:	CYCLE 27							
		CYCLE 26	GE	CUSTOMER	CYCLE 27				
		RESOLVED	PROPOSED	PROPOSED	RESOLVED	CUSTOMER	CUSTOMER	GE	GE
PARAMETER DESCRIPTION	UNITS	VALUE	VALUE	VALUE	VALUE	REFERENCE	COMMENTS	REFERENCE	COMMENTS
1.5 PRINCIPAL PLANT EQUIPMENT PARAMETERS									
A. MAIN STEAMLINE									
1. AVERAGE LENGTH (VESSEL TO INBOARD MSIV)	FT	96.2	96.2		96.2	17			
2. COMBINED VOLUME (VESSEL TO INBOARD MSIV)	FT**3	675.2	675.2		675.2	17			
3. AVERAGE LENGTH (INBOARD MSIV TO TSV)	FT	157.5	157.5		157.5	17			
4. COMBINED VOLUME (INBOARD MSIV TO TSV)	FT**3	1105.8	1105.8		1105.8	17			
B. BYPASS STEAMLINE									
1. AVERAGE LENGTH (HEADER/TAP TO BPV)	FT	100.8	100.8		100.8	17			
2. COMBINED VOLUME (HEADER/TAP TO BPV)	FT**3	92.2	92.2		92.2	17			
C. NUMBER OF TURBINE DRIVEN RFP	#	0	0		0				
D. NUMBER OF MOTOR DRIVEN RFP	#	2	2		2	66			
E. RFP COMBINATIONS SUPPORTING RATED POWER									
1. COMBINATION 1	#TD	0	0		0				
	#MD	2	2		2	66			
2. COMBINATION 2	#TD								
	#MD								
F. MAXIMUM RFP RUNOUT FLOW AT FW DESIGN PRESSURE									
1. FW DESIGN PRESSURE	PSIG	1060.0	1060.0		1060.0	4			
2. MAXIMUM RFP RUNOUT FLOW (COMBINATION)	% RATED	114.9	114.9		114.9		5, 35		
G. MAXIMUM RFP RUNOUT FLOW AT RLA DOME PRESSURE									
1. RLA DOME PRESSURE	PSIG	1025.0	1025.0		1025.0	43			
2. MAXIMUM RFP RUNOUT FLOW (COMBINATION)	% RATED	116.8	116.8		116.8		5, 35		
H. DESIGN CAPACITY OF HIGH PRESSURE SYSTEMS									
(AT RATED DOME PRESSURE)									
1. RCIC	GPM	400.0	400.0		400.0	40, 52			
2. HPCI	GPM	3000.0	3000.0		3000.0	53			
3. HPCS	GPM								
4. FOR PLANTS WITH HPCI, PERCENTAGE OF 1.5.H.2 FLOW TO FW SPARGI	%	100.0	100.0		100.0	42,53			
I. MINIMUM TEMPERATURE OF CONDENSATE STORAGE TANK WATER	DEG F	40.0	40.0		40.0		42		

Table 15.0-3

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OPL-3									
Transient Protection Parameters Verification For									
Reload Licensing Analyses	Plant:	Duane Arnold							
	Project:	CYCLE 27							
		CYCLE 26	GE	CUSTOMER	CYCLE 27				
		RESOLVED	PROPOSED	PROPOSED	RESOLVED	CUSTOMER	CUSTOMER	GE	GE
PARAMETER DESCRIPTION	UNITS	VALUE	VALUE	VALUE	VALUE	REFERENCE	COMMENTS	REFERENCE	COMMENTS
1.6 TURBINE CONTROL COMPONENT CHARACTERISTICS									
A. NUMBER OF BYPASS VALVES (BPV)	#	2	2		2	73			
B. TOTAL BPV CAPACITY (AT RATED TURBINE THROTTLE PRESSURE)	% RATED	20.60	20.60		20.60	61			
C. BPV DELAY (BPV FAST OPENING)	SEC	0.10	0.10		0.10	4			
(FROM EVENT INITIATION TO START OF BPV OPENING)									
D. TOTAL RESPONSE TIME OF BPV (80% of rated BPV flow)	SEC	0.30	0.30		0.30	4			
E. TYPE OF TURBINE CONTROL	MHC/EHC	EHC	EHC		EHC	61			
F. MODE OF TCV OPERATION (FULL ARC/PARTIAL)	FA/PA	3x1	3x1		3x1	74			
G. NUMBER OF TURBINE CONTROL VALVES (TCV)	#	4	4		4	62			
H. TCV CHARACTERISTIC LIFT CURVE	FIG. 1.6-1	see Figure 1.6-1							
I. TURBINE STEAM FLOW CHARACTERISTICS									
TOTAL STEAM FLOW vs. TCV POSITION	FIG. 1.6-2	see Figure 1.6-2							
J. RESPONSE TIME OF TCV FAST CLOSURE (MINIMUM)	MSEC	150.00	150.00		150.00	4	6		
(FULL STROKE/EHC PLANTS ONLY)									
K. RESPONSE TIME OF TSV FAST CLOSURE (MINIMUM)	MSEC	100.00	100.00		100.00	4			
L. RESPONSE TIME OF TCV SERVO CLOSURE (FULL STROKE EHC PLANTS)									
1. TCV 1 FULL STROKE	SEC	2.50	2.50		2.50	4	70		
2. TCV 2 FULL STROKE	SEC	2.50	2.50		2.50	4	70		
3. TCV 3 FULL STROKE	SEC	2.50	2.50		2.50	4	70		
4. TCV 4 FULL STROKE	SEC	2.50	2.50		2.50	4	70		

Table 15.0-3

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OPL-3									
Transient Protection Parameters Verification For									
Reload Licensing Analyses	Plant:	Duane Arnold							
	Project:	CYCLE 27							
		CYCLE 26 RESOLVED VALUE	GE PROPOSED VALUE	CUSTOMER PROPOSED VALUE	CYCLE 27 RESOLVED VALUE	CUSTOMER REFERENCE	CUSTOMER COMMENTS	GE REFERENCE	GE COMMENTS
PARAMETER DESCRIPTION	UNITS								
1.6 TURBINE CONTROL COMPONENT CHARACTERISTICS (CONT.)									
M. TCV POSITION (MEASURED AT RATED POWER)									
1. FULL ARC									
1.1 TCV POSITION (VALVES 1-4)	% OPEN								
1.2 TURBINE THROTTLE PRESSURE	PSIG								
2. PARTIAL ARC									
2.1 TCV 1 POSITION	% OPEN	100.00	100.00		100.00	83	78		
2.2 TCV 2 POSITION	% OPEN	100.00	100.00		100.00	83	78		
2.3 TCV 3 POSITION	% OPEN	100.00	100.00		100.00	83	78		
2.4 TCV 4 POSITION	% OPEN	40.00	40.00		40.00	83	7,78		
2.5 TURBINE THROTTLE PRESSURE	PSIG	968.00	968.00		968.00		67,77		
N. MAXIMUM COMBINED FLOW LIMITER SETPOINT	% RATED	125.0	125.0		125.0	36			
1.7 LOAD REJECTION/RECIRCULATION SYSTEM LOGIC									
A. RECIRCULATION SYSTEM (RS) POWER SOURCE									
(MAIN GENERATOR (MG)/OFF-SITE (OS))									
1. LOOP A	MG/OS	MG	MG		MG	56	57		
2. LOOP B	MG/OS	MG	MG		MG	56	57		
B. IS MAIN GENERATOR POWER SOURCE									
INTERRUPTED FOLLOWING LOAD REJECTION ?	YES/NO	YES	YES		YES		28		
C. IF YES:									
1. TRANSFERRED OR SHED ?	TRANS/SHED	SHED	SHED		SHED		28		
2. RESPONSE TIME OF TRANSFER OR SHEDDING	SEC	0.23	0.23		0.23	57,58	28,58		
(FROM EVENT INITIATION TO COMPLETION									
OF TRANSFER OR SHEDDING)									
D. MAX CORE POWER WHERE A TCV FAST CLOSURE WILL NOT									
OCCUR DUE TO THE PLU SETTING (EHC PLANTS) OR OTHER SETTINGS	% RATED	40.00	40.00		40.00	59	59		

Table 15.0-3

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OPL-3									
Transient Protection Parameters Verification For									
Reload Licensing Analyses	Plant:	Duane Arnold							
	Project:	CYCLE 27							
		CYCLE 26	GE	CUSTOMER	CYCLE 27				
		RESOLVED	PROPOSED	PROPOSED	RESOLVED	CUSTOMER	CUSTOMER	GE	GE
PARAMETER DESCRIPTION	UNITS	VALUE	VALUE	VALUE	VALUE	REFERENCE	COMMENTS	REFERENCE	COMMENTS
2. PARAMETERS OF SPECIAL INTEREST AND/OR RELATED TO THE									
VALIDATION OF THE FSAR AND THE RELOAD ANALYSES									
2.1 INITIAL OPERATING CONDITIONS									
A. VESSEL WATER LEVEL (WIDE RANGE AT RATED POWER)	IN AVZ	535.50	535.50		535.50	54,55	45		
2.2 SCRAM PARAMETERS									
A. FLOW REFERENCE SCRAM:									
1. APRM FLUX SCRAM/THERMAL POWER SCRAM?	AFS/TPS	AFS	AFS		AFS	45,46	60		
2. SAFETY GRADE?	YES/NO	YES	YES		YES	45,46			
3. SCRAM SETPOINT INTERCEPT AT ZERO DRIVE FLOW	% RATED	72.18	72.18		72.18	6	8		
ANALYSIS BASIS									
4. RESPONSE TIME OF DRIVE FLOW SENSOR	SEC	1.00	1.00		1.00		62		
B. LOW WATER LEVEL (L3) INSTRUMENTATION	NR/WR	NR	NR		NR		63		
(NARROW RANGE/WIDE RANGE)									
C. L3 SCRAM SETPOINT	IN AVZ	507.00	507.00		507.00	20,55	64		
ANALYSIS BASIS									
D. L3 TRIP LEVEL SENSOR TIME CONSTANT	SEC	1.00	1.00		1.00	60	65		
E. TURBINE POWER/SCRAM BYPASS SETPOINT	% RATED	26.00	26.00		26.00	21	66		
ANALYSIS BASIS									
F. LOW CONDENSER VACUUM SCRAM SETPOINT									
1. NTSP	IN HG VAC						9		
2. ANALYSIS BASIS	IN HG VAC						9		

Table 15.0-3

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OPL-3									
Transient Protection Parameters Verification For									
Reload Licensing Analyses	Plant:	Duane Arnold							
	Project:	CYCLE 27							
		CYCLE 26	GE	CUSTOMER	CYCLE 27				
		RESOLVED	PROPOSED	PROPOSED	RESOLVED	CUSTOMER	CUSTOMER	GE	GE
PARAMETER DESCRIPTION	UNITS	VALUE	VALUE	VALUE	VALUE	REFERENCE	COMMENTS	REFERENCE	COMMENTS
2.3 OTHER TRANSIENT PROTECTION PARAMETERS									
A. L8 TRIP FUNCTIONS									
1. RCIC TRIP	YES/NO	YES	YES		YES	37			
2. RCIC AUTO RESET	YES/NO	YES	YES		YES	37	10		
3. HPCI/HPCS TRIP	YES/NO	YES	YES		YES	38			
4. HPCI/HPCS AUTO RESET	YES/NO	YES	YES		YES	38	10		
5. LPCI/LPCS TRIP	YES/NO	NO	NO		NO				
B. L3 TRIP SETPOINT ANALYSIS BASIS	IN AVZ	507.00	507.00		507.00	20			
C. L3 TRIP FUNCTIONS									
1. SCRAM	YES/NO	YES	YES		YES	81, 82			
2. RECIRCULATION SYSTEM DRIVE FLOW RUNBACK	YES/NO	NO	NO		NO	79, 80			
3. ISOLATION (EXCEPT MSIV)	YES/NO	YES	YES		YES		29, 33		
4. RECIRCULATION SYSTEM PUMP MOTOR TRIP	YES/NO	NO	NO		NO				
D. L2 TRIP SETPOINT ANALYSIS BASIS	IN AVZ	447.30	447.30		447.30	22			

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OPL-3									
Transient Protection Parameters Verification For									
Reload Licensing Analyses	Plant:	Duane Arnold							
	Project:	CYCLE 27							
		CYCLE 26	GE	CUSTOMER	CYCLE 27				
		RESOLVED	PROPOSED	PROPOSED	RESOLVED	CUSTOMER	CUSTOMER	GE	GE
PARAMETER DESCRIPTION	UNITS	VALUE	VALUE	VALUE	VALUE	REFERENCE	COMMENTS	REFERENCE	COMMENTS
2.3 OTHER TRANSIENT PROTECTION PARAMETERS (CONT.)									
E. L2 TRIP FUNCTIONS									
1. RECIRCULATION SYSTEM LFMG SET TRIP	YES/NO	NO	NO		NO				
2. RECIRCULATION SYSTEM DRIVE MOTOR TRIP	YES/NO	YES	YES		YES	86,87			
3. MSIV CLOSURE	YES/NO	NO	NO		NO	77	76		
4. START HPCI/HPCS	YES/NO	YES	YES		YES	75			
5. START RCIC	YES/NO	YES	YES		YES	76			
6. INITIATE ADS (PARTIAL LOGIC)	YES/NO	NO	NO		NO				
7. INITIATE LPCI/LPCS (PARTIAL LOGIC)	YES/NO	YES	YES		YES	78			
F. L1 TRIP SETPOINT ANALYSIS BASIS	IN AVZ	350.00	350.00		350.00	23			
G. L1 TRIP FUNCTIONS									
1. INITIATE ADS (PARTIAL LOGIC)	YES/NO	YES	YES		YES	88			
2. INITIATE LPCI/LPCS (PARTIAL LOGIC)	YES/NO	YES	YES		YES	78			
3. MSIV CLOSURE	YES/NO	YES	YES		YES	77	76		
H. LOW TURBINE THROTTLE PRESSURE SETPOINT	PSIG	800.0	800.0		800.0	24	11		
(MSIV FAST CLOSURE)									
I. HIGH DOME PRESSURE SETPOINT/ISOLATION CONDENSERS	PSIG								
(INITIATES OPERATION) ANALYSIS BASIS									
J. TOTAL CAPACITY/ISOLATION CONDENSERS	%RATED								

Table 15.0-3

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OPL-3									
Transient Protection Parameters Verification For									
Reload Licensing Analyses	Plant:	Duane Arnold							
	Project:	CYCLE 27							
		CYCLE 26	GE	CUSTOMER	CYCLE 27				
		RESOLVED	PROPOSED	PROPOSED	RESOLVED	CUSTOMER	CUSTOMER	GE	GE
PARAMETER DESCRIPTION	UNITS	VALUE	VALUE	VALUE	VALUE	REFERENCE	COMMENTS	REFERENCE	COMMENTS
2.4 RELIEF AND BYPASS VALVE PARAMETERS									
A. AVAILABILITY OF BPV FAST OPENING MODE	YES/NO	YES	YES		YES		12		
(LOSS OF AUXILIARY POWER)									
B. AVAILABILITY OF POWER ACTUATED RELIEF VALVES	YES/NO	N/A	N/A		N/A	4			
(FIRST LIFT) (LOSS OF AUXILIARY POWER)									
2.5 PLANT EQUIPMENT PARAMETERS									
A. RATE OF LOSS OF CONDENSER VACUUM(FAILED CW PUMPS)	IN HG/SEC	2.00	2.00		2.00	4			
B. LOW CONDENSER VACUUM PROTECTION SETPOINTS									
1. INITIATE TSV FAST CLOSURE ANALYSIS BASIS	IN HG VAC	20.0	20.0		20.0	25,63			
2. INITIATE BPV AND MSIV FAST CLOSURE ANALYSIS BASIS	IN HG VAC	13.0	13.0		13.0	4, 25			
C. RATE OF CHANGE OF RECIRCULATION M-G SET	%/SEC	3	3		3	64	30		
COUPLER SCOOP TUBE (MAXIMUM)									
D. FW HEATER BYPASS VALVE STROKE TIME (MINIMUM)	MIN	0.27	0.27		0.27	26, 27	13		
2.6 LOSS OF STATOR COOLING (LOSC) PARAMETERS									
A. TURBINE TRIP AFTER LOSC	YES/NO	NO	NO		NO	72		1	
B. TURBINE-GENERATOR RUNBACK RATE	%/SEC	0.357	0.357		0.357		69	1	1
C. FINAL TURBINE CAPACITY FOLLOWING RUNBACK	% RATED	25.0	25.0		25.0	72		1	
D. RECIRCULATION FLOW PARAMETERS									
1. RECIRCULATION PUMP TRIP FOLLOWING LOSC	YES/NO	NO	NO		NO	72		1	
2. RECIRCULATION RUNBACK FOLLOWING LOSC	YES/NO	NO	NO		NO	72		1	
2.1 RECIRCULATION RUNBACK RATE	%/SEC								
2.2 FINAL CORE FLOW AFTER RECIRC RUNBACK	% RATED								

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OPL-3									
Transient Protection Parameters Verification For									
Reload Licensing Analyses	Plant:	Duane Arnold							
	Project:	CYCLE 27							
		CYCLE 26	GE	CUSTOMER	CYCLE 27				
		RESOLVED	PROPOSED	PROPOSED	RESOLVED	CUSTOMER	CUSTOMER	GE	GE
PARAMETER DESCRIPTION	UNITS	VALUE	VALUE	VALUE	VALUE	REFERENCE	COMMENTS	REFERENCE	COMMENTS
3.0 OTHER PARAMETERS OF SPECIFIC INTEREST (up to 80 characters)									
1. LPCI loop select recirculation riser differential pressure nominal setpoint									
2.3.E.8: Isolate RWCU (See Customer Comment 34)	YES/NO	YES	YES		YES		34		
2.3.K: L4 Trip Function (with less than 2 feed pumps in operation)									
2.3.K.1: Recirc System Drive Flow Runback	YES/NO			YES	YES	79,80,85			
2.3.K.2: L4 Trip Setpoint Analysis Basis	IN AVZ			530.5	530.5	79,80,85			

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	OPL-3
	Transient Protection Parameters Verification For
	Duane Arnold
	Reload Licensing Analyses
	GE REFERENCES
1	0000-0156-9405-R0, Evaluation of Loss of Stator Water Cooling for Duane Arnold Energy Center
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	OPL-3
	Transient Protection Parameters Verification For
	Duane Arnold
	Reload Licensing Analyses
	GE COMMENTS
1	The runback rate is based on the input provided in Reference 1 of GE Report 0000-0156-9405-R0 (GE Reference 1). No plant data was available on the actual runback rate; instead, the rate is calculated based on trip avoidance during the load runback.
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	OPL-3
	Transient Protection Parameters Verification For
	Duane Arnold
	Reload Licensing Analyses
	CUSTOMER REFERENCES
1	DAEC UFSAR, Figure 15.0-3, Reactor Heat Balance - Rated Power
2	Deleted
3	DAEC UFSAR, Figure 15.0-4, Reactor Heat Balance - 102% Power
4	TDP-0087, Revision 9, OPL-3 Design Guide.
5	Deleted
6	DAEC document CAL-E02-002, rev. 1, APRM setpoint calculation.
7	DAEC document CAL-E93-036, Revision 2, RPV Hi pressure setpoint calculation.
8	DAEC document CAL-E94-005, rev. 3, MSIV 10% closed setpoint calculation.
9	DAEC document CAL-E95-008, rev. 2, TSV 10% closed setpoint calculation.
10	DAEC document CAL-E94-008, rev. 2, TCV fast closure setpoint calculation.
11	DAEC UFSAR 7.2.1.2.3.
12	DAEC document CAL-E92-024, rev. 0, RPV Hi pressure (ATWS RPT/ARI) setpoint calculation.
13	DAEC document CAL-E93-025, rev. 1, RPV Hi level (L8) setpoint calculation.
14	DAEC document ARP 1C05A(D-1), rev. 86.
15	NG-17-0252 Cycle 27, Reload 26 OPL3 Reference 15 Vendor Letter
16	Deleted
17	DRF L12-00647-1 Duane Arnold OPL3/Transient Basedeck - Supplement 1
18	Deleted
19	Deleted
20	DAEC document CAL-E92-010, rev. 5, RPV Lo level (L3) setpoint calculation.
21	DAEC document CAL-E93-021, rev. 4, HPT inlet pressure for 26% trip bypass setpoint calculation.
22	DAEC document CAL-E93-026, rev. 3, RPV Lo-Lo level (L2) setpoint calculation.
23	DAEC document CAL-E93-016, rev. 3, RPV Lo-Lo-Lo level (L1) setpoint calculation.
24	DAEC document CAL-E93-003, rev. 5, Main Steam Lo pressure setpoint calculation.
25	DAEC document CAL-E93-004, rev. 2, Main Condenser vacuum setpoint calculation.
26	DAEC document BECH-E200<1473>, rev. 2, MOV data list for MO1473.
27	DAEC document BECH-E200<1546>, rev. 1, MOV data list for MO1546.
28	DAEC UFSAR 7.2.1.3.
29	DAEC UFSAR 7.2.3.2
30	DAEC Tech Specs SR 3.6.1.3.5
31	DAEC Tech Specs 2.1.2
32	DAEC UFSAR Table 5.2-1
33	DAEC document CAL-E94-003, Rev. 2, LLS Pressure Switch Actuation Setpoint Calculation
34	DAEC document APED-B21-074<1>, rev. 1, Nuclear Boiler System Sheet
35	Deleted
36	DAEC UFSAR, 15.1.7
37	DAEC UFSAR, 7.4
38	DAEC UFSAR, 7.3
39	DAEC UFSAR, 15.1.1.3
40	DAEC UFSAR Table 5.4-3
41	Deleted
42	DAEC UFSAR, 6.3
43	APED-A41-029 REAC SYS HEAT BALANCE RATED (TASK T0100) (GE document GE-NE-A22-00100-01-01, Rev. 0, DAEC Asset Enchancement Program, Task T0100: Nominal Reactor Heat Balance.)
44	APED-A41-030 REAC SYS HEAT BALANCE 102% RATED AND OFF RATED (TASK T0101) (GE document GE-NE-A22-00100-02-01, Rev. 0, DAEC Asset Enchancement Program, Task T0101: Offrated Reactor Heat Balance.)
45	APED-C51-049 NEUTRON MON SYSTEM (GE Document No. 22A1473), Rev 8
46	APED-C71-020 REAC PROTECTION SYSTEM (GE Document No. 22A1382), Rev 4
47	STP 3.3.1.1-18 MSIV LIMIT SWITCH CALIBRATION AND INSPECTION, Rev 22
48	APED-B21-3379-001 PILOT OPERATED RELIEF VALVE Sheet 3, Rev 12; Sheet 4, Rev 14; and Sheet 5 Rev 12

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	OPL-3
	Transient Protection Parameters Verification For
	Duane Arnold
	Reload Licensing Analyses
	CUSTOMER REFERENCES
49	Duane Arnold Energy Center Asset Enhancement Program Task T0315: SRV Setpoint Tolerance Monitoring Program Review, R1, June 2000
50	Deleted.
51	Deleted.
52	Duane Arnold Energy Center Asset Enhancement Program, Project Task Report T0309: Reactor Core Isolation Cooling System, GE-NE-A22-00100-16-01, October 2000
53	Duane Arnold Energy Center Asset Enhancement Program, Project Task Report T0404, High Pressure Coolant Injection System, GE-NE-A22-00100-26-01, October 2000.
54	OI644 – Operating Instructions for Condensate and Feedwater Systems, Revision 178. Section 3.5
55	APED-B11-084, Sheet 1, Reactor Assembly, Revision 9
56	BECH E001, Sheet 1, Single Line Diagram Station Connections, Revision 40.
57	APED-B31-6073-17, Design Specification Data Sheet for the Recirculation Pump Trip System, Revision 2.
58	APED –C71-022, Design Specification for the Reactor Protection System, Revision 2.
59	Maintenance Procedure GMP-TEST-39, Turbine Load Unbalance Functional Test, Revision 5.
60	APED-C71-023, Design Specification for the Reactor Protection System. Revision 7.
61	Duane Arnold Energy Center Asset Enhancement Program, Project Task Report T0502: Pressure Control System, GE-NE-A22-00100-35-01, August 2000.
62	BECH-M103 <Sheet 1>, P&ID MAIN STEAM TURBINE STOP AND CONTROL VALVES, Revision 44
63	BECH-M424, IDS PRESSURE SWITCHES, Rev 67.
64	EC 156019, “Recirc MG Set Actuator Replacement”.
65	M001-N32-009, Sheet 2, Rev 20, SCHEMATIC WIRING DIAGRAM ALARM AND TRIP (EHC).
66	APED-C31-008, Sheet 1, ELEM DIAG FEEDWATER CONTROL SYSTEM, Revision 35.
67	Deleted.
68	GE-NE-A22-00100-60-01, Revision 0, August 2000, DAEC Asset Enhancement Program, Task T0902: Anticipated Transient Without Scram (ATWS).
69	APED-C11-065, CONTROL ROD DRIVE SYS, Revision 9.
70	APED-B21-013, RELIEF VALVES MAIN STEAM PIPING, Revision 6.
71	APED-C71-021 Rev.5 Reactor Protection System Design Specification (GE 22A1382AU)
72	NEE Letter NF-13-065, Duane Arnold Energy Center Loss of Stator Cooling Evaluation DIR.
73	DAEC Tech Specs Bases B3.7.7
74	DAEC UFSAR 10.2.2
75	APED-E41-006, Sheet 3, Rev 27, HIGH PRESSURE COOLANT INJECTION SYSTEM ELEMENTARY DIAGRAM
76	APED-E51-009, Sheet 2, Rev 25, ELEM DIAG RCIC SYS
77	APED-A71-003, Sheet 6, Rev 50, ELEM DIAG NSSSS <B21>
78	APED-E11-007, Sheet 4, Rev 41, RESIDUAL HEAT REMOVAL SYS ELEM DIAG
79	APED-C31-008, Sheet 4, Rev 39, ELEM DIAG FEEDWATER CONTROL SYS
80	APED-C31-022, Sheet 11, Rev 3, Feedwater Control System (Constant Speed Motor Driven Pump)
81	APED-C71-004, Sheet 6, Rev 33, REAC PROTECT SYS ELEM DIAG
82	APED-C71-004, Sheet 7, Rev 34, REAC PROTECT SYS ELEM DIAG
83	EC 155793 HP TURBINE REPLACEMENT
84	Deleted.
85	APED-C31-021 FEEDWATER CONTROL SYSTEM, Revision 5
86	BECH-E120, Sheet 20A, Rev 2, ATWS SYSTEM CHANNEL A SCHEME 1Q304
87	BECH-E120, Sheet 21A, Rev 7, ATWS SYSTEM CHANNEL B SCHEME 1Q404
88	APED-B21-018, Sheet 2, Rev 21, ELEM DIAG AUTO DEPRESSURE SYS
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	OPL-3
	Transient Protection Parameters Verification For
	Duane Arnold
	Reload Licensing Analyses
	CUSTOMER COMMENTS
1	Based on 57 * (8.554/8.352)**2
2	Deleted
3	This Nominal Trip Setpoint value from the setpoint calculation is higher than the Nominal Trip Setpoint value of 1055 psig in ARP 1C05B(C-4).
4	The 60 psid criterion will be used for domains with operable turbine bypass valves. The 25 psid criterion will be used for domains with inoperable turbine bypass valves.
5	Installed hardware not capable of delivering this flow rate (condensate pumps are limiting). However, the transient analyses assume this capability for conservatism in the "FWCF - Max Demand" event.
6	Letter, NG-01-0259 from B. Hopkins to T. Orr identifies 150 msec for TCV 1, TCV2 and TCV3 from the full open position and 75 msec for TCV 4 from the full open position.
7	Recent operational data at 1912 MWth show that CV-4 reaches a maximum position of approximately 45 percent open, however, normal position is approximately 40 percent.
8	This is the Analytical Limit. The Allowable Value is 67.70% and the Nominal Trip Setpoint is 65.36%.
9	Not a direct scram.
10	HPCI/RCIC restart waits for L2.
11	This is the Analytical Limit. The Allowable Value is 821 psig and the Nominal Trip Setpoint is 850 psig.
12	BPV accumulators will maintain BPV operation for a short period of time after a loss of EHC power.
13	Minimum value of 0.27 minutes based on 3.5 inch nominal valve size and 13 inches/minute stem speed. Note that this valve only bypasses FW Heaters #1 and #2 and LP drains, not the whole system.
14	deleted
15	PSV 4407, Steamline D, LL set
16	PSV 4401, Steamline A, LL set
17	PSV 4400, Steamline A, ADS
18	PSV 4402, Steamline B, ADS
19	PSV 4405, Steamline C, ADS
20	PSV 4406, Steamline D, ADS
21	PSV 4403, Steamline B
22	PSV 4404, Steamline C
23	Closing NTSP is set at 96% of the NTSP defined in section 1.4
24	From TDP-0087 Rev. 9: " “Some plants initiate a TCV fast closure nearly simultaneous with TSV closure. If this trip is not delayed (within 0.1 sec) the TCV closing event could be more limiting than the TSV closing event. This effect is insignificant for partial arc plants since the full TCV stroke time is longer than the TSV stroke time.”
25	Deleted
26	Per OTH026975, DAEC document CAL-E93-036, Rev. 2, RPV Hi pressure setpoint calculation, the NTSP for vessel dome pressure scram setpoint has been changed to 1067.0 psig.
27	Deleted
28	Non-essential busses are tripped on opening of H, I breaker or Generator Lock-out. Open system transfer would occur, but recirc pumps will not auto-restart. Recirc pumps are interrupted on main generator load reject. Pumps are shed. Note: This assumes EOC RPT trip function is out of service. EOC RPT trip would occur on TCV fast closure and/or turbine stop valve less than 90 percent open.
29	Reactor Water Clean-up isolates on Level 2.
30	Runout to max speed is 35 seconds for full stroke (0 - 100%). Runback to min speed is 27 seconds (100% - 20%). Bounding time is 3 percent per second.
31	Deleted
32	Deleted
33	Page 20, Item 2.3.C.3: This item excludes MSIV and Reactor Water Cleanup (RWCU).
34	Page 21, Item 2.3.E: Propose adding new line 8: Isolate RWCU; Units YES/NO; YES for Customer Proposed -> has been moved to Section 3.0.
35	Values are enveloped by existing and replacement feed pumps being installed under EC 156026.
36	Deleted

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	OPL-3
	Transient Protection Parameters Verification For
	Duane Arnold
	Reload Licensing Analyses
	CUSTOMER COMMENTS
37	Per TS 3.4.3 Bases: SRV/SV setpoints are established to ensure that the ASME Code limit on peak reactor pressure is satisfied. The ASME code specifications require the lowest safety valve setpoint to be at or below vessel design pressure (1250 psig) and the highest safety valve to be set so that the total accumulated pressure does not exceed 110% of the design pressure for overpressure conditions. The transient evaluations in the UFSAR are based on these setpoints, but also include the additional uncertainties of 3% (high) error in the nominal setpoint to provide an added degree of conservatism.
38	UFSAR Table 5.2-1 provides valve type, number of valves, set pressure (nominal) and capacity of 103% of set pressure.
39	Based on 1040 psia - 14.7 psi/atm
40	102% of nominal power 1912 MWt
41	Based on 1055 psia - 14.7 psi/atm
42	The Condensate Storage Tanks are heated by the Condensate Storage Tank Heater 1E015 during cold weather operations to prevent the water in the tanks from freezing.
43	The core flow measurement system calibration was completed in December 2012 following startup from RFO23. Data was obtained using OSI PI to take snapshot data for 30 minutes each day at 0000 for the month of January 2013 (which is immediately following the core flow calibration with rated flow and power conditions). Data was excluded on 1/10/13 and 1/27/13 when the plant was not at rated conditions between 0000 and 0030 that day (Ref 1). Data was compiled on a spreadsheet. The average values for the month were used as the customer proposed value. This data is from Cycle 24 at a cycle exposure of 1455.5 MWD/ST on 01/31/13. Recirc speeds are taken at the pump.
44	Flow Control Valve Position Loop A and Loop B is N/A for DAEC since DAEC is an MG Set plant and does not use flow control valves.
45	Per Ref. 54, normal vessel water level is maintained between 186 in and 195 in TAF. Per Ref 55, TAF is 344.94 in above vessel 0 value. Assume the mid-point of the desired water level range (190.5 in) and add the TAF elevation, the normal reactor water level would be as follows: 190.5 in + 344.94 in = 535.44 in AVZ (above Vessel zero). The current 535.5 in AVZ is acceptable.
46	RPT will occur on load reject or turbine trip via RPT breakers 1A501, 1A502, 1A601, and 1A602.
47	Per OPL-3 Procedure Specification (Ref.4), the original GEH Design Spec delay time is ≤ 0.175 seconds for MG Set Plants. The procedure specification states that the maximum specification value should be used as a conservative basis and these should be consistent with plant Technical Specifications. Response time of 0.175 seconds is proposed. Same value as used in past OPL-3 submittals.
48	ATWS trip function will activate Recirc Pump Trip when vessel dome pressure exceeds the specified setpoint.
49	Per the GEH OPL-3 Procedure Specification (Ref. 4), the nominal trip set point (NTSP) is used for all applicable RLA transient events. A utility may elect to use the most limiting setpoint value for licensing analysis to include the TSL, AV, or AL. Per Customer Reference 12 on the Cycle 24 OPL-3, DAEC CAL-E92-024 (Ref. 12) was used to provde the requested value. Based on the Conclusions and Recommendations of Ref. 12, the Analytical Limit (AL) is used for the requested RLA Value.
50	Per the GEH OPL-3 Procedure Specification (Ref. 4), this requested setpoint value is the Tech Spec upper limit. Per Ref. 12, the TSUL value is ≤ 1154.2 psig.
51	DAEC uses MG Sets for varying Recirc Pump speed. LFMG sets are used in flow control plants (BWR 5/6). DEAC does not have LFMGs installed.
52	The DAEC ATWS recirculation system trip is not completed/performed at the MG Set Drive Motor Breaker.
53	The DAEC ATWS recirculation system trip is not completed/performed at the MG Set Field Breaker.
54	The DAEC ATWS recirculation system trip is completed/performed at the Pump Drive Motor Breakers 1A501, 1A502, 1A601, and 1A602 RPT Breakers.
55	The DAEC design does not initiate an auto start of any of these systems.
56	DAEC design does not limit feedwater flow following the initiation of an ATWS high dome pressure trip of the recirculation system.

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	OPL-3
	Transient Protection Parameters Verification For
	Duane Arnold
	Reload Licensing Analyses
	CUSTOMER COMMENTS
57	As shown in Ref. 56, the DAEC MG Set Drive motor breakers are supplied power from 4160 V busses 1A1 and 1A2. With the generator on line, the 1A1 and 1A2 busses are supplied power from the main generator via the Auxiliary transformer. Both Loop A and B MG Sets are supplied by the main generator (MG) as described.
58	Per Ref. 57, the interruption time for the Recirculation pump trip (RPT) breakers is 135 milliseconds. Per Ref 12 below the TCV Fast closure trip sensor response time is 30 milliseconds. This accounts for a total of 165 milliseconds. The value provided in the Cycle 23 OPL-3 is 230 milliseconds. Per Ref 58 Section 4.1.8, a maximum operating time from opening of a sensor contact to and including opening of the contacts on the main trip actuators be less than 50 milliseconds. If this time is conservatively added to the values above, a total response time of 215 milliseconds would occur. The current 230 milliseconds appears to be a good conservative value.
59	Per Ref. 59, the power load unbalance circuitry adjusts as needed and verifies that the PLU will activate at above 40% power (core power). At core powers of 40% and below, the TCV fast closure will not occur due to the setup of the PLU unit.
60	DAEC does not have the Thermal Power Scram (TPS) function. Instead, DAEC has the APRM Flux Scram (AFS) function referenced to drive flow.
61	Deleted.
62	GE OPL-3 Procedure Specification (Ref. 4) indicates that the drive flow sensor for the flow referenced scram is typically 1.0 sec. A search of GE Recirculation System Documentation (APED-B31) could not find a specific flow sensor response time. Based on the GE Procedure Specification, the 1.0 sec time will be used as the input.
63	DAEC uses LIS 4592A-D as the (L3) low level scram trip function sensors. These instruments are narrow range instruments.
64	Requests the lower limit (Analytical Limit) of the L3 setpoint scram setpoint. Ref. 20, provides the lower limit (Analytical Limit) for the L3 low water level scram setpoint for instruments LIS4592A-D. The AL is 162.5 inches RWL. The requested value is in inches AVL. Per Ref. 55, TAF is 344.94". L3 Trip (Inches AVZ) = 162.5 in. + 344.94 in. L3 Trip = 507.4 In. AVZ. Cycle 24 OPL-3 indicates a value of 507.0 in. The difference is small and the 507.0 in value is Good. Note that Assumption 4.9 of Ref. 20, discusses the process measurement error to account for the Bernoulli effect. GEH analysis has shown that for DAEC, the consequence of a delay in the scram on low level is insignificant. Therefore this error is not incorporated into the L3 trip value noted above.
65	Per Ref. 60 (Section 4.6.5.8), the trip sensor time constant is 1.0 second.
66	Per Ref. 21, the analytical limit (AL) for the core thermal power setpoint is 26% rated thermal power.
67	Per OSI PI data reviewed during Cycle 25 operation.
68	EHC Customer Trips have no time delay.
69	This value is noted in GE reference 1 however the utility cannot validate this value.
70	The GE manual for the OPL-3 (Ref. 4) stated that typical values were between 2.5-10 seconds so customer is selecting lowest of the typical values to be conservative.
71	The customer references for this item is References 32, 4, 48, 34, 49, and 70. This list of reference numbers could not fit in the Customer Reference field.
72	The customer references for this item is References 32, 4, 33, 48, 34, 49, and 70. This list of reference numbers could not fit in the Customer Reference field.
73	The correct value is 571.6 not 571.600. The spreadsheet form will not allow the customer to format the cell to specify four significant digits. GE should modify this field in the resolved OPL-3.
74	It was determined by site engineering that Cycle 24 recirc pump data applies to Cycles 26 and 27. Thus this cycle number was not changed.
75	Deleted.
76	MSIV closure occurs on Reactor Low Low Low Water Level (L1) not Reactor Low Low Level (L2).
77	It should be noted that with the 120% reactor uprate, the new rated throttle pressure is 983 psig however the plant has been operating at 968 psig.
78	Per EC 155793 (Ref 83), CV1-3 were restored to full lift operation, and CV4 was restricted to the original stroke limit of 40% per the new partial arc operation

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

1 - Plant Operational Parameters									
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**
A.	**Operational Parameters-Rated Conditions**								
1	Core thermal power-Nominal	MWt	1912	2		1912	1		1912
2	Core thermal power-Appendix K	MWt	1951	4	1	1951	2	1	1951
3	Vessel steam dome pressure-Nominal	psia	1040	2		1040	1		1040
4	Vessel steam dome pressure-Appendix K	psia	1055	2		1055	2		1055
5	Vessel steam output-Nominal	Mlbm/hr	8.352	9		8.352	1		8.352
6	Vessel steam output-Appendix K	Mlbm/hr	TBD by heat balance			8.554	2		TBD by heat balance
7	Core flow	Mlbm/hr	49.00	2		49.0	1		49.0
8	Recirculation drive flow-Loop A	Mlbm/hr	11.2	4		11.2	4	2	11.2
9	Recirculation drive flow-Loop B	Mlbm/hr	11.2	4		11.2	4	2	11.2
10	Feedwater temperature-Nominal	°F	431.4	4		431.4	1		431.4
11	Feedwater temperature-Appendix K	°F	TBD by heat balance			433.8	2		TBD by heat balance
12	Appendix K uncertainty on PLHGR	%	2		1	2	3		2

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

1 - Plant Operational Parameters									
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**
B.	**Alternate Operation Mode Parameters-[Increased Core Flow, ICF]**								
1	Core thermal power-Nominal	MWt	1912	2		1912	14		1912
2	Core thermal power-Appendix K	MWt	1951	4	1	1951	14	3	1951
3	Vessel steam dome pressure-Nominal	psia	1040	2		1040	14		1040
4	Vessel steam dome pressure-Appendix K	psia	1055	2		1055	14		1055
5	Vessel steam output-Nominal	Mlbm/hr	TBD by heat balance			8.356	14		TBD by heat balance
6	Vessel steam output-Appendix K	Mlbm/hr	TBD by heat balance			8.620	14		TBD by heat balance
7	Core flow	Mlbm/hr	51.45	3	2	51.45	14		51.45
8	Recirculation drive flow-Loop A	Mlbm/hr	11.76	4	3	11.76	14	4	11.76
9	Recirculation drive flow-Loop B	Mlbm/hr	11.76	4	3	11.76	14	4	11.76
10	Feedwater temperature-Nominal	°F	TBD by heat balance			431.5	14		TBD by heat balance
11	Feedwater temperature-Appendix K	°F	TBD by heat balance			433.8	14		TBD by heat balance

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

1 - Plant Operational Parameters									
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**
C.	**Alternate Operation Mode Parameters - [MELLLA]**								
1	Core thermal power-Nominal	MWt	1658	2	30	1658	2		1658
2	Core thermal power-Appendix K	MWt	1692		31	1692	2	1	1692
3	Vessel steam dome pressure-Nominal	psia	1040	2		1040	2		1040
4	Vessel steam dome pressure-Appendix K	psia	1055	2		1055	2		1055
5	Vessel steam output-Nominal	Mlbm/hr	TBD by heat balance			TBD by heat balance			TBD by heat balance
6	Vessel steam output-Appendix K	Mlbm/hr	TBD by heat balance			TBD by heat balance			TBD by heat balance
7	Core flow	Mlbm/hr	39.2	5	33	39.2	2	55	39.2
8	Recirculation drive flow-Loop A	Mlbm/hr	8.96		34	8.96	2	56	8.96
9	Recirculation drive flow-Loop B	Mlbm/hr	8.96		34	8.96	2	56	8.96
10	Feedwater temperature-Nominal	°F	TBD by heat balance			TBD by heat balance			TBD by heat balance
11	Feedwater temperature-Appendix K	°F	TBD by heat balance			TBD by heat balance			TBD by heat balance

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

1 - Plant Operational Parameters									
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**
D.	**Alternate Operation Mode Parameters-[Single Loop Operation, SLO]**								
1	Core thermal power-Nominal	MWt	1278	5	6	1278	13	8	1278
2	Core thermal power-Appendix K	MWt	1304	5	7	1304	13	9	1304
3	Vessel steam dome pressure-Nominal	psia	1040	2		1040	2, 3, 13		1040
4	Vessel steam dome pressure-Appendix K	psia	1055	2		1055	2, 3, 13		1055
5	Vessel steam output-Nominal	Mlbm/hr	TBD by heat balance			6.163	2	11	TBD by heat balance
6	Vessel steam output-Appendix K	Mlbm/hr	TBD by heat balance			6.306	2	12	TBD by heat balance
7	Core flow	Mlbm/hr	25.96	5		25.96	2, 13		25.96
8	Recirculation drive flow-Loop A	Mlbm/hr	NA	1	8	NA		10	NA
9	Recirculation drive flow-Loop B	Mlbm/hr	NA	1	8	NA		10	NA
10	Feedwater temperature-Nominal	°F	TBD by heat balance			402.4	2	11	TBD by heat balance
11	Feedwater temperature-Appendix K	°F	TBD by heat balance			404.5	2	12	TBD by heat balance

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

1 - Plant Operational Parameters									
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**
E.	**Pre-EPU Operational Parameters**								
1	Core thermal power-Nominal	MWt	1658	2	30	1658	2		1658
2	Core thermal power-Appendix K	MWt	1692		31	1692	2	1	1692
3	Vessel steam dome pressure-Nominal	psia	1040	2		1040	2		1040
4	Vessel steam dome pressure-Appendix K	psia	1055	2		1055	2		1055
5	Vessel steam output-Nominal	Mlbm/hr	TBD by heat balance			TBD by heat balance			TBD by heat balance
6	Vessel steam output-Appendix K	Mlbm/hr	TBD by heat balance			TBD by heat balance			TBD by heat balance
7	Core flow	Mlbm/hr	49.00	2	32	49.0	2		49.0
8	Recirculation drive flow-Loop A	Mlbm/hr	11.2	2		11.2	2		11.2
9	Recirculation drive flow-Loop B	Mlbm/hr	11.2	2		11.2	2		11.2
10	Feedwater temperature-Nominal	°F	TBD by heat balance			TBD by heat balance			TBD by heat balance
11	Feedwater temperature-Appendix K	°F	TBD by heat balance			TBD by heat balance			TBD by heat balance

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

2. Emergency Diesel Generators (LPCS/LPCI)									
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**
A.	Initiating signals								
1	Low water level	Level#	1	6		1	6	13	1
2	High drywell pressure	Yes/No	YES	6		YES	6	13	YES
B.	Delay time to process initiation signal (T_{SPD})	seconds	TBD by customer		20	--	6	14	--
C.	Maximum delay time from EDG start signal until bus is at rated voltage (T_{DG})	seconds	TBD by customer		21	--	6	14	--

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

3. Low Pressure Coolant Injection (LPCI) System									
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**
A.	Initiating signals								
1	Low water level	Level#	1	6		1	3	13	1
2	High drywell pressure	Yes/No	YES	6		YES	3	13	YES
3	Low vessel pressure permissive (pump start logic)	psig	NA			NA		15	NA
4	Timer delay for sustained low water level	minutes	NA			NA		15	NA
B.	Delay time to process initiation signal (T_{SPD} on Fig. 1)	seconds	TBD by customer		20	--	6	14	--
C.	Maximum vessel pressure at which pumps can inject flow (pressure associated with T_{CIPH} on Fig. 1)	psid (vessel to drywell)	197	6		197	6	16	197
D.	Minimum flow delivered to vessel								
1	Vessel pressure at which flow rates listed below are quoted	psid (vessel to drywell)	20	6		20	6	16	20
2	For one LPCI pump injecting into one recirculation loop	gpm	NA			NA		17	NA
3	For two LPCI pumps injecting into one recirculation loop	gpm	8224	6	9, 22, 24	8224	3, 6	18	8,224
4	For three LPCI pumps injecting into one recirculation loop	gpm	10557	6	9, 22, 24	10,557	3, 6	18	10,557

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

3. Low Pressure Coolant Injection (LPCI) System									
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**
5	For four LPCI pumps injecting into one recirculation loop	gpm	13330	6	9, 22, 24	13,330	3,6	18	13,330
6	One LPCI pump into shroud	gpm	NA			NA		17	NA
7	Two LPCI pumps into shroud	gpm	NA			NA		17	NA
8	Three LPCI pumps into shroud	gpm	NA			NA		17	NA
E.	Minimum flow at 0 psid (vessel-to-drywell)								
1	For one LPCI pump injecting into one recirculation loop	gpm	NA			NA		17	NA
2	For two LPCI pumps injecting into one recirculation loop	gpm	TBD by customer		10	--		19	--
3	For three LPCI pumps injecting into one recirculation loop	gpm	TBD by customer		10	--		19	--
4	For four LPCI pumps injecting into one recirculation loop	gpm	TBD by customer		10	--		19	--
5	One LPCI pump into shroud	gpm	NA			NA		17	NA
6	Two LPCI pumps into shroud	gpm	NA			NA		17	NA
7	Three LPCI pumps into shroud	gpm	NA			NA		17	NA
F.	Maximum delay time from bus at rated voltage until power available for pump start. (T_{CIPA} on Fig. 1)	seconds	TBD by customer		11, 23	--	3, 6	14	--
G.	Maximum delay time from pump start until pump is at rated speed (T_{CIPR} on Fig. 1)	seconds	TBD by customer		11	--	3, 6	14	--

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

3. Low Pressure Coolant Injection (LPCI) System									
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**
H.	LPCI Injection Valves								
1	Maximum delay time from bus at rated voltage until power available at injection valve (T_{CIPV} on Fig. 1)	seconds	TBD by customer			--	3, 6	14	--
2	Pressure at which injection valve may open (pressure permissive associated with T_{CIPP} on Fig. 1)	psig (vessel)	350	6		350	6		350
3	Maximum injection valve stroke time – opening (T_{CIIV} on Fig. 1)	seconds	28	6		28	6		28
I.	Recirculation discharge valves								
1	Maximum delay time from bus at rated voltage until power available at discharge valve (T_{CIPC} on Fig. 1)	seconds	TBD by customer			--		14	--
2	Pressure at which discharge valve may close (TCIPD on Fig. 1)	psig (vessel)	TBD by customer			900 psig	18, 19	20	900
3	Discharge valve stroke time – closing (TDV on Fig. 1)	seconds	30	6		30	3, 13		30

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

3. Low Pressure Coolant Injection (LPCI) System									
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**
4	Discharge bypass valve stroke time – closing (not shown on Fig. 1)	seconds	NA		24	Infinite	20, 21	21	Infinite
J.	Minimum flow bypass (MFB) valve								
1	Normal position of MFB valve at system startup	Open/ Closed	TBD by customer			CLOSED	7, 22, 23	22	CLOSED
2	System flow at which MFB valve is signaled to close	gpm	TBD by customer			2000	7, 22, 23	22	2,000
3	MFB valve stroke time	seconds	TBD by customer			15.6	24		15.6
4	MFB flow rate	gpm	TBD by customer			2000	9		2,000
K.	Minimum detectable break size for Loop Selection Logic	ft^2	0.5	6		0.5	3	54	0.5
L.	Total system delay time from initiating signal until the system is ready to inject	seconds	50	6		50	3	14	50

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

4. Core Spray (CS)/Low Pressure Core Spray (LPCS) System									
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**
A.	Initiating signals								
1	Low water level	Level#	1	6		1	3	13	1
2	High drywell pressure	Yes/No	YES	6		YES	3	13	YES
3	Low vessel pressure permissive	psig	NA			NA	-	15	NA
4	Timer delay for sustained low water level	minutes	NA			NA		15	NA
B.	Delay time to process initiation signal (T_{SPD} on Fig. 2)	seconds	TBD by customer		20	50	4	14	--
C.	Maximum vessel pressure at which pumps can inject flow (pressure associated with T_{CSPH} on Fig. 2)	psid (vessel to drywell)	264	6		264	6	16	264
D.	Minimum flow delivered to vessel								
1	Vessel pressure at which flow rate listed below is quoted	psid (vessel to drywell)	113	6		113	6	16	113
2	Minimum flow at vessel pressure (one loop)	gpm	2718	6	14	2718	6		2,718
E.	Minimum flow at 0 psid (vessel-to-drywell), 1 pump	gpm	3173	6	14	3173	6	16	3,173
F.	Maximum delay time from bus at rated voltage until power available for pump start. (T_{CSPA} on Fig. 2)	seconds	TBD by customer		13, 25	--	6	14	--

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

4. Core Spray (CS)/Low Pressure Core Spray (LPCS) System									
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**
G.	Maximum delay time from pump start until pump is at rated speed (T_{CSPR} on Fig. 2)	seconds	TBD by customer		13	--	6	14	--
H.	CS/LPCS Injection Valve(s)								
1	Maximum delay time from bus at rated voltage until power available at injection valve (T_{CSPV} on Fig. 2)	seconds	TBD by customer			--	6		--
2	Pressure at which injection valve may open (pressure permissive associated with T_{CSPP} on Fig. 2)	psig (vessel)	350	6		350	6	16	350
3	Maximum injection valve stroke time – opening (T_{CSIV} on Fig. 2)	seconds	18	6	15	18	6		18
I.	Minimum flow bypass (MFB) valve								
1	Normal position of MFB valve at system startup	Open/ Closed	TBD by customer			OPEN	10	23	OPEN
2	System flow at which MFB valve is signaled to close	gpm	TBD by customer			--	12	23	--
3	MFB valve stroke time	seconds	TBD by customer			--	24	24	--
4	MFB flow rate	gpm	TBD by customer			--	12	23	--

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

4. Core Spray (CS)/Low Pressure Core Spray (LPCS) System									
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**
J.	Total system delay time from initiating signal until the system is ready to inject	seconds	37	6		37	3	14	37

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

6. High Pressure Coolant Injection (HPCI) System										
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**	**Resolved Notes**
A.	Initiating signals									
1	Low water level	Level#	2	6		2	6, 25	13, 25	2	
2	High drywell pressure	Yes/No	Yes	6		YES	6, 25	13	YES	
B.	Delay time to process initiating signal (T_{SPD})	seconds	TBD by customer		20	1	17		-	
C.	Operating pressure range									
1	Maximum	psid (vessel to drywell)	1120.3	6	17	1120.3	6, 25, 27	26	1,120.3	
2	Minimum	psid (vessel to drywell)	150.3	6	17	150.3	6	27	150.3	
D.	Minimum flow over pressure range in Item C	gpm	2700	6		2700	6		2,700	
E.	Maximum allowed delay time from initiating signal to pump at rated flow, injection valve wide open and bypass valve closed	seconds	45	6		45	6		45	
F.	Steam flow over operating pressure range									
1	Maximum	lbm/hr	125000	6		125000	6, 27		125,000	
2	Minimum	lbm/hr	55000	6		55000	6, 27		55,000	

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

6. High Pressure Coolant Injection (HPCI) System										
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**	**Resolved Notes**
G.	Maximum time delay from initiating signal to start of steam supply valve opening	seconds	TBD by Customer		16	29	6	28	25	2
H.	Steam supply valve opening stroke time	seconds	TBD by Customer		16	16-20	24		20	1
I.	HPCI flow at minimum operating pressure diverted to core spray	gpm	NA			N/A			NA	
J.	Total system delay time from initiating signal until the system is ready to inject	seconds	45	6		45	6		45	

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

9. Automatic Depressurization System (ADS)										
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**	**Resolved Notes**
A.	Initiating signals									
1	Low water level	Level#	1	6		1	3, 6	29	1	
2	High drywell pressure	Yes/No	TBD by Customer			No	3, 6	29	No	
3	High drywell pressure bypass timer delay for sustained low water level (T_{BT} on Fig. 3)	seconds	TBD by Customer			NA		17	NA	
4	ECCS ready permissive	Yes/No	TBD by Customer			YES	56, 59, 60, 61		YES	
B.	Delay time to process initiating signal (T_{SPD} on Fig. 3)	seconds	TBD by customer		20	--		30	--	3
C.	Total number of relief valves with ADS function	#	4	6		4	28	31	4	
D.	Total number of relief valves with ADS function assumed in analysis	#	3	6		4	3, 6	32	4	
E.	Pressure at which flow capacity listed below is quoted	psig (vessel)	1125	6		1125	3, 6, 28	33	1,125	
F.	Minimum flow rate for one valve open at above listed pressure	lbm/hr	800,000	6		800,000	3, 6, 28	34	800,000	

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

9. Automatic Depressurization System (ADS)										
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**	**Resolved Notes**
G.	ADS timer delay from initiating signal completed to the time valves are opened (T_{ST} on Fig. 3)									
1	Nominal	seconds	TBD by customer		26	120	30, 57, 58	35	120	
2	Analytical limit	seconds	132	6		132	3, 6, 28		132	
H.	Valve pressure setpoints									
1	ADS close on vessel pressure	psig	50			25-50	31	36	50	
2	ADS reopen on vessel pressure	psig	100			100	31	36	100	
3	ADS reclose on vessel pressure	psig	50		19	25-50	31	36	50	
I.	Break used as basis for ADS bypass timer delay									
1	Location		NA				NA		NA	
2	Break isolation signal		NA				NA		NA	
3	MSIV stroke time	seconds	NA				NA		NA	

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

10. In-Vessel Leakage Rates									
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**
A.	LPCI leakage								
1	Total leakage flow	gpm	600	6	18	600	3, 4, 6	37	600
2	Pressure at which leakage flow is defined	psid	20	6		20	3, 4, 6	16	20
B.	Jet pump leakage								
1	Total leakage flow, with water level at top of jet pumps	gpm	TBD by Customer		16	600	3, 4, 6, 17	38	600
C.	CS/LPCS leakage - Principally through vent hole of T-joint, or CS/LPCS header and riser cracks, or CS/LPCS related repairs								
1	Total leakage flow	gpm	100	6		--	3, 4, 6	39	100
2	Pressure at which leakage flow is defined	psid	113	6		--	3, 4, 6, 17	39	113
D.	HPCS leakage - Principally through vent hole of T-joint, or HPCS header and riser cracks, or HPCS related repairs								
1	Total leakage flow	gpm	NA			NA		40	NA
2	Pressure at which leakage flow is defined	psid	NA			NA			NA
E.	Leakage allowance for shroud cracks, internal modifications and repairs								
1	Leakage flow	gpm	TBD by Customer		16	0	3, 4, 6, 17	41	0

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

10. In-Vessel Leakage Rates									
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**
2	Core flow at which leakage flow is defined	% of rated	TBD by Customer		16	0	3, 4, 6, 17	41	0
3	Elevation of lowest core shroud crack	inches AVZ	TBD by Customer		16	0	3, 4, 6, 17	41	0
F.	Leakage allowance for access hole cover cracks, internal modifications and repairs								
1	Leakage flow	gpm	TBD by Customer		16	0	3, 4, 6, 17	41	0
2	Core flow at which leakage flow is defined	% of rated	TBD by Customer		16	0	3, 4, 6, 17	41	0

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

11. Miscellaneous Inputs										
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**	**Resolved Notes**
A.	Normal water level at rated power (indicated level)	inches	535.5	6	27	530.94 - 539.94	32, 33	42	535.5	
B.	Water level setpoints									
1	Level 8-High Level	inches AVZ	TBD by Customer		16	572.0	34	43	572.0	
2	Level 7-High Level Alarm (bulk/indicated level)	inches AVZ	TBD by Customer		16	539.94	35		539.94	
3	Level 4-Low Level Alarm (bulk/indicated level)	inches AVZ	TBD by Customer		16	530.9	32, 35		530.9	
4	Level 3-Low Level (Scram level) (indicated level)	inches AVZ	507	6	28, 29	507.44	36		507.44	
5	Level 2-Low Low Level	inches AVZ	447.3	6		447.74	38		447.74	
6	Level 1-Low Low Low Level	inches AVZ	350.0	6		350.44	3, 39		350.44	
C.	Steam dryer pressure drop	psid	0.43	9		0.6	14, 17		0.6	4
D.	MSIV isolation-initiation signal									
1	Low water level	Level #	1	6		1	55		1	
2	Low steam line pressure	psig	TBD by Customer		16	850	54		850	
3	High steam line flow	% of rated	TBD by Customer		16	140	28		140	
E.	MSIV signal delay (from initiating event to start of valve motion)	seconds	0.5	6		0.5	6		0.5	

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

11. Miscellaneous Inputs										
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**	**Resolved Notes**
F.	MSIV closure time									
1	Minimum closing time	seconds	3	1		3	40	44	3	
2	Maximum closing time	seconds	5	6		5	41	44	5	
G.	Feedwater pump coastdown (from initial value to zero flow)	seconds	5	6		5		45	5	
H.	Time constant for recirculation pump coastdown	seconds	3	6		3		45	3	
I.	Number of pilot-actuated Safety/Relief Valves (SRVs) in group									
1	Group A		1	7		1	28	46	1	
2	Group B		1	7		1	28	46	1	
3	Group C		2	7		2	28	46	2	
4	Group D		2	7		2	28	46	2	
1a	Opening of Group A	psig	1110.0	7		1110	31	47	1,110	
1b	Closing of Group A	psig	1065.6	7		1065.6	43, 44, 45	48	1,065.60	
2a	Opening of Group B	psig	1120.0	7		1120	31	47	1,120	
2b	Closing of Group B	psig	1075.2	7		1075.2	43, 44, 45	48	1,075.20	
3a	Opening of Group C	psig	1130.0	7		1130	31	47	1,130	
3b	Closing of Group C	psig	1084.8	7		1084.8	43, 44, 45	48	1,084.80	
4a	Opening of Group D	psig	1140.0	7		1140	31	47	1,140	
4b	Closing of Group D	psig	1094.4	7		1094.4	43, 44, 45	48	1,094.40	

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

11. Miscellaneous Inputs										
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**	**Resolved Notes**
K.	Number of low-low set SRVs in Group									
1	Group A		NA			1	32		1	
2	Group B		NA			1	32		1	
3	Group C		NA			NA			NA	
L.	Opening/closing pressure setpoints of low-low set SRVs									
1a	Opening of Group A	psig	NA			1035 (1059.9)	32	49	1,059.90	
1b	Closing of Group A	psig	NA			915 (920.9)	32	49	920.9	
2a	Opening of Group B	psig	NA			1030 (1054.9)	32	49	1,054.90	
2b	Closing of Group B	psig	NA			910 (915.9)	32	49	915.9	
3a	Opening of Group C	psig	NA			NA			NA	
3b	Closing of Group C	psig	NA			NA			NA	
M.	Low-low set logic	Yes/No	NA			YES	48		YES	
N.	Pilot-actuated SRV capacity									
1	SRV capacity at (100+ACC)% of reference pressure	lbm/hr	829,000	7		829,000	17, 46	50	829,000	
2	Reference pressure	psig	1080	7		1080	17, 46		1,080	
3	Overpressure Accumulation Factor (ACC)	%	3	7		3	43		3	

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

11. Miscellaneous Inputs										
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**	**Resolved Notes**
O.	Additional Pilot-actuated SRV opening/closing parameters									
1	Time delay before opening of pilot-actuated SRVs	seconds	0.2	7		</= 0.2 seconds	31	51	0.2	
2	Time constant of SRV opening/closing	seconds	0.2	7		</= 0.2 seconds	31	51	0.2	
P.	Number of Spring Safety Valves (SSVs)									
1	Group A		NA			2	17, 46	51	2	
2	Group B		NA			NA			NA	
3	Group C		NA			NA			NA	
Q.	Opening/closing setpoint of SSVs									
1a	Opening of Group A	psig	1240	7		1240	17, 42, 48	52	1,240	
1b	Closing of Group A	psig	TBD by Customer			1129-1202	45	53	1,202	1
R.	SSV capacity at opening setpoint									
1	Group A	Mlbm/hr	0.6421			0.6421	17, 46		0.6421	
S.	ECCS make-up water temperature	°F	120	6		120	49, 50		120	
T.	Operator action time	seconds	600	1		600	51		600	
U.	High drywell pressure setpoint	psig	2.3	6		2.3	52, 53		2.3	

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

12 - Others									
No.	**Parameter**	**Units**	**Proposed by GEH**	**GEH References**	**GEH Notes**	**Proposed by Customer**	**Customer References**	**Customer Notes**	**Resolved for Analysis**

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

13 - GEH References	
No.	**Reference**
1	AG-0019 Rev 3, Analysis Guide OPL-4 Design Guide
2	Duane Arnold Energy Center Asset Enhancement Program Task T0407: ECCS-LOCA SAFER/GESTR, GE-NE-A22-00100-29-01 R0, September 2000.
3	Safety Analysis Report for Duane Arnold Energy Center Increased Core Flow, NEDC-33439P, Revision 3, August 2009
4	Safety Analysis Report for Duane Arnold Energy Center Extended Power Uprate, NEDC-32980P, Revision 1, April 2001
5	Duane Arnold Energy Center Asset Enhancement Program Task T0201: Power/Flow Map, GE-NE-A22-00100-04-01 R0, February 2000
6	Resolved DAEC AEP OPL-4/OPL-5 Forms, Revision 1, September 2000
7	Resolved DAEC OPL-3 Form for Cycle 23, Revision 1, June 2010
8	The GESTR-LOCA and SAFER Models for the Evaluation fo the Loss-Of-Coolant Accident Volume III, SAFER/GESTR Application Methodology, NEDC-23785-1-PA, October 1984 (Jet Pump Plant - SAFER)
9	Duane Arnold Energy Center Asset Enhancement Program Task T0304: Reactor Internal Pressure Differences, GE-NE-A22-00100-11-01 R0, August 2000
10	Evaluation of Steam Flow Induced Error (SFIE) Impact on the L3 Setpoint Analytic Limit, GEH-NE-0000-0077-4603-R1, October 2008
11	The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance, Part 3 - Application Methodology, NEDC-33258P-A, Rev.1, September 2010

Table 15.0 - 4
ECCS/LOCA Analysis Input Parameters (Form OPL-4)

14 - GEH Notes	
No.	**Note**
1	All Appendix K calculations apply a 2% PLHGR uncertainty and a corresponding 2% core thermal power increase. Note that the LPU Appendix K power equals 1912*1.02 = 1950.24 MWt, rounded up (Ref.1) to 1951 MWt.
2	The licensed maximum core flow at rated power is 105%. 51.45 Mlbm/hr core flow is 105% of 49.0 Mlbm/hr.
3	11.76 Mlbm/hr corresponds to 105% of 11.2 Mlbm/hr.
4	Deleted.
5	Deleted.
6	The SLO point is 66.79% of LPU, i.e 1,912*0.6679=1,277.0248 MWt, rounded up (Ref.1) to 1,278 MWt.
7	The thermal power for Appendix K is equal 1,278*1.02=1,303.56 MWt, rounded up (Ref.1) to 1,304 MWt.
8	Recirculation drive flow is not applicable to the SLO operating domain.
9	This value has not been reduced by the leakage from Item 10.A.1. This is to remain consistent with the previous OPL-4. The 600 GPM is the default value for leakage and applies to 20 psid. These leakages are on a per-loop basis.
10	LPCI pump flow values were not quoted at 0 psid in pervious analyses and are therefore not developed here.
11	Maximum allowable time delay from initiating signal to pumps at rated speed is TSPD + TDG + TCIPA + TCIPR.
12	Deleted.
13	Maximum allowable time delay from initiating signal to pumps at rated speed is TSPD + TDG + TCSPA + TCSPR.
14	This value has not been reduced by the leakage from Item 10.C.1. This is to remain consistent with the previous OPL-4. The 100 GPM is the default value for leakage and applies to 113 psid. These leakages are on a per-loop basis.
15	The LOCA analysis assumes no core spray flow until the valve is fully open.
16	There is no data within the prior analyses for this item. Therefore, this item is left for the customer to complete.
17	Values in Reference 6 are in the psia range and converted to psig (psid) by subtracting 1 atm – 14.7 psi.
18	This leakage is not included in LPCI flows listed in Section 3.
19	ADS reclose on vessel pressure is assumed to be the same as ADS close on vessel pressure.

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

20	The value for this parameter was not specified within the prior analysis, but was included in an overall assumed LPCI, LPCS and HPCI time delays. Per AG-0019 (Reference 1) a typical value is 1 second.
21	The value for this parameter was not specified within the prior analysis, but was included in an overall assumed LPCI and LPCS time delays. Per AG-0019 (Reference 1) a typical value is 10 seconds.
22	15% of LPCI flow would be lost through the bypass line. Number of LPCI Pumps: 2, 3, 4. The LPCI flow would enter the vessel: With Bypass Valve Closed: 9,675; 12,420; 15,682. With Bypass Valve Open: 8,224; 10,557; 13,330.
23	The value for this parameter was not specified within the prior analysis, but was included in an overall assumed LPCI time delay of 50 seconds.
24	The reactor recirculation system discharge valve is assumed to be open during the LOCA event.
25	The value for this parameter was not specified within the prior analysis, but was included in an overall assumed LPCS time delay of 37 seconds.
26	The typical value is 120 seconds.
27	The bulk water level (519.2 inches) value is 16.3 inches below the indicated (sensed) water level to account for a steam dryer delta P (Item 11.C).
28	The bulk water level (490.7 inches) value is 16.3 inches below the indicated (sensed) water level to account for a steam dryer delta P (Item 11.C).
29	Level 3 does not include adjustment for the Steam Flow Induced Error (SFIE) (Reference 10).
30	104.1% of ORTP (1593 MWT).
31	All Appendix K calculations apply a 2% PLHGR uncertainty and a corresponding 2% core thermal power increase. Note that the LPU Appendix K power equals 1658*1.02 = 1691.16 MWt, rounded up (Ref.1) to 1692 MWt.
32	100 % of rated core flow.
33	39.2 Mlbm/hr corresponds to 80% of 49.0 Mlbm/hr.
34	8.96 Mlbm/hr corresponds to 80% of 11.2 Mlbm/hr.

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

15 - Customer References	
No.	**Reference**
1	Duane Arnold Energy Center Asset Enhancement Program Task T0100: Nominal Reactor Heat Balance, GE-NE-A22-00100-01-01 R0, April 2000.
2	Duane Arnold Energy Center Asset Enhancement Program Task T0101: Offrated Reactor Heat Balance, GE-NE-A22-00100-02-01 R0, June 2000.
3	Duane Arnold Energy Center Asset Enhancement Program Task T0407: ECCS-LOCA SAFER/GESTR, GE-NE-A22-00100-29-01 R0, September 2000.
4	NEDO-32980P Safety Analysis Report for Duane Arnold Energy Center Extended Power Uprate Revision 0. NEDC-32980P GE PROPRIETARY INFORMATION CLASS III Figure 1-2 EPU Heat Balance @ 102% EPU RTP.
5	Duane Arnold Energy Center Asset Enhancement Program Task T0304: Reactor Internal Pressure Differences, GE-NE-A22-00100-11-01 R0, August 2000.
6	NEDC-31310P, Supplement, Rev. 1, “Duane Arnold Energy Center SAFER/GESTR LOCA Analysis”, September 1993.
7	DAEC Drawing BECH-M120 P&ID Residual Heat Removal System, Rev.65.
8	Intentionally left blank.
9	DAEC Calculation CAL-M91-007 MEDP PRESSURE, FLOW, AND TEMPERATURE DETERMINATION, Rev.4.
10	DAEC Drawing P&ID BECH-M121 P&ID Core Spray System, Rev.38.
11	Intentionally left blank.
12	DAEC Calculation CAL-M92-030 MEDP, PRESSURE, FLOW, TEMPERATURE DETERMINATION, Rev.1.
13	Duane Arnold Energy Center Asset Enhancement Program Task T0201: Power/Flow Map, GE-NE-A22-00100-04-01 R0, February 2000.
14	NEDC-33439P, Rev.0, DRF 0000-0077-5085, November 2008, Safety Analysis Report for Duane Arnold Energy Center Increased Core Flow.
15	GENE-637-034-1093, DRF A00-05703, CLASS III, October 1993, DAEC SAFER/GESTR-LOCA, Loss-of-Coolant Accident Analysis, Engineering Report.
16	Intentionally left blank.
17	AG-0019 Rev 3, Analysis Guide OPL-4 Design Guide.
18	DAEC Drawing APED-E11-007<6> Residual Heat Removal System Elementary Diagram, Rev.28.
19	DAEC Drawing APED-E11-007<9> Residual Heat Removal System Elementary Diagram, Rev.26.
20	DAEC UFSAR 15.2.1.1 A 1 d - Loss of Coolant Accidents, Rev.20, 8/09.
21	DAEC UFSAR 7.3.1.1.2.4 - LPCI System Instrumentation and Control, Rev.17, 10/03.
22	P&ID BECH-M119 Residual Heat Removal System, Rev.83.
23	P&ID BECH-E121<054A> Reactor Core Cooling Systems <RHR>, Rev. 5.
24	ASME Valve Stroke Time Databook, Rev. 60, 02/14/2012.
25	Resolved DAEC AEP OPL-4/OPL-5 Forms, Revision 1, September 2000. (Also, see UFSAR Table 15.0-04, ECCS Data for LOCA Analysis, Rev.17, 10/03).
26	Intentionally left blank.

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27	DAEC Letter HG-00-1332, R. McGee (Alliant Energy) to W.F. Farrell (GE), "Transmittal of DIR T0309 RCIC and DIR T0404 HPCI," July 31, 2000.
28	Duane Arnold Energy Center Asset Enhancement Program Task T0300: Nuclear Boiler System, GE-NE-A22-00100-07-01 R0, September 2000
29	DAEC Calculation CAL-E94-003 Low Low Set Pressure Switches Actuation Setpoint, Rev.2.
30	DAEC Technical Specifications Table 3.3.5.1-1, ECCS Instrumentation, Amendments 223 and 245.
31	DAEC Calculation CAL-MC-003C Main Steam Line Pressure, Rev.0.
32	NEE Nuclear Fuels Letter, NF-12-065, DAEC Cycle 24 Customer Proposed OPL-3 Form, February 16, 2012.
33	DAEC Operating Instruction OI-644 Condensate and Feedwater Systems, Rev.120.
34	DAEC Calculation CAL-E93-025 Setpoint NR RPV Level High HPCI & RCIC Trip, Rev.1
35	DAEC Annunciator Response Procedure ARP 1C05A Reactor Control, Rev.70.
36	DAEC document CAL-E92-010, Low Reactor Water Level Scram & ADS Confirmatory, Rev.5.
37	Intentionally left blank.
38	DAEC CAL-E93-026 Lo-Lo Reactor Water Level 119'5", HPCI, RCIC, ATWS, Rev.3.
39	DAEC CAL-E93-016 Reactor Low Low Low Water Level - CS, LPCI, PDIS, Rev.3.
40	DAEC Technical Specifications Surveillance Requirement 3.6.1.3.5 "Primary Containment Isolation Valves (PCIVs)", Amendments 223, 230, 234, and 276.
41	GE Report GE-NET2300752-00-01-R2, "Duane Arnold Energy Center Containment Analysis," July 1998.
42	Duane Arnold Energy Center Asset Enhancement Program Task T0315: SRV Setpoint Tolerance Monitoring Program Review, R1, June 2000
43	DAEC Calculation CAL-M92-020 Main Steam Safety Valve Min. Discharge.
44	GE Specification 21A9207, General Requirements for Valves, Rev. 3.
45	DAEC Drawing APED-B21-066 General Requirements for Valves, Rev.4.
46	DAEC UFSAR Table 5.2-1 Nuclear System Safety and Relief Valves, Rev.18, 10/04.
47	TDP-0087, OPL-3 Design Guide, Revision 3.
48	DAEC Technical Specifications 3.4.3 SRVs and SVs, Amendments 223 and 228.
49	GE report NEDC-22082-P, "Duane Arnold Energy Center Suppression Pool Temperature Response," March 1982.
50	GE report NEDC-22082-P, "Duane Arnold Energy Center Suppression Pool Temperature Response," Supplement 1, April 1984.
51	DAEC Design Basis Document, DBD-A61-002, Nuclear Safety Criteria for BWRs, Rev.0. (Issued for Use per DDC-1742, 02-08-91)
52	DAEC Calculation CAL-E95-010 High Drywell Pressure ECCS Initiation Setpoint, Rev.2.
52	DAEC Calculation CAL-E95-009 High Drywell Pressure Scram & Isolation Setpoint, Rev.2.
53	DAEC Calculation CAL-E93-003 Main Steam Line Low Press Isol Setpoint PS-1014, Rev.5.
54	DAEC Calculation CAL-E93-016 Reactor Low Low Low Water Level - CS, LPCI, PDIS, Rev.3.
55	DAEC Drawing APED-B21-018 <1> Elem Diag Auto Depressure Sys., Rev.25.
56	DAEC Letter NG-95-2633 "ADS Time Delay Setting Reference Updated Final Safety Analysis Report, NEDC-31310 93-08-01, STP 42B012-CY Rev 4, DAEC Tech Specs, Emergency Operating Procedure", 08/16/1995.
57	DAEC Letter NG-96-1196, "Meeting Minutes TSIP Setpoint Calculation Implementation Issues PSC Room H", 05/20/1996.
58	DAEC Drawing APED-B21-018 <2> Elem Diag Auto Depressure Sys., Rev.21.
59	DAEC Drawing APED-B21-018 <3> Elem Diag Auto Depressure Sys., Rev.26.
60	DAEC Drawing APED-B21-018 <3A> Elem Diag Auto Depressure Sys., Rev.2.

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

16 - Customer Notes	
No.	**Note**
1	All Appendix K calculations apply a 2% PLHGR uncertainty and a corresponding 2% core thermal power increase. Note that the LPU Appendix K power equals 1912*1.02 = 1950.24 MWt, rounded up (Ref.1) to 1951 MWt. (Ref.17)
2	Value of Wd/2 (Figure 1-1 of NEDC32980P)
3	Consistent with GEH Note 1. However, note that Cycle 24 OPL-3 used 1950.2.
4	Proposed value is (Wd X 1.05)/2
5	Proposed value is (Wd X 0.99)/2
6	State Point: 100% power, 99% flow
7	State Point: 102% power, 99% flow
8	The SLO point is 66.79% of LPU, i.e 1,912*0.6679=1,277.0248 MWt, rounded up (Ref.17) to 1,278 MWt.
9	The thermal power for Appendix K is equal 1,278*1.02=1,303.56 MWt, rounded up (Ref.17) to 1,304 MWt.
10	Recirculation drive flow is not applicable to the SLO operating domain. Not required per GE OPL-4 Design Guide AG-0019 Rev 3 Sheet 12.
11	Value was taken from a statepoint of 77% power (0.77 X 1658 MWth = 1277 MWth) and 53% core flow from Ref. (2).
12	Value was taken from a statepoint of 78.5% power (0.785 X 1658 MWth = 1302 MWth) and 53% core flow from Ref. (2).
13	Provided in the Section 11 Miscellaneous.
14	The time intervals between individual events, as listed in LPCI, CS, ADS Initiation Logic are not available. Total delay time of 50 seconds for LPCI, 37 seconds for CS and 45 seconds for HPCI initiations were used in Reference 6. Value given is the bounding cumulative time (accounting for D/G start, pump start, inject valve opening and recirc discharge valve closing times).
15	Not installed at Duane Arnold.
16	A constant drywell pressure of 14.7 psia is assumed in the analysis.
17	Not part of plant design.
18	Since the valve in the 4 inch bypass line around the reactor recirculation system discharge valve is assumed to be open during the LOCA event. 15% of LPCI flow would be lost through the bypass line (Reference 12). The LPCI flow that would enter the vessel is shown below: No. of LPCI Pumps — With Bypass Valve Closed — With Bypass Valve Open 2 — 9,675 — 8,224 3 — 12,420 — 0,557 4 — 15,682 — 13,330
19	Use pump curves that were used in Section 3.D.

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

20	Following initiation of LPCI Loop Select Logic (Hi DW Pressure [Permissive2] or Low-Low Reactor Water Level [Level 1] on Figure 1), a 900 psig (vessel) permissive is inserted if both recirculation pumps are not running at initiation. No pressure permissive is inserted if both recirculation pumps are running at initiation.
21	Although the Recirculation pump discharge bypass valve receives a closed signal following a LOCA, the Recirculation pump discharge bypass valve is assumed not to close during the LOCA event. Analysis has been conducted that demonstrate that the acceptance criteria of 10CFR50.46 are still met is the recirculation pump discharge bypass valve remains open in the unbroken (selected) loop.
22	Under a low flow condition of less than 2000 gpm, the valve will receive an open signal after a 10-second time delay if either loop pump is running as sensed by pump breaker contacts. If loop flow is not established above 2000 gpm within the first 10 seconds of a pump start, the minimum flow valve will open to prevent the longterm effects of overheating the pump and possible cavitation. The minimum flow valve will close if respective loop flow increases above 2,000 gpm or if both RHR pumps A and C (B or D for Loop B) are secured as sensed by breaker contacts. To avoid tripping its breaker, there is a two-second time delay in the valve opening/closing logic such that the open (close) signal will not be applied to the valve until it has been closed (open) for a nominal two seconds.
23	CS minimum flow valve (which is normally open) opens on a low flow signal of 300 gpm, and closes on a high flow signal of 600 gpm. It will auto open whenever a low flow signal is present. To avoid tripping its breaker, there is a two-second time delay in the valve opening/closing logic such that the open (close) signal will not be applied to the valve until it has been closed (open) for a nominal two seconds.
24	Closing time is greater than or equal to (≥) 6.9 seconds and less than or equal to (≤) 11.3 seconds. Opening time is greater than or equal to (≥) 5.8 seconds and less than or equal to (≤) 9.6 seconds. ASME
25	inches AVZ (Above reactor vessel reference zero)
26	Coversion from psia to psid: 1135 psia - 14.7 psi = 1120.3 psid
27	Coversion from psia to psid: 165 psia - 14.7 psi = 150.3 psid
28	Per GEH's recommendations to customer this value was tabulated by taking the value in Section 6E (Maximum allowed delay time from initiating signal to pump at rated flow, injection valve wide open and bypass valve closed) and subtracting the value in Section 6H (Steam supply valve opening stroke time) = 45 seconds - 16 seconds = 29 seconds.
29	Initiation signals for ADS is Low-low-low water level at 350 inches above vessel 0 and 132 seconds via the ADS Timer.
30	See values provided Section 9G.
31	Also note that CAL-E94-003 (Ref.29) lists the S/RV number for PSV4407/1 with pressure switches for Hi/Close, Lo/Open and channel either A or B.
32	Refer to OPL-5 for sensitivity study for one ADS valve out of service.
33	Ref. 28 Table 1-1.
34	Ref.28 Table 3-1 specifies ADS capability (4 ADS S/RVs at 1125 psid) is 3.2 Mlb/hr and total as 0.8x10E6.
35	Ref.30 provides an allowable value of < or = 125 seconds. The nominal setpoint is 120 seconds per Ref. 57.

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

36	Valve capable of remote manual opening at any pressure above 100 psig and staying open, once opened, until pressure decreases to 50 psig. For pressure relief valve operation, valve shall reclose when inlet pressure falls 25-50 psi below set point. Per Page 4 of 7 of Ref.31.
37	The LPCI flow is reduced by 600 GPM at 20 psid to take into account the leakage flow. For 2 LPCI pumps injecting into one recirculation loop, the flow = 8224 (Reference 6) 600 = 7624 (gpm). For 3 LPCI pumps injecting into one recirculation loop, the flow = 10557 (Reference 2) 600 = 9957 (gpm). For 4 LPCI pumps injecting into one recirculation loop, the flow = 13330 (Reference 6) 600 = 12730 (gpm)
38	Value is 600 gpm. Jet pump slip joint leakage is not included in this value because its affect on peak cladding temperature is small.
39	The core spray flow is reduced by 100 GPM at 113 psid for each core spray system to take into account the leakage flow. The proposed core spray flow = 2718 (Reference 6) - 100 = 2618 (gpm).
40	There is no leakage allowance for cracks, etc.
41	There is no allowance for cracks, etc. There have been no identified cracks in the core shroud, and no modifications or repairs.
42	535.5 inches used in Ref.32.
43	571.6 per Ref. 37. Rounded up to 572 inches.
44	Ref.28 specifies MSIV closure valve stroke time as 3 to 5 seconds, no change from Pre-EPUP.
45	See Section 18 "Assumptions and Initial Conditions".
46	Ref.28 lists the Groups as 1 through 4, not A through D. So therefore Group 1 = A, 2 = B, and so on.
47	Ref.28 lists the Groups as 1 through 4, not A through D. So therefore Group 1 = A, 2 = B, and so on. NOTE: Table 1-1 of FTR T0300 shows different values as analytical limits. Also reference Nominal Septoint, As Found Low, As Found High and Analytical Limits for SRVs and SVs in Ref.42.
48	Per Ref.43, closing pressure is >/= 96% of set pressure per Ref.44 and Ref.45.
49	Due to calibration drift issues with the existing pressure switches, we have generated a new setpoint calculation (DAEC document CAL-E94-003, Rev. 3, LLS Pressure Switch Actuation Setpoint Calculation) that will support a Tech Spec change to modify the LLS setpoints. The new proposed values are shown in parentheses. Because this Tech Spec change may be implemented sometime during Cycle 24, please use the more limiting values for a given analysis. These same values were reflected in the Cycle 24 OPL-3 form. Note that these values are also different from what is listed in the current (Cycle 23) UFSAR Table 15.0-4 (OPL-4), so that change will need to be evaluated as well.
50	Total vessel steam flow is 8352 Mlbm/hr at 1040 psia dome pressure per Ref.1.
51	Refer to Ref.32 for a complete listing of SRV/SSV data, customer references, and customer comments in accordance with Ref. 47.
52	Safety Valve analytical limit for set pressure is 1277.2 psig per Table 1-1 in Ref.28. Rounded up to 1280 psig. Nominal Setpoint is 1240 psig per Ref.42, which specifies a reference of DAEC Tech Spec Surveillance Requirement 3.4.3.1, Also see Ref.32 and Ref.43.

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ECCS/LOCA Analysis Input Parameters (Form OPL-4)

53	The reclosing setpoint is set at 6% below the opening setpoint of 1240 psig with +/- 3% tolderance. There is a 91 to 97% tolerance on the reclosing setpoint per Ref.59. Thus 1240 psig * 0.91 = 1128.4 psig and 1240 psig * 0.97 = 1202.8 psig. Values are rounded conservatively for a range of 1129 - 1202 psig.
54	As of 02/23/2012, DAEC is evaluating the LPCI loop select break size value to determine if there is any margin that can be attained that will allow a modification to pressure switch settings. Please confirm with NEE prior to proceeding with this number in the analysis.
55	39.2 Mlbm/hr corresponds to 80% of 49.0 Mlbm/hr
56	8.96 Mlbm/hr corresponds to 80% of 11.2 Mlbm/hr

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Table 15.0 - 4

ECCS/LOCA Analysis Input Parameters (Form OPL-4)

17 - Resolved Notes	
No.	**Note**
1	The higher value is conservative.
2	45 seconds - 20 seconds (6.H) = 25 seconds.
3	This value is included in the 132 analytical delay time in Item 9.G.2.
4	This value is calculated in GEH Reference 3. Analysis assumed a thermal power of 1912 MWt (LOCA analysis basis) with a steam flow of 8.356 Mlbm/hr.

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Table 15.0 - 4
ECCS/LOCA Analysis Input Parameters (Form OPL-4)

18 - Analysis Assumptions and Initial Conditions

The ECCS performance evaluation will use the SAFER/PRIME-LOCA Application Methodology, i.e. a SAFER/GESTR-LOCA Methodology (Reference 8) with PRIME implementation approved by the Nuclear Regulatory Commission (NRC) (Reference 11). The analyses assumptions summarized here are in compliance with 10CFR50 Appendix K.

ECCS Performance Analysis Assumptions

The following assumptions are used to confirm that the ECCS design is capable of mitigating all postulated LOCA events.

(a) A break occurs in any steam or liquid line which forms part of the primary reactor coolant pressure boundary (10CFR50, Appendix K).

(b) Coincident with the LOCA, offsite power is assumed become unavailable as a limiting condition. Consequently, the limiting condition, either availability or unavailability of offsite power, must be evaluated (10CFR50, Appendix A, General Design Criteria 35).

(c) A single component within the ECCS network fails coincident with the LOCA (10CFR50, Appendix K).

(d) RCIC system is not part of the DAEC ECCS.

Break Location and Size

The ECCS performance evaluation will consider the break of any pipe that forms part of the reactor coolant pressure boundary. This can include break sizes ranging from the maximum recirculation suction line break (2.523 sq. ft) down to 0.05 sq. ft. The maximum recirculation suction line break area consists of 2.127 sq. ft. from the vessel nozzle, 0.38 sq. ft. from the jet pumps, and 0.016 sq. ft. from the bottom head drain. The non-recirculation line breaks will also be evaluated. This includes the feedwater line, core spray line, and main steam line. Since these break locations are not limiting, only the maximum break size will be analyzed for these locations. A summary of break sizes to be evaluated for the SAFER/PRIME-LOCA analysis (Reference 6) is included in the table below.

Break Location	Break Area (ft^2)
Recirculation Suction Line	2.523*
Recirculation Discharge Line	NA
Core Spray Line	0.21
Feedwater Line	0.51
Steam Line (Inside Containment)	1.77
Steam Line (Outside Containment)	1.77

*Contributing Areas

2.127 ft2 (one 19.75" ID recirc suct nozzle)

0.016 ft2 (one 1.689" ID BHD nozzle)

0.380 ft2 (eight 2.95" ID jet pump nozzles)

The maximum break area for each break location is calculated using the minimum flow (cross-sectional) area in each possible flow path from the point of the break to inside the vessel. The maximum design tolerances are used in the calculation of the minimum flow area in order to ensure conservative values.

The core spray line and feedwater break areas are determined using the minimum pipe flow areas in their respective flow paths. The use of the minimum pipe flow area rather than the total flow area of the sparger nozzles ensures a conservative flow area value.

For steam lines inside containment, the initial break area prior to MSIV closure is comprised of minimum flow areas of the steam line nozzle/safe end and the steam flow limiter. Following MSIV closure, the break area is reduced to the minimum steam line nozzle/safe end flow area.

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Table 15.0 - 4

ECCS/LOCA Analysis Input Parameters (Form OPL-4)

18 - Analysis Assumptions and Initial Conditions
For steam lines outside containment, the initial break area prior to MSIV closure is comprised of the minimum flow area of all four of the steam flow limiters. Following MSIV closure, the break area is reduced to zero. **Assumption related to loss of Offsite Power** It is assumed that the loss of offsite power causes a trip of the reactor recirculation pump at the beginning of the event. For SAFER analyses, a time constant of 3 seconds is assumed for coastdown. **Additional Assumptions for ECCS Analysis** Two sets of initial reactor operating conditions are used for the standard ECCS performance evaluation. The nominal and upper bound calculations use the parameters designated "Nominal" in Section 1; while the parameters designated "Appendix K" are used in the Appendix K calculations. The initial reactor water level is assumed to be at normal water level for all nominal and upper bound calculations and for large (>1.0ft^2) recirculation line break Appendix K calculations. The initial reactor water level is assumed to be at the low water level scram setpoint for all small break. The high drywell pressure trip is assumed at time zero for all large breaks inside the containment. The reactor is assumed to scram on the high drywell pressure at time zero for all large breaks inside containment. The reactor is assumed to scram on low water level at time zero for all small break inside containment. The reactor is assumed to scram on the MSIV isolation due to high steam line flow at time zero for all steam line breaks outside the containment. The ECCS is assumed to initiate on high drywell pressure at time zero for all large breaks inside the containment. The ECCS is assumed to initiate at the appropriate low reactor water level for breaks outside the containment. The feedwater pumps are assumed to trip at the beginning of the event. The feedwater pumps are conservatively assumed to linearly coast down from the initial value to zero in 5 seconds. The feedwater flow is stopped immediately (no coastdown) for a feedwater line break. The hot channel dryout time will be calculated for the maximum recirculation line break. The hot channel dryout times will be estimated for recirculation line breaks between 1.0 ft^2 and the maximum. For break areas less than 1.0 ft^2, the core is expected to remain in nucleate boiling until core uncovery occurs. The drywell pressure is assumed to be constant at 14.7 psia throughout the event.

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Table 15.0 - 4

ECCS/LOCA Analysis Input Parameters (Form OPL-4)

18 - Analysis Assumptions and Initial Conditions
The high steam line flow trip is not explicitly modeled in SAFER. The analysis assumes that the high flow signal occurs at time zero. A full sized steam line break is assumed to occur at time zero, resulting in critical flow occurring immediately at the flow limiters. The flow limiters are typically sized for about 200% of original rated steam flow and the high flow trip signal is usually set to about 140% rated steam flow. Since critical flow occurs immediately and since the critical flow rate is much higher than the high flow trip setpoint, this assumption is reasonable for the SAFER steam line break analysis. In addition, the steam line break cases are not limiting with regards to peak cladding temperature and the temperature results are not sensitive to small changes (5-10 second) in the MSIV closure time. The ECCS-LOCA evaluation assumes LOOP coincident with the LOCA and assumes that the ECCS start times are dictated by the diesel generator start times.

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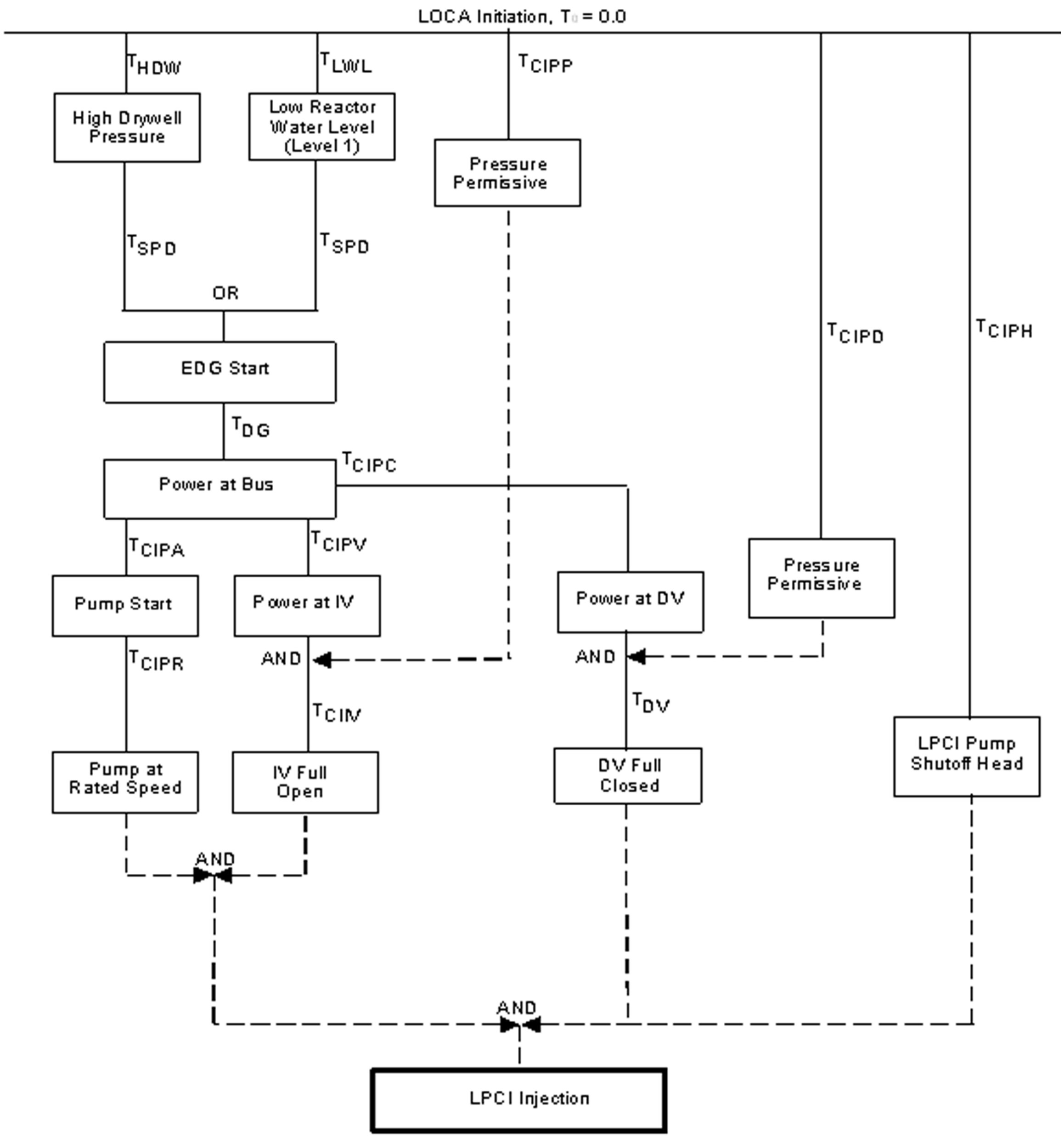
Table 15.0 - 4

ECCS/LOCA Analysis Input Parameters (Form OPL-4)

20 - Figure 1



Note: Delay times refer to solid lines between events. Dotted lines show logic without time delay. This logic diagram assumes a LOOP and is a simplification of true ECCS logic providing a conservative approach to the LOCA analysis.

Figure 1. Initiation Logic Diagram for Low Pressure Coolant Injection (LPCI) System

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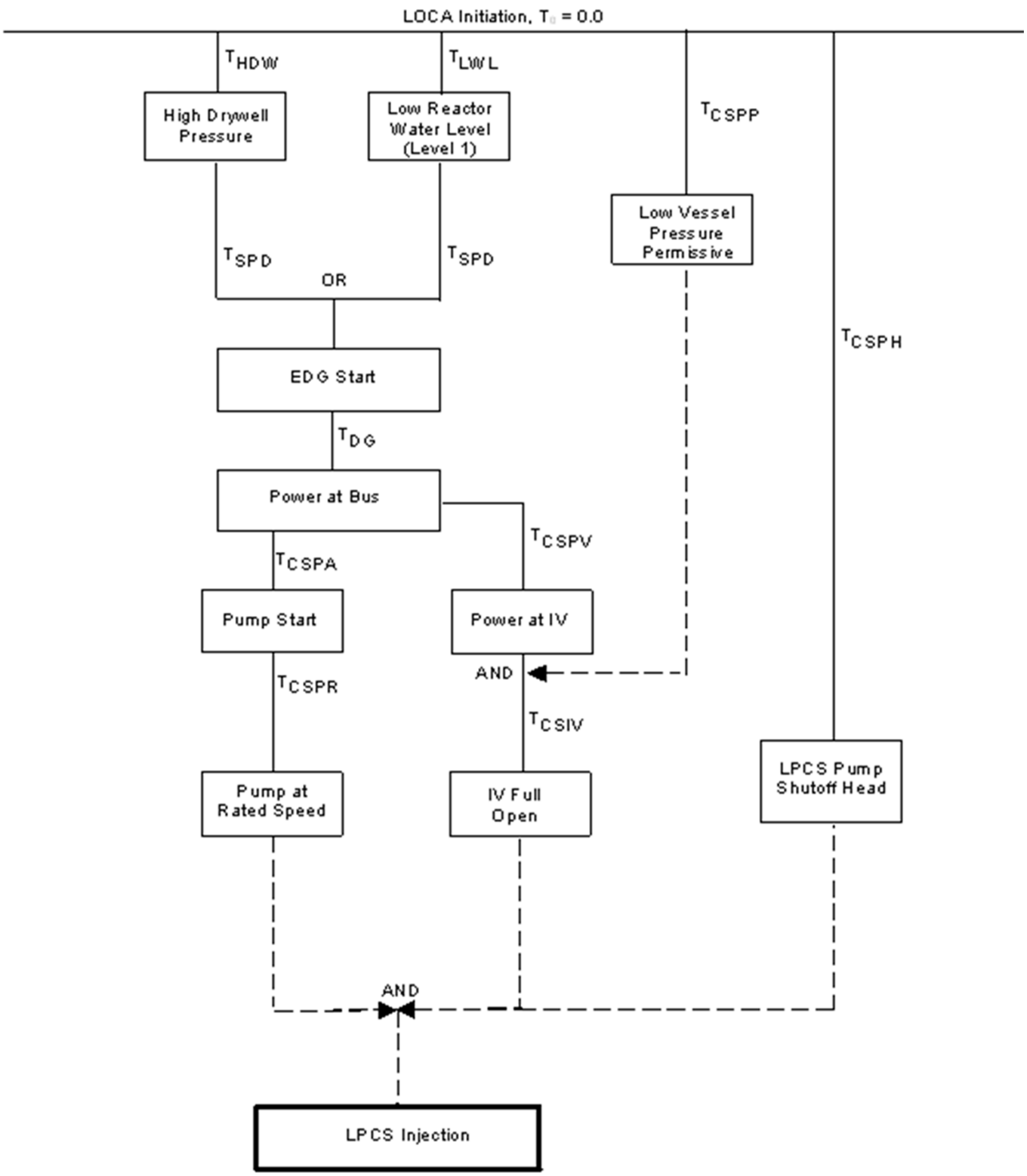
Table 15.0 - 4
ECCS/LOCA Analysis Input Parameters (Form OPL-4)

21 - Figure 2



Note: Delay times refer to solid lines between events. Dotted lines show logic without time delay. This logic diagram assumes a LOOP and is a simplification of true ECCS logic providing a conservative approach to the LOCA analysis.

Figure 2. Initiation Logic Diagram for Low Pressure Core Spray (LPCS) System

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Table 15.0 - 4

ECCS/LOCA Analysis Input Parameters (Form OPL-4)

22 - Figure 3

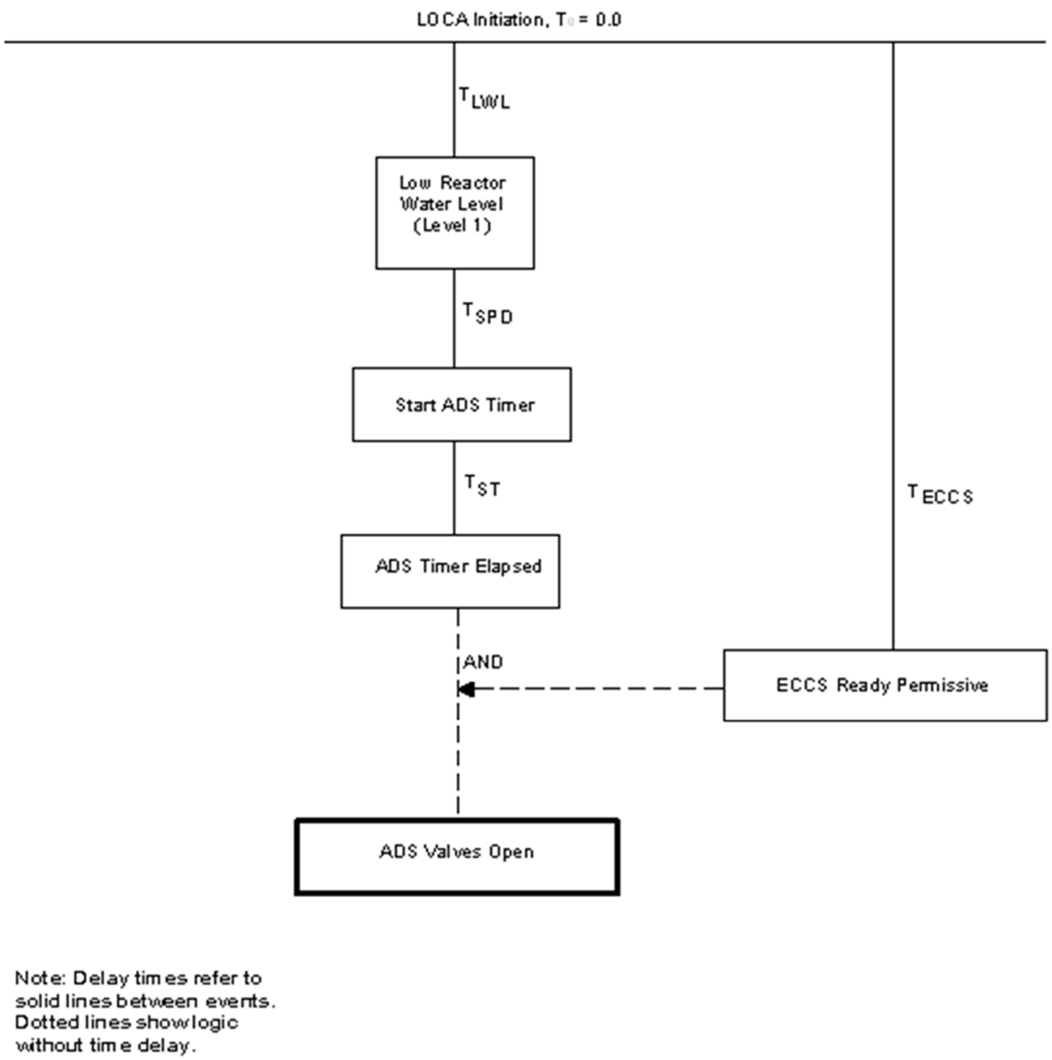

Figure 3. Initiation Logic Diagram for Automatic Depressurization System (ADS)

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Table 15.0 - 4

ECCS/LOCA Analysis Input Parameters (Form OPL-4)

23 - Figure 4

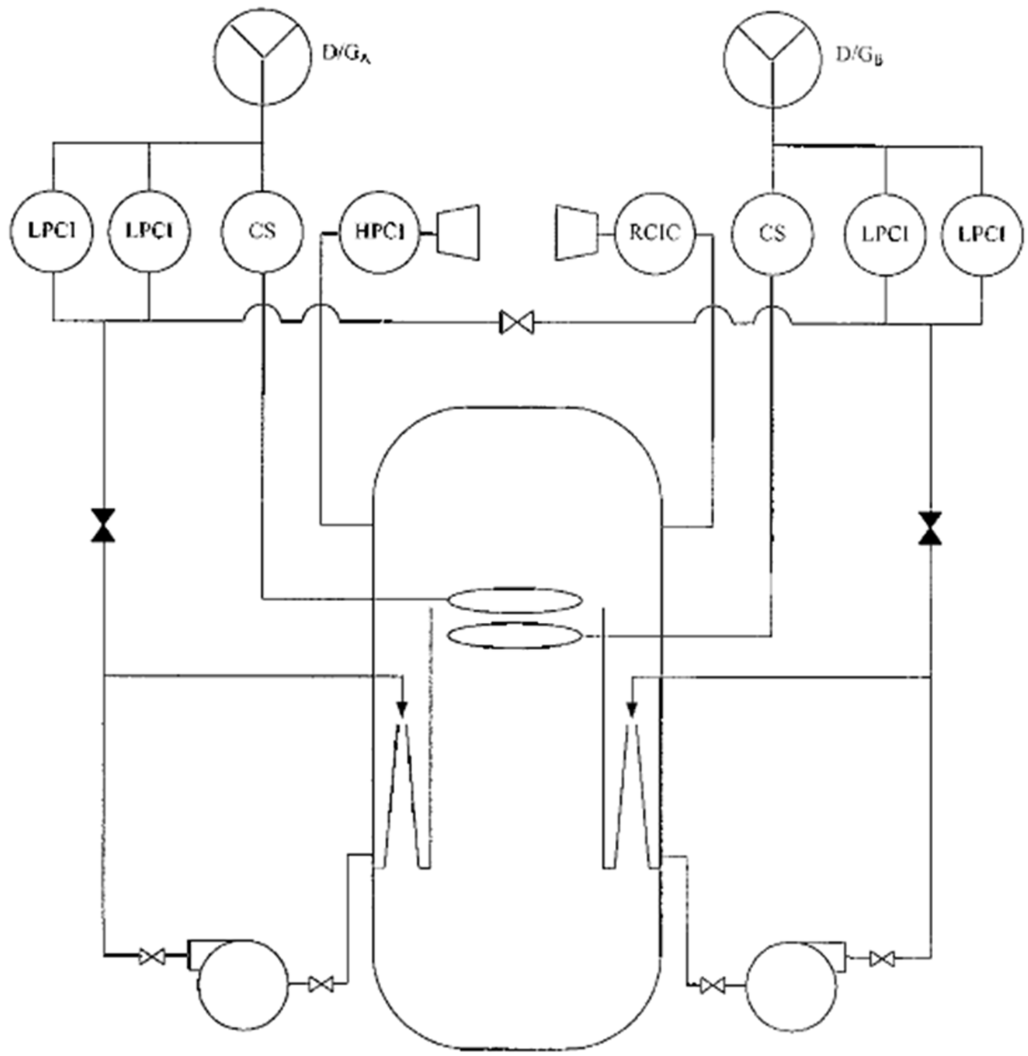

Figure 4. ECCS Configuration

Table 15.0 - 5

2012-020 **OPL-5 - Verification of Emergency Core Cooling System Single Failure Evaluation for SAFER/PRIME Analysis(1)**

2012-020 The table below shows the various combinations of Automatic Depressurization System (ADS), High Pressure Coolant Injection (HPCI) System, Core Spray (CS) System and Low Pressure Coolant Injection (LPCI) system which might be operable in an assumed design basis accident situation. In performing the ECCS performance analysis with SAFER/PRIME, GEH will assume that no postulated single active component will result in less than certain minimum combinations of systems remaining operable.

The following single, active failures will be considered in the ECCS performance evaluation:

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Assumed Failure(1)	Recirculation Suction Break Systems Remaining(2)
Battery	ADS, 1 CS, 2 LPCI
LPCI Injection Valve (LPCI IV)	ADS, 2 CS, HPCI
Diesel Generator (D/G)	ADS, 1 CS, HPCI, 2 LPCI
HPCI	ADS, 2 CS, 4 LPCI

Notes for OPL-5

(1) Other postulated failures are not specifically considered because they all result in at least as much ECCS capacity as one of the above assumed failures.
(2) Systems remaining, as identified in this table, are applicable to all non-ECCS line breaks. For a LOCA from an ECCS line break, the systems remaining are those listed, less the ECC system in which the break is assumed.
2012-020 (3) One ADS valve OOS case to be performed as a sensitivity study. This will not be part of licensing basis results.

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
1.	**Reactor Vessel**						
a.	***Initial Power***			(1)			
	1. Short-Term Response						
	Case 1 - 102% Licensed Power Uprate	MWt	1950	1			1950
	Case 1a - 102% Licensed Power Uprate	MWt	1950	1			1950
	Case 2 - 102% RTP	MWt	1691	1, (2)			1691
	Case 3 - 102% RTP/ MELLL	MWt	1691	1			1691
	Case 4 - 102% ORTP/ MELLL	MWt	1625	1, (2)			1625
	Case 5 - 102% of 47.5% LPU (near natural circulation point)	MWt	926	1, (3)			926
	Case 6 - 102% SLO/ MELLL	MWt	1303	1		(1)	1303
	Case 7 – 102% LPU (MELLL)	MWt	1950	1			1950
	2. Long-Term Response						
	a. All cases except SBO - 102% Uprated Power	MWt	1950	1			1950
	b. SBO - 100% Uprated Power	MWt	1912	1, (4)			1912
b.	Initial Core Flow			(6)			

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
	1. Short-Term Response*						
	Case 1 - 100% rated core flow	Mlb/hr	49.0	1			49.0
	Case 1a - 100% rated core flow	Mlb/hr	49.0	1			49.0
	Case 2 - 100% rated core flow	Mlb/hr	49.0	1			49.0
	Case 3 - 79.7% rated core flow	Mlb/hr	39.1	1	39.1	(2)	39.1
	Case 4 - 75% rated core flow	Mlb/hr	36.8	1			36.8
	Case 5 - 29% rated core flow	Mlb/hr	14.2	1			14.2
	Case 6 - 53% rated core flow (SLO)	Mlb/hr	26.0	1			26.0
	Case 7 – 99% rated core flow	Mlb/hr	48.5	1			48.5
c.	***Feedwater Temperature at Vessel Inlet***			(1), (6)			
	1. Short-Term Response (M3CPT/LAMB) – All short-term cases except Case 1a The feedwater temperature is not input to the GE M3CPT analysis since break flows are externally generated with the LAMB analyses.						

* GE did an evaluation of the short-term response at Increased Core Flow (105% of Rated Core Flow) conditions (Reference 15.0-60).

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
	1a. Short-Term Response (M3CPT/HEM) – Case 1a						
	The feedwater temperature is not input to the GE M3CPT analysis since no feedwater addition to the vessel is assumed.						
	2. Long-Term Response (SHEX)						
	a. All cases except SBO	°F	433.4	7, (6)			433.4
	b. SBO	°F	431.4	7, (6)			431.4
d.	***Decay Heat Model***			(1), (6)			
	1. Short-Term Response - (M3CPT/LAMB) – All short-term cases except Case 1a		ANS 5 + 20%	(8)	ANS 5.1 + 20% per 10CFR50 App. K	(3)	ANS 5.1 + 20% per 10CFR50 App. K
	1a. Short-Term Response – (M3CPT/HEM) – Case 1a		May-Witt	12			May-Witt
	2. Long-Term Response - (SHEX)		ANS 5.1 + 2σ	(5)			ANS 5.1 + 2σ

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
	3. SBO – (SHEX)			15, (4)	ANS 5.1 nominal	(4)	ANS 5.1 nominal
	4. Fuel Bundle Average Enrichment (for ANS 5.1+2σ and ANS 5.1 Nominal decay heat)	%	4.25	29, (31)			4.25
	5. End-of-Cycle Core Average Exposure (for ANS 5.1+2σ and ANS 5.1 Nominal decay heat)	GWt days /Short Ton	31.7	29, (31)			31.7
	6. Core Average Time at Power (Irradiation Time) (for ANS 5.1+2σ and ANS 5.1 Nominal decay heat)	Year	3.5	29, (31)			3.5
e.	***Initial Dome Pressure***			(1)			
	1. All cases except short-term Cases 5 & 6	psia	1055	2, 4, 6			1055
	2. Short-term Cases 5 and 6	psia	From Reference 2 heat balance	2, (6)		(5)	From Reference 27 heat balance
	3. SBO	psia		15, (4)	1040	(6)	1040

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
f.	Vessel volumes						
	1. Total vessel free volume	ft^3	10521	3, (6)	Open item	(7)	10521
	2. Vessel liquid volume						
	a. Subcooled	ft^3	4331	3, (6)	Open item	(7)	4331
	b. Saturated	ft^3	1989	3, (6)	Open item	(7)	1989
h.	Vessel related masses						
	1. Liquid mass in main steam lines to the inboard isolation valve	lbm	0	4			0
	2. Liquid Mass in one recirculation loop	lbm	15,738	4			15,738
	3. Liquid mass in the RHR shutdown line to the first normally closed valve	Ibm	350	4			350
	4a. Mass of RPV internals structure (excluding fuel and fuel assembly)	lbm	344600	3, (6)	Open item	(8)	344600
	4b. Mass of RPV, including top head but excluding vessel skirt.	lbm	777300	3, (6)	Open item	(8)	777300

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
	4c. Mass of RPV connected piping						
	1. Mass of recirculation piping for both loops	lbm	181789	5			181789
	2. Mass of RHR/LPCI shutdown piping	lbm	1630	5			1630
	3. Mass of Main Steam Lines to second isolation valve	lbm	88192	5			88192
	4. Mass of LPCI, CS, HPCI and RCIC lines to first normally closed valve	lbm	79810	5, (11)			79810
	5. Mass of fuel and fuel assembly	lbm	243000	3, (6), (26)	Open item	(9)	267300
i.	LOCA Break area						
	1. Short-Term Response						
	a. DBA Cases 1 and 1a (for peak pressure and temperature evaluation)	ft^2	2.523	4, (8), (9)			2.523
	b. DBA Case 2 (for Ref. 8 evaluation)	ft^2	2.523	4, (8), (9)			2.523
	2. Long-Term Response						
	a. DBA (also DBA for NPSH)	ft^2	2.523	4			2.523

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
	b. Small Steam Line Breaks	ft^2	Four cases: 0.01, 0.1, 0.25 and 1.0	9, (10)			Four cases: 0.01, 0.1, 0.25 and 1.0
j.	LOCA Break elevation (from bottom of vessel)						
	1. Recirculation Suction Line	ft	10.11975	24, (6)			10.11975
	2. Steam Line	ft	50.945	24, (6)			50.945
k.	Break critical flow model						
	1. Short-Term cases (except Case 1a) 1a. Short-term Case 1a 2. Long-Term cases		Slip HEM HEM	conservative 12 (32)			Slip HEM HEM
l.	Elevation for Level 1 (low pressure ECCS & ADS trips) (from bottom of vessel)	in	350.0	7			350.0
m.	Elevation for Level 2 (high pressure ECCS trip) (from bottom of vessel)	in	447.3	7			447.3

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
n.	Elevation for Level 8 (ECCS high level trips) (from bottom of vessel)	in	571.6	7			571.6
o.	Turbine steam flow rate	Mlb/hr	From Ref. 2 & 27 heat balance	2, 27 (6)		(10)	From Ref. 2 & 27 heat balances (27)
p.	Time at which MSIVs start to close						
	1. DBA	sec	0.5	23			0.5
	2. All other cases	sec	0.5	Same as DBA			0.5
q.	Time at which MSIVs are completely closed						
	1. DBA	sec	3.5	23			3.5
	2. All other cases	sec	3.5	Same as DBA			3.5
2.	**Drywell/Vent System**						
a.	***Total drywell free volume*** (including vent system)	ft^3	130000	4, (1)			130000
b.	***Initial drywell pressure***			(1)			

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
	1. DBA	psig	2.3	4			2.3
	2. DBA, IBA & SBA for Ref. 8 evaluation	psig	0.59	8			0.59
	3. NPSH	psig	0.5	4			0.5
	4. SBO	psig		15, (4)	0.7	(11)	0.7
	5. All other cases	psig	2.3	Same as DBA			2.3
c.	***Initial drywell temperature***			(1)			
	1. DBA	°F	135	4			135
	2. DBA, IBA & SBA for Ref. 8 evaluation	°F	135	8			135
	3. NPSH	°F	150	4	135	(12)	135
	4. All other cases	°F	135	Same as DBA			135
d.	***Initial drywell relative humidity***			(1)			
	1. DBA	%	20	4			20
	2. DBA, IBA & SBA for Ref. 8 evaluation	%	20	8			20
	3. NPSH	%	100	4			100
	4. Small Steam Line Breaks	%		9	100	(13)	100
	5. All other cases	%	20	Same as DBA			20
e.	***Number of downcomers***		48	4, (1)			48

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
f.	Inside Diameter of each downcomer	ft	1.958	4		(14)	1.958
g.	Drywell holdup volume	ft^3	1955	4			1955
h.	Drywell pool surface area (in contact with drywell airspace)	ft^2	1248	4			1248
i.	***Submergence of downcomers***			(1)			
	1. Low water level	ft	3.026	4			3.026
	2. High water level	ft	3.359	4			3.359
j.	Loss coefficient for vent system (including downcomer exit loss)		4.65	4		(15)	4.65
k.	Additional parameters						
	1. Number of main vents		8	4			8
	2. Number of vent header miter bends		16	4			16
	3. Main vent I.D.	ft	4.75	4			4.75
	4. Vent header I.D.	ft	3.5	4			3.5
	5. Angle of main vent with horizontal	deg	23.823	4			23.823

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
	6. Angle of first downcomer miter bend with horizontal (Note: Downcomers are straight down from the vent header)	deg	90	4		(16)	90
l.	Duration of drywell temperature EQ profile	days	400	9	120		120
3.	**Wetwell/Suppression Pool**						
a.	***Initial suppression pool volume***			(1)			
	1. Low water level (LWL)	ft^3	58900	4			58900
	2. High water level (HWL)	ft^3	61500	4			61500
b.	***Initial suppression pool temperature***			(1)			
	1. All cases except DBA for Reference 8 evaluation	°F	95	4			95
	2. DBA for Reference 8 evaluation	°F	81	8			81
c.	***Initial wetwell free airspace volume***			(1)			
	1. Short-Term Response - High water level (HWL)	ft^3	96670	4	94070	(17)	94070

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
	2. Long-Term Response - Low water level (LWL)	ft^3	94070	4	96670	(18)	96670
d.	***Initial wetwell airspace pressure***			(1)			
	1. DBA	psig	2.3	4			2.3
	2. DBA, IBA & SBA for Ref. 8 evaluation	psig	0.59	8			0.59
	3. NPSH	psig	0.5	4			0.5
	4. SBO	psig		15, (4)	0.7	(19)	0.7
	5. All other cases	psig	2.3	Same as DBA			2.3
e.	***Initial wetwell airspace temperature***			(1)			
	1. DBA	°F	95	4			95
	2. DBA for Ref. 8 evaluation	°F	81	8			81
	3. All other cases	°F	95	Same as DBA			95
f.	***Initial wetwell airspace relative humidity***			(1)			
	1. DBA and DBA for NPSH	%	100	4			100
	2. All other cases	%	100	Same as DBA			100

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
g.	Suppression pool surface area (in contact with suppression chamber airspace)	ft^2	7763	4			7763
h.	Torus major radius	ft	49.333	4			49.333
i.	Torus cross-sectional (minor) radius	ft	12.833	4			12.833
j.	Maximum allowable containment mass leakage (for NPSH case)	% / day	5.0	4			5.0
k.	Acceptable peak suppression pool temperature for DBA	°F	IES to provide		281		281
l.	Acceptable peak suppression pool temperature for SBO	°F	Heat Capacity Temperature Limit	Figure 1 of Reference 15	HCTL for EPU shall be used	(20)	HCTL for EPU (to be provided by IES)
4.	**SRV**						
a.	Flow loss coefficient for each SRV line (include entrance & exit losses)		N/A	(12)			

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
b.	Minimum number of SRV openings in normal set before two SRVs are switched to low-low set		1	22		(21)	1
c.	Minimum flow area of each SRV line (for liquid discharge)	ft^2	N/A	(12)			
d.	Suppression pool temperature above which vessel controlled cooldown is initiated	°F	120	10			120
e.	Time delay in starting controlled vessel cooldown from the time of reaching 120°F suppression pool temperature (includes operator action, valve stroke time, etc.)	sec	0	(28)			0
f.	Vessel controlled cooldown rate using SRVs	°F/hr	100	Standard value			100
g.	Vertical elevation drop from SRV entrance at main steam line to SRV quenchers	ft	IES to provide		57.4		57.4
h.	SRV quenchers initial submergence at LWL	ft	IES to provide		6.125		6.125

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
i.	Opening and closing setpoints, and number of SRVs, for each group in normal set	psia	Table 1	7, 21			Table 1
j.	Opening and closing setpoints, and number of SRVs, for each group in low-low set	psia	Table 1	22			Table 1
k.	Number of SRVs available for pressure and temperature control		4	10		(22)	4
5.	**HPCI**						
a.	Vessel water level (Level 2) below which HPCI is automatically actuated	in	447.3	22		(23)	447.3
b.	Vessel water level (Level 8) above which HPCI is automatically shut off	in	571.6	7			571.6
c.	Vessel pressure below which credit is taken for HPCI shut off	psia	165	22		(23)	165
d.	Maximum suppression pool liquid volume above which suppression pool replaces CST as water source for HPCI	ft^3	Not used	(13)			

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
e.	Drywell pressure for actuation of HPCI	psig	$\geq$ 2.0	22	2.3	(23)	2.3
f.	If high drywell pressure and high vessel water level coexist, HPCI will cycle between level 2 and Level 8 (Yes or No)		No	14	Yes		Yes
g.	HPCI flow rates vs RPV pressure	gpm psia	3000 165-1170	17 17		(24) (24)	3000 165-1135
h.	Suction from CST (Yes or No)		Yes for SBO; No for all other cases				Yes for SBO; No for all other cases
i.	HPCI turbine steam flow rates vs RPV pressure	lbm/hr psia	Table 5	17		(24)	Table 5
j.	Maximum HPCI delay time	sec	30	22	45	(23)	45
k.	Submergence of HPCI pump suction strainer at LWL	ft	IES to provide	(16)	2.677		2.677
6.	**RHR/LPCI**						
a.	***Heat exchanger K-value (per HX)***			(1)			

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
1.	DBA (LPCI, 1 RHR loop, 1 RHR pump (4800 gpm), 2 RHR SW pumps (4080 gpm total), 1 HX)	Btu/sec-°F	135	4, (15)			135
2.	IBA & SBA (Reference 8 evaluation) (Pool cooling, 1 RHR loop, 1 RHR pump (4800 gpm total), 2 RHR SW pumps (4080 gpm), 1 HX)	Btu/sec-°F	135	Same as DBA			135
3.	NPSH (Containment spray, 1 RHR loop, 1 RHR pump (4800 gpm), 2 RHR SW pumps (5200 gpm total), 1 HX)	Btu/sec-°F	141	4, (15)			141
4.	Steam line breaks (Containment spray, 1 RHR loop, 1 RHR pump (4800 gpm), 2 RHR SW pumps (4080 gpm total), 1 HX)	Btu/sec-°F	135	Same as DBA			135
5.	Drywell bypass leakage (Containment spray, 1 RHR loop, 1 RHR pump (4800 gpm), 2 RHR SW pumps (4080 gpm total), 1 HX)	Btu/sec-°F	135	Same as DBA			135

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
	6. NUREG-0783 (Pool cooling and shutdown cooling, 2 RHR loops, 4 RHR pumps (two in each loop, 19200 gpm total), 4 RHR SW pumps (8160 gpm total), 2HX)	Btu/sec-°F	142 (per HX)	Same as Case 1 (All ECCS) of Reference 4		(25)	142 (per HX)
b.	***Heat exchanger initiation time***			(1)			
	1. DBA	sec	600	4			600
	2. IBA & SBA (Ref. 8 evaluation)	sec	600	Same as DBA			600
	3. NPSH	sec	600	4			600
	4. Steam line breaks	sec	600	(18)			600
	5. Drywell bypass leakage	sec	Time of cont. spray initiation	Appendix B			Time of cont. spray initiation
c.	***Service water temperature***	°F	95	4, (1)			95
d.	***Drywell spray initiation time***			(1)			
	1. Steam line breaks (other than 0.01 ft^2)	sec	600	9			600
	2. Steam line break (0.01 ft^2)	sec		9	1800	(26)	1800
	3. DBA for NPSH	sec	600	4			600
	4. Drywell bypass leakage	sec	TBD	Appendix B			

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
e.	***Wetwell spray initiation time***			(1)			
	1. NPSH	sec	600	4			600
	2. Steam line breaks	sec	600	9, (18)			600
	3. Drywell bypass leakage	sec	TBD	Appendix B			
f.	***Drywell spray flow rate (1 RHR pump)***			(1)			
	1. NPSH	gpm	4560	4			4560
	2. SLBs & drywell bypass leakage	gpm	4560	Same as NPSH			4560
g.	***Wetwell spray flow rate (1 RHR pump)***			(1)			
	1. NPSH	gpm	240	4			240
	2. SLBs & drywell bypass leakage	gpm	240	Same as NPSH			240
h.	Average vertical distance between drywell spray nozzles and bottom of drywell	ft	IES to provide		49.105 (upper header) 27.438 (lower header)	(27)	38.27 (29)
i.	Average drywell spray droplet diameter	ft	0.002	26			0.002

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
j.	Average vertical distance between wetwell spray nozzles and suppression pool surface at LWL	ft	IES to provide		13.432	(27)	13.432
k.	Average wetwell spray droplet diameter	ft	0.002	26			0.002
l.	***Number of pumps***			(1)			
	1. DBA, IBA & SBA						
	a. t < 600 sec		2	4			2
	b. t ≥ 600 sec		1	4			1
	2. Short-term NPSH (t ≤ 600 sec)		4	4			4
	3. Long-term NPSH & steam line breaks						
	a. Before 600 sec		2	4			2
	b. After 600 sec		1	4			1
	4. Station blackout		0	No RHR is available			0
	5. Drywell bypass leakage					(28)	

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
	a. Before initiation of containment sprays				2		2
	b. After initiation of containment sprays				1		1
	6. NUREG-0783		4	2 RHR loops (Case 1B)			4
m.	RHR/LPCI flow rate (per pump)						
	1. DBA, IBA & SBA	gpm	4800	4			4800
	2. Short-term NPSH	gpm	6500	4			6500
	3. Long-term NPSH						
	a. t < 600 sec	gpm	6500	4			6500
	b. t $\geq$ 600 sec	gpm	4800	4			4800
	4. Steam line breaks	gpm	4800	Same as DBA			4800
	5. Drywell bypass leakage	gpm	4800	Same as SLBs			4800
	6. NUREG-0783	gpm	4800	Same as SLBs			4800
n.	RHR/LPCI pump heat (per pump)	hp	600	4			600

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
o.	LPCI Shutoff head	psid	197	22		(23)	197
p.	Total LPCI time delay	sec	40	22	50	(23)	50
q.	Drywell pressure above which drywell sprays can operate	psig	$\geq$ 2.0	22	2.3	(23)	2.3
r.	Wetwell pressure above which automatic wetwell spray can operate	psia	Not used	(19)			
s.	Wetwell pressure below which wetwell spray will be turned off	psia	Not used	(19)			
t.	Drywell pressure above which LPCI will automatically be actuated	psig	$\geq$ 2.0	22	2.3	(23)	2.3
x.	Vessel water level (Level 1) below which LPCI will automatically be actuated	in	447.3	22	350.0	(23)	350
y.	Vessel water level (Level 8) above which LPCI will be shut off	in	571.5	7	571.6		571.6

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
z.	Minimum bypass flow per LPCI pump in injection mode	gpm	250	13, (30)	0		250
aa.	Submergence of RHR pump suction strainers at LWL	ft	IES to provide	(16)	1.195		1.195
bb.	Maximum shutdown cooling rate	ºF/hr	IES to provide	28		(25)	80
7.	**Low Pressure Core Spray**						
a.	***Number of pumps***			(1)			
	1. DBA, IBA & SBA						
	a. t < 600 sec		1	4			1
	b. t ≥ 600 sec		1	4			1
	2. Short-term NPSH (t ≤ 600 sec)		2	4			2
	3. Long-term NPSH & steam line breaks						
	a. Before 600 sec		1	4			1
	b. After 600 sec		1	4			1
	4. Station blackout		0	No CS is available			0

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
	5. Drywell bypass leakage		1	Same as SLBs before 600 sec			1
	6. NUREG-0783		2	Both CS pumps are available			2
b.	CS flow rate (per pump)						
	1. DBA, IBA & SBA	gpm	3100	4			3100
	2. Short-term NPSH	gpm	4500	4			4500
	3. Long-term NPSH						
	a. t < 600 sec	gpm	4500	4			4500
	b. t ≥ 600 sec	gpm	3100	4			3100
	4. Steam line breaks	gpm	3100	Same as DBA			3100
	5. Drywell bypass leakage	gpm	3100	Same as SLBs			3100
	6. NUREG-0783	gpm	3100	Same as SLBs			3100

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
c.	Core spray pump heat (per pump)	hp	700	4			700
d.	Shutoff head	psid	264	22		(23)	264
e.	Total pump time delay	sec	27	22	37	(23)	37
f.	Drywell pressure above which CS will automatically be actuated	psig	$\geq$ 2.0	22	2.3	(23)	2.3
g.	Vessel water level (Level 1) below which CS will automatically be actuated	in	447.3	22	350.0	(23)	350
h.	Vessel water level (Level 8) above which CS will be shut off	In	571.6	7			571.6
i.	Submergence of CS pump suction strainers at LWL	ft	IES to provide	(16)	1.902		1.902
j.	Minimum bypass flow per pump	gpm	312	18, (30)			312
8.	**RCIC**						
a.	Vessel water level (L2) below which RCIC is automatically actuated	in.	447.3	7		(29)	447.3

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
b.	Vessel water level (L8) above which RCIC is automatically shut off	in.	571.6	7		(30)	571.6
c.	Rated flow	lbm/sec	55.6	7	55.15	(31)	55.15
d.	Time delay	sec	30	7, 15, (7)			30
e.	Turbine steam flow rate vs vessel pressure						
	1. Below 165 psia	lbm/hr	0	20			0
	2. At 165 psia	lbm/hr	6650	20			6650
	3. At 1055 psia	lbm/hr		20	20600	(32)	20600
	4. At 1135 psia	lbm/hr	21500	20			21500
9.	**SEHR (European BWR6 Only)**		N/A				
10.	**Feedwater**						

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
2014-006 a.	***Feedwater system liquid and metal masses***	lbm		4, (25)	"OPL-4a (AEP) FW system masses.xls'	(33)	Appendix A
b.	***Amount of hot inventory available for injection***	lbm	GE to determine	(1), (24)		(34)	
c.	***Corresponding enthalpy vs mass***	Btu/lbm lbm	GE to determine	(1), (24)		(34)	
d.	Flow rate	Mlb/hr	To be determined from Ref. 2 heat balance	2		(34)	To be determined from Ref. 2 & 27 heat balances
11.	**Closed Cooling Loop**		N/A				
12.	**RWCU**		Not Modeled	(20)			
13.	**Upper Pool (Mark III Only)**		N/A				

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
14.	**CRD**		Not Modeled	(21)			
15.	**Wetwell-to-Drywell Vacuum Breakers**						
a.	Pressure difference between wetwell and drywell for vacuum breakers to be fully open	psid	0.35	4			0.35
b.	Total loss coefficient of each vacuum breaker line (per valve system)		2.41	4			2.41
c.	Total flow area of one vacuum breaker line (per valve system)	ft^2	1.396	4			1.396
d.	Number of valve systems						
	1. NPSH			4	7		7
	2. All other cases			4	6		6
16.	**Reactor Building-to-Wetwell Vacuum Breakers**		Not modeled				
17.	**Isolation Condenser**		N/A				
18.	**Reactor Building**		Not modeled				

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
19.	**Drywell-to-Wetwell Bypass Leakage**						
a.	***Acceptable effective bypass leakage area (A/√K)***	ft^2	0.11	16, (22)			0.11
b.	***Suppression chamber pressure at which operator will be alerted to the existence of a bypass leakage path***	psig	10	16, (22)	35	(35)	35
c.	***Operator action time***	sec	600	16, (22)			600
20.	**CST (not modeled except SBO)**			(1)			
a.	***Available mass for vessel makeup***	lbm	75,000		620482	(36)	620482
b.	***Water temperature***	°F	90	20	100	(37)	100
21.	**Other plant unique parameters**		None				
22.	**Initial drywell heat sources (for SBO evaluation)**						
a.	Reactor pressure vessel	Btu/sec	125.6	15, 25, (6)			125.6

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

No	Parameter	Unit	Proposed by GE	GE References (Notes)	Proposed by Customer (IES)	IES References (Notes)	Resolved for Analysis
b.	Recirc pump casing	Btu/sec	8.3	15, 25, (6)			8.3
c.	Recirc piping	Btu/sec	73.6	15, 25, (6)			73.6
d.	SRVs	Btu/sec	33.6	15, 25, (6)			33.6
e.	Biological shield	Btu/sec	13.9	15, 25, (6)			13.9
f.	CRD piping (scram)	Btu/sec	111.1	15, 25, (6)			111.1
g.	Feedwater piping	Btu/sec	39.5	15, 25, (6)			39.5
h.	RHR piping	Btu/sec	23.3	15, 25, (6)			23.3
i.	RWCU piping	Btu/sec	16.7	15, 25, (6)			16.7
j.	RCIC Piping	Btu/sec	28.2	15, 25, (6)			28.2
k.	LPCS piping	Btu/sec	7.0	15, 25, (6)			7.0
l.	HPCI piping	Btu/sec	8.8	15, 25, (6)			8.8
m.	SLCS piping	Btu/sec	5.3	15, 25, (6)			5.3
n.	Unidentified leakage	Btu/sec	61.2	15, 25, (6)			61.2
o.	SRV leakage	Btu/sec	102.9	15, 25, (6)			102.9
p.	SRV discharge lines	Btu/sec	95.8	15, 25, (6)			95.8
q.	Miscellaneous	Btu/sec	166.1	15, 25, (6)			166.1

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

TABLE 1 - SRV Setpoint Data Proposed by GE

Normal Set Mode

Group	Number of SRVs	Opening Setpoint (psia)[7]	Closing Setpoint (psia)[21]
1	1	1158.0	1111.7*
2	1	1168.3	1121.6*
3	2	1178.6	1131.5*
4	2	1188.9	1141.3*
5**	2	1291.9	1240.2*

* 96% of opening setpoint, per Reference 21.

** Safety Valves.

Low-Low Set Mode

Group	Number of SRVs	Opening Setpoint (psia)[22]	Closing Setpoint (psia)[22]
1	1	1028.7	939.7
2	1	1033.7	944.7

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

TABLE 2 - Drywell Heat Sinks Proposed by GE (Reference 4, Attachment C, Page C-1)

Sink No.	Sink Description	Total Exposed Surface Area (ft^2)	Material	Shell Thickness (ft)	Exposure	Mass (lb)
1	Zone I	2490	Steel	0.114	1	139091
2	Zone II	2330	Steel	0.063	1	71927
3	Zone III	5370	Steel	0.063	1	165772
4	Zone IV	4650	Steel	0.125	1	284813
5	LOCA Vent System	8920	Steel	0.021	1, 2	91787

* Exposure –

1 = Drywell atmosphere

2 = Wetwell atmosphere

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

TABLE 3 - Wetwell Heat Sinks Proposed by GE (Reference 4)

Sink No.	Sink Description	Total Exposed Surface Area (ft^2)*	Material	Total Thickness (ft)	Exposure	Mass (lb)*
1	Upper Torus Shell	-	Steel	0.042	Wetwell airspace	-
2	Lower Torus Shell (not modeled in analysis since it is conservative for the suppression pool temperature response)	-	Steel	0.045	Suppression Pool	-

* Surface area and mass of the heat sinks will be calculated based on the given torus dimensions.

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

TABLE 4 - Thermophysical Properties of Passive Heat Sink Materials Proposed by GE (Note 23)

Material	Density $\left(\frac{\text{lbm}}{\text{ft}^3}\right)$	Specific Heat $\left(\frac{\text{BTU}}{\text{lbm }^\circ\text{F}}\right)$	Thermal Conductivity $\left(\frac{\text{BTU}}{\text{hr ft }^\circ\text{F}}\right)$
Carbon Steel	490	0.11	26

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

TABLE 5 - HPCI Turbine Steam Flow Rates vs RPV Pressure (Reference 17)

RPV Pressure (psia)	HPCI Turbine steam flow rates (lbm/hr)
0	0*
165	52000
450	**
550	**
650	**
750	**
850	**
1055	112000
1135	123000

* Turbine steam flow rate is 0 for RPV pressures below 165 psia.

** Between RPV pressures of 165 psia and 1055 psia, HPCI turbine flow rate will be obtained by linear interpolation.

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

GE Notes

(1) Input parameters of most significance are denoted in ***bold and italic*** fonts with 10% gray shading. If IES does not provide a proposed value, the GE proposed value will be taken as the resolved value and will be used in the analysis.

(2) ORTP = 1593 MWt; RTP = 1658 MWt, LPU = 1912 MWt.

(3) The actual power/flow point to be analyzed may not exactly correspond to the natural circulation point but will closely approach it. This is due to computational difficulties with the GE LAMB code at very low core flow conditions. However, the results can still be used to address the expected containment response at this point. Also, the MELLL point of 102% power and 99% of core flow is effectively covered by Case 1 (102% power/100% core flow).

(4) Station blackout is not considered a design basis event, therefore more realistic input values and initial conditions can be used in analysis. Also, the use of 100% rated power is consistent with the Reference 15 analysis.

(5) Short-term analysis decay heat is that used to calculate the shutdown power for the LOCA (LAMB) analysis; long-term analysis uses shutdown power based on Duane Arnold plant-specific ANS 5.1 + 2σ for 24-month cycles. This decay heat is documented in the GE letter from C.L. Martin to W.F. Farrell, "Decay Heat Table for Duane Arnold Power Uprate Equilibrium Cycle," NSA-00-002, February 17, 2000.

(6) Value is listed for information only. No IES confirmation is required.

(7) Values used in the GE Reference 15 analysis.

(8) The required inputs are obtained and used per the LAMB blowdown analytical methodology.

(9) Not required for short-term analysis; obtained from externally calculated (LAMB) break flows and enthalpies.

(10) Based on GE's experience, these break sizes will provide sufficient envelopes to establish a drywell temperature EQ profile. These break areas were also used in the Reference 9 analysis.

(11) The total mass of LPCI, CS, HPCI and RCIC lines to first normally closed valve is 64000 + 10200 + 4750 + 860 = 79810.

(12) This input value is required only for cases with liquid discharges through the SRVs. Since the current analyses do not involve liquid discharge through the SRVs, this value is not used.

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

(13) CST is used only for the SBO analysis. It will be assumed that, when the suppression pool level rises to a certain value and automatically initiating the switch of HPCI suction from the CST to the suppression pool, the operator will override it and keep the HPCI suction at the CST until the available CST water inventory is depleted.

(14) The CST system (including suction line) is not considered safety grade. Therefore, only for the non-design-basis event of an SBO is credit taken for the CST water.

(15) The RHR HX K-value is based on the pump configurations for the different cases to be analyzed. The RHR HX K-values given in Item 6.a are those given in Reference 4 and need IES re-confirmation.

(16) This value is needed to determine whether a NUREG-0783 analysis with the EPU is necessary. If the SRV quenchers are located above the top of the ECCS pump suction strainers, then a NUREG-0783 analysis is not necessary. If the SRV quenchers are located below the top of the ECCS pump suction strainers, then a NUREG-0783 analysis will be performed.

(17) During an SBO, it is assumed that no containment heat removal is available during the coping duration.

(18) GE report MDE-14-0186, January 1986 (Duane Arnold Energy Center Drywell Temperature Analysis) assumes that the wetwell spray is initiated at 300 seconds. To be conservative, it will be assumed in the current analysis that the wetwell spray will be initiated at 600 seconds, consistent with other manual actions.

(19) The drywell and wetwell sprays will be assumed to operate continuously throughout the event.

(20) Water mass in RWCU is very small and may be neglected for containment analysis.

(21) It is conservative to assume that the CRD flow stops at time 0, since the CRD water is relatively cold.

(22) Analysis will be performed to see if the current drywell bypass leakage capability ($A/\sqrt{K}$) of 0.11 ft^2 remains valid with the assumed operator actions and with the EPU. If not, revision of certain operator actions may be necessary.

(23) Typical values used in other BWR evaluations.

(24) The feedwater masses and enthalpies will be evaluated as part of the long-term analysis. The feedwater enthalpies will consider the power uprate conditions.

(25) IES to confirm Attachment B of Reference 4.

(26) 10% has been added to the GE proposed value of 243000 lbm to cover potential variations of future fuels. See GE response to IES comment #9 in Appendix D.

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

(27) It takes a finite period of time for the MSIVs to fully close. During this period, steam continues to flow to the condenser. The steam flow rates specified in References 2 and 27 are for the given pressure. During the time prior to the MSIV closure, the reactor pressure may change (either up or down). In such cases, the steam flow rate at a given pressure is obtained by linear interpolation or extrapolation from the rated point and the point of 0 pressure.

(28) The assumption of early vessel depressurization is conservative. A vessel depressurization results in the release of the sensible energy to the suppression pool and therefore higher suppression pool temperature. For the steam line breaks, an early vessel depressurization leads to a higher spray temperature and therefore a higher drywell and wetwell airspace temperature. For the SBO, an early vessel depressurization ensures that the vessel will be depressurized down to the desired pressure by the end of the coping duration.

(29) Use average of 49.105 ft and 27.438 ft. The containment response is not sensitive to this input value.

(30) The minimum bypass flows for LPCI and CS and modeled so that, when the vessel pressure is above the shutoff head, the LPCI/CS system cannot inject into the vessel; instead the flow will return to the suppression pool via the minimum bypass flow. This is to insure that the pump heat will be added to the containment system, making the analysis conservative.

(31) The values for the fuel bundle average enrichment, end-of-cycle core average exposure and core average time at power used for the Reference 23 analysis are 4.0%, 33.7 GWt days/short ton and infinite irradiation, respectively. Fuel bundle average enrichment and end-of-cycle core average exposure have small impact on the decay heat. However, the core average time at power has a significant effect on the decay heat. The assumption of an infinite core average time at power in the Reference 23 analysis is conservative. The use of 3.5 years for the core average time at power for the EPU analysis is in line with the current and expected future fuel configuration at DAEC and is consistent with GE's analysis approach for other BWR plants.

(32) For the long-term cases, the break flow model has negligible impact on the containment response.

GE References

1. DAEC-AEP Task Report T0201, "Power Flow Map", GE report GE-NE-A22-00100-04-01, Rev. 0, February 2000.
2. DAEC-AEP Draft Task Report T0100, "Heat Balance", GE report GE-NE-A22-00100-01, March 1, 2000.
3. DAEC vessel-related volumes, performed in 1983 and documented in DRF A13-92 (GE internal document).

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

4. Duane Arnold OPL-4A, dated 12/11/97, approved by IES via IES Letter No. NG-97-2145, from S. Huebsch (IES) to E.G. Thacker (GE), dated December 13, 1997.

5. “Containment Data,” IES document APED-B11-194, Rev. 2 (GE document 22A5745, Rev. 2).

6. “Heat Balance Inputs from OPL-4a,” E-mail from R. McGee (IES) to W.F. Farrell (GE), 3/3/2000.

7. “DAEC AEP Resolved OPL-3,” Letter from W. Farrell (GE) to R. McGee (IES), GEDA-AEP-100 Rev. 1, February 21, 2000.

8. “Mark I Containment Program Plant Unique Load Definition Duane Arnold Energy Center Unit 1,” GE report NEDO-24571, Rev. 1, March 1982.

9. “Duane Arnold Energy Center Drywell Temperature Analysis,” GE report MDE-14-0186, January 1986, a copy of which is contained in DRF T23-00620 (GE internal document).

10. “Duane Arnold Energy Center Suppression Pool Temperature Response,” GE report NEDC-22082-P, March 1982. Also, Supplement 1, April 1984.

11. “Duane Arnold Energy Center Power Uprate,” GE report NEDC-30603-P, Class III, May 1984.

12. Moody, F.J., “Maximum Flow Rate of A Single Component, Two-Phase Mixture ,” Transaction of the ASME, Volume 87, Series C, 1966.

13. RHR Process Diagram, IES drawings APED-E11-008<1>, Rev. 6, and APED-E11-008<2>, Rev. 4.

14. “Functional Control Document,” IES drawing APED-E41-012 (Sheet 2), Rev. 7, 11/7/92.

15. “Duane Arnold Energy Center Evaluation of the Containment Response to A Station Blackout Event,” GE report NEDC-31783, May 1990.

16. DAEC UFSAR Sections 6.2.1.3.4 and 6.2.1.3.5.

17. HPCI Process Diagram, IES drawing APED-E41-002, Rev. 5.

18. “Core Spray Process Diagram,” IES drawing APED-E21-001, Rev. 2 (GE drawing 161F267CA, Rev. 1).

19. GE response to TSD T0404 Comments on Revised TSD – 3/22/00.

20. RCIC Process Diagram, IES drawing APED-E51-003, Rev. 5.

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)

21. “Nuclear Boiler System Data Sheet,” IES document APED-B21-074<1>, Rev. 0.
22. “Approval of Resolved OPL-4/5 Forms for DAEC-AEP,” Letter from R. McGee (IES) to W.F. Farrell (GE), NG-00-0531, March 28, 2000.
23. “Duane Arnold Energy Center Containment Analysis,” GE report GE-NE-T2300752-00-01-R2, July 1998.
24. SAFER04 Basedeck on the GE BWR Engineering Data Bank (GE internal).
25. “Duane Arnold SBO Containment Analysis,” DRF T23-00662 (GE internal document).
26. “Duane Arnold Containment Analysis,” DRF T23-00752 (GE internal document).
27. “Duane Arnold Energy Center Task T0101 – Offrated Reactor Heat Balance,” GE Report GE-NE-A22-00100-02-01, Rev. 0, March 2000.
28. “Transmittal of DIR T0310 RHR System,” Letter from R. McGee (IES) to W.F. Farrell (GE), NG-00-0591, April 4, 2000.
29. “Decay Heat Table for Duane Arnold Power Uprate Equilibrium Cycle,” GE internal letter from C.L. Martin to W.F. Farrell, NSA-00-002, February 17, 2000.

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)
Appendix A - Feedwater Mass and Temperature Data (1912 MWt)

	Proposed by GE					Resolved for Analysis			
	OPL-4a (Dec. 1997)					OPL-4a (DAEC AEP)			
	Metal Mass (lbm)	Fluid Mass (lbm)	Temp (deg F)	Fluid Spec Vol (ft^3/lbm)		Temp (deg F)	Fluid Spec Vol (ft^3/lbm)	Fluid Mass (lbm)	Metal Mass (lbm)
					RPV				
	131,000	36,400	419.0	0.018805		431.3	0.018990	36045	131,000
	161,800	24,000	393.1	0.018433	#6 FWH	407.9	0.018640	23733	161,800
	142,700	34,400	367.1	0.018095		384.4	0.018320	33976	142,700
2014-006	14,500		366.0		RFP	383.3			67,400**
	54,300	44,200	364.8	0.018158		382.1	0.018380	43666	54,300
2014-006	118,000	28,200	340.2	0.017860	#5 FWH	356.4	0.018050	27903	199,000**
	10,800	10,000	315.5	0.017595		330.6	0.017760	9907	10,800
2014-006	103,400	18,800	296.7	0.017407	#4 FWH	310.8	0.017550	18647	165,600**
	4,500	4,100	277.8	0.017230		291.0	0.017350	4072	4,500
2014-006	148,800	32,000	241.4	0.016921	#3 FWH	253.3	0.017020	31814	237,600**
	45,000	36,900	205.0	0.016625		215.5	0.016730	36668	45,000
	169,000	29,400	186.9	0.016538	#2 FWH	196.8	0.016601	29288	169,000
	6,100	5,600	168.8	0.016433		178.0	0.016480	5584	6,100
	120,000	22,800	152.2	0.016341	#1 FWH	178.0*	0.016480	22608	120,000
	13,200	12,200	135.5	0.016258		178.0*	0.016480	12036	13,200
	84,600	16,600	124.7	0.016209	Drain cooler	178.0*	0.016480	16327	84,600
	60,900	56,000	113.9	0.016166		178.0*	0.016480	54933	60,900

Table 15.0 - 6
Containment Analysis Input Parameters (OPL-4A)
Appendix A - Feedwater Mass and Temperature Data (1912 MWt)

75,400	73,000	113.9	0.016166	Demin tank	178.0*	0.016480	71609	75,400
25,700	23,500	113.9	0.016166		114.7	0.016169	23496	25,700
25,000	2,600	113.3	0.016163	SJAE	113.8	0.016165	2600	25,000
23,100	18,600	112.7	0.016161		112.9	0.016162	18599	23,100
11,000	1,300	112.4	0.016160	SPE	112.6	0.016160	1300	11,000
24,400	15,000	112.0	0.016158		112.3	0.016159	14999	24,400
110,000	25,600	112.0	0.016158	Cond pump	112.2	0.016159	25598	110,000
12,500	21,400	112.0	0.016158		112.0	0.016159	21400	12,500
	575,600	112.0		Condenser	112.0		575600	

* Temperatures for these locations are not available, so the higher temperature of 178°F is conservatively assumed.

2014-006 ** Metal mass updated to reflect new equipment installed after the DAEC EPU containment analysis was performed.

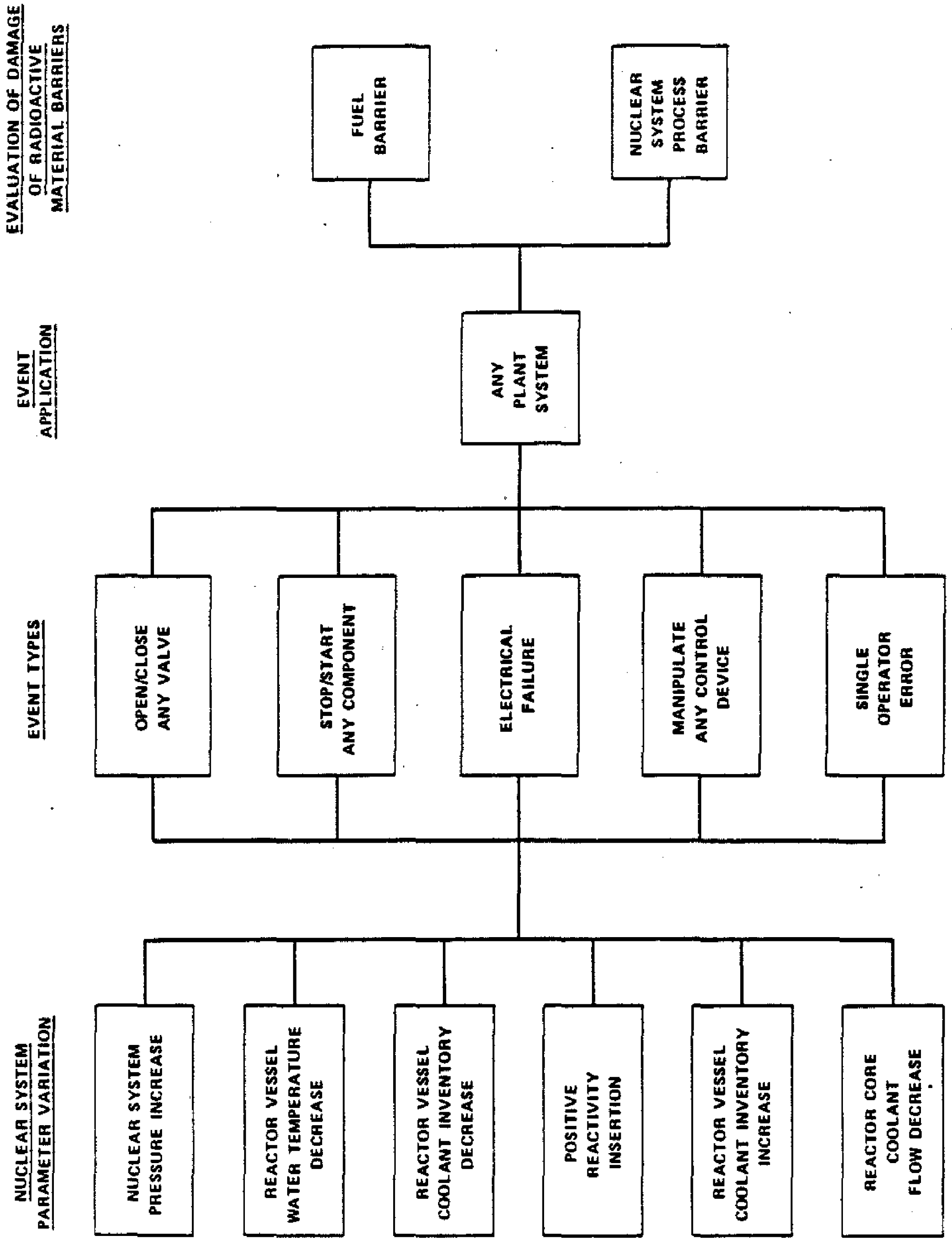
2013-014

Table 15.0-7
Design Basis Limit for Fission Product Barrier (DBLFPB) Summary Table

The controlling numerical values established during the licensing review are presented throughout the UFSAR for any parameter(s) used to determine the integrity of the fission product barrier. These limits have three key attributes: 1) The parameter is fundamental to the barrier's integrity, and, 2) the limit is expressed numerically (not via a description of functional requirements), and, 3) the limit is identified in the UFSAR (or a vendor topical report incorporated by reference in the UFSAR.) These limits are summarized below.

Boundary	Design Bases Parameter	Limit	Reference
Fuel Cladding	MCPR**	1.10 for two loop operation 1.12 for single loop operation	Tech Spec 2.1.1.2
	Maximum average Planar Linear Heat Generation Rate	See latest Revision of COLR	COLR
	Linear Heat Generation Rate	See latest Revision of COLR	COLR
	Fuel Enthalpy	170 cal/gm for transients (Rod Withdrawal Error), 280 cal/gm for accidents (Control Rod Drop Accident)	NEDE-24011-P-A
	Clad Strain	1% plastic strain	UFSAR Chapters 3 and 15 TS Bases 3.2.1
	Fuel Burnup	70GWD/MTU peak pellet exposure	NEDC-32868P
	Clad Temperature*	2200°F	10CFR50.46(b)(1)
	Clad Oxidation*	17% local 1% overall maximum hydrogen generation	10CFR50.46(b)(2) 10CFR50.46(b)(3)
RCS Boundary	Pressure** ***	Reactor Steam Dome Pressure ≤ 1335 psig	Tech Spec 2.1.2
	Stresses	ASME Code compliance for normal, upset, emergency and faulted conditions, as appropriate for the event.	UFSAR 3.2.3
	Heat-up/Cool-down**	≤ 20°F/hour ≤ 100°F/hour	Tech Spec 3.4.9 Curve A Tech Spec 3.4.9 Curves B and C
Containment	Pressure***	56 psig	UFSAR 3.8.2.1.1, 6.2.1.1.2.2
	Temperature***	281°F (design)	UFSAR 3.8.2.1.1, 6.2.1.1.2.2

*Controlled by another CFR, therefore 10 CFR 50.59 does not apply. **Controlled by Tech Specs, therefore 10 CFR 50.59 does not apply. ***62 psig at 281°F is the Containment acceptance criteria for ATWS and 1500 psig is the Reactor Steam Dome pressure acceptance criteria for ATWS.



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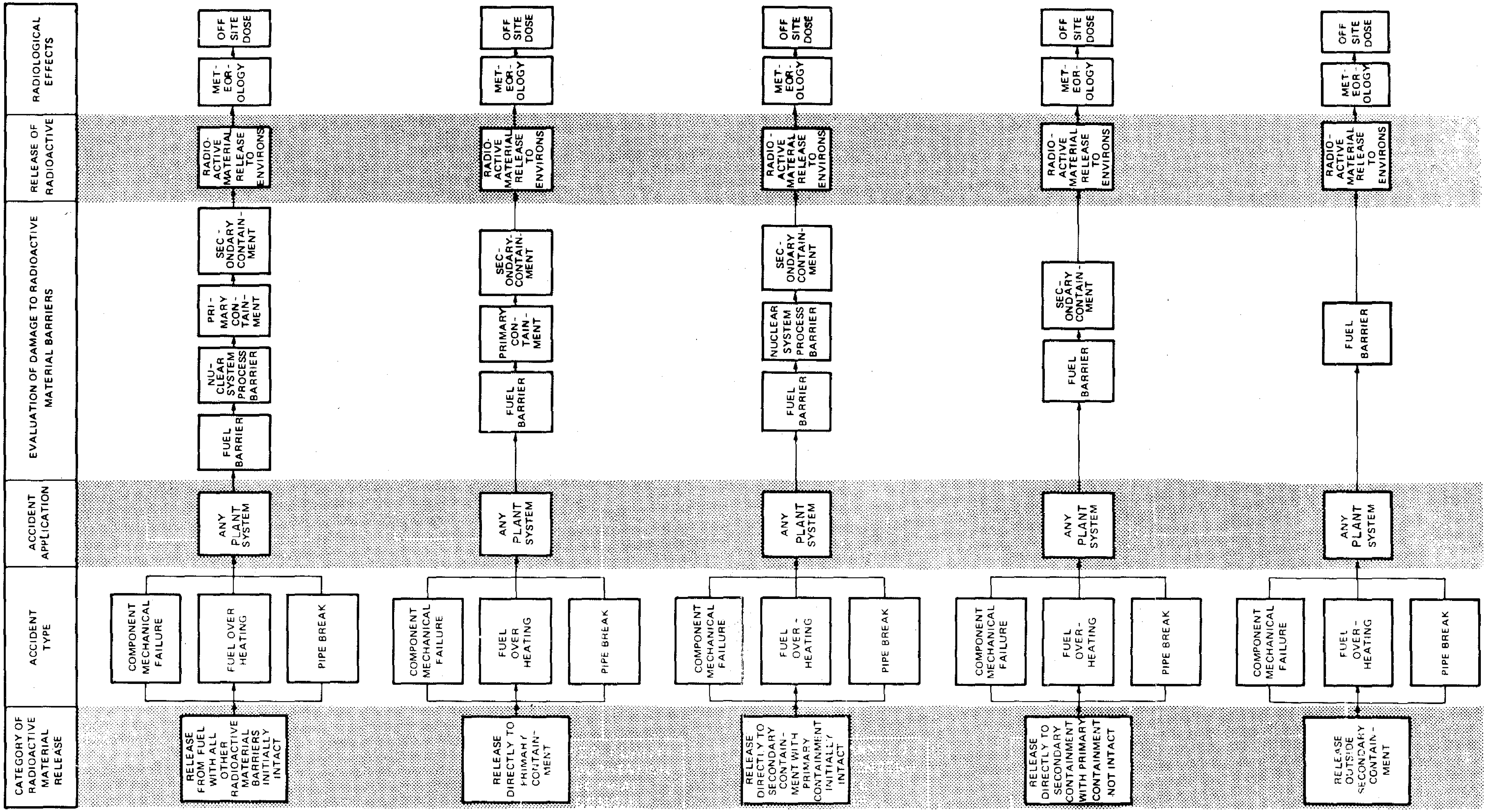
Method for Identifying and Evaluating Abnormal Operational Transients

Figure 15.0-1



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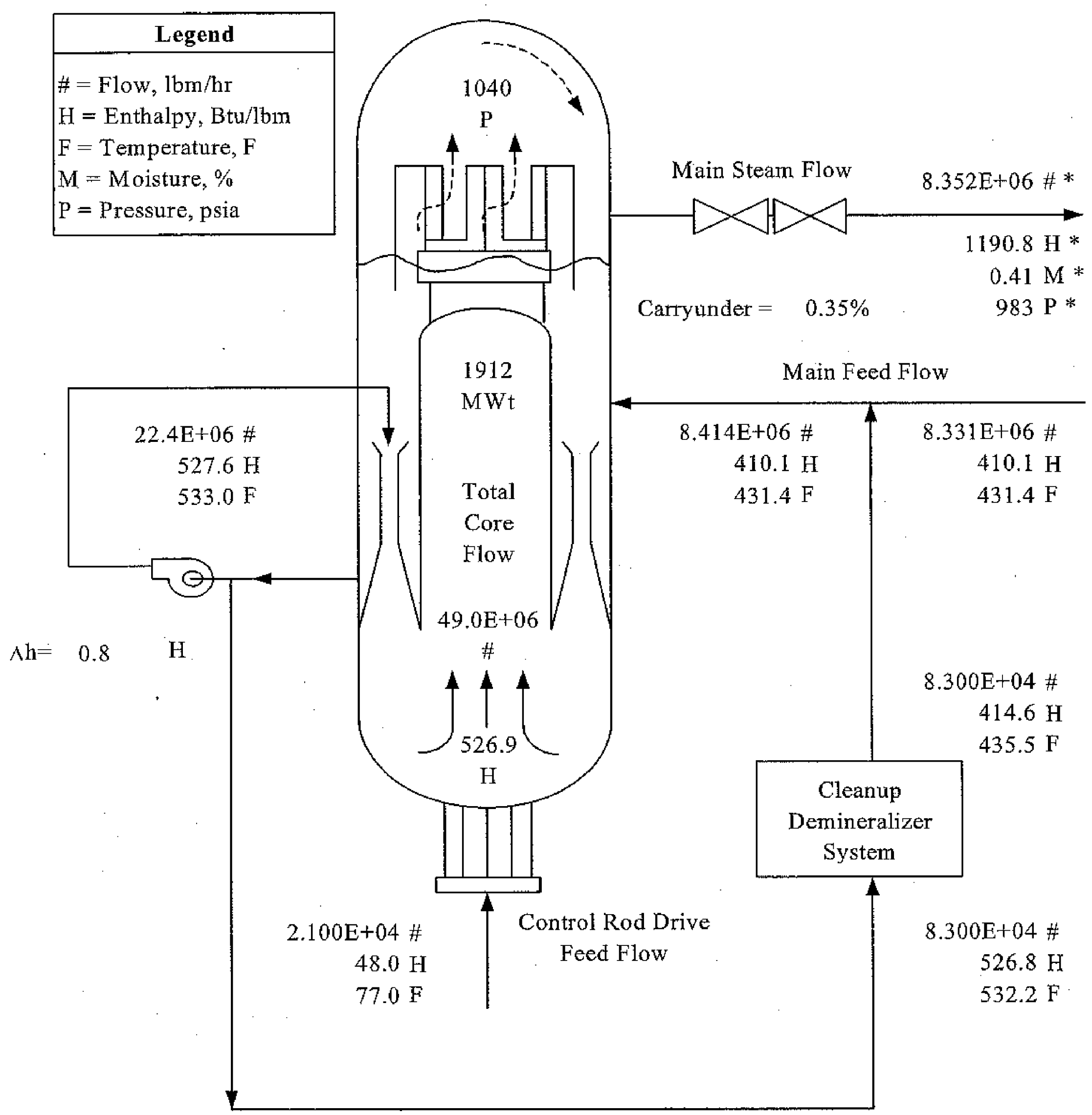
Method for Identifying and Evaluating Accidents

Figure 15.0-2



* Conditions at upstream side of TSV

Core Thermal Power	1912.0
Pump Heating	5.1
Cleanup Losses	-2.7
Assumed System Losses	-1.1
Turbine Cycle Use	1913.3 MWt

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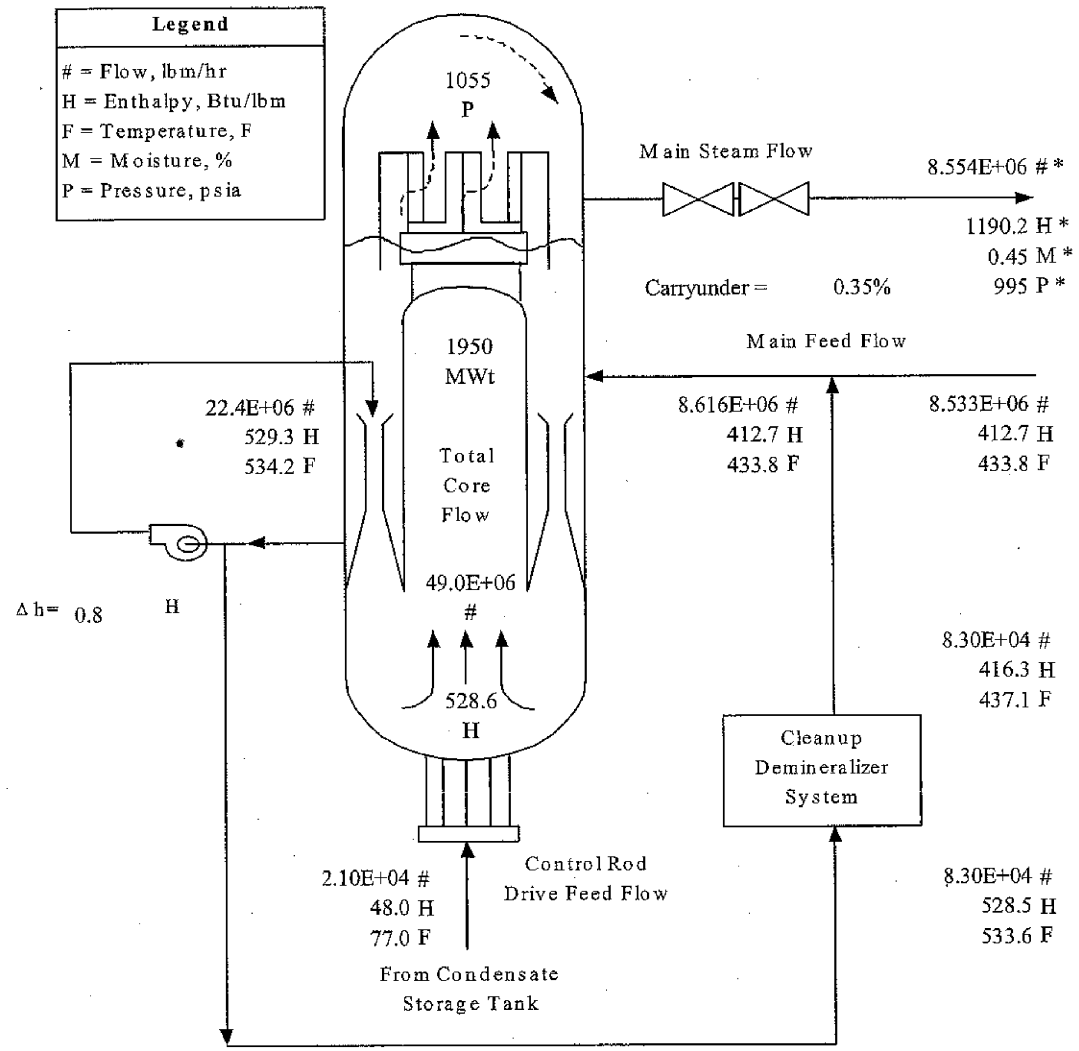
Reactor Heat Balance –
Rated Power and Rated Core Flow

Figure 15.0-3



* Conditions at upstream side of TSV

Core Thermal Power	1950.0
Pump Heating	5.1
Cleanup Losses	-2.7
Other System Losses	-1.1
Turbine Cycle Use	1951.3 MWt

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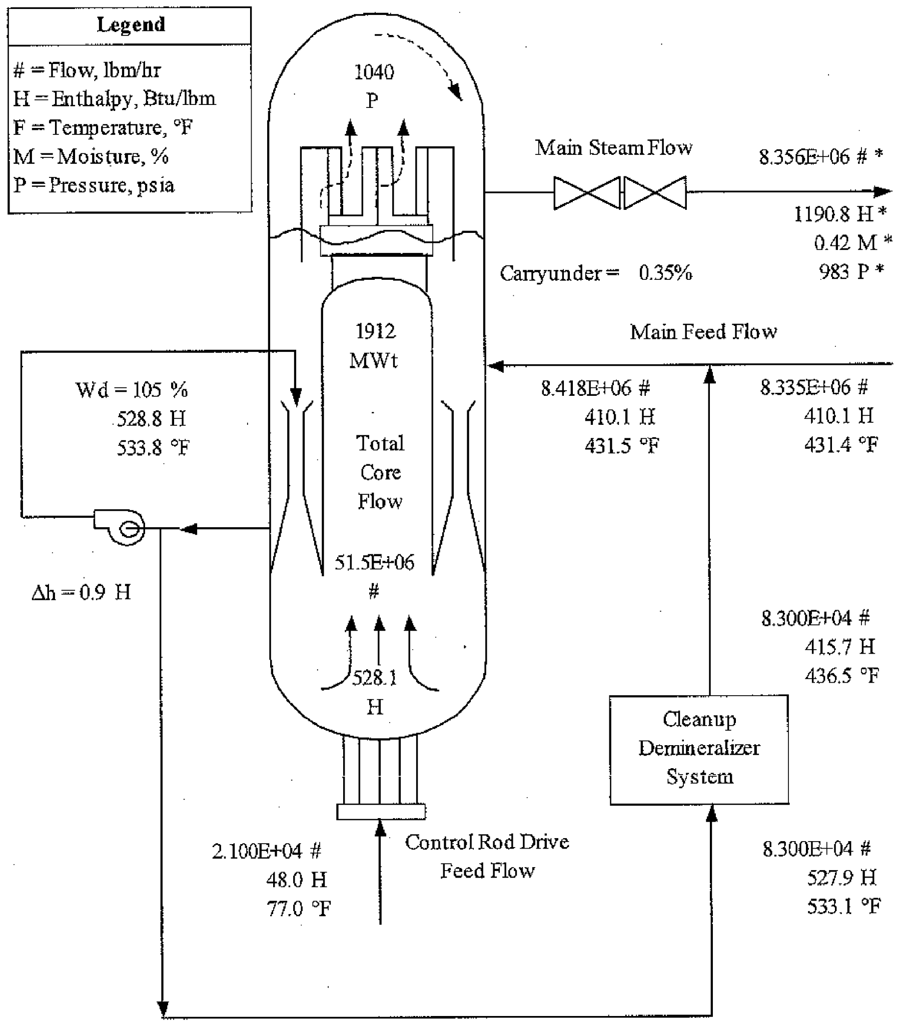
Reactor Heat Balance –
102% Power and Rated Core Flow

Figure 15.0-4



*Conditions at upstream side of TSV

Core Thermal Power	1912.0
Pump Heating	5.9
Cleanup Losses	-2.7
Other System Losses	-1.1
Turbine Cycle Use	1914.1 MWt

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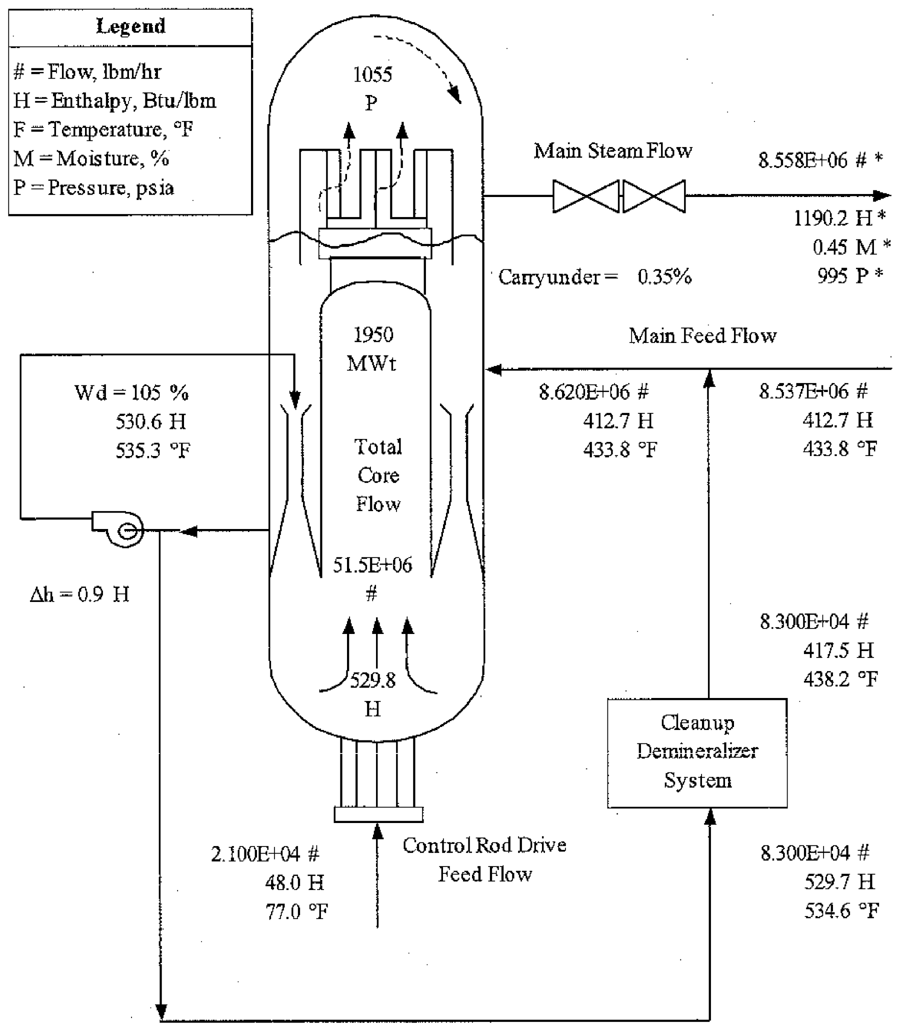
Reactor Heat Balance –
Rated Power and 105% Core Flow

Figure 15.0-5



*Conditions at upstream side of TSV

Core Thermal Power	1950.2
Pump Heating	5.9
Cleanup Losses	-2.7
Other System Losses	-1.1
Turbine Cycle Use	1952.3 MWt

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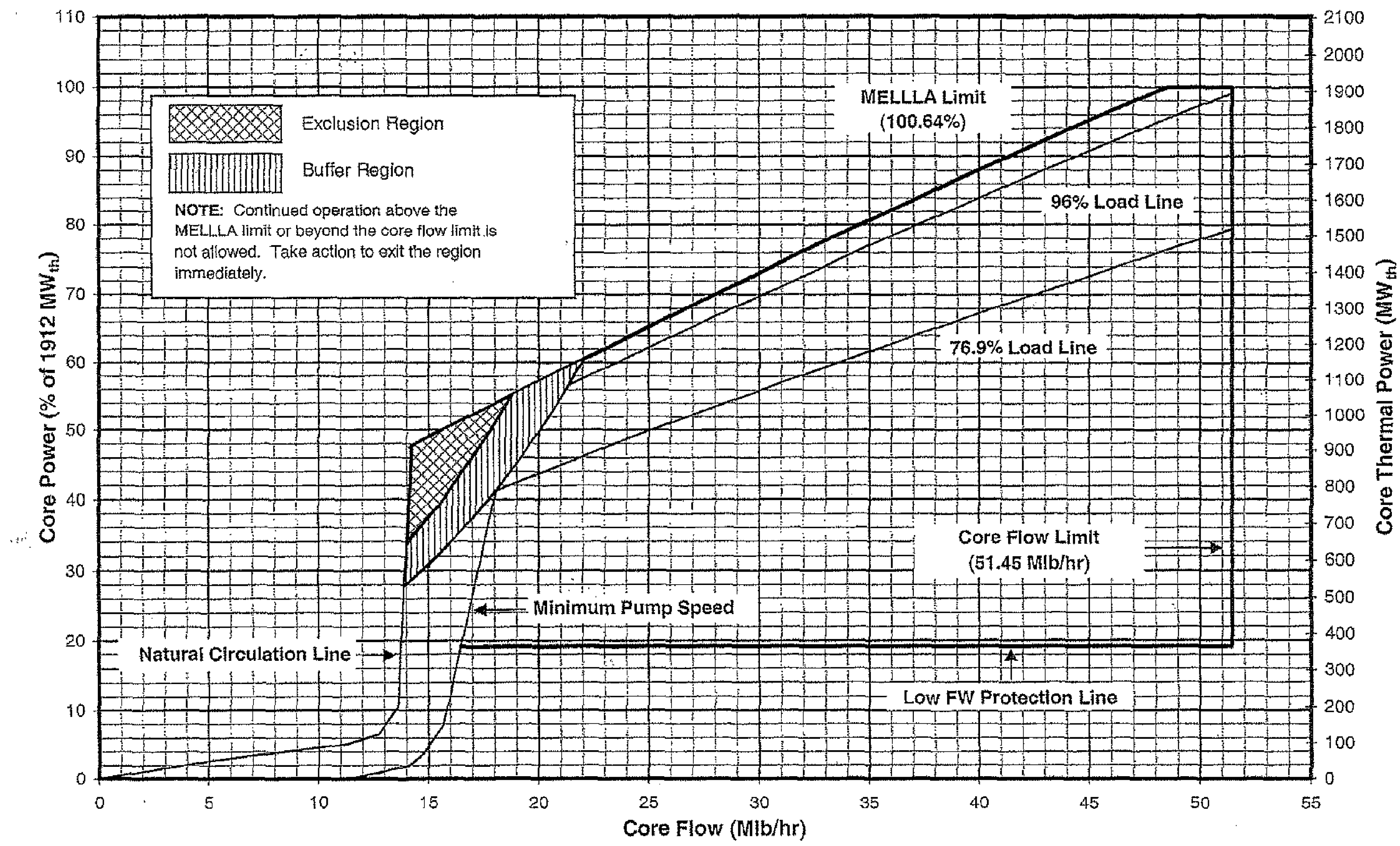
Reactor Heat Balance –
102% Power and 105% Core Flow

Figure 15.0-6



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Reactor Power-to-Flow Map

Figure 15.0-7

15.1 TRANSIENTS

This Section of the UFSAR contains the event descriptions, methods of analysis, assumptions, and analytical results of that subset of plant events classified as Transients (sometimes referred to as Abnormal Operating Transients (AOTs) or Abnormal Operational Occurrences (AOOs)) (See Section 15.0.2). The plant response to each Transient will be discussed in terms of the impact on the fission product barriers specifically, the fuel cladding and reactor coolant pressure boundary. Because these events generally do not lead to fuel failures or direct challenges to either the primary or secondary containments, those fission product barriers are not evaluated for Transients. Also, the methods used, and assumptions made, in the individual event analyses are specifically adjusted to provide conservative results for the specific event, it is recognized that between each Transient, there may not be full coherence between the various evaluations performed. For example, different initial conditions/values may be used in the evaluation of the fuel and that used in the analysis of the reactor coolant pressure boundary for the Main Steamline Isolation Valve Closure Transient. Or, different computer codes may be better suited to one type of event over another. For example, a slow moving transient may be better evaluated using a series of calculations with a steady state model than using a dynamic model that is better suited to fast moving events. Thus, each event description will describe the methods, inputs and assumptions used and will highlight any uniqueness in that evaluation.

15.1.1 TRANSIENTS 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 three events that result in the most severe transients in this category are the following:

1. Feedwater controller failure - maximum demand.

2. Loss of feedwater heating.

3. Inadvertent HPCI actuation.

15.1.1.1 Feedwater Controller Failure - Maximum Demand

<u>Description of Event</u>

a) Initiator:
A postulated failure in the feedwater control logic creates a demand in the feedwater flow to the maximum runout value of both feed pumps.

b) Sequence of Events:

The plant is operating at 100% power and 105% core flow, when there is a maximum demand signal generated by the feedwater level control system causing the feedwater regulating valves to go to full open and feedwater flow increases to 115% of rated flow (pump runout condition). Because there is an initial mismatch between steamflow and feedwater flow, the inlet subcooling to the core increases because the feedwater flow is not sufficiently heated by the feedwater heaters. This causes reactor power to increase due to the collapse of the voids in the core. Also, the mismatch causes the reactor water level to increase to the High Level trip setpoint (Level 8), which trips both feedwater pumps and the main turbine. (Note: at this point, the transient response becomes essentially a turbine trip with bypass from slightly higher than rated power.) The turbine trip signal to the EHC system causes the turbine stop valves begin to close. Upon reaching the 90% open (nominal) point, as sensed by valve position switches, a reactor trip signal (Scram) is initiated, along with an end-of-cycle recirculation pump trip (EOC-RPT). Control rods begin to insert and the recirculation pumps begin to coast down, both of which help turn around the power increase generated by the collapsing of the voids in the core from the pressure increase due to the loss of a steam path with the closing of the stop valves. Because the turbine bypass valves do not have the capacity to accommodate the initial steam flow, reactor pressure increases and SRVs lift to relieve the pressure. This will arm the Low-Low Set logic and control the SRVs opening and closing setpoints. Control rods are fully inserted to terminate the power increase.

Long-term response (beyond the explicit analyzed period): Turbine bypass valves will control the pressure. HPCI and/or RCIC initiate to maintain reactor vessel level, as needed. Operators take manual control and guide the plant to a cold shutdown condition.

c) Single Failure/Operator Error (as applicable):

None

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):

Short term: Feedwater and Main Turbine trip (Level 8 trip), RPS trip (TSV closure), EOC-RPT (TSV closure), Control Rod Scram, Recirculation Pump Trip, SRVs open/close.

Long-term: Low-low set logic armed (SRV open and High Pressure Scram), Turbine bypass valves open to control reactor pressure, HPCI/RCIC initiations (low-low RPV level). The Operators intervene only after the initial transient is over and guide the plant to a stable condition.

Event Category & Acceptance Criteria

This is an Anticipated Operational Occurrence – a single transient of moderate-to-infrequent frequency (feedwater controller failure to maximum runout flow conditions) with no other equipment failures or Operator errors.

Fuel SAFDLs (Section 15.0.4) shall not be exceeded.

RPV Pressure shall remain within ASME Upset limits.

Methods

a) Calculation Tools & Computer Codes:
Primary Code – ODYN, using GEMINI methods. (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-3 (Table 15.0-3). No unique inputs for this analysis.

c) Key Assumptions:
Plant is initially at rated thermal power and 105% core flow.
TSVs close in a linear ramp over 0.1 seconds.
No Operator Actions are assumed during the initial transient response.

Results

a) Barrier Performance and Comparison to Acceptance Criteria:

Fuel Performance: Fuel thermal limits (MCPR, LHGR (MOP and TOP) and APLHGR) all remain within their respective acceptance limits. See the current cycle's Supplemental Reload Licensing Report (SRLR) for actual values.

Reactor Pressure Boundary Performance: Reactor vessel pressure remains well below the 1375 psig ASME acceptance limit. See the current cycle's SRLR for actual values.

Containment Performance: The containment is not challenged by this event and is not explicitly analyzed.

b) Known Sensitivities:

This event has been analyzed assuming no high level trip of the Main Turbine and Feedwater pumps (Ref. 15.0-39). This analysis demonstrates that the results of this event are not sensitive to the Level 8 setpoint, provided that the initial power level increase due to the increase in subcooling has stabilized prior to the turbine trip.

This event is very sensitive to the vessel pressure change. Changes in plant equipment characteristics, e.g., valve stroke times, setpoint changes, steamline lengths and volumes, etc., that will cause the pressurization rate and/or peak vessel pressure to increase will have a negative impact on the event results.

As discussed in Section 15.0.9, the results of this transient are sensitive to the initial power level, as the impact of the total runout flow is more pronounced at lower powers (i.e., incremental increase in feedwater flowrate). The fuel thermal limits are adjusted to account for this by the use of the MCPRp and MAPFACp/LHGRFACp multipliers from the ARTS (APRM/RBM/Technical Specification) program.

To support certain equipment being out of service during the operating cycle, (Ref. FRED form in Section 15.7) additional analysis of this event is performed assuming that equipment is not Operable. The results of these equipment out-of-service conditions are found in the SRLR for the current cycle.

c) Uncertainties in Results:

The plant performance is analyzed to 95%/95% confidence levels using GEMINI methods.

Conclusion

a) Statement of Acceptability:
This event meets all the fission product barrier performance criteria for an Abnormal Operating Transient.

b) Known Conservatisms/Margins:
Allowance for 2% core thermal over-power (accounted for in the GEMINI methods).

End-of-cycle core conditions are assumed.
Conservative control rod scram times are used.
Bounding TSV closure time.
Bounding SRV opening setpoints (+3% tolerance to nominal settings)

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
This is considered to be a Limiting Event and is re-analyzed as part of the reload analysis for each operating cycle.

15.1.1.2 Loss of Feedwater Heating

Description of Event

a) Initiator:
A number of various failure modes can lead to a loss of feedwater heating. For the purposes of this evaluation, we do not specify the exact failure mode, but only that there is a loss of heating that results in a 100 °F reduction in feedwater temperature.

b) Sequence of Events (NOT a time line):

There is a loss of feedwater heating that results in a slow decrease in feedwater temperature. This increases the inlet subcooling to the core, which in turn, cause the power level to increase due to the collapse of the voids from the colder water. The power level increases until a new steady state condition is achieved when the increase in steam flow to the turbine equilibrates to the new power level.

c) Single Failure/Operator Error (as applicable):

None

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
It is assumed that the Operator notices the indication of increasing reactor power, steam flow, etc. and takes control of the event to lower the power back to within the licensed loadline/thermal power level by lowering recirculation flow and/or inserting control rods.

Event Category & Acceptance Criteria

This is an Anticipated Operational Occurrence – a single transient of moderate frequency (loss of feedwater heating) with no other equipment failures or Operator errors.

Fuel SAFDLs (Section 15.0.4) shall not be exceeded.

RPV Pressure shall remain within ASME Upset limits.

Methods

a) Calculation Tools & Computer Codes:
The primary code used to perform this analysis is the GE 3-D Core Simulator (PANACEA). (See Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):

This event is analyzed at 100% power/100% flow (Note: no 2% allowance for overpower is used in this analysis).
Cycle-specific core loading (FRED form). No other unique inputs are used.

c) Key Assumptions:
There is no scram signal generated by this event.

Results

a) Comparison to Acceptance Criteria:

Fuel Performance: Fuel thermal limits (MCPR, LHGR (MOP and TOP) and APLHGR) all remain within their respective acceptance limits. See the current cycle's Supplemental Reload Licensing Report (SRLR) for actual values.

Reactor Pressure Boundary Performance: Reactor vessel pressure remains essentially unchanged in this event due to it being a very slow transient.

Containment Performance: The containment is not challenged by this event and is not explicitly analyzed.

b) Known Sensitivities:

The results of this event are mildly sensitive to the magnitude of the feedwater temperature change and the results are most limiting at rated conditions. This event is analyzed at BOC, MOC and EOR conditions (see 15.7).

c) Uncertainties in Results:

The plant performance is analyzed to 95%/95% confidence levels using GEMINI methods.

Conclusion

a) Statement of Acceptability:
This event meets all the fission product barrier performance criteria for an Anticipated Operational Occurrence.

b) Known Conservatisms/Margins:
The assumed 100 °F change in feedwater temperature is bounding for any single failure within the Feedwater system.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
This is a Limiting Event and is evaluated as part of the cycle-specific reload analysis.

15.1.1.3 Inadvertent HPCI Actuation

Description of Event

a) Initiator:

For the purposes of this evaluation, we do not specify the exact cause of the HPCI actuation; only that there is an inadvertent injection to the vessel from the HPCI system.

b) Sequence of Events (NOT a time line):

There is an inadvertent initiation of the HPCI system that injects colder water to the reactor vessel, via the Feedwater System. The Feedwater level control instrumentation compensates for the increase in level due to the additional inventory from the HPCI injection by reducing the feedwater flow. However, there is a decrease in feedwater temperature. Similar to the Loss-of-Feedwater Heating event, this temperature decrease increases the inlet subcooling to the core, which in turn, causes the power level to increase due to the collapse of the voids from the colder water. The power level increases until a new steady state condition is achieved when the increase in steam flow to the turbine equilibrates to the new power level.

c) Single Failure/Operator Error (as applicable):

None

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
The Feedwater level control system reacts to the increasing vessel level due to the HPCI injection and reduces feedwater flow to compensate.
It is assumed that the Operator notices the indication of the HPCI initiation and takes action to secure the injection. They also react to the increasing reactor power, steam flow, etc. and take control of the event to lower the power back to within the licensed loadline/thermal power level by lowering recirculation flow and/or inserting control rods.

Event Category & Acceptance Criteria

This is an Anticipated Operational Occurrence – a single transient of moderate frequency (inadvertent HPCI injection) with no other equipment failures or Operator errors.

Fuel SAFDLs (Section 15.0.4) shall not be exceeded.

RPV Pressure shall remain within ASME Upset limits.

Methods

a) Calculation Tools & Computer Codes:
Primary Code – REDY, using GEMINI methods. (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items)
See OPL-3 (Table 15.0-3)
HPCI injects at 3000 gpm.

c) Key Assumptions:
Plant is initially at 102% of rated thermal power and rated core flow.
There is no scram signal generated by this event.

Results

a) Comparison to Acceptance Criteria:

Fuel Performance: Fuel thermal limits (MCPR, LHGR (MOP and TOP) and APLHGR) all remain within their respective acceptance limits. See the current cycle's Supplemental Reload Licensing Report (SRLR) for confirmation.

Reactor Pressure Boundary Performance: Reactor vessel pressure remains essentially unchanged in this event due to it being a very slow transient.
Containment Performance: The containment is not challenged by this event and is not explicitly analyzed.

b) Known Sensitivities:

The results of this event are mildly sensitive to the magnitude of the HPCI injection flowrate (i.e., feedwater temperature change) and the results are most limiting at rated conditions. This event is analyzed at BOC, MOC and EOR conditions (see 15.7).

c) Uncertainties in Results:

The plant performance is analyzed to 95%/95% confidence levels using GEMINI methods.

Conclusion

a) Statement of Acceptability:
This event meets all the fission product barrier performance criteria for an Anticipated Operational Occurrence.

b) Known Conservatisms/Margins:
Allowance for 2% core thermal over-power.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
This is a non-limiting event and is only evaluated as part of the cycle-specific reload analysis to confirm that it remains bounded by the Loss-of-Feedwater Heating event.

15.1.2 TRANSIENTS 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. Plant systems have been surveyed to identify event within each system that could result in the rapid reduction of steam flow. The survey revealed the following events:

1. Generator 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.

6. Pressure regulator malfunction causing turbine control valves to close.

7. Loss of Offsite Power

A consideration of Events 4-6 above shows that turbine bypass valve failure, loss of condenser vacuum, and pressure regulator malfunction are specific cases of the first two event types. A failure of the turbine bypass valves to open when required is analyzed as the most severe form of a turbine or generator trip. A loss of condenser vacuum causes turbine stop valve closure and turbine bypass valve closure; thus, a loss of vacuum is a turbine trip without bypass. Pressure regulator malfunctions that result in turbine steam flow shutoff and a nuclear system pressure rise are mild forms of a generator load rejection. Thus, all of the effects of these events are included in the effects described for the generator load rejection and turbine trip.

The loss-of-offsite power (LOOP) is a complex sequence of events that occurs when the plant loses all auxiliary power. A loss of auxiliary power is an event that deenergizes all buses that supply power to the unit auxiliary equipment, such as recirculation pumps, condensate pumps, and circulating water pumps. Such an event can result if faults or trips occur in the auxiliary power distribution system.

The reactor is subjected to a complex sequence of events when the plant loses all auxiliary power. Estimates of the responses of the various reactor systems (assuming a loss of the auxiliary transformer) provide the following sequence:

1. The recirculation pumps are tripped with normal coastdown times.

2. Independent main steam isolation valve closure and scram are initiated

because of the loss of power to their respective solenoids, i.e., RPS M/G set trip.

3. Motor driven feedwater pumps are tripped.

An alternative transient results if there is a loss of all electrical connections to the grids external to the plant. The same sequence as above would be followed except that the reactor would also experience a generator load rejection and its associated scram at the beginning of the transient. Consequently, this transient can be characterized as either a closure of all MSIVs or generator load rejection event.

15.1.2.1 Generator Load Rejection (Turbine Control Valve Fast Closure)

A loss of generator load causes the turbine-generator to increase in speed. The turbine speed and acceleration protection systems and the power load unbalance circuitry in the electrohydraulic controller quickly close the turbine control valves to shut off the steam supply to the turbine, thus avoiding excessive turbine overspeed. Several variations in the Load Rejection transients are possible according to the assumptions made concerning the initial power level and the turbine bypass system. These cases are discussed individually.

15.1.2.1.1 <u>Main Generator Load Rejection with Bypass Vales (LRWBP) – High Power</u>

<u>Description of Event</u>

a) Initiator:
A non-mechanistically caused trip of the main generator.

b) Sequence of Events:
The plant is operating at 100% power, when the EHC system receives a trip signal and the turbine control valves begin to close. Upon actuation of the Turbine Control Valve (TCV) fast closure, as sensed by EHC oil pressure, a reactor trip signal (Scram) is initiated, along with an end-of-cycle recirculation pump trip (EOC-RPT). Control rods begin to insert and the recirculation pumps begin to coast down. Both of which help turn around the power increase generated by the collapsing of the voids in the core from the pressure increase due to the partial loss of a steam path. Although the bypass valves open, their steam flow capacity is not enough initially to control the pressure. So, reactor pressure increases and SRVs lift to relieve the pressure. Control rods are fully inserted to terminate the power increase.
Long-term response (beyond the explicit analyzed period): Low-low set logic armed (SRV open and High Pressure Scram), Turbine bypass valves open to control reactor pressure, Feedwater and Condensate pumps are used to maintain reactor level. Operators take manual control and guide the plant to a cold shutdown condition using normal shutdown procedures.

e) Single Failure/Operator Error (as applicable):
None

f) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
Short term: RPS trip (TCV fast closure), EOC-RPT (TCV fast closure), Control Rod Scram, Recirculation. Pump Trip, SRVs open/close.

Long-term: Low-low set logic armed (SRV open and High Pressure Scram), Turbine bypass valves open to control reactor pressure, Feedwater/Condensate pumps operate. The Operators intervene only after the initial transient is over and guide the plant to a stable condition.

Event Category & Acceptance Criteria

This is an Anticipated Operational Occurrence – an expected operational transient of moderate frequency (load rejection) with no other equipment failures or Operator errors.

Fuel SAFDLs (Section 15.0.4) shall not be exceeded.

RPV Pressure shall remain within ASME Upset limits.

Methods

a) Calculation Tools & Computer Codes:
Primary Code – ODYN, using GEMINI methods. (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-3 (Table 15.0-3). No unique inputs for this analysis.

c) Key Assumptions:
Plant is initially at rated thermal power and core flow.
TCVs operate in two admission mode (3x1 mode – TCVs 1, 2 & 3 move together and TCV 4 is the controlling valve.)
TCV RPS trip is based upon an assumed response time of 0.03 secs from beginning of TCV fast closure to the trip of the RPS relay.
No Operator Actions are assumed during the initial transient response.

Results

d) Barrier Performance and comparison to Acceptance Criteria:
Fuel Performance: Fuel thermal limits (MCPR, LHGR (MOP and TOP) and APLHGR) all remain within their respective acceptance limits. This is a non-limiting event and is not re-analyzed as part of the reload process. See Table 15.0-1 for comparison of event response to the bounding events.
Reactor Pressure Boundary Performance: Reactor vessel pressure remains well below the 1375 psig ASME acceptance limit. See Table 15.0-1 for comparison of event response to the bounding events.

Containment Performance: The containment is not challenged by this event and is not explicitly analyzed.

e) Known Sensitivities:

This event is moderately sensitive to the vessel pressure change. Changes in plant equipment characteristics, e.g., valve stroke times, setpoint changes, steamline lengths and volumes, etc., that will cause the pressurization rate and/or peak vessel pressure to increase will have a negative impact on the event results.

c) Uncertainties in Results:

The plant performance is analyzed to 95%/95% confidence levels using GEMINI methods.

Conclusion

d) Statement of Acceptability:
This event meets all the fission product barrier performance criteria for an Abnormal Operating Transient.

e) Known Conservatisms/Margins:
Allowance for 2% core thermal over-power is accounted for in the analysis methods.
End-of-cycle core conditions are assumed.
Conservative control rod scram times are used.
Bounding TCV (faster) closure time.
Bounding SRV opening setpoints (+3% tolerance)

f) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
This is considered to be a non-Limiting Event and is not re-analyzed as part of the reload analysis for each operating cycle. See Table 15.0-1 for comparison of event response to the bounding events.

15.1.2.1.2 Main Generator Load Rejection with Bypass Valve Failure (LRNBP) – High Power

Description of Event

a) Initiator:
A non-mechanistically caused trip of the main generator with failure of the bypass valves to open.

b) Sequence of Events:
The plant is operating at 100% power, when the EHC system receives a trip signal and the turbine control valves begin to close. Upon actuation of Turbine Control Valve (TCV) fast closure, as sensed by EHC oil pressure, a reactor trip signal (Scram) is initiated, along with an end-of-cycle recirculation pump trip (EOC- RPT). Control rods begin to insert and the recirculation pumps begin to coast down. Both of which help turn around the power increase generated by the collapsing of the voids in the core from the pressure increase due to the loss of a steam path, due to the failure of the turbine bypass valves to open. Reactor pressure increases and SRVs lift to relieve the pressure. Control rods are fully inserted to terminate the power increase.

Long-term response (beyond the explicit analyzed period): Low-low set logic armed (SRV open and High Pressure Scram) and cycles the SRVs to control reactor pressure, Feedwater and Condensate pumps eventually are no longer available, as condenser inventory is depleted, HPCI and/or RCIC initiate to maintain reactor vessel level, as needed. Operators take manual control and guide the plant to a cold shutdown condition.

c) Single Failure/Operator Error (as applicable):
Failure of Turbine Bypass Valves to open upon demand.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
Short term: RPS trip (TCV fast closure), EOC-RPT (TCV fast closure), Control Rod Scram, Recirculation Pump Trip, SRVs open/close.

Long-term: Low-low set logic armed and controls reactor pressure, FW/Condensate pumps operate, HPCI/RCIC initiations (low-low RPV level). The Operators intervene only after the initial transient is over and guide the plant to a stable condition.

Event Category & Acceptance Criteria

This is an Abnormal Operating Transient – an expected operational transient of moderate frequency (load rejection), but with the additional single failure of the bypass valves failing to open.

Fuel SAFDLs (Section 15.0.4) shall not be exceeded.

RPV Pressure shall remain within ASME Upset limits.

Methods

a) Calculation Tools & Computer Codes:
Primary Code – ODYN, using GEMINI methods. (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-3 (Table 15.0-3). No unique inputs for this analysis.

c) Key Assumptions:
Plant is initially at rated thermal power and 105% core flow.
TCVs operate in two admission mode (3x1 mode – TCVs 1, 2 & 3 move together and TCV 4 is the controlling valve.)
TCV RPS trip is based upon an assumed response time of 0.03 secs from beginning of TCV fast closure to the trip of the RPS relay.
No Operator Actions are assumed during the initial transient response.

Results

a) Barrier Performance and comparison to Acceptance Criteria:
Fuel Performance: Fuel thermal limits (MCPR, LHGR (MOP and TOP) and APLHGR) all remain within their respective acceptance limits. See the current cycle's Supplemental Reload Licensing Report (SRLR) for actual values.

Reactor Pressure Boundary Performance: Reactor vessel pressure remains well below the 1375 psig ASME acceptance limit. See the current cycle's SRLR for actual values.

Containment Performance: The containment is not challenged by this event and is not explicitly analyzed.

b) Known Sensitivities:

This event is very sensitive to the vessel pressure change. Changes in plant equipment characteristics, e.g., valve stroke times, setpoint changes, steamline lengths and volumes, etc., that will cause the pressurization rate and/or peak vessel pressure to increase will have a negative impact on the event results.

To support certain equipment being out of service during the operating cycle, (Ref. FRED form in Section 15.7) additional analysis of this event is performed assuming that equipment is not Operable. The results of these equipment out-of-service conditions are found in the SRLR for the current cycle.

c) Uncertainties in Results:

The plant performance is analyzed to 95%/95% confidence levels using GEMINI methods.

Conclusion

a) Statement of Acceptability:
This event meets all the fission product barrier performance criteria for an Abnormal Operating Transient.

b) Known Conservatisms/Margins:
Allowance for 2% core thermal over-power is accounted for in the analysis methods.
End-of-cycle core conditions are assumed.
Conservative control rod scram times are used.
Bounding (faster) TCV closure time.
Bounding SRV opening setpoints (+3% tolerance)

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
This is considered to be a Limiting Event and is re-analyzed as part of the reload analysis for each operating cycle.

15.1.2.1.3 Generator Load Rejection from Low Power without Bypass

a) Initiator:
A non-mechanistically caused trip of the main generator with failure of the bypass valves to open.

b) Sequence of Events:

Note: Under the ARTS program (APRM/RBM/Technical Specification), this event is analyzed as a series of four cases representing the combination of thermal power and core flow: 21.7% power (representing the lowest power level for monitoring of fuel thermal limits), 26% power (representing the bypass of the direct scram signal on the TCVs), 50% core flow and 105% core flow. The results of these four cases help to define the MCPRp, LHGRFACp and MAPFACp limits in the Core Operating Limits Report (COLR). The limiting case of 26% power/105% flow is discussed below.

The plant is operating at 26% power at 105% rated core flow, when the EHC system receives a trip signal and the turbine control valves begin to close. Because this power level is below the bypass for the direct scram signal on the TCVs, no direct scram is generated. Reactor pressure increases due to the loss of a steam path, as a result of the failure of the turbine bypass valves to open. The pressure quickly reaches the scram setpoint and control rods begin to insert, which help turn around the power increase generated by the collapsing of the voids in the core from the pressure increase. Reactor pressure increases and SRVs lift to relieve the pressure. Control rods are fully inserted to terminate the power increase.

Long-term response (beyond the explicit analyzed period): Low-low set logic cycles the SRVs to control reactor pressure. Feedwater and Condensate pumps eventually trip, as condenser inventory is depleted. HPCI and/or RCIC initiate to maintain reactor vessel level, as needed. Operators take manual control and guide the plant to a cold shutdown condition using normal plant shutdown procedures.

c) Single Failure/Operator Error (as applicable):
Failure of Turbine Bypass Valves to open upon demand.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
Short term: RPS trip (Rx Dome Pressure), Control Rod Scram, Recirc. Pump Trip/coast down, SRVs open/close.
Long-term: Low-low set logic trip, FW/Condensate pumps trip, as condenser inventory is depleted, HPCI/RCIC initiations (low-low RPV level). The Operators intervene only after the initial transient is over and guide the plant to a stable condition.

Event Category & Acceptance Criteria

This is an Abnormal Operating Transient – an expected transient (load rejection) with the additional single failure of the bypass valves failing to open.

Fuel SAFDLs (Section 15.0.4) shall not be exceeded.

RPV Pressure shall remain within ASME Upset limits.

Methods

a) Calculation Tools & Computer Codes:
Primary Code – ODYN, using GEMINI methods. (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-3 (Table 15.0-3). No unique inputs for this analysis.

c) Key Assumptions:
Plant is initially at 21.7% of rated and also evaluated at 26% thermal power and both power levels are evaluated at both 105% and 50% core flowspace.
TCVs operate in the two admission mode (3x1 mode – TCVs 1, 2 & 3 move together and TCV 4 is the controlling valve. Because this event is at low power, TCVs 1, 2 & 3 will be partially open and TCV 4 will be closed.
No Operator Actions are assumed during the initial transient response.

Results

a) Barrier Performance and comparison to Acceptance Criteria:

Fuel Performance: Fuel thermal limits (MCPR, LHGR (MOP and TOP) and APLHGR) all remain within their respective acceptance limits. The results of this event are used to develop the ARTS off-rated limits. See the current cycle's Supplemental Reload Licensing Report (SRLR) for actual values.

Reactor Pressure Boundary Performance: Reactor vessel pressure remains well below the 1375 psig ASME acceptance limit. Because the peak pressure is directly proportional to the initial power level, the peak pressure from this event is not evaluated.

Containment Performance: The containment is not challenged by this event and is not explicitly analyzed.

b) Known Sensitivities:

This event is very sensitive to the vessel pressure change. Changes in plant equipment characteristics, e.g., valve stroke times, setpoint changes, steamline lengths and volumes, etc., that will cause the pressurization rate and/or peak vessel pressure to increase will have a negative impact on the event results.

To support certain equipment being out of service during the operating cycle, additional analysis of this event is performed assuming that equipment is not

Operable (Ref. FRED form in Section 15.7). The results of this equipment out-of-service condition are found in the SRLR for the current cycle.

c) Uncertainties in Results:

The plant performance is analyzed to 95%/95% confidence levels using GEMINI methods.

Conclusion

a) Statement of Acceptability:
This event meets all the fission product barrier performance criteria for an Abnormal Operating Transient.

b) Known Conservatisms/Margins:
End-of-cycle core conditions are assumed.
Conservative control rod scram times are used.
Bounding SRV opening setpoints (+3% tolerance)
Bounding (faster) TCV closure time.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
2014-003 This is considered to be a Non-Limiting Event and is confirmed as part of the reload analysis for each operating cycle.

15.1.2.2 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. Several variations in the turbine trip transients are possible according to the assumptions made concerning the initial power level and the turbine bypass system. These cases are discussed individually.

15.1.2.2.1 Main Turbine with Bypass (TTWBP) – High Power

Description of Event

a) Initiator:
A non-mechanistically caused trip of the main turbine.

b) Sequence of Events:
The plant is operating at 100% power, when the EHC system receives a trip signal and the turbine stop valves begin to close. Upon reaching the 90% open (nominal) point, a reactor trip signal (Scram) is initiated, along with an end-of-cycle recirculation pump trip (EOC-RPT). Control rods begin to insert and the recirculation pumps begin to coast down. Both of which help turn around the power increase generated by the collapsing of the voids in the core from the

pressure increase due to the partial loss of a steam path. Although the bypass valves open, their steam flow capacity is not enough initially to control the pressure. So, reactor pressure increases and SRVs lift to relieve the pressure. Control rods are fully inserted to terminate the power increase.
Long-term response (beyond the explicit analyzed period): Low-low set logic armed (SRV open and High Pressure Scram), Turbine bypass valves open to control reactor pressure, Feedwater and Condensate pumps are used to maintain reactor water level. Operators take manual control and guide the plant to a cold shutdown condition.

c) Single Failure/Operator Error (as applicable):
None

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
Short term: RPS trip (TSV closure), EOC-RPT (TSV closure), Control Rod Scram, Recirculation Pump Trip, SRVs open/close.
Long-term: Low-low set logic armed, Turbine Bypass Valves open, Feedwater/Condensate pumps operate. The Operators intervene only after the initial transient is over and guide the plant to a stable condition.

<u>Event Category & Acceptance Criteria</u>

This is an Anticipated Operational Occurrence – an expected operational transient of moderate frequency (turbine trip) with no other equipment failures or Operator errors.

Fuel SAFDLs (Section 15.0.4) shall not be exceeded.

RPV Pressure shall remain within ASME Upset limits.

Methods

a) Calculation Tools & Computer Codes:
Primary Code – ODYN, using GEMINI methods. (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-3. No unique inputs for this analysis.

c) Key Assumptions:
Plant is initially at rated thermal power and core flow.
TSVs close in a linear ramp over 0.1 seconds.
No Operator Actions are assumed during the initial transient response.

Results

a) Barrier Performance and comparison to Acceptance Criteria:

Fuel Performance: Fuel thermal limits (MCPR, LHGR (MOP and TOP) and APLHGR) all remain within their respective acceptance limits. See the current cycle's Supplemental Reload Licensing Report (SRLR) for actual values.

Reactor Pressure Boundary Performance: Reactor vessel pressure remains well
2014-003 below the 1375 psig ASME acceptance limit.

Containment Performance: The containment is not challenged by this event and is not explicitly analyzed.

b) Known Sensitivities:

This event is very sensitive to the vessel pressure change. Changes in plant equipment characteristics, e.g., valve stroke times, setpoint changes, steamline lengths and volumes, etc., that will cause the pressurization rate and/or peak vessel pressure to increase will have a negative impact on the event results.

To support certain equipment being out of service during the operating cycle, additional analysis of this event is performed assuming that equipment is not Operable. Specifically, EOC-RPT is assumed to not to function. The results of this equipment out-of-service condition are found in the SRLR for the current cycle.

c) Uncertainties in Results:

The plant performance is analyzed to 95%/95% confidence levels using GEMINI methods.

Conclusion

a) Statement of Acceptability
This event meets all the fission product barrier performance criteria for an Anticipated Operational Occurrence.

b) Known Conservatisms/Margins:
Allowance for 2% core thermal over-power is accounted for in the analysis methods.
End-of-cycle core conditions are assumed.
Conservative control rod scram times are used.
Bounding (faster) TSV closure time.
Bounding SRV opening setpoints (+3% tolerance)

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
2014-003 This is considered to be a Non-Limiting Event and is bounded by TTNBP event.

15.1.2.2.2 Main Turbine Trip with Bypass Valve Failure (TTNBP) – High Power

Description of Event

a) Initiator:
A non-mechanistically caused trip of the main turbine with failure of the bypass valves to open.

b) Sequence of Events:
The plant is operating at 100% power, when the EHC system receives a trip signal and the turbine stop valves begin to close. Upon reaching the 90% open (nominal) point, a reactor trip signal (Scram) is initiated, along with an end-of-cycle recirculation pump trip (EOC-RPT). Control rods begin to insert and the recirculation pumps begin to coast down. Both of which help turn around the power increase generated by the collapsing of the voids in the core from the pressure increase due to the loss of a steam path, due to the failure of the turbine bypass valves to open. Reactor pressure increases and SRVs lift to relieve the pressure. Control rods are fully inserted to terminate the power increase. Long-term response (beyond the explicit analyzed period): Low-low set logic cycles the SRVs to control reactor pressure. Feedwater and Condensate pumps eventually are no longer available, as condenser inventory is depleted. HPCI and/or RCIC initiate to maintain reactor vessel level, as needed. Operators take manual control and guide the plant to a cold shutdown condition.

c) Single Failure/Operator Error (as applicable):
Failure of Turbine Bypass Valves to open upon demand.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures): Short term: RPS trip (TSV closure), EOC-RPT (TSV closure), Control Rod Scram, Recirculation Pump Trip, SRVs open/close.

Long-term: Low-low set logic armed, FW/Condensate pumps operate, HPCI/RCIC initiations (low-low RPV level). The Operators intervene only after the initial transient is over and guide the plant to a stable condition.

Event Category & Acceptance Criteria

This is an Abnormal Operating Transient – an expected operational transient of moderate frequency (turbine trip), but with the additional single failure of the bypass valves failing to open.

Fuel SAFDLs (Section 15.0.4) shall not be exceeded.

RPV Pressure shall remain within ASME Upset limits.

Methods

a) Calculation Tools & Computer Codes:
Primary Code – ODYN, using GEMINI methods. (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-3 (Table 15.0-3). No unique inputs for this analysis.

c) Key Assumptions:
Plant is initially at rated thermal power and 105% core flow.
TSVs close in a linear ramp over 0.1 seconds.
No Operator Actions are assumed during the initial transient response.

Results

a) Barrier Performance and comparison to Acceptance Criteria:

Fuel Performance: Fuel thermal limits (MCPR, LHGR (MOP and TOP) and APLHGR) all remain within their respective acceptance limits. See the current cycle's Supplemental Reload Licensing Report (SRLR) for actual values.

Reactor Pressure Boundary Performance: Reactor vessel pressure remains well below the 1375 psig ASME acceptance limit. See the current cycle's SRLR for actual values.

Containment Performance: The containment is not challenged by this event and is not explicitly analyzed.

b) Known Sensitivities:

This event is very sensitive to the vessel pressure change. Changes in plant equipment characteristics, e.g., valve stroke times, setpoint changes, steamline lengths and volumes, etc., that will cause the pressurization rate and/or peak vessel pressure to increase will have a negative impact on the event results.

To support certain equipment being out of service during the operating cycle, additional analysis of this event is performed assuming that equipment is not Operable (Ref. FRED form in Section 15.7). The results of this equipment out-of-service condition are found in the SRLR for the current cycle.

Note: a special case of the TTNBP was analyzed, which assumes the direct RPS signal off the TSV closure is failed, i.e., the RPS trip (Scram) comes off the resulting high neutron flux (APRM trip) instead. The results of this special case confirmed that the MSIV closure with direct scram failure is the limiting event for ASME vessel overpressure analysis. See MSIV-F event (Section 15.1.2.3.2).

c) Uncertainties in Results:

The plant performance is analyzed to 95%/95% confidence levels using GEMINI methods.

Conclusion

a) Statement of Acceptability
This event meets all the fission product barrier performance criteria for an Abnormal Operating Transient.

b) Known Conservatisms/Margins:
Allowance for 2% core thermal over-power is accounted for in the analysis methods.
End-of-cycle core conditions are assumed.
Conservative control rod scram times are used.
Bounding (faster) TSV closure time.
Bounding SRV opening setpoints (+3% tolerance)

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
This is considered to be a Limiting Event and is re-analyzed as part of the reload analysis for each operating cycle.

15.1.2.2.3 Main Turbine Trip with Bypass Valve Failure (TTNBP) – Low Power

Description of Event

a) Initiator:
A non-mechanistically caused trip of the main turbine with failure of the bypass valves to open.

b) Sequence of Events:

Note: Under the ARTS program (APRM/RBM/Technical Specification), this event is analyzed as a series of four cases representing the combination of thermal power and core flow: 21.7% power (representing the lowest power level for monitoring of fuel thermal limits), 26% power (representing the bypass of the direct scram signal on the TSVs), 50% core flow and 105% core flow. The results of these four cases help to define the MCPRp, LHGRFACp and MAPFACp limits in the Core Operating Limits Report (COLR). The limiting case of 26% power/105% flow is discussed below.

The plant is operating at 26% power at 105% rated core flow, when the EHC system receives a trip signal and the turbine stop valves begin to close. Because this power level is below the bypass for the direct scram signal on the TSVs, no direct scram is generated. Reactor pressure increases due to the loss of a steampath, as a result of the failure of the turbine bypass valves to open. The pressure quickly reaches the scram setpoint and control rods begin to insert, which help turn around the power increase generated by the collapsing of the voids in the core from the pressure increase. Reactor pressure increases and SRVs lift to relieve the pressure. Control rods are fully inserted to terminate the power increase.
Long-term response (beyond the explicit analyzed period): Low-low set logic cycles the SRVs to control reactor pressure. Feedwater and Condensate pumps eventually are no longer available, as condenser inventory is depleted. HPCI and/or RCIC initiate to maintain reactor vessel level, as needed. Operators take manual control and guide the plant to a cold shutdown condition.

c) Single Failure/Operator Error (as applicable):
Failure of Turbine Bypass Valves to open upon demand.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
Short term: RPS trip (High Dome Pressure), Control Rod Scram, SRVs open/close.
Long-term: Low-low set logic armed and controls pressure, FW/Condensate pumps operate, HPCI/RCIC initiations (low-low RPV level). The Operators intervene only after the initial transient is over and guide the plant to a stable condition.

Event Category & Acceptance Criteria

This is an Abnormal Operating Transient – an expected transient (turbine trip) with the additional single failure of the bypass valves failing to open.

Fuel SAFDLs (Section 15.0.4) shall not be exceeded.

RPV Pressure shall remain within ASME Upset limits.

Methods

a) Calculation Tools & Computer Codes:

Primary Code – ODYN, using GEMINI methods. (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items): OPL-3 (Table 15.0-3). No unique inputs for this analysis.

c) Key Assumptions:
Plant is initially at 21.7% of rated and also evaluated at 26% thermal power and both power levels are evaluated at both 105% and 50% core flow.
TSVs close in a linear ramp over 0.1 seconds.
No Operator Actions are assumed during the initial transient response

Results

a) Barrier Performance and comparison to Acceptance Criteria:

Fuel Performance: Fuel thermal limits (MCPR, LHGR (MOP and TOP) and APLHGR) all remain within their respective acceptance limits. The results of this event are used to develop the ARTS off-rated limits. See the current cycle's Supplemental Reload Licensing Report (SRLR) for actual values.

Reactor Pressure Boundary Performance: Reactor vessel pressure remains well below the 1375 psig ASME acceptance limit. Because the peak pressure is directly proportional to the initial power level, the peak pressure from this event is not evaluated.

Containment Performance: The containment is not challenged by this event and is not explicitly analyzed.

b) Known Sensitivities:

This event is very sensitive to the vessel pressure change. Changes in plant equipment characteristics, e.g., valve stroke times, setpoint changes, steamline lengths and volumes, etc., that will cause the pressurization rate and/or peak vessel pressure to increase will have a negative impact on the event results.

To support certain equipment being out of service during the operating cycle, additional analysis of this event is performed assuming that equipment is not Operable (Ref. FRED form in Section 15.7). The results of this equipment out-of-service condition are found in the SRLR for the current cycle.

c) Uncertainties in Results:

The plant performance is analyzed to 95%/95% confidence levels using GEMINI methods.

Conclusion

a) Statement of Acceptability
This event meets all the fission product barrier performance criteria for an Abnormal Operating Transient.

b) Known Conservatisms/Margins:
End-of-cycle core conditions are assumed.
Conservative control rod scram times are used.
Bounding TSV closure time.
Bounding SRV opening setpoints (+3% tolerance)

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
2014-003 This is considered to be a Non-Limiting Event and is confirmed as part of the reload analysis for each operating cycle.

15.1.2.3 Main Steam Line Isolation Valve Closure

Automatic circuitry or operator action can initiate the closure of the main steam line isolation valves. Position switches on the valves initiate a scram if valves in three or more main steam lines are less than 90% open and the mode switch is in RUN. However, reactor protection system logic does permit the test closure of one valve without initiating a scram from the position switches. These cases were investigated separately.

15.1.2.3.1 Closure of All Main Steam Line Isolation Valves (MSIV) – High Power

Description of Event

a) Initiator:
A spurious trip that causes all the MSIVs to rapidly close.

b) Sequence of Events:
The plant is operating at 100% power, when a trip signal causes all the MSIVs to rapidly close. Upon reaching the 90% open (nominal) point, a reactor trip signal (Scram) is initiated. Control rods begin to insert. The scram is fast enough to turn around the power increase generated by the collapsing of the voids in the core from the pressure increase due to the loss of a steam path, due to the closure of the MSIVs. Reactor pressure increases and SRVs lift to relieve the pressure. The pressure reaches the ATWS-RPT setpoint, which trips the reactor recirculation pumps and they begin to coastdown.

Long-term response (beyond the explicit analyzed period): Low-low set logic cycles the SRVs to control reactor pressure, Feedwater and Condensate pumps eventually are no longer available, as condenser inventory is depleted, HPCI and/or RCIC initiate to maintain reactor vessel level, as needed. Operators take manual control and guide the plant to a cold shutdown condition using normal shutdown procedures.

c) Single Failure/Operator Error (as applicable):
None

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
Short term: RPS trip (MSIV closure), Control Rod Scram, ATWS-RPT Recirculation Pump Trip (Reactor Pressure), SRVs open/close.

Long-term: Low-low set logic arm and control pressure, FW/Condensate pumps operate, HPCI/RCIC initiations (low-low RPV level). The Operators intervene only after the initial transient is over and guide the plant to a stable condition.

<u>Event Category & Acceptance Criteria</u>

This is an Anticipated Operational Occurrence – an expected operational transient of moderate frequency (MSIV closure), with no other equipment failures or Operator errors.

Fuel SAFDLs (Section 15.0.4) shall not be exceeded.

RPV Pressure shall remain within ASME Upset limits.

Methods

a) Calculation Tools & Computer Codes:
Primary Code – ODYN, using GEMINI methods. (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-3 (Table 15.0-3). No unique inputs for this analysis.

c) Key Assumptions:
Plant is initially at rated thermal power and core flow.
MSIVs close in a linear ramp over 3.0 seconds.
No Operator Actions are assumed during the initial transient response.

<u>Results</u>

a) Barrier Performance and comparison to Acceptance Criteria:

Fuel Performance: Fuel thermal limits (MCPR, LHGR (MOP and TOP) and APLHGR) all remain within their respective acceptance limits. Because of the direct scram, there is no power increase as a result of this event.

Reactor Pressure Boundary Performance: Reactor vessel pressure remains well below the 1375 psig ASME acceptance limit. The pressure increase is much less severe than other pressurization events, especially those with loss of bypass capacity.

Containment Performance: The containment is not challenged by this event and is not explicitly analyzed.

b) Known Sensitivities:

This event is only mildly sensitive to changes in plant parameters. Changes in plant equipment characteristics, such as slower scram speed/time, MSIV closure speed and MSIV scram setpoint, will cause the pressurization rate and/or peak vessel pressure to increase, which will begin to balance the negative reactivity from the direct scram.

c) Uncertainties in Results:

The plant performance is analyzed to 95%/95% confidence levels using GEMINI methods.

Conclusion

a) Statement of Acceptability
This event meets all the fission product barrier performance criteria for an Abnormal Operating Occurrence.

b) Known Conservatisms/Margins:
Allowance for 2% core thermal over-power is accounted for in the analysis methods.
End-of-cycle core conditions are assumed.
Conservative control rod scram times are used.
Bounding (faster) MSIV closure time.
Bounding SRV opening setpoints (+3% tolerance)

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):

This is considered to be a Non-Limiting Event and is not re-analyzed as part of the reload analysis for each operating cycle.

15.1.2.3.2 Closure of All Main Steam Line Isolation Valves with Direct Scram Failure (MSIVF) – High Power

Description of Event

a) Initiator:
A spurious trip that causes all the MSIVs to rapidly close. However, the direct scram signal from the MSIV position switches is assumed to fail.

Note the MSIV closure with direct scram failure, which is a special case of the MSIV transient, is analyzed as the limiting event for ASME vessel overpressure analysis. Fuel barrier performance is not analyzed as part of this event.

b) Sequence of Events:
The plant is operating at an overpower condition (102% of rated), when a trip signal causes all the MSIVs to rapidly close. The power increases quickly by the collapsing of the voids in the core from the pressure increase due to the loss of a steam path from the closure of the MSIVs. Upon reaching the APRM 120% high flux (nominal) trip point, a reactor trip signal (Scram) is initiated. Control rods begin to insert and terminate the power increase. The pressure reaches the ATWS-RPT setpoint, which trips the reactor recirculation pumps and they begin to coastdown. Reactor pressure continues to increase and SRVs, and potentially the Spring Safety Valves (SSVs), lift to relieve the pressure.
Long-term response (beyond the explicit analyzed period): Low-low set logic armed (SRV open and High Pressure Scram) and cycles the SRVs to control reactor pressure. Feedwater and Condensate pumps eventually are no longer available, as condenser inventory is depleted. HPCI and/or RCIC initiate to maintain reactor vessel level, as needed. Operators take manual control and guide the plant to a cold shutdown condition.

c) Single Failure/Operator Error (as applicable):
MSIV direct scram from the position switches is assumed to fail.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
Short term: RPS trip (APRM Flux - High), Control Rod Scram, ATWS-RPT Recirculation Pump Trip (Reactor Pressure), SRVs (and SSVs) open/close.

Long-term: Low-low set logic armed (SRV open and High Pressure Scram) and cycles the SRVs to control reactor pressure, FW/Condensate pumps operate, HPCI/RCIC initiations (low-low RPV level). The Operators intervene only after the initial transient is over and guide the plant to a stable condition.

Event Category & Acceptance Criteria

This is an Abnormal Operating Transient – an expected operational transient of moderate frequency (MSIV closure), with the additional single failure of the MSIV direct scram signal.

RPV Pressure shall remain within ASME Upset limits.

Methods

a) Calculation Tools & Computer Codes:
Primary Code – ODYN. (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-3 (Table 15.0-3). No unique inputs for this analysis.

c) Key Assumptions:
Plant is initially at 102% thermal power and 105% core flow.
Reactor Dome Pressure is initially at the corresponding value of 1055 psia.
MSIVs close in a linear ramp over 3.0 seconds.
MSIV direct position scram fails.
No Operator Actions are assumed during the initial transient response.

Results

a) Barrier Performance and comparison to Acceptance Criteria:

Fuel Performance: Fuel thermal limits (MCPR, LHGR (MOP and TOP) and APLHGR) are not analyzed for this event.

Reactor Pressure Boundary Performance: Reactor vessel pressure remains below the 1375 psig ASME acceptance limit. See the current cycle's Supplemental Reload Licensing Report (SRLR) for results.

Containment Performance: The containment is not challenged by this event and is not explicitly analyzed.

b) Known Sensitivities:

This event is only moderately sensitive to changes in plant parameters. Changes in plant equipment characteristics, such as slower scram speed/time, faster MSIV closure speed and APRM flux scram setpoint, will cause the pressurization rate and/or peak vessel pressure to increase, which will begin to balance the negative reactivity from the scram.

c) Uncertainties in Results:

Application of the ASME Upset limit as the acceptance criterion compensates for the uncertainties in the analysis methods and input values.

Conclusion

a) Statement of Acceptability:
This event meets the high vessel pressure performance criteria for an ASME Upset event.

b) Known Conservatisms/Margins:
Allowance for 2% core thermal over-power.
End-of-cycle core conditions are assumed.
Conservative control rod scram times are used.
Bounding (faster) MSIV closure time.
Failure of the MSIV direct position scram.
Bounding SRV opening setpoints (+3% tolerance).
Application of ASME Upset limits as an acceptance criteria, when the Code would allow the application of Emergency limits (pressure < 1500 psig).

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
This is considered to be a special case for demonstrating compliance to the ASME vessel overpressure protection requirements and is analyzed as part of the reload analysis for each operating cycle.

15.1.2.3.3 Closure of One Main Steam Line Isolation Valve (1 MSIV) – High Power

Description of Event

a) Initiator:
A spurious trip (or Operator error) causes one of the MSIVs to rapidly close. However, because only one MSIV closes, the direct scram signal from its position switch will not cause a trip, by design.

b) Sequence of Events:
The plant is operating at rated conditions (100% of rated power and core flow), when a trip signal (or Operator error) causes one MSIV to fast close. The remaining three steamlines cannot compensate for the closed steamline and vessel pressure increases. The power increases quickly from the collapsing of the voids in the core from the pressure increase. Upon reaching the APRM 120% high flux (nominal) trip point, a reactor trip signal (Scram) is initiated. Control rods begin to insert and terminate the power increase. The steamflow is rebalanced between the remaining steamlines. Feedwater and pressure control systems react to the

dynamic changes. Reactor pressure continues to increase and SRVs lift to relieve the pressure. The turbine bypass valves also open to control the pressure. Long-term response (beyond the explicit analyzed period): Low-low set logic cycles the SRVs to control reactor pressure when the decay heat load is high and then the TCV and Bypass Valves control the pressure once decay heat load is decreased. Feedwater and Condensate pumps maintain reactor vessel level. Operators take manual control and guide the plant to a cold shutdown condition.

c) Single Failure/Operator Error (as applicable):
None

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
Short term: RPS trip (APRM Flux - High), Control Rod Scram, SRVs open/close.
Long-term: Low-low set logic Low-low set logic armed (SRV open and High Pressure Scram) and cycles the SRVs to control reactor pressure. The Operators intervene only after the initial transient is over and guide the plant to a stable condition using normal shutdown procedures.

Event Category & Acceptance Criteria

This is an Anticipated Operational Occurence – an expected operational transient of moderate frequency (single MSIV closure), with no other equipment failures or Operator errors.

Fuel SAFDLs (Section 15.0.4) shall not be exceeded.

RPV Pressure shall remain within ASME Upset limits.

Methods

a) Calculation Tools & Computer Codes:
Primary Code – ODYN, using GEMINI methods. (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-3 (Table 15.0-3). No unique inputs for this analysis.

c) Key Assumptions:
Plant is initially at rated thermal power and rated core flow.
The MSIV closes in a linear ramp over 3.0 seconds.
No Operator Actions are assumed during the initial transient response.

Results

a) Barrier Performance and comparison to Acceptance Criteria:

Fuel Performance: Fuel thermal limits (MCPR, LHGR (MOP and TOP) and APLHGR) all remain within their respective acceptance limits. This is a non-limiting event as the change in fuel thermal limits is bounded by the other pressurization events.

Reactor Pressure Boundary Performance: Reactor vessel pressure remains well below the 1375 psig ASME acceptance limit. The pressure increase is much less severe than other pressurization events, especially those with loss of bypass capacity.

Containment Performance: The containment is not challenged by this event and is not explicitly analyzed.

b) Known Sensitivities:

This event is only moderately sensitive to changes in plant parameters. Changes in plant equipment characteristics, such as slower scram speed/time, MSIV closure speed and APRM flux scram setpoint, will cause the pressurization rate and/or peak vessel pressure to increase, which will begin to balance the negative reactivity from the scram.

c) Uncertainties in Results:

The plant performance is analyzed to 95%/95% confidence levels using GEMINI methods.

Conclusion

a) Statement of Acceptability:
This event meets all the fission product barrier performance criteria for an Anticipated Operational Occurrence.

b) Known Conservatisms/Margins:
Allowance for 2% core thermal over-power is accounted for in the analysis methods.
End-of-cycle core conditions are assumed.
Conservative control rod scram times are used.
Bounding (faster) MSIV closure time.
Bounding SRV opening setpoints (+3% tolerance).

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
This is considered to be a Non-Limiting Event and is not re-analyzed as part of the reload analysis for each operating cycle.

15.1.3 TRANSIENTS 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 pumps.

15.1.3.1 Recirculation Flow Control Failure - Decreasing Flow

Several varieties of recirculation flow control malfunctions can cause a decrease in core coolant flow. A master controller could malfunction in such a way that a zero speed signal is generated for both recirculation pumps. The recirculation flow control system is provided with a speed demand limiter that is set so that this situation cannot be more severe than the simultaneous tripping of both recirculation pumps. A simultaneous trip of both recirculation pumps is evaluated in Section 15.1.3.3. The master controller has been removed, thus, this is no longer a credible event at the DAEC.

The remaining recirculation flow controller malfunction is one in which the speed controller for one recirculation pump motor-generator (M-G) set fails in such a way that the speed controller output signal changes in the direction of zero speed. This transient is similar to the trip of one recirculation pump (evaluated in Section 15.1.3.2). However, the pump speed reduction is slower than that resulting from the opening of a field breaker so that the event is bounded by the single recirculation pump trip.

15.1.3.2 Trip Of One Recirculation Pump

NOTE: the information in the following section is historical in nature and was not updated as part of the Extended Power Uprate Project. It is being presented here to show basic plant response and parametric trends only.

Description of Event

g) Initiator:
A malfunction occurs that cause one of the main reactor recirculation pumps to trip (e.g., opening the motor-generator set generator field circuit breaker opens) while the reactor is operating at rated power/flow conditions on the highest permissible loadline.

h) Sequence of Events (NOT a time line):
Short term: A malfunction causes one of the main reactor recirculation pumps to trip with the reactor at rated power/core flow conditions. This event is assumed to

occur on the highest allowable loadline, causing the final power level to be maximized. There is a sudden, swell in water level due to increased voiding in the core. Thus, reactor power initially goes down. The level swell is small and does not reach the high level trip setpoint (Level 8). The level control system quickly compensates for the increase in water level by closing down on the feedwater regulating valves. The reactor stabilizes at new steady, state conditions. This is a very mild transient on the fuel and vessel. The Operators take control of the plant and maintain the water level and pressure at the new conditions. The plant is licensed to operate in Single Loop Operation.

Note: a special case of this event is analyzed in Section 15.3.4 – Thermal-Hydraulic Stability.

i) Single Failure/Operator Error (as applicable):
None

j) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
No trips or other actuations occur during this event.
The Operators intervene only after the initial transient is over and maintain the plant in a stable condition.

Event Category & Acceptance Criteria:

This is an Anticipated Operational Occurrence – an expected operational transient of moderate frequency (single recirculation pump trip), with no other equipment failures or Operator errors.

Fuel SAFDLs (Section 15.0-4) shall not be exceeded.

RPV Pressure shall remain within ASME Upset limits.

Methods

d) Calculation Tools & Computer Codes:
Primary Code – ODYN, using GEMINI methods. (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

e) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-3 (See Table 15.0-3)

f) Key Assumptions:
Plant is initially at rated thermal power and core flow on the highest allowable loadline.
Initial vessel level swell does not reach the Level 8 trip point.

Results

d) Comparison to Acceptance Criteria:

Fuel Performance: Fuel thermal limits (MCPR, LHGR (MOP and TOP) and APLHGR) all remain within their respective acceptance limits. Because of the initial level swell (void increase), there is no power increase above the initial value as a result of this event. Steady State operation in Single Loop is allowed by Technical Specifications, provided that the appropriate adjustments are made in the fuel thermal limits, see the current cycle's Core Operating Limits Report (COLR).

Reactor Pressure Boundary Performance: Reactor vessel pressure remains well below the 1375 psig ASME acceptance limit. The reactor pressure decreases during this event.

Containment Performance: The containment is not challenged by this event and is not explicitly analyzed.

e) Known Sensitivities:

The tripping of the M/G set drive motor breaker, instead of the generator field circuit breaker, would maintain the M/G set in the dynamic response, such that its inertia would lessen the recirculation flow decrease and overall plant response.

f) Uncertainties in Results:

The amount of water level swell is the key variable. If the swell reaches the high level trip point, then a turbine trip and feedwater pump would occur.

Conclusion

g) Statement of Acceptability:
This event meets all the fission product barrier performance criteria for an Anticipated Operational Occurrence.

h) Known Conservatisms/Margins:
The off-rated power and flow multipliers for the fuel thermal limits (ARTS) are conservatively derived and provide margin to the actual operating limits at the final steady state operating conditions.

i) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
This is considered to be a Non-Limiting Event and is not re-analyzed as part of the reload analysis for each operating cycle. However, Single Loop Operation is re-validated as part of each reload analysis.

15.1.3.3 Trip Of Two Recirculation Pumps

This transient primarily evaluated the fuel thermal margin maintained by the rotating inertia of the recirculation system drive equipment. The inertia from the recirculation flow control system M-G sets is included because no single event can simultaneously open the generator field circuits of both M-G sets. This transient results if the power supply to both M-G sets is lost, the most-likely cause of which would be a loss-of-offsite power (LOOP), which is discussed in Section 15.1.2. A special case of this event is discussed in Section 15.3.4.

15.1.4 TRANSIENTS REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Events that result directly in rapid power increases and core reactivity are included in this section. 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.

15.1.4.1 Rod Withdrawal Error At Power

Description of Event

k) Initiator:
With the reactor operating at high power (rated power and core flow), the Operator selects the highest worth, fully-inserted control rod and begins to continuously withdraw it.

l) Sequence of Events:
The reactor is operating at high power (> 85% of rated), when the Operator selects a fully inserted control rod, with the highest rod worth, and begins to continuously withdraw it at the maximum withdrawal speed of 3.6 inches/sec. The local power in the adjacent fuel assemblies begins to increase. When the increase in power begins to approach the thermal limits for those bundles, the Local Power Range Monitors (LPRMs) will alarm. The Operator ignores these alarms and continues to withdraw the control rod. The Rod Block Monitor (RBM) also monitors the local power change and alarms when the change in power reaches the setpoint. The Operator ignores the alarm and continues to withdraw the control rod. At the 108% rod block setpoint (Analytical Limit), the RBM generates a rod block to prevent further rod withdrawal before the local power increase can violate the fuel thermal limits.

m) Single Failure/Operator Error (as applicable):
The Operator ignores the LPRM and RBM alarms and continues to withdraw the control rod. The most-responsive channel of the RBM is assumed to be inoperable. Random LPRM failures are assumed in the analysis.

n) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures)
RBM rod block at the high power (>85% of rated) setpoint (>108%).

Event Category & Acceptance Criteria

This is an Abnormal Operating Transient – an expected operational transient (continuous rod withdrawal), but with the Operator error (ignores the LPRM and RBM alarms) and the additional equipment failure of one channel of RBM.

Fuel SAFDLs shall not be exceeded.

Methods

g) Calculation Tools & Computer Codes:
The primary code used to perform this analysis is the GE 3-D Core Simulator (PANACEA), with input to another code (GROMT) that simulates the RBM system. (See Table 15.0-2 for complete listing, code versions and NRC acceptance).

h) Inputs (Reference common list in 15.0, and/or include event-specific items):
Cycle-specific fuel bundle designs (FRED form – see Section 15.0.7).
RBM setpoint (FRED form).

i) Key Assumptions:
Reactor is operating at high power/flow.
Error rod is assumed to be the highest worth control rod in the core.
The fuel assemblies adjacent to the error rod are initially operating at the maximum allowable fuel thermal limits (Operating Limits).
The “most responsive” channel of RBM is not available/operable during the event.

Results

g) Comparison to Acceptance Criteria:

Fuel Performance: Fuel thermal limits (MCPR, LHGR (MOP and TOP) and APLHGR) all remain within their respective acceptance limits. See the current cycle’s Supplemental Reload Licensing Report (SRLR) for actual values.

Reactor Pressure Boundary Performance: Reactor vessel pressure is not challenged by this event and is not explicitly analyzed.
Containment Performance: The containment is not challenged by this event and is not explicitly analyzed.

h) Known Sensitivities:

The results are somewhat sensitive to the initial core power; hence, the power-dependent setpoints for the RBM system.

i) Uncertainties in Results

The plant performance is analyzed to 95%/95% confidence levels using GEMINI methods. The original generic RWE analysis performed for the APRM, RBM and Technical Specification (ARTS) program (Ref. 15.0-57), which determined the power-dependent RBM setpoints, was done to 95%/95% confidence levels.

Conclusion

j) Statement of Acceptability:
This event meets all the fission product barrier performance criteria for an Abnormal Operating Transient.

k) Known Conservatisms/Margins:
The control rod pattern is manipulated in the analysis to generate the high worth rod and localized conditions of the adjacent fuel assemblies operating at the allowable thermal limits as an initial condition of the event.
Maximum allowable control rod withdrawal speed is used.
A random distribution of LPRM failures is assumed.

l) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
This is a Limiting Event and is evaluated as part of the cycle-specific reload analysis. This evaluation is done to confirm the original generic RWE analysis done for the ARTS program.

15.1.4.2 Rod Withdrawal Error At Startup

Description of Event

a) Initiator:
The reactor is critical and in the startup range when the Operator makes a selection error (out-of-sequence rod) and continuously withdraws the control rod with the highest rod worth from the fully inserted position.

b) Sequence of Events:
With the reactor critical and operating in the startup range, the Operator makes a selection error of an out-of-sequence control rod and continuously withdraws the highest worth control rod in the core from the full-in position at the maximum withdrawal speed (3.6 inches/sec). The Rod Worth Minimizer (RWM) is not functioning and the second Licensed Operator does not catch the out-of-sequence control rod selection and withdrawal. The core power reaches the scram setpoint of the Intermediate Range Neutron Monitor (IRM), and the scram inserts the control rods, including the error rod, and stops further increases in core power.

c) Single Failure/Operator Error (as applicable):

The most responsive channel (nearest to the control rod) of the IRM system is assumed to fail/not operable (bypassed).

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
The operable IRM channels trip and generates an RPS scram (dependent on IRM range, trip setpoint of either 40/40 or 125/125 of scale (Analytical Limits)). Control Rods scram at Technical Specification insertion speed.
The second Licensed Operator does not catch the out-of-sequence control rod selection and withdrawal.

Event Category & Acceptance Criteria

This is an Abnormal Operating Transient – an expected operational transient (rod withdrawal error), but with the additional single failure (most-responsive IRM channel fails).

Note: in reality, this is a highly unlikely event, as multiple equipment failures (RWM and IRM), coupled with multiple Operator errors (selection error and second Licensed Operator error) have to occur for this event to happen.

Fuel SAFDLs shall not be exceeded. In particular, the peak fuel enthalpy shall be < 170 cal/gm.

Methods

a) Calculation Tools & Computer Codes:
This is a generic analysis and is not done on a plant-specific basis (Ref. 15.0-2).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
This is a generic analysis and is not done on a plant-specific basis (Ref. 15.0-2).

c) Key Assumptions:
Reactor is critical and in the startup range, below the low power setpoint of the RWM. Error rod is assumed to be the highest worth control rod in the core.
The most responsive channel (nearest channel) of IRM is not available/operable during the event.

Results

a) Comparison to Acceptance Criteria:

Fuel Performance: Peak fuel enthalpy is well below the limit of 170 cal/gm.

Reactor Pressure Boundary Performance: Reactor vessel pressure is not challenged by this event and is not explicitly analyzed.

Containment Performance: The containment is not challenged by this event and is not explicitly analyzed.

b) Known Sensitivities

The result of this event is primarily sensitive to the rod worth of the error rod.

c) Uncertainties in Results

Use of conservative assumptions in the evaluation are intended to bound the uncertainties in the final results.

<u>Conclusion</u>

a) Statement of Acceptability:
This event meets all the fission product barrier performance criteria for an Abnormal Operating Transient.

b) Known Conservatisms/Margins:
Maximum allowable control rod withdrawal speed is used.
The error control rod is assumed to be fully withdrawn, when in fact, the scram will terminate the withdrawal after only partial withdrawal.
The most-responsive IRM channel is not available (bypassed), which delays the scram.
Banked Position Withdrawal Sequence (BPWS), which is programmed into the RWM limits control rod worth to well below the value assumed in this evaluation.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents)
This is a non-Limiting Event and is not evaluated as part of the cycle-specific reload analysis.

15.1.4.3 Control Rod Removal Error During Refueling

The nuclear characteristics of the core ensure 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. The 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 that 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.

15.1.4.4.1 Fuel Assembly Insertion Error During Refueling-Inadvertent Criticality

The core is designed such that it can be made subcritical under the most reactive conditions with the strongest control rod fully withdrawn. The refueling shutdown margin is determined each cycle by using a 3-D, safety-related BWR simulator code as referenced in Table 15.0-2. Refueling shutdown margin is confirmed to meet Technical Specification LCO 3.1.1. 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. Because of the above-mentioned constraints, there is no analysis required for this event.

15.1.4.4.2 Fuel Loading Error – Mislocated Bundle

Description of Event

a) Initiator:
During the fuel reloading process, two bundles are misloaded into the core in the opposite core locations (i.e., swapped).

b) Sequence of Events:
During the fuel reloading process, two bundles are loaded into the core in the opposite core locations. These mislocated bundles are not discovered during the core loading verification process and the reactor is started up and operates at rated power and flow with the bundles in the wrong locations for the entire fuel cycle.

c) Single Failure/Operator Error (as applicable)
The verification of the core loading does not catch the mislocated bundle error.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures)
None.

Event Category & Acceptance Criteria

This is an Abnormal Operating Transient – an expected operational transient (mislocated bundle), but with the Operator/Reactor Engineer error (fails to catch the mislocated bundles during the various core loading verifications.).

Fuel SAFDLs shall not be exceeded.

Methods

a) Calculation Tools & Computer Codes
The primary code used to perform this analysis is the GE 3-D Core Simulator (PANACEA). (See Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items)
Cycle-specific fuel bundle designs (FRED form-see Section 15.0.7).

c) Key Assumptions
A high power bundle is swapped with a low power bundle in a core cell that maximizes the power increase on the high power bundle.
The core cell containing the mislocated high power bundle is not a location directly monitored by an LPRM string or a TIP monitor (i.e., separated by at least one fuel bundle from the detectors).

Results

a) Comparison to Acceptance Criteria:

Fuel Performance: Fuel thermal limits (MCPR, LHGR (MOP and TOP) and APLHGR) all remain within their respective acceptance limits.

Reactor Pressure Boundary Performance: Reactor vessel pressure is not challenged by this event and is not explicitly analyzed.

Containment Performance: The containment is not challenged by this event and is not explicitly analyzed.

b) Known Sensitivities

The results of this event are dependent upon the mismatch of the bundle powers and core loading locations involved.

c) Uncertainties in Results

The plant performance is analyzed to 95%/95% confidence levels using GEMINI methods.

Conclusion

a) Statement of Acceptability
This event meets all the fission product barrier performance criteria for an Abnormal Operating Transient.

b) Known Conservatisms/Margins
Use of TIP adaptive core monitoring methods would reduce the impact if the mislocated bundle were in a monitored location (highly likely), as the local power would be adjusted to maintain the fuel within thermal limits and there would be no impact from the misloading.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents)
2014-003 This is a non-Limiting Event and is confirmed as part of the cycle-specific reload analysis.

15.1.4.4.3 Fuel Loading Error – Rotated Bundle

<u>Description of Event</u>

a) Initiator:
During the fuel reloading process, a bundle is misloaded into the core in the proper core location, but is rotated in orientation by either 90° or 180°, whichever produces the worst result.

b) Sequence of Events:
During the fuel reloading process, a bundle is misloaded into the core in the proper core location, but is rotated in orientation by either 90° or 180°, whichever produces the worst result. This misoriented bundle is not discovered during the core loading verification process and the reactor is started up and operates the entire cycle with the bundle in the wrong orientation for the entire fuel cycle.

c) Single Failure/Operator Error (as applicable):
The verification of the core loading does not catch the error of the misoriented bundle.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
None.

Event Category & Acceptance Criteria

This is an Abnormal Operating Transient – an expected operational transient (misoriented bundle), but with the Operator/Reactor Engineer error (fails to catch the misoriented bundle during the various core loading verifications.).

Fuel SAFDLs shall not be exceeded.

Methods

a) Calculation Tools & Computer Codes:
The primary code used to perform this analysis is the GE 3-D Core Simulator (PANACEA). (See Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
Cycle-specific fuel bundle designs (FRED form-see Section 15.0.7).

c) Key Assumptions:
A high power bundle is rotated in a core cell that maximizes the power increase on the rotated bundle (i.e., maximizes the rotated R-factor on the bundle).
To account for the fact that the misoriented bundle may be tilted and not seated correctly, a bias of 0.02 ΔCPR is added to the results to account for the variable water gap.

Results

a) Comparison to Acceptance Criteria:

Fuel Performance: Fuel thermal limits (MCPR, LHGR (MOP and TOP) and APLHGR) all remain within their respective acceptance limits. See the current cycle's Supplemental Reload Licensing Report (SRLR) for actual values.
Reactor Pressure Boundary Performance: Reactor vessel pressure is not challenged by this event and is not explicitly analyzed.

Containment Performance: The containment is not challenged by this event and is not explicitly analyzed.

b) Known Sensitivities:

The results of this event are dependent upon the pin-to-pin peaking factor of the bundle (rotated R-factor) and its core loading location.

c) Uncertainties in Results:

The plant performance is analyzed to 95%/95% confidence levels using GEMINI methods. A bias of 0.02 ΔCPR is added to the results to account for the uncertainty due to a variable water gap if the fuel assembly not seated correctly and is tilted toward the control rod (preferential direction).

Conclusion

a) Statement of Acceptability:
This event meets all the fission product barrier performance criteria for an Abnormal Operating Transient.

b) Known Conservatisms/Margins:
There are five separate visual indications of proper bundle orientation. Operating experience has shown that these indications are readily visible during the fuel loading process. Thus, the actual probability of this event is quite low.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents)
This is a Limiting Event and is evaluated as part of the cycle-specific reload analysis.

15.1.5 TRANSIENTS RESULTING IN AN INCREASE IN CORE FLOW

Events that result in an increase in core coolant flow cause a decrease in core void fraction and a corresponding increase in neutron flux and reactor power.

The following are the identified events in this category:

a) Startup of an Idle Recirculation Pump

b) Recirculation Flow Controller Failure – Increasing Flow (Fast)

c) Recirculation Flow Controller Failure – Slow Flow Runout

15.1.5.1 Startup Of An Idle Recirculation Pump

NOTE: the information in the following section is historical in nature and was not updated as part of the Extended Power Uprate Project. It is being presented here to show basic plant response and parametric trends only.

Description of Event

a) Initiator:
The plant is initially operating in Single Loop Operation (SLO), when the Operator starts up the idle recirculation pump without pre-warming the coolant in the loop, as required by procedures and Technical Specifications.

b) Sequence of Events (NOT a time line):

Case a) The plant is initially operating in SLO at 68% power (pre-Uprate) and 48% core flow, when the Operator starts up the idle recirculation pump without pre-warming the loop. The pump discharge valve is opened. The resulting surge in core flow, with the accompanying decrease in inlet subcooling from the slug of colder water in the idle loop, collapses the voids in the core and reactor power increases. However, the resulting increase in core power/neutron flux is not great enough to cause an APRM flow-biased scram. Reactor vessel water level and pressure are only slightly affected by this event.

Case b) The plant is initially operating in SLO at 55% power (pre-Uprate) and 38% core flow, when the Operator starts up the idle recirculation pump without pre-warming the loop. The pump discharge valve is opened. The resulting surge in core flow, with the accompanying decrease in inlet subcooling from the slug of colder water in the idle loop, collapses the voids in the core and reactor power increases. In this case, the resulting increase in neutron flux/core power is sufficient to reach the APRM flow-biased scram setting and a reactor scram occurs. Again, reactor water level and pressure are only slightly affected by this transient.

c) Single Failure/Operator Error (as applicable):
None

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):

Case a) Feedwater and Pressure control systems respond to these changes and maintain vessel level and pressure within normal operating limits.

Case b) RPS trip on reaching the APRM flow-biased scram trip setpoint and Control Rod Scram. Feedwater and Pressure control systems respond to these changes and maintain vessel level and pressure within normal operating limits.

<u>Event Category & Acceptance Criteria:</u>

This is an Anticipated Operational Occurrence – a single transient of moderate frequency (idle recirculation pump start with inadequate loop warmup) with no other equipment failures or Operator errors.

Fuel SAFDLs shall not be exceeded.

RPV Pressure shall remain within ASME Upset limits.

Methods

a) Calculation Tools & Computer Codes:
This is an historical event analysis. The computer code/version used to perform this evaluation is no longer used. Any future re-analysis would be a change in methods.

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
This is an historical event analysis. The inputs used are no longer valid, due to Extended Power Uprate and other plant changes since the original analysis was conducted.

c) Key Assumptions:
Recirculation pump starts up in 8 seconds from closing the generator field breaker. The recirculation discharge valve stroke time is 30 secs.
The temperature difference between the reactor and the coolant in the idle loop is 100°F.

Results

a) Comparison to Acceptance Criteria:

Fuel Performance: Fuel thermal limits (MCPR, LHGR (MOP and TOP) and APLHGR) all remain well within their respective acceptance limits.

Reactor Pressure Boundary Performance: Reactor vessel pressure is not significantly affected by this event and is well within limits.

Containment Performance: The containment is not challenged by this event and is not explicitly analyzed.

b) Known Sensitivities:

The results of this event are highly sensitive to the initial conditions. The two cases analyzed represent the bounding results. Case a) represents the highest power level achievable in SLO (pre-Uprate). At this higher power and core flow, the resulting core ΔP is sufficiently high to cause part of the running recirculation loop flow to bypass the core and cause reverse flow in the idle loop jet pumps. Thus, when the idle loop is started up, there is an initial resistance of this reverse flow to be overcome, which limits the impact of the surge of cooler water from the idle loop. However, in Case b), the power and flow are maximized to be just below the point where reverse flow occurs in the idle loop jet pumps. Thus, when the idle loop is started up, the flow in the idle loop is in the forward direction, which causes a greater initial impact, as the surge of cooler water goes immediately into the core, causing a higher spike in neutron flux and resulting scram to be generated.

c) Uncertainties in Results:

Use of conservative assumptions (e.g., ΔT greater than allowed by Tech Specs) in the evaluation are intended to bound the uncertainties in the final results.

Conclusion

a) Statement of Acceptability:
This event meets all the fission product barrier performance criteria for an Anticipated Operational Occurrence.

b) Known Conservatisms/Margins:
This analysis used a change in coolant temperature of 100°F, whereas TS limit this to 50°F. Per SIL No. 517, the licensing basis for this event was changed to allow the 50°F change in temperature as part of the ARTS program.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
This is a non-Limiting Event and is not evaluated as part of the cycle-specific reload analysis.

15.1.5.2 Recirculation Flow Controller Failure – Increasing Flow (Fast)

Description of Event

a) Initiator:
A failure in the recirculation flow controller causes one recirculation pump to increase speed at maximum rate.

b) Sequence of Events (NOT a time line):
The plant is initially at 55.7% core thermal power and 39% of rated core flow – two pump minimum speed at the highest allowable core loadline. A failure within the recirculation flow control system causes one of the recirculation pumps to increase speed at the maximum rate. The pump runs out to the MG set scoop tube lockup position. Core flow increases and initially reduces core voiding, leading to an increase in neutron flux/reactor power. The increase is large enough to cause a reactor scram on high neutron flux. The control rods insert and terminate the event. Initially reactor level drops, due to the mismatch between steam flow and feedwater flow. The feedwater control system reacts and increases feed flow to recover level. Reactor pressure is only mildly affected by this event and the pressure control system easily maintains pressure to the initial value.

c) Single Failure/Operator Error (as applicable):
<u>None</u>

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
RPS trips (high neutron flux-fixed and Control Rod Scram. Feedwater Control system reacts to the steam flow-feed flow mismatch and corrects the decreasing water level. The Operators intervene only after the initial transient is over and guide the plant to a stable condition.

<u>Event Category & Acceptance Criteria:</u>

This is an Anticipated Operational Occurrence – an expected operational transient of moderate frequency (recirculation pump speed increase) with no other equipment failures or Operator errors.

Fuel SAFDLs shall not be exceeded.

RPV Pressure shall remain within ASME Upset limits.

Methods

a) Calculation Tools & Computer Codes:

Primary Code – ODYN, using GEMINI methods. (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items): OPL-3 (Table 15.0-3).

c) Key Assumptions:
The controller increases pump speed at the maximum rate of 25%/sec.
MG set scoop tube positioner set at 102.5% speed (nominally 1710 RPM).
Note: this evaluation is for EPU and was not re-performed for ICF, which permits core flow up to 105% of rated flow.

Results

a) Comparison to Acceptance Criteria:

Fuel Performance: Fuel thermal limits (MCPR, LHGR (MOP and TOP) and APLHGR) all remain within their respective acceptance limits. The calculated TOP (37.99%) exceeded the TOP limit of 28%. However, when the ARTS MAPFACp multiplier is applied for the off-rated condition, the adjusted TOP is within limits.

Reactor Pressure Boundary Performance: Reactor vessel pressure remains well below the 1375 psig ASME acceptance limit.

Containment Performance: The containment is not challenged by this event and is not explicitly analyzed.

b) Known Sensitivities:

Small changes in the initial power/flow point have a negligible impact on the results.

The increase in maximum pump flow to 105% for ICF does not impact these results as the scram terminates the event prior to the recirculation pump reaching the 102.5% speed in the original analysis.

c) Uncertainties in Results:

The plant performance is analyzed to 95%/95% confidence levels using GEMINI methods.

Conclusion

a) Statement of Acceptability:

This event meets all the fission product barrier performance criteria for an Anticipated Operational Occurrence.

b) Known Conservatisms/Margins:
The recirculation flow controllers have a clamp (lockup) on rate of change at 40%/minute (0.67%/sec).
End-of-cycle core conditions are assumed.
Conservative control rod scram times are used.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
This is a non-Limiting Event and is not evaluated as part of the cycle-specific reload analysis. It was included in the analyses for Extended Power Uprate, as required by the NRC to confirm that it remained a non-limiting event after the uprate.

15.1.5.3 Recirculation Flow Controller Failure – Slow Flow Runout

Description of Event

a) Initiator:
A failure in the recirculation flow controller causes one (or both) recirculation pump(s) to increase speed at a slow rate.

b) Sequence of Events (NOT a time line):
The plant is initially at 55.7% core thermal power and 39% of rated core flow – two pump minimum speed at the highest allowable core loadline. A failure within the recirculation flow control system causes one (or both) of the recirculation pumps to increase speed at a slow rate. The pump runs out to the MG set scoop tube lockup position. Core flow increases and initially reduces core voiding, leading to an increase in neutron flux/reactor power. The increase is slow enough such that the core conditions are in quasi-equilibrium. The feedwater control system maintains vessel water level. Reactor pressure is not affected by this event and the pressure control system easily maintains pressure at the initial value.

c) Single Failure/Operator Error (as applicable):
None

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
Feedwater Control system and Pressure Control System maintain vessel level and pressure constant, as this is a slow event. The Operators intervene only after the initial transient is over and guide the plant to a stable condition.

Event Category & Acceptance Criteria:

This is an Anticipated Operational Occurrence – an expected operational transient of moderate frequency (recirculation pump speed increase) with no other equipment failures or Operator errors.

Fuel SAFDLs shall not be exceeded.

RPV Pressure shall remain within ASME Upset limits.

Methods

a) Calculation Tools & Computer Codes:
Primary Code – ISCOR (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-3 (Table 15.0-3).
Cycle-specific fuel bundle designs (FRED form-see Section 15.0.7).

c) Key Assumptions:
The rate of change in recirculation pump speed/core flow is slow enough that the reactor is in a quasi-steady state condition throughout the event.
MG set scoop tube positioner set at 102.5% speed (nominally 1710 RPM). Note: this evaluation is for EPU and was not re-performed for ICF, which permits core flow up to 105% of rated flow.

Results

a) Comparison to Acceptance Criteria:
Fuel Performance: Fuel thermal limits (MCPR, LHGR (MOP and TOP) and APLHGR) all remain within their respective acceptance limits. The slow recirculation flow increase event is the basis for the ARTS $MCPR_f$ limits. Because ARTS MCPRf curves exist for 107% of rated core flow, ICF implementation does not impact these results.

Reactor Pressure Boundary Performance: Reactor vessel pressure remains well below the 1375 psig ASME acceptance limit.

Containment Performance: The containment is not challenged by this event and is not explicitly analyzed.

b) Known Sensitivities:

Small changes in the initial power/flow point have a negligible impact on the results.

c) Uncertainties in Results:

The plant performance is analyzed to 95%/95% confidence levels using GEMINI methods.

Conclusion

a) Statement of Acceptability:
This event meets all the fission product barrier performance criteria for an Anticipated Operational Occurrence.

b) Known Conservatisms/Margins:
The resulting $MCPR_f$ multiplier is based upon an MCPR Safety Limit of 1.08. End-of-cycle core conditions are assumed.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
This is a Limiting Event, but is not evaluated as part of the cycle-specific reload analysis. It was included in the analyses for Extended Power Uprate, as required by the NRC to confirm the $MCPR_f$ multiplier.

15.1.6 TRANSIENTS RESULTING IN AN INCREASE IN REACTOR COOLANT INVENTORY

There are no events in this category analyzed in the original FSAR with the possible exception of the feedwater controller failure, which is discussed in Section 15.1.1.1.

15.1.7 TRANSIENTS RESULTING IN A REACTOR VESSEL COOLANT INVENTORY DECREASE

Transients that result directly in a decrease of reactor vessel coolant inventory are those that either restrict the normal flow of fluid into the vessel, increase the removal of fluid from the vessel, or are the result of the inadvertent opening of a relief or safety valvereactor coolant pressure boundary. Events identified in this category are the following:

1. Pressure regulator failure.
2. Inadvertent opening of a relief or safety valve.
3. Loss of feedwater flow.
4. Trip of One Feedwater Pump.

15.1.7.1 Pressure Regulator Failure - Open

NOTE: the information in the following section is historical in nature and was not updated as part of the Extended Power Uprate Project. It is being presented here to show basic plant response and parametric trends only.

Description of Event

a) Initiator:
A failure of either the primary or back-up pressure regulator occurs at rated conditions, sending a signal to the turbine control and turbine bypass valves to open to the maximum combined flow limit setpoint.

b) Sequence of Events (NOT a time line):

Short term: Either the primary or backup pressure regulator fails to the full open position. This causes the turbine control valves to open to full flow and turbine bypass valve to fully open, because the maximum combined flow limiter setpoint in the EHC system is set at ≥ 125% of rated steamflow. This sudden increase in steamflow causes the vessel pressure to drop. The decrease in pressure causes a sudden swell in water level due to increased voiding in the core. Thus, reactor power initially goes down. The level swell is large enough to reach the high level trip setpoint (Level 8), causing a turbine trip and feedwater pump trip. The turbine trip will initiate the RPS trip (scram) and EOC-RPT trip of the recirculation pumps upon turbine stop valve closure. However, the pressure regulator failure causes the bypass valves to be fully open, so the resulting pressure increase from the turbine trip is mild. In addition, the turbine trip was initiated from less than the initial power level, which also reduces the impact of the turbine trip. Thus, only one or two SRVs are needed to open to regulate the vessel pressure. The turbine inlet pressure eventually drops below the low pressure setpoint and the MSIVs close on low steamline pressure to preclude an unacceptable cooldown rate on the reactor pressure vessel. This essentially terminates the event.

Long-term response (beyond the explicit analyzed period): Low-low set logic cycles the SRVs to control reactor pressure. Feedwater and Condensate pumps eventually trip, as condenser inventory is depleted. HPCI and/or RCIC initiate to maintain reactor vessel level, as needed. Operators take manual control and guide the plant to a cold shutdown condition using normal shutdown procedures.

c) Single Failure/Operator Error (as applicable):
None

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
Short term: Turbine and Feedwater pump trip on high vessel level – Level 8 RPS trip (TSV closure), Control Rod Scram, MSIV closure on low steamline pressure (< 850 psig in RUN – nominal), SRV open and close. Long-term: Low-low set logic trip, FW/Condensate pumps trip, HPCI/RCIC initiations (low-low RPV level). The Operators intervene only after the initial transient is over and guide the plant to a stable condition.

Event Category & Acceptance Criteria:

This is an Anticipated Operational Occurrence – an expected operational transient of moderate frequency (pressure regulator failure), with no other equipment failures or Operator errors.

Fuel SAFDLs shall not be exceeded.

RPV Pressure shall remain within ASME Upset limits.

Methods

a) Calculation Tools & Computer Codes:
Primary Code – ODYN, using GEMINI methods. (Note: because this analysis is historical, an earlier NRC-approved version of the code was used).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-3 (See Table 15.0-3)
EHC Maximum Combined Flow Limiter setting of 130% of rated steamflow was used in the actual analysis.

c) Key Assumptions:
Plant is initially at rated thermal power and core flow.
MSIVs close in a linear ramp over 3.0 seconds.

Results

a) Comparison to Acceptance Criteria:

Fuel Performance: Fuel thermal limits (MCPR, LHGR (MOP and TOP) and APLHGR) all remain within their respective acceptance limits. Because of the initial level swell (void increase) and direct scram, there is no power increase above the initial value as a result of this event.

Reactor Pressure Boundary Performance: Reactor vessel pressure remains well below the 1375 psig ASME acceptance limit. The pressure increase is much less severe than other pressurization events, especially those with loss of bypass capacity. Also, the MSIV closure on low steamline pressure precludes an unexceptable cooldown rate on the pressure vessel.

Containment Performance: The containment is not challenged by this event and is not explicitly analyzed.

b) Known Sensitivities:

The initial depressurization rate of the vessel (MCFL setting) sets the amount of level swell experienced.

Sensitivity studies were performed as part of the resolution of GE SIL 502 (Ref. 15.0-58) that demonstrated that the event is not very sensitive to MSIV stroke time (10 second versus 5 seconds) and only mildly sensitive to MSIV isolation pressure setpoint (800 psig versus 850 psig).

c) Uncertainties in Results:

The plant performance is analyzed to 95%/95% confidence levels using GEMINI methods.

Conclusion

a) Statement of Acceptability:
This event meets all the fission product barrier performance criteria for an Anticipated Operational Occurrence.

b) Known Conservatisms/Margins:
End-of-cycle core conditions are assumed.
Conservative control rod scram times are used.
Maximum Technical Specification MSIV closure time.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):

This is considered to be a Non-Limiting Event and is not re-analyzed as part of the reload analysis for each operating cycle.

15.1.7.2 Inadvertent Opening Of A Safety/Relief Valve

NOTE: the information in the following section is historical in nature and was not updated as part of the Extended Power Uprate Project. It is being presented here to show basic plant response and parametric trends only.

Description of Event

a) Initiator:
A malfunction occurs that cause one Safety/Relief Valve (S/RV) to open while the reactor is operating at rated power/flow conditions.

b) Sequence of Events (NOT a time line):
Short term: A malfunction causes one S/RV to open with the reactor at rated power/core flow conditions. There is a sudden, short increase in steamflow, which causes the vessel pressure to drop. The decrease in pressure causes a sudden swell in water level due to increased voiding in the core. Thus, reactor power initially goes down. The level swell is small and does not reach the high level trip setpoint (Level 8). The pressure regulator quickly compensates for the drop in reactor pressure by closing down on the turbine control valves. The reactor stabilizes at new steady, state conditions. This is a very mild transient on the fuel and vessel.

Longterm (beyond the explicit analyzed period): The Operators take control of the plant and attempt to close the open S/RV. If the valve can not be closed, the reactor is brought to a cold shutdown condition using normal operating procedures. Containment cooling is initiated to handle the steam flow to the suppression pool from the open S/RV.

c) Single Failure/Operator Error (as applicable):
None

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
No trips or other actuations occur during this event.
The Operators intervene only after the initial transient is over and guide the plant to a stable condition.

Event Category & Acceptance Criteria:

This is an Anticipated Operational Occurrence – an expected operational transient of moderate frequency (open S/RV), with no other equipment failures or Operator errors.

Fuel SAFDLs shall not be exceeded.

RPV Pressure shall remain within ASME Upset limits.

Methods

a) Calculation Tools & Computer Codes:
Primary Code – ODYN, using GEMINI methods. (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-3 (See Table 15.0-3)

c) Key Assumptions:
Plant is initially at rated thermal power and core flow.
Initial vessel level swell does not reach the Level 8 trip point.

Results

a) Comparison to Acceptance Criteria:
Fuel Performance: Fuel thermal limits (MCPR, LHGR (MOP and TOP) and APLHGR) all remain within their respective acceptance limits. Because of the initial level swell (void increase), there is no power increase above the initial value as a result of this event.

Reactor Pressure Boundary Performance: Reactor vessel pressure remains well below the 1375 psig ASME acceptance limit. The reactor pressure decreases during this event.

Containment Performance: The containment is not challenged by this event and is not explicitly analyzed.

b) Known Sensitivities:

The initial depressurization rate of the vessel sets the amount of level swell experienced.

c) Uncertainties in Results:

The amount of level swell is the largest uncertainty and would have the most impact on the results, especially if the Level 8 trips were reached.

Conclusion

a) Statement of Acceptability:
This event meets all the fission product barrier performance criteria for an Anticipated Operational Occurrence.

b) Known Conservatisms/Margins:
None

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
This is considered to be a Non-Limiting Event and is not re-analyzed as part of the reload analysis for each operating cycle.

15.1.7.3 Loss Of Feedwater Flow

2016-016 A loss of feed water 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.

Feedwater control system failures or feedwater pump trips can lead to partial or complete loss of feedwater flow. An interlock with the recirculation flow control system reduces the recirculation pumps to about 20% rated speed. The decrease in core flow moderates the decrease in actual reactor vessel water level. Startup of the RCIC and HPCI systems occur when sensed level reaches Level 2 to provide makeup flow to the vessel. If vessel level reaches the Level 1 trip, MSIVs will close to minimize loss of coolant.

Either the RCIC or HPCI system can maintain adequate water level for initial core cooling and to restore and maintain water level. Loss of feed water is a design basis for the RCIC system. The HPCI system injection capability exceeds that of RCIC. The HPCI system, therefore, acts as a backup and can be considered redundant to RCIC for this transient, since it bounds the RCIC system injection rate.

The following analysis is provided in response to NUREG-0737 (TMI Action Items) action to evaluate anticipated transients with worst single failure.

Description of Event

a) Initiator:
A failure in the feedwater control system (or other cause) trips both Feedwater pumps with the reactor at rated thermal power/core flow conditions.

b) Sequence of Events (NOT a time line):
Both Feedwater pumps trip and begin to coastdown. Reactor vessel level begins to drop with the loss of feedwater. Recirculation pumps begin to runback to minimum speed upon the loss of both feedwater pumps after 15 seconds. The recirculation pump runback helps initially moderate the level drop. A reactor scram occurs when level reaches Level 3. Level continues to decrease to Level 2 and RCIC is initiated and starts to inject in about 30 seconds. Water level inside the vessel will begin to recover when the injected flow exceeds the steamflow from vessel (decay heat). If water level in the downcomer region doesn't recover fast enough, the level may reach the Level 1 trips (Low Pressure ECCS, ADS and Group I isolation - MSIV closure). Low Pressure ECCS will not actually inject, as the reactor pressure remains high. The Operators would monitor recovering level and inhibit ADS actuation before the 2 minute timer expires. The MSIV closure would initially collapse the water level, but there is sufficient margin to maintain ample core coverage. The S/RVs would cycle (Low-Low Set) to maintain pressure after the vessel isolation. The Operators would take over and guide the plant to a stable condition and eventually to cold shutdown.

c) Single Failure/Operator Error (as applicable):
The HPCI system is assumed to not operate.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
Recirculation pump runback to minimum speed, RPS trip on low level (Level 3) , Control Rod Drive (Scram), RCIC initiation at low-low level (Level 2).
If water level reaches low-low-low (Level 1), CS and LPCI initiation, ADS timer start, and Group I isolation (MSIV closure).

<u>Operators will inhibit ADS actuation if water level reaches the Level 1 trip</u>

<u>Event Category & Acceptance Criteria:</u>

This is an Abnormal Operating Transient – an expected operational transient of moderate frequency (loss of feedwater flow), but with the additional single failure (HPCI fails to operate).

Because this is a unique analysis, to satisfy NUREG-0737, Item II.K.3.44, the acceptance safety criterion is that the RCIC system is able to maintain reactor vessel level above the Top of Active Fuel (TAF).

There is a second, operational, acceptance criterion of Level 1 trip avoidance.

Methods

a) Calculation Tools & Computer Codes:

Primary Code – SAFER (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-4 (see Table 15.0-4)
OPL-3 (See Table 15.0-3)
Condensate Store Tank water temperature of 125°F.

c) Key Assumptions:
Initial water level is the normal operating water level.
The feedwater flow linearly ramps to zero in 5 seconds.
Recirculation pumps trip, with a 5 second coastdown, at Level 3 (Scram) to simulate the actual pump runback to minimum speed.

Results

a) Comparison to Acceptance Criteria:
For the safety criterion, the minimum water level inside the core shroud (i.e., above the core) remains several feet above the Top of Active Fuel.
However, there is some probability that the level outside the shroud (i.e., indicated level for vessel instrumentation) will not remain above the Level 1 trip point. Thus, conservatively, we assume that the operational acceptance criterion is not met. This is found to be acceptable, given the likelihood of the event occurring, plus, taking credit for Operator actions, would minimize the impact of not meeting the Level 1 criterion.

b) Known Sensitivities:

The results of this event are sensitive to the decay heat (initial power level) and the initial water level (vessel inventory), which is influenced by the steam dryer pressure drop.

c) Uncertainties in Results:

The calculated water level is adjusted downward by 1 foot to account for known biases between the code calculation and actual plant tests.
Instrument uncertainties dictate whether water level reaches Level 1 (i.e., nominal trip setpoint (NTSP) versus Spurious Trip Avoidance).

Conclusion

a) Statement of Acceptability:
The safety criterion is satisfied. As discussed above, the operational criterion may not be satisfied.

b) Known Conservatisms/Margins:
A 2% overpower allowance is added (initial power is 102% of rated).
RCIC flowrate is assumed to only be 98% of rated.
No credit for CRD flow to the vessel.
Decay Heat based upon ANS 5.1 (1979) + 10%.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
This is a unique event, required by NUREG-0737 (TMI Action Items) and is not part of the normal reload transient analysis. It was re-analyzed as part of Extended Power Uprate.

15.1.7.4 – Trip Of One Feedwater Water Pump

a) Initiator:
A failure in the feedwater control system (or other cause) trips one Feedwater pump, with the reactor at rated thermal power/core flow conditions.

b) Sequence of Events (NOT a time line):
One Feedwater pump trips and begin to coastdown. Reactor vessel level begins to drop with the loss of feedwater. Recirculation pumps begin to runback to 45% speed upon the loss of one feedwater pump with vessel level at Level 4 (low alarm). The recirculation pump runback helps initially moderate the level drop. Feedwater level control opens the Feedwater Regulating Valves (FRV) to attempt to compensate for the lowering level. Water level in the downcomer region doesn't recover fast enough and a reactor scram occurs when level reaches the Level 3 trip point. The scram also contributes to the level decrease due to void collapse. The level most-likely will reach the Level 2 trips (HPCI and RCIC initiation) and they will start to inject in about 30 seconds. Water level inside the vessel begins to recover with this additional injected flow. The Operators would take over and guide the plant to a stable condition and eventually to cold shutdown, using normal plant procedures.

c) Single Failure/Operator Error (as applicable):
<u>None</u>

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
Recirculation pump runback to 45% speed (only one Feedpump running), RPS trip on low level (Level 3), Control Rod Drive (Scram), HPCI and

RCIC initiation at low-low level (Level 2).

Event Category & Acceptance Criteria:

This is an Anticipated Operational Transient – an expected operational transient of moderate frequency (single Feed pump trip), with no additional single failures or Operator errors.

Because this is a unique analysis, performed as part of the Extended Power Uprate program, there are only operational, acceptance criteria applied. They are based upon trip avoidance – both Level 3 (Scram) and Level 2 (HPCI and RCIC initiation).

Methods

a) Calculation Tools & Computer Codes:
Primary Code – ODYN (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-3 (See Table 15.0-3)
Feedwater Controller settings for a lead time constant of 0.5 and the lag time constant is 5.0, with proportional gain at 110% and Integral gain at 0.2 resets/min.
Recirculation System controller settings: runback rate of 12%/sec, with proportional gain at 333% and Integral gain at 10 resets/min.

c) Key Assumptions:
Power and core flow are at rated conditions (100%).
Initial water level is the normal operating water level.
The tripped Feed pump linearly ramps to zero flow in 3 seconds.

Results

a) Comparison to Acceptance Criteria:

For the first operational criterion, the water level decreases to well below the Level 3 trip point and the Scram is not avoided.

For the second criterion, there is some probability that the level will remain above the Level 2 nominal trip setpoint (NTSP), but the Spurious Trip Avoidance (STA) setting is exceeded by several inches. Thus, conservatively, we assume that this operational acceptance criterion is not met. This is found to be acceptable taking credit for Operator actions, would minimize the impact of not meeting the Level 2 criterion.

b) Known Sensitivities:

As part of this study, a sensitivity case was run, increasing the recirculation pump runback to 35% to determine if enough margin to Level 2 could be obtained to provide sufficient confidence that the trip could be avoided. While margin to the Level 2 NTSP was gained, it was not sufficient to avoid the STA point. Noteworthy is the fact that the settings on the Feedwater controllers regulating the FRVs have little impact on the event results.

The change in initial power level due to EPU causes the results to be slightly more severe than at pre-EPU conditions. However, the acceptance criterion for the pre-EPU case was also not met.

c) Uncertainties in Results:

Instrument uncertainties dictate whether water level reaches Level 2 (i.e., nominal trip setpoint (NTSP) versus Spurious Trip Avoidance).

Conclusion

a) Statement of Acceptability:
As discussed above, the operational criterion for Level 3 (Scram) avoidance is not met and the Level 2 avoidance (HPCI and RCIC initiation) may not be satisfied. However, the Operators' ability to manage the event is not unacceptably compromised.

b) Known Conservatisms/Margins:
The setpoint methodology has inherent conservatism in it as the results provide for 95% probability at 95% confidence statistical limits.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):

This is a unique event, which was re-analyzed as part of Extended Power Uprate and is not part of the normal reload transient analysis.

15.2 ACCIDENTS

This Section of the UFSAR contains the event descriptions, methods of analysis, assumptions, and analytical results of that subset of plant events classified as Accidents (See Section 15.0.2). The plant response to each Accident will be discussed in terms of the impact on the various fission product barriers and off-site dose consequences. Because the methods used, and assumptions made, in the analyses are specifically adjusted to provide conservative results for the specific fission product barrier being challenged, it is recognized that within each Accident, there may not be full coherence between the various evaluations performed. For example, different values (correlations) for the decay heat following plant shutdown may be used in the Loss-of-Coolant Accident (LOCA) evaluation of the fuel and that used in the analysis of the primary containment response to the LOCA. Consequently, the evaluation of the response of the Reactor (fuel and vessel), Containment (Drywell, Torus/Suppression Pool, Reactor Building) and Radiological Consequences (Offsite, Control Room, Technical Support Center) will be discussed individually within each Accident discussion.

15.2.1 – LOSS-OF-COOLANT ACCIDENTS

Loss-of-Coolant Accidents (LOCA) are those postulated events that are a result of a non-mechanistic rupture of any pipe attached to the Reactor Pressure Vessel (RPV), up to, and including the double-ended guillotine break of the largest pipe in the Reactor Recirculation System, as required by 10 CFR 50.46 and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 35. The latter, being the most challenging to the plant systems and fission product barriers, is generally referred to as the Design Basis Accident (DBA) LOCA (DBA-LOCA).

The specific locations and break sizes analyzed are as follows:

- Reactor Recirculation Piping (Suction and Discharge)
  - Large Breaks
  - Intermediate Breaks
  - Small Breaks*
- Core Spray Line Break
- Feedwater Line Break
- Main Steamline Break – Inside Primary Containment
- Main Steamline Break – Outside Primary Containment

* Note: there is a special category of small break LOCA, the break of an Instrument Line attached to the RPV, which is analyzed separately (See Section 15.2.2).

15.2.1.1 – Reactor Recirculation Pipe Breaks

Because the Reactor Recirculation System piping is both the largest in size and attached at the lowest elevation to the RPV, i.e., below the top of the fuel, this category of piping represents the biggest challenge to the Emergency Core Cooling Systems (ECCS) and fission product barriers. Hence, the LOCA analyses focus on breaks in this piping.

A) Large Breaks

In the spectrum of break sizes, Large breaks are those typically in cross-sectional area from 1.0 ft^2 up to the DBA case of 2.523 ft^2, which includes the recirculation pump suction pipe, jet pump nozzle (pump discharge side) and Reactor Bottom Head drain line flow areas.

<u>1) Reactor Response</u>

This discussion centers on the response of the fuel and the RPV to the Large Break LOCA event.

Note: there is a specific set of thermal-hydraulic calculations of various events, including LOCAs, performed to derive loads for the stress analysis of the vessel internals (Section 3.9.5.2), this evaluation is discussed in Section 15.3.5.

<u>Description of Event</u>

a) Initiator: This event is initiated by an instantaneous, non-mechanistic, double-ended, guillotine break of the reactor recirculation pump suction pipe at the nozzle on the RPV.

b) Sequence of Events: Coincident with the initiation of the break, a complete Loss-of-Offsite Power (LOOP) is assumed to occur, in accordance with GDC 35. Reactor coolant begins to exit the vessel rapidly into the Drywell at the critical mass flux and reactor vessel water level begins to drop, as does the reactor pressure. The reactor is assumed to scram immediately. The Emergency Diesel Generators (EDGs) start on the LOOP condition and all loads are stripped off the Essential AC busses. The non-Essential busses are lost, leading to a loss of Feedwater and a Reactor Recirculation Pump coastdown. As the RPV level reaches the various level setpoints, ECCS systems are actuated (a conservative assumption to delay injection), Vessel isolation signals are generated (Containment isolations are discussed in the Containment Response below), and Low Pressure Coolant Injection (LPCI) loop select logic actuates to determine which recirculation loop is broken and closes the recirculation pump discharge valve in the non-broken loop (the pump discharge bypass valve is conservatively assumed to remain open). If the plant had previously been operating in single loop recirculation mode, loop select logic would trip the running recirculation pump and effect a short time delay to allow it to coastdown prior to its selecting the “broken” recirculation loop (See Chapter 7.3.1.1.2.4 for a complete explanation of LPCI loop select logic). The reactor level continues to drop and uncovers the fuel, which begins to heat up. High Pressure Coolant Injection (HPCI) starts, but before it can come up to speed and inject to the RPV, reactor pressure has decreased below its operational range and it isolates. Once the EDGs are up to speed, its output breaker closes in on the Essential AC busses, and the low pressure ECCS pumps (and other essential loads) are sequenced onto the busses, the pumps start and their minimum flow bypass valves open. The Automatic Depressurization System (ADS) actuation logic initiates on lowering RPV level and ECCS pumps running, but because the RPV depressurizes through the break prior to the ADS 2 minute time delay expiring, the valves never actually open. Once the reactor pressure decreases to their respective permissive setpoints, the injection valves for Core Spray and LPCI (based upon the "chosen" loop by loop select logic), open and allow injection to begin to the RPV. The injection refills the lower vessel plenum area and the water level inside the core shroud rises and terminates the fuel heatup. Water level is maintained at the top of the jet pumps and long-term recovery mode is entered. The analysis of the fuel and RPV response is terminated at this point.

c) Single Failure/Operator Error: The foregoing narrative did not include a discussion of the single failure of an active component, which is required by GDC 35. Depending upon the methodology used and assumptions made in doing the analysis, different single failures can lead to the limiting response on the fuel. This is further discussed in the Methodology section below. The OPL-5 form (Table 15.0-5) details the various single failures assumed in the analysis. Historically, either the loss of Division II of 125VDC or the failure of the LPCI injection valve to open is the limiting single failure for the DAEC Large Break LOCA analysis.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures): No Operator Actions are assumed in this evaluation. Reactor scrams (high Drywell pressure), EDG starts (either LOOP (undervoltage) or high Drywell pressure) and loads ("dead" buss permissive), Feedwater and Recirculation pumps coastdown on LOOP condition, HPCI actuates on low-low RPV level (conservatively ignore high Drywell pressure), LPCI loop select logic actuates on low-low RPV level and low RPV pressure and chooses the "broken" loop and closes the recirculation discharge valve in the non-broken loop, based upon recirculation loop differential pressure (assuming not in single loop operation), ADS initiation on low and low-low-low RPV levels, with confirmation signal on ECCS pump running, which starts 2 minute time delay, Core Spray and LPCI pumps start signal on low-low-low RPV level and timers sequence the pumps onto the AC busses and the LPCI minimum flow bypass valves open on high pump discharge pressure and close on high flow (dP), the normally-open CS minimum flow valves close on high flow (dP), CS and LPCI injection valves open on low RPV pressure permissive signals.

<u>Event Category & Acceptance Criteria</u>

This is a Design Basis Accident (DBA), presenting the most challenge to the ECCS capability.

Fuel shall remain within 10 CFR 50.46 limits as follows:

Peak Cladding Temperature (PCT) shall remain ≤ 2200 °F;
Maximum Cladding Oxidation shall not exceed 17% of the total cladding thickness;
Maximum Hydrogen Generation shall not exceed 1% metal-water reaction;
Coolable Geometry shall be maintained;
Long-term Cooling shall be ensured to remove decay heat.

There are no acceptance criteria for the RPV, as this event assumes a breach of the RPV as the initiating event.

Methods

2012-020 a) Calculation Tools & Computer Codes: SAFER, GESTR-LOCA, PRIME-LOCA, LAMB and TASC (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

As part of General Electric's methodology for complying with 10 CFR 50.46 and Appendix K, a statistical approach is used, which relies on the combination of calculations using both "nominal" inputs and assumptions for ECCS performance, fuel parameters, decay heat model, etc. and those meeting the strict requirements
2012-020 of "Appendix K." See References 15.0-4, 44, and 62 for a complete discussion.

2012-020 2015-004 2017-001 The ECCS-LOCA GNF2 analysis is based on the SAFER/PRIME LOCA methodology (Reference 62).

b) Inputs: The primary set of plant inputs used in the LOCA analysis is provided on the OPL-4 and OPL-5 forms (Tables 15.0-4 and 15.0-5).

c) Key Assumptions:
There is a simultaneous LOOP with the LOCA condition.
There is a single active failure. Both a loss of Division II of 125 VDC or LPCI Injection Valve Failure are evaluated to determine which failure gives the limiting response on the fuel.

In addition, to the assumed single failure above, we also assume that the Recirc. discharge bypass valve in the "selected loop" fails to close. This is due to Environmental Qualification issues (See Section 6.3.2.2.4 and Reference 15.0-4). The reactor scrams immediately, ignoring Control Rod scram time. Only Decay and Sensible Heat are considered.

ECCS initiation is on RPV level. The Drywell Pressure signal is ignored. Limiting assumptions on fuel exposure, peaking factors, power shape, initial thermal limits are made.

ECCS Injection water is assumed to be at 88 Btu/lb_m (120 °F).

Note: for the special case of single recirculation loop operation, the initial core thermal power/core flow is assumed to be 1277 MWth (66.8% rated power) and 25.96 Mlb/hr (53% rated flow). This corresponds to the maximum power on the MELLLA boundary, assuming a maximum core flow corresponding to 102.5% recirculation pump speed in the operating pump. Because this analysis was not re-performed as part of the Increased Core Flow analysis, administrative limits are in place to assure this initial condition remains valid.

Results

Figures 15.2-1 thru 3 show the vessel (water level and pressure) and fuel (PCT for the high power fuel bundle) response to the DBA. NOTE: These are "typical" response curves and are not DAEC-specific. For DAEC-specific curves, see
2012-020 Reference 15.0-4, 44, and 62.

Nominal Case

Table 15.2-1 gives the PCT results for each fuel type for the DAEC.

Appendix K Case

Table 15.2-1 gives the PCT results for each fuel type for the DAEC.

a) Conformance to Acceptance Criteria:

Using the GE methodology for statistically combining the nominal and Appendix K results to form the "Licensing Basis PCT (LBPCT)" value, which is used to demonstrate conformance to the 2200 °F PCT limit of 10 CFR 50.46, we get the Table 15.2-1 results for each fuel type. As we can see, the resulting LBPCTs are well below the limit. In addition, we see from the Table that the local oxidation fraction and metal-water reaction limits are also met. Thus, a coolable geometry is maintained, as well.

To show compliance to the long-term core cooling criteria, we need to demonstrate that either: 1) the core is fully reflooded to the Top of Active Fuel (TAF); OR 2) that we are reflooded to a level equal to the top of the jet pumps AND we have at least one Core Spray pump available for cooling. This is to ensure that sufficient cooling is available either by total submergence or by the combination of partial submergence and spray cooling. In the short-term response, there is enough steam cooling from the partially submerged fuel to cool the upper part of the fuel bundle without reliance on CS. However, as the decay heat dissipates, there may not be enough steam cooling effects to maintain sufficient cooling in the upper part of the bundle to preclude significant cladding oxidation

over the long-term, especially if the fuel axial power shape prior to the accident was heavily “top-peaked.” Hence, we must rely upon CS to meet the acceptance criteria for long-term cooling requirements of §50.46.
For the case of single recirculation loop operation, a special multiplier (<1.0) is determined and applied to the operating limit on Linear Heat Generation Rate (LHGR) and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) to ensure that the single loop results (PCT) remain bounded by the two recirculation loop operating cases. This multiplier is found in the COLR for each operating cycle.

b) Sensitivities:

Sensitivity cases have been done (References 15.0-50 and 51) that show how the PCT varies with reflood time, which is also proportional to the total ECCS injection flow rate. The longer the reflood time (i.e., the lower the total injection flowrate), the higher the resulting PCT. These studies also conclude that neither HPCI nor ADS performance is consequential in large break LOCA cases, as the vessel depressurizes itself prior to either system becoming effective in responding to the accident.

As part of implementation of Increased Core Flow (ICF) (Ref. 15.0-60), the reactor response to the DBA-LOCA was evaluated at an initial condition of 105% of rated core flow. Because the downcomer subcooling is reduced over that at 100% rated core flow (i.e., the water is warmer), the resulting break flowrate during the LOCA blowdown phase is less than the 100% rated flow case presented above and consequently, core uncovery occurs later. Later core uncovery results in a lower PCT. Hence, the rated core flow case is bounding over the ICF case.

c) Uncertainties in Results:

2012-020 The statistical approach described in Ref. 15.0-4, 44, and 62 is used to address the various uncertainties in both the plant specific inputs, as well as the computer model and its internal correlations used to calculate the results. The result of this calculation is the “Upper Bound PCT (UBPCT),” which represents the 95^{th} percentile of the calculation distribution considering the uncertainties. The LBPCT must always bound the UBPCT, i.e., give a more-conservative (higher) PCT result. The UBPCT results for each fuel type are shown in Table 15.2-1. As can be seen, the LBPCTs are higher than their corresponding UBCPTs for each fuel type. Thus, this licensing criterion for the GE methodology is met.

2) Containment Response

The response of the Primary Containment to the DBA-LOCA is broken up into the "short-term" response and the "long-term" response. In addition, the Primary Containment response is also further sub-divided into analyses for calculation of peak pressure and temperature response, for containment parameters used in the ECCS pump net positive suction head (NPSH) calculations and for containment parameters used in the Mark I containment loads evaluation. Different methods, inputs and assumptions are used to ensure conservative results for the various cases.

The evaluation of the Secondary Containment (Reactor Building) is considered as part of the Radiological response to the DBA-LOCA below.

Containment Evaluation for Peak Pressure and Temperature Response

Description of Event

a) Initiator: The same as for the Reactor evaluation above.

b) Sequence of Events:

Event Sequence for Short-Term Response

1. The plant is operating at 102% of 120% ORTP (i.e., 1950 MWt) when a double-ended Recirculation suction line break occurs.
2. Drywell pressurizes rapidly, and downcomers clear, driving steam and non-condensables to the suppression pool via the vents/downcomers.
3. Steam condenses in the suppression pool, while non-condensables exit the suppression pool and enter the wetwell airspace.
4. Peak drywell pressure and peak drywell-to-wetwell differential pressure occur.

5. The Torus-to-Drywell vacuum breakers open and the non-condensables return to the Drywell and equilibrate the pressure between the Torus airspace and the Drywell.

Event Sequence for Long-Term Response (UFSAR Case 4)

1. The plant is operating at 102% of 120% ORTP (i.e., 1950 MWt) when a double-ended recirculation suction line break occurs. There is also a concurrent loss of offsite power and only minimum diesel power is available. Reactor scrams.
2. For the first 10 minutes (600 seconds) following the accident, two LPCI pumps (in one RHR loop) at a flow rate of 4800 gpm/pump and one CS pump at 3100 gpm inject into the vessel.
3. At 10 minutes (600 seconds), operator activates the RHR heat exchanger in the operating RHR loop. One RHR pump at 4800 gpm is re-aligned so

that flow goes through the heat exchanger before injecting into the vessel. The other RHR pump is shutdown. This configuration is maintained throughout the accident.

4. After 10 minutes (600 seconds), the CS pump is maintained at 3100 gpm.

c) Single Failure/Operator Error:
The same as for the Reactor evaluation above.

d) Key Equipment Response:
Reactor scrams (assumed on High Drywell Pressure), MSIVs close, EDG starts and loads (LOOP), RHR and CS pumps start (High DW pressure), Torus-to-Drywell vacuum breakers open/close, Operators secure one RHR pump and manually load the RHR Service Water (RHRSW) pumps to initiate cooling with the RHR heat exchanger after 10 minutes.

<u>Event Category & Acceptance Criteria</u>

This is a Design Basis Accident (DBA), presenting the most challenge to the Primary Containment's capability.

The Primary Containment response to the DBA-LOCA shall remain within the containment design pressure (56 psig) and temperature (281°F) for the shell. The Primary Containment is designed for 100% humidity.

<u>Methods</u>

a) Calculation Tools & Computer Codes: Short-Term Response: M3CPT code with the HEM break flow model. Long-Term Response: SHEX code with the HEM break flow model. (See Table 15.0-2 for complete listing, code versions and NRC acceptance.)

b) Inputs: The primary set of plant inputs used in the containment analysis is provided on the OPL-4a form (Tables 15.0-6).

c) Key Assumptions:
<u>Key Assumptions for Short-Term Response</u>

a. The power level for the power/flow point analyzed includes an additional 2% power, consistent with Regulatory Guide 1.49.

b. The DBA-LOCA is the instantaneous double-ended guillotine break of the recirculation suction line at the reactor vessel nozzle safe-end to pipe weld. The effective break area is 2.523 ft^2, which includes the bottom head drain line.

c. The shutdown power fractions include metal-water reaction energy and May-Witt decay heat (which includes the fuel relaxation energy).

d. No credit is taken for the passive structural heat sinks.

e. The initial vent submergence and the suppression pool volume correspond to the Technical Specifications (TS) High Water Level (HWL) to maximize the containment pressure response.

f. Initial conditions for drywell pressure, wetwell pressure and suppression pool temperature are based on limiting (e.g., analytical, TS) values.

g. The wetwell airspace is in thermal equilibrium with the suppression pool at all times to maximize the containment pressure and temperature response.

h. Only 6 wetwell-to-drywell vacuum breakers are assumed to be active.

<u>Key Assumptions for Long-Term Response (UFSAR Case 4)</u>

1. The power level for the power/flow point analyzed includes an additional 2% power, consistent with Regulatory Guide 1.49.

2. The shutdown power fractions include fuel relaxation energy, metal-water reaction energy and ANS 5.1 +2sigma decay heat for fuel applicable up to GE14 with 24-month fuel cycle.

3. Concurrent with the postulated LOCA, a loss of offsite power occurs.

4. Only minimum diesel power is available. This results in only one RHR loop with one heat exchanger available for containment cooling, starting at 10 minutes (600 seconds).

5. RHR heat exchanger performance is based on one RHR pump (4800 gpm) and two RHRSW pumps (4080 gpm total).

6. The portion of the feedwater inventory at a temperature higher than the peak suppression pool temperature, after absorbing additional energy from the feedwater piping as it flows toward the vessel, is injected into the vessel. This assumption is used to maximize the suppression pool temperature. This hot portion of the feedwater inventory is transferred to the vessel regardless of the availability considerations of feedwater and condensate pumps.

7. Heat and mass transfer from the suppression pool to the wetwell airspace is determined mechanistically.

8. The DBA-LOCA is the instantaneous double-ended guillotine break of the recirculation suction line at the reactor vessel nozzle safe-end to pipe

weld. The effective break area is 2.523 ft^2, which includes the bottom head drain line.

9. The initial suppression pool water volume corresponds to the TS Low Water Level (LWL) to maximize the suppression pool temperature response.

10. Initial conditions for drywell pressure, wetwell pressure and suppression pool temperature are based on limiting (e.g., analytical, TS) values.

11. Passive heat sinks in the drywell, wetwell airspace and suppression pool are conservatively neglected to maximize the suppression pool temperature. Heat transfer from the primary containment to the reactor building is also conservatively neglected.

12. Drywell fan coolers are inactive.

13. Operating Core Spray and LPCI/RHR pumps have 100% of their motor horsepower rating converted to pump heat which is added either to the Reactor Pressure Vessel (RPV) liquid or suppression pool water. This assumption is used to maximize the suppression pool temperature response.

14. Main Steam Isolation Valves (MSIVs) start closing at 0.5 seconds and close completely at 3.5 seconds.

15. Only 6 wetwell-to-drywell vacuum breakers are assumed to be active.

<u>Results</u>

a) Conformance to Acceptance Criteria:
<u>Short-Term Response</u>
The peak drywell pressure of 45.7 psig is well below the containment design pressure of 56 psig.

The peak drywell gas temperature of 292.9ºF exceeds the drywell shell design temperature of 281ºF, but only for a very short duration (less than twenty seconds). Since the drywell shell is initially at 135ºF and the peak drywell gas temperature is only 12ºF above the drywell shell design temperature, the drywell shell temperature is expected to remain below the design temperature because of its large heat capacity and the short period during which the drywell gas temperature exceeds the design temperature. This is validated by the SBO containment analysis results, which calculated the actual shell heat-up. The SBO results (Section 15.3.2.1) show a difference between the peak drywell shell temperature and the peak drywell gas temperature of more than 23ºF. Therefore, the peak drywell shell temperature remains below the drywell shell design temperature of 281ºF for the short-term DBA-LOCA case.

<u>Long-Term Response (UFSAR Case 4)</u>
The peak wetwell pressure of 36.5 psig is well below the containment design pressure of 56 psig.

The peak wetwell airspace temperature of 236.5°F is well below the containment design temperature of 281°F. The peak wetwell airspace temperature of 236.5°F occurs during the initial blowdown period and is a result of compression effects modeled in SHEX. This peak value is not to be expected during a DBA-LOCA due to the vigorous mixing of the suppression pool water and wetwell airspace during the initial air carryover. The long-term peak wetwell airspace temperature of 215.3°F, which occurs near the time of the peak suppression pool temperature, provides a more accurate measure of the peak wetwell temperature.
The peak suppression pool temperature of 215.3°F is well below the containment design temperature of 281°F.

b) Sensitivities:
<u>Short-term Response:</u>

<u>Extended Power Uprate (EPU)/Maximum Extended Load Line Limit Analysis (MELLLA)</u>

For the EPU containment evaluation, initial calculations were performed using the M3CPT containment model (M3CPT05A) with break flows calculated using the LAMB break flow model (LAMB08) to cover the EPU power/flow map, including MELLLA. These calculations were performed to assess the sensitivity of the short-term DBA-LOCA containment response to the different initial power/flow conditions. The LAMB break flow model was use for these initial calculations since it has more detailed modeling of the vessel and recirculation lines relative to the simple single node M3CPT vessel model. This more detailed modeling in LAMB allowed for a better prediction of trends with changes in vessel conditions (such as subcooling); M3CPT cannot easily model the change in vessel subcooling that occurs with a reactor operating in off-rated conditions. The LAMB break flows for this sensitivity study were calculated using the Moody SLIP critical flow model instead of the HEM critical flow model used in M3CPT. Use of the Moody SLIP model instead of the HEM was considered acceptable for the EPU evaluations because the Power/Flow state points were very close (i.e., only a 1% flow difference between rated flow and MELLLA statepoints) and this was a comparison, sensitivity study only, not the design basis calculation.

The calculations were then performed with these LAMB break flows (mass and enthalpy) input into M3CPT to give the short-term containment temperature and pressure response. These results determined that differences in the calculated response between the rated condition and the off-rated (MELLLA) condition are essentially the same (i.e., within the assumed accuracy of the model). Based on the results using the LAMB break flows, the license basis calculations for the DAEC short-term DBA-LOCA containment analysis were done at the rated power/flow condition, using the M3CPT vessel break flow prediction (HEM) along with the M3CPT containment model, which is the NRC approved method.

Increased Core Flow (ICF)

For the ICF analysis, the LAMB break flows were calculated using the HEM critical flow model instead of the SLIP critical flow model used in the EPU analysis above, as the flow difference is greater for ICF (5%). Comparison of the LAMB (HEM)/M3CPT initial break flow enthalpy for both cases shows the enthalpy for the ICF case is slightly higher (less subcooling) than that of the rated (EPU) case. If this ICF case break flow and enthalpy were used with the M3CPT vessel and containment model, the higher enthalpy (less subcooling) would be expected to result in slower pressurization of the drywell and slightly lower peak drywell pressure than the EPU case. Therefore the results of the EPU analysis are considered to be bounding for ICF.

EPU Long-Term Response with Containment Sprays (UFSAR Case 3)
Same event sequence and key assumptions as UFSAR Case 4 above, except that RHR flow is re-aligned from vessel injection to containment spray when the RHR heat exchanger is activated at 10 minutes (600 seconds).

The peak wetwell pressure of 36.5 psig and peak wetwell airspace temperature of 236.5ºF are the same as the UFSAR Case 4 results, and well below the respective design limits (56 psig and 281ºF). The peak wetwell airspace temperature of 236.5°F occurs during the initial blowdown period and is a result of compression effects modeled in SHEX. This peak value is not to be expected during a DBA-LOCA due to the vigorous mixing of the suppression pool water and wetwell airspace during the initial air carryover. The long-term peak wetwell airspace temperature of 212.7°F, which occurs near the time of the peak suppression pool temperature, provides a more accurate measure of the peak wetwell temperature.
The peak suppression pool temperature of 212.7°F is well below the containment design temperature of 281°F.

2014-006

EPU Long-Term Response with Heavier Feedwater Equipment

New feedwater pumps (RFP) and heaters (#5 FWH, #4 FWH, #3 FWH) have more metal mass than previously evaluated in the long term containment analysis for EPU. Consistent with the assumption that hot feedwater is transferred to the vessel after absorbing additional energy from the feedwater piping (i.e., metal mass), potential increase in peak suppression pool temperature (UFSAR Case 4) is estimated based on additional metal energy (relative to the coldest feedwater being heated) and the mass of water in containment (i.e., initial suppression pool

2014-006 water inventory and hot feedwater added). Estimated increase in peak suppression pool temperature is approximately 1°F, which is insignificant compared to the margin of more than 65°F between calculated peak suppression pool temperature of 215.3°F and the containment design temperature of 281°F.

As shown in UFSAR Figures 5.4-15 Sheet 1 and 5.4-15 Sheet 2, available margin for adequate Core Spray and RHR pump NPSH is also substantial relative to effects from an estimated increase in peak suppression pool temperature of approximately 1°F.

Increased Core Flow (ICF)

As part of implementation of Increased Core Flow (ICF) (Ref. 15.0-60), the containment response to the DBA-LOCA was evaluated at an initial condition of 105% of rated core flow. Long-term heatup of the suppression pool following a DBA-LOCA is governed by the capability of the Residual Heat Removal system to remove decay heat and sensible energy in the vessel and piping. The decay heat depends upon the initial reactor rated power level, which remains unchanged with ICF. The sensible energy for the long-term containment analysis is conservatively based on the saturated temperature at the normal operating reactor pressure, which is also unchanged with ICF. Therefore, the long-term containment response due to EPU is applicable for operation in the ICF domain.

c) Uncertainties in Results:
The use of the 102% power level and conservative values for the decay heat generation (ANS 5.1-1979 +2sigma) and conservative inputs to the calculation all contribute to compensate for any uncertainties in the calculation methodology.

Containment Evaluation for ECCS Pump Net Positive Suction Head (NPSH)

A special case of the above long-term response (UFSAR Case 4) is evaluated with containment sprays to generate the suppression pool temperature and wetwell pressure (using the SHEX code with the HEM break flow model) for input into the ECCS pump net positive suction head (NPSH) evaluations. The inputs and assumptions used have been modified from the above (for UFSAR Case 4) to give conservative results with respect to NPSH.

Event Sequence for Short-Term NPSH Evaluation

This case is analyzed to 600 seconds, which is the time period assumed for no operator actions. After 600 seconds, the event sequence corresponds to that for the long-term NPSH evaluation below.

1. The plant is operating at 102% of 120% ORTP (i.e., 1950 MWt) when a double-ended recirculation suction line break occurs. There is also a

concurrent loss of offsite power and maximum diesel power is available.
Reactor scrams.

2. All 4 LPCI pumps at the maximum runout flow rate of 6500 gpm/pump and both CS pumps at the maximum runout flow rate of 4500 gpm/pump inject into the vessel.

<u>Key Assumptions for Short-Term NPSH Evaluation</u>

The assumptions listed above for UFSAR Case 4 are applicable, with the following exceptions:

1. Minimum initial drywell and wetwell pressures, and maximum initial drywell relative humidity are assumed. This is done to minimize the mass of non-condensables in the containment.

2. Maximum diesel power is available so that all LPCI and CS pumps are available for vessel injection.

3. Passive heat sinks in the drywell and wetwell airspace are modeled.

4. Containment leakage effects are considered.

5. All 7 wetwell-to-drywell vacuum breakers are assumed to be active.

<u>Event Sequence for Long-Term NPSH Evaluation</u>

This case is analyzed to 40,000 seconds, which is beyond the time of peak suppression pool temperature.

1. The plant is operating at 102% of 120% ORTP (i.e., 1950 MWt) when a double-ended recirculation suction line break occurs. There is also a concurrent loss of offsite power and only minimum diesel power is available. Reactor scrams.

2. For the first 10 minutes (600 seconds) following the accident, two LPCI pumps (in one RHR loop) at the maximum runout flow rate of 6500 gpm/pump and one CS pump at the maximum runout flow rate of 4500 gpm inject into the vessel.

3. At 10 minutes (600 seconds), operator activates the RHR heat exchanger in the operating RHR loop. One RHR pump at 4800 gpm is re-aligned so that flow goes through the heat exchanger before discharging to the drywell and wetwell in the form of drywell and wetwell sprays. The other RHR pump is shutdown. This configuration is maintained throughout the accident.

4. After 10 minutes (600 seconds), the CS pump is reduced to 3100 gpm.

Key Assumptions for Long-Term NPSH Evaluation

The assumptions listed above for UFSAR Case 4 are applicable, with the following exceptions:

1. Minimum initial drywell and wetwell pressures, and maximum initial drywell relative humidity are assumed. This is done to minimize the mass of non-condensables in the containment.

2. RHR heat exchanger performance is based on one RHR pump (4800 gpm) and two RHRSW pumps (5200 gpm total).

3. Containment cooling is achieved by operating the RHR loop, with heat exchanger, in the containment spray mode (drywell and wetwell sprays) instead of the vessel injection mode.

4. 95% of 4800 gpm goes to the drywell spray and 5% of 4800 gpm goes to the wetwell spray.

5. Passive heat sinks in the drywell and wetwell airspace are modeled.

6. Containment leakage effects are considered.

7. All 7 wetwell-to-drywell vacuum breakers are assumed to be active.

Containment Results for the NPSH Evaluations

For the short-term NPSH evaluation, the suppression pool temperature at 10 minutes (600 seconds) is 164.3°F, and the corresponding wetwell pressure is 4.4 psig (accounting for the containment leakage effects).

For the long-term NPSH evaluation, the peak suppression pool temperature is 209.2°F, and the corresponding wetwell pressure is 13.3 psig (accounting for the containment leakage effects).

Mark I Containment Loads Evaluation for DBA-LOCA

A special case of the above short-term response is evaluated to generate the key containment parameters (using the M3CPT code with the HEM break flow model) for input into the Mark I containment loads evaluation. The inputs and assumptions used have been modified from the above (short-term response) to give conservative results with respect to containment loads.

The containment loads evaluated include:

1. Vent Thrust
2. Pool Swell
3. Condensation Oscillation (CO)
4. Chugging

The Safety Relief Valve (SRV) loads, including thrust loads on the SRV discharge line and T-quencher, air bubble loads on the submerged pool boundaries and air bubble drag loads on the submerged structures, are evaluated for both initial and subsequent SRV actuations.

Methods

The LOCA dynamic loads are defined generically for Mark I plants as part of the Mark I containment program (References 15.0-53 and 15.0-55) and are discussed in the Mark I Plant Unique Load Definition (PULD; Reference 15.0-52). Applicability to DAEC is described in the Plant Unique Analysis Report (PUAR; Reference 15.0-56). The significant loads occur in the first few seconds for a DBA-LOCA. The M3CPT vessel model, together with the initial conditions given in the PULD, is used to generate the drywell pressurization rate at vent clearing to evaluate the pool swell loads, and the drywell and wetwell pressures and temperatures at the time of vent clearing, the time of peak drywell-wetwell pressure difference, and the time of peak vent mass flow to calculate the vent thrust loads.

1. Vent thrust loads are calculated using the equations documented in the Load Definition Report (LDR) (Reference 15.0-53). Where the vent thrust loads exceed the previously defined values, a conservative approach is taken such that all stresses and stress ratios given in the PUAR are increased by double the percentage of the increase in vent thrust loads. Additionally, it is assumed that the resultant stresses and displacements increase linearly with the vent thrust loads, which is conservative because more than the vent thrust loads influence the controlling load combinations.

2. The pool swell loads are quantified based on the drywell pressurization rate, which is calculated from the M3CPT results.

3. The CO loads are quantified based on the Root Mean Square (RMS) pressure, which is calculated from the M3CPT results.

4. The chugging loads are evaluated by comparing the containment response parameters (M3CPT results) to the Full Scale Test Facility (FSTF) test conditions (Reference 15.0-55).

Key Assumptions

The assumptions listed above for short-term response are applicable, with the following exceptions:

1. Initial conditions for drywell pressure, wetwell pressure and suppression pool temperature are based on nominal values, as specified in the PULD (Reference 15.0-52).

Results

The vent thrust loads are calculated to be up to 5% higher than the plant specific values calculated during the Mark I containment program. The stress and displacement ratios remain less than one after increasing the values documented in the PUAR by 10% (i.e., twice the 5% increase in vent thrust loads). Therefore, the vent thrust loads are acceptable.

The pool swell loads defined in the PULD are based on pool swell tests conducted using DAEC specific parameters in the Mark I containment Quarter Scale Test Facility (QSTF) (Reference 15.0-54). A key input parameter to the QSTF tests used in developing the test conditions is the initial drywell pressurization rate up to the time of vent clearing. The test pressurization rate used for the DAEC QSTF tests is 27.8 psi/sec. The scale factor used to model the DAEC containment parameter is 0.302 (Reference 15.0-54). The corresponding full-scale value for pressurization rate is obtained by dividing the test value of 27.8 psi/sec by the square root of the scale factor. The resulting full-scale test pressurization rate is therefore $27.8/\sqrt{0.302} = 50.6$ psi/sec. The peak pressurization rate calculated from the M3CPT results is 42.9 psi/sec, which is bounded by the DAEC test value used in defining the DAEC pool swell loads.

The CO load defined in the PULD is based on the RMS pressure calculated from the M3CPT simulation of the FSTF tests. The RMS pressure calculated from the M3CPT results is 1.21 RMS psi, which is bounded by the RMS pressure calculated from the M3CPT simulation of the FSTF test (1.60 RMS psi). The chugging loads defined in the PULD are based on the FSTF tests, which were run for a range of blowdown and containment conditions developed to bound all Mark I plants. The steam mass flux, suppression pool temperature and air content from analysis are within the range indicated in Figure 6.2.1-3 of Reference 15.0-55. Therefore, the chugging loads defined in the PULD remain bounding.

b) Sensitivities:

Increased Core Flow (ICF)

The LOCA containment dynamic loads evaluation is based upon the results of the short-term DBA-LOCA analysis. The LOCA dynamic loads considered for ICF operation include pool swell, condensation oscillation, chugging, and vent system thrust loads. Results of the containment dynamic loads evaluation show that all containment loads remain within the limits previously defined for Extended Power Uprate (Ref. 15.0-60).

3) Radiological Response

Description of Event

Design of Emergency Core Cooling Systems to meet 10 CFR 50.46 requirements assures that design basis accidents will not result in significant fuel damage or radiological consequences. During the reactor siting process, 10 CFR 100.11 requires that structures, systems and components used to mitigate radiological consequences shall be analyzed assuming a fission product release based upon a major accident. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of significant quantities of fission products. DAEC has adopted the Accident Source Term in accordance with 10 CFR 50.67. Dose Consequences Analysis for Design Basis Accidents has been performed using the Guidelines of Regulatory Guide 1.183 “Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors.”

a) Initiator:
With regard to radiological consequences, a large-break LOCA is assumed as the design basis case for evaluating the performance of release mitigation systems and the containment and for evaluating the proposed siting of a facility.

b) Sequence of Events:

Time	Event
0	Accident Begins (Coolant Release)
2 min	Gap Fission Product Release Begins
4 min	Automatic Control Room Ventilation to Emergency Mode on High Radiation
5 min	Secondary Containment Negative pressure Drawdown Complete
24 min	Manual Initiation of Standby Liquid Control (SLC) Injection per Severe Accident Guidelines
30 min	Manual Technical Support Center Ventilation to Emergency Mode
32 min	Early-In-Vessel Fission Product Release Begins
110 min	Fission Product Release from Fuel Ends
120 min	SLC Injection Complete
2017-010 24 hr	MSIV Leakage Rate Reduced to 57% of the design leakage rate
30 days	End of Analyzed Accident Scenario

c) Single Failure/Operator Error:
Safety-related systems and components that provide active functions to prevent or mitigate radiological releases are designed to be single-failure proof. Operator manual actions are not credited during the initial 10 minutes of an accident. At the

beginning of the event, a loss of offsite power is assumed which results in the loss of reactor building ventilation that maintains secondary containment at a negative pressure with respect to the outside atmosphere. Radiological ground-level releases from primary containment and ECCS leakage during the ensuing positive pressure period (5 minute delay until SGTS establishes a negative pressure of 0.25 inches of water between secondary containment and the atmosphere) are the most significant contribution to offsite and onsite dose consequences.

d) Key Equipment Response:
The following systems, structures, and components prevent and/or mitigate radiation releases as described below:

Primary Containment reduces radiological releases and dose through retention, deposition, and decay mechanisms.

Primary Containment Isolation reduces radiological releases and dose by closing off release paths and minimizing leakage from primary containment.

Main Steam Isolation reduces radiological releases and dose by closing off release paths through the Main Steam system and minimizing leakage that bypasses Secondary Containment.

The MSIV Leakage Transport Path collects and transports MSIV leakage to the Condenser for holdup and decay.

The Main Condenser provides holdup and decay of radioisotopes from MSIV leakage.

Secondary Containment establishes a secondary barrier and holdup volume for primary containment and ECCS leakage to reduce radiological releases and dose.

The Standby Gas Treatment System (SGTS) provides negative pressure control for the secondary containment to minimize radiological leakage to the environs. The SGTS also provides filtration using activated charcoal filters and HEPA filters to remove radioisotopes to reduce the concentration of radioisotopes released to the offgas stack.

The Offgas Stack Release Path provides an elevated release point to promote atmospheric dispersion and reduce dose from filtered releases from the SGTS.

The Control Room provides an enclosed, shielded control center for operation and monitoring of plant systems during accident mitigation.

Control Building HVAC provides automatic isolation for the control room to reduce radiological dose to operators.

The Standby Filter Units provide positive pressure and filtered ventilation to the control room to minimize inleakage and reduce dose to operators.

The Technical Support Center provides an enclosed, shielded center for personnel providing technical support during accident mitigation and recovery.

The Technical Support Center HVAC provides filtered intake and recirculation cleanup filtration to reduce dose to personnel manning the support center.

The Standby Liquid Control System provides a pH buffering function that will minimize the potential for re-evolution of radioiodine from the suppression pool.

<u>Event Category and Acceptance Criteria</u>

With regard to radiological consequences, a large-break LOCA is assumed as the design basis case for evaluating the performance of release mitigation systems and the containment and for evaluating the proposed siting of a facility.

The acceptance criteria from 10 CFR 50.67 are:

- An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE). An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) TEDE.
- Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

The radiological consequences to personnel in the Technical Support Center are evaluated using the acceptance criterion issued in Section 8.2. of Generic Letter 83-11 “Supplement 1 to NUREG-0737 – Requirements for Emergency Response Capability,” dated December 17, 1982.

- Adequate radiation protection is provided to assure that radiation exposure to any person working in the Technical Support Center would not exceed 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Methods

The radiological consequences of design basis accidents were analyzed using the methods and guidelines of RG 1.183 "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors."

a) Calculation Tools and Computer Codes
The following computer tools and calculational methods were used during analysis of radiological consequences.

Source Term Inventory
The ORIGEN2 code (Reference Table 15.0-2), which is a widely used Oak Ridge National Laboratory code used in the production and decay of radioactive material, was used by General Electric (GE) in the calculation of plant-specific fission product inventories which bound the effect of two year fuel cycles, power operation at 1950 MWt (102% of 1912 MWt), and anticipated fuel designs. ORIGEN2 was controlled within the GE software quality assurance program.

Atmospheric Dispersion
Atmospheric dispersion factors (CHI/Q's) were calculated with the ARCON96 computer code. The ARCON96 code calculates relative concentrations in plumes from nuclear power plants at the control room and Technical Support Center air intakes and accounts for the effects of building wakes. The ARCON96 code was verified and validated in accordance with the DAEC Software Quality Assurance Program.

Atmospheric dispersion factors for offsite dose consequences and for atmospheric fumigation conditions in the vicinity of the offgas stack were calculated with the PAVAN code. The PAVAN code was verified and validated in accordance with the DAEC Software Quality Assurance Program.

Radiological Dose
The MicroShield code, a point kernel integration code, was used for general purpose gamma shielding analysis. This version of the code has been verified and validated in accordance with the DAEC Software Quality Assurance Program.

The RADTRAD computer code, is a radiological consequence analysis code used to estimate radiological source transport, removal, decay, and post-accident doses
2017-010 at plant offsite locations, the control room, and Technical Support Center. The code was verified and validated in accordance with the DAEC Software Quality
2017-010 Assurance Program and/or an approved vendor program.

Suppression Pool pH Response
The calculation methodology was based on the approach outlined in NUREG/CR-5950, "Iodine Evolution and pH Control", Published December 1992.

b) Key Assumptions and Inputs

Source Term Inventory
The following table summarizes inputs used in the development of the accident source term inventory.

24-Month Fuel Cycle Data	**Inputs in ORIGEN2 Calculation of Fission Product Inventory**	**DAEC Cycle**
Fuel Type	GE12	GE14
Total Effective Full Power Days (EFPD)	1500	1132
Core Average Discharge Burnup (GWD/MT)	55	45.03
Initial Bundle Mass of Uranium (kg)	180	179.48
Initial Core Average Enrichment U-235 wt%	4.2	4.15
Core Average Bundle Power (MW/bundle)	4.78	5.19

The fission product inventory was provided by General Electric and consisted of a pre-existing analysis of fission product inventories from higher burnups associated with 24-month cycles. The inventory used a bounding value of exposure of 1500 EFPD that exceeds the DAEC cycle value of 1132 EFPD. The ORIGEN2 extended-burnup library designated BWRUE was used in the calculation. The analysis used an average enrichment value of 4.2 wt% that exceeds the predicted DAEC average enrichment of 4.15 wt%. A core average burnup of 55 GWD/MTU was assumed which exceeds the DAEC predicted average burnup of 45 GWD/MTU. The analysis assumed GE12 fuel and a core average bundle power of 4.78 MW/bundle. DAEC operation at extended power
2017-010 uprate conditions will employ GE14 and/or GNF2 fuel with a core average bundle power of 5.19 MW/bundle. The effect of the different fuel type and higher bundle power are offset by the conservative nature of the other inputs. The analysis provides a bounding fission product inventory.

Atmospheric Dispersion

Atmospheric dispersion coefficients were calculated, for each source/receptor release path, based on site-specific meteorology data collected between January 1997 and December 1999.

Analyzed Release paths:

- Reactor Building (RB) to the Exclusion Area Boundary (EAB)
- RB to the Low Population Zone (LPZ)
- RB to the control room (CR)
- RB to the Technical Support Center (TSC)
- Offgas Stack to the EAB
- Offgas Stack to the LPZ

- Offgas Stack to the CR
- Offgas Stack to the TSC
- Turbine Building Exhaust to the CR
- Main Condenser to the CR
- Main Condenser to the TSC

Design Inputs:

Plant grade elevation	757’
Control room air intake elevation	[redacted]
Control building roof elevation	[redacted]
Reactor building roof elevation	[redacted]
Reactor building exhaust vent elevation	[redacted]
Height of RB above grade	43 meters
Cross sectional area of RB	17313 ft^2 (1609 m^2)
Turbine building roof elevation	[redacted]
Turbine building exhaust elevation	[redacted]
Off gas stack height	100 meters
Off gas stack grade elevation	[redacted]
Lower elevation of wind sensors	33’ (10 meters)
Upper elevation of wind sensors	156’ (47.5 meters)

Radiological Dose

1. All analyses were performed at 1950 MWt (102% of 1912 MWt) in accordance with USNRC Regulatory Guide 1.49

2. The release source term was developed using the 60 isotope subset found in the BWR default inventory described in NUREG/CR-6604, April 1998 and Supplement 1, June 8, 1999 “RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation.”

3. Assumed positive pressure period (PPP) for secondary containment pressure drawdown is 5 minutes.

4. Following the PPP, all primary containment leakage is filtered via the SBGTS.

5. 100% of the LOCA source term releases are assumed to enter the Drywell and/or Torus Suppression Pool

6. A 2%/day primary to secondary containment leakage rate was assumed.

7. ESF systems were assumed to leak at a rate of 1.5 gpm into the Reactor Building - about 15 times higher than plant leakage surveillance procedure limits.

8. Containment and ECCS leakage rates were assumed constant throughout the 30-day duration of the postulated accident.

9. Main Steam pathway leakage of 200 scfh was assumed to leak directly into the environment via the main steam lines (including the inboard MSIV drain line) and the condenser during the first 24 hours of the accident. This value was assumed to decrease to 100 scfh after 24 hours per the guidelines in Appendix A, of RG 1.183.

10. The Main Steam pathway leakage is routed to the Main Condenser by the Leakage Transport pathway.

11. The Condenser is credited for reducing the release of Main Steam pathway leakage through holdup, deposition, and decay mechanisms.

12. Releases from the condenser are assumed as ground releases from approximately the center of the Turbine Building without crediting any further holdup or delays.

13. The CR is automatically isolated and placed in the Emergency ventilation mode upon an air intake radiation monitor isolation signal. For analysis purposes, the accident activity was allowed to enter the CR for up to four (4) minutes at the normal ventilation flow rate of 3150 cfm

14. The TSC is isolated by operator action. A conservative operator isolation at 30 minutes was assumed.

15. Prior to isolation, the TSC normal ventilation flow rate is 900 cfm. Sensitivity analyses of the post-LOCA dose to the TSC were performed at
2017-010 assumed unfiltered TSC inleakage rate of 67.5, 500, and 1000 cfm.

16. Drywell natural deposition was simulated using the 10th percentile data for the Power's natural deposition model in the RADTRAD code.

17. Main steam line pipe deposition was simulated using the RADTRAD
2017-010 code's Brockmann – Bixler pipe deposition model.

18. Activity deposition in the plant condenser was estimated using the DAEC condenser deposition filter efficiency calculated by GENE documented in CAL-M94-010 Revision 1, "Plant-Specific Radiological Dose Calculations".

19. The SBGT Bypass Flow assumed is the maximum 0.1% allowed by plant Technical Specifications.

20. An aerosol removal efficiency of 99% was used for the SBGT, SFU and TSC ventilation system HEPA filters.

21. SBGT charcoal filter removal efficiency for elemental and organic iodine is 99%, consistent with RG 1.52 for a 6 inch charcoal bed with humidity control.

22. CR and TSC charcoal filter removal efficiencies for elemental iodine and for organic iodine are 90% and 30% respectively, consistent with Regulatory Guide 1.52 for a 2 inch charcoal bed without humidity control.

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23. Analysis accounts for error in the RADTRAD Powers natural deposition model.

24. Analysis addresses commitment No. 2 to the NRC via letter NG-01-0382 to include only horizontal piping with the use of the RADTRAD Brockmann-Bixler pipe deposition model.

<u>Suppression Pool pH Response</u>

1. Core concrete aerosols are basic materials produced from the interaction of molten core materials with the concrete inside primary containment. For DBA LOCAs core damage is assumed to be arrested after the early in-vessel release phase. Thus, these chemicals were not considered within this assessment.

2. Minimum SP volume to be utilized for the post-LOCA assessment of SP pH is 58,900 ft^3. The maximum SP volume, without the relatively small Sodium Pentaborate addition from the SLCS, to be utilized for the post-LOCA assessment of SP pH is 68,312 ft^3 [61,500 + 6187 + 625].

3. SLCS data assumed a volume of 2500 gallons of 11.8 wt% $Na_2O*5B_2O_3*10H_2O$ injected at 26 gpm within two hours of the onset of the accident.

4. Approximately 1-minute after the LOCA begins, ECCS Core Spray & LPCI pumps are available to draw suction from the Torus SP. At least 7038 gpm (1-LPCS pump at 2718 gpm and 1-RHR pump at 4320 gpm, will be circulating from the SP to the reactor vessel and/or spray system and spilling back into the SP via the vent/downcomer system. Based on the pool and coolant inventory of 68,312 ft^3, this ECCS flow represents approximately one complete exchange of the volume per 1.2 hours. On this basis, complete mixing is assumed such that a single SP pH can be applied.

5. The allowable SP pH range is 6.8 to 7.3. The allowable reactor coolant pH ranges from 5.6 to 8.6. To conservatively simplify this SP pH assessment, the initial SP pH will be conservatively considered at the minimum reactor coolant pH value so that no SP pH change need be considered due to the released reactor coolant mixing with the SP inventory. The minimum reactor coolant pH of 5.6 specified is obtained from a depressurized reactor coolant sample.

6. The cesium that is not in the chemical form of CsI is assumed to exit the RCS in the form of cesium hydroxide (CsOH) and be deposited into the SP. The rate of modeled CsOH deposition into the SP could be reasonably based on the 10% Powers model for natural deposition of aerosols as utilized within the RADTRAD computer code. However, since the Powers model only addresses deposition from the primary containment vapor region, simplifying assumptions are made for CsOH deposition. Initially, all CsOH was assumed to reach the SP. In the resulting calculation, SP pH fell below 7.0. This showed that the effect of CsOH alone is insufficient to maintain a basic pH. Therefore, in the final analysis, (addition of Sodium Pentaborate), conservatively, no credit for CsOH deposition into the SP is taken.

7. The 30-Day Integrated Dose for the DAEC Suppression Pool water, the Primary Containment (DW) air gamma plus bremsstrahlung, and the DW air beta were developed. These integrated dose values are based on TID-14844 source terms at the DAEC with an assumed power level of 1950 MWt (102% of 1912 MWt). These integrated dose profiles were curve fit for inclusion in the pH calculations. Comparison of the curve fit dose values versus the reference dose values given in drywell EQ dose calculations are quite good. To conservatively ensure that the curve fit dose value at any time point greater than 0.53661-hours (0.5 hour + 121 seconds) is conservatively bounded, a margin factor of 1% is applied to all curve fit integrated dose values.

8. It is expected that the initial effects on pool pH will come from rapid fission product transport and formation of CsOH. As radiolytic production of nitric acid and hydrochloric acid proceeds and these acids are transported to the pool over the first hours of the event the pH would become more acidic. The buffering effect of SLCS injection within two hours is assumed to be sufficient to offset the effects of these acids as they are transported to the pool. As stated above, the CsOH is not credited for long term pool pH. This assumption is consistent with previous NRC staff conclusions that for the first two hours of a DBA, the iodine source term behavior and its transport within the drywell will be independent of iodine reevolution and pH control.

9. The cable mass in the DAEC DW was identified from the DAEC cable and raceway database as ~23,000 lbm excluding the copper conductor mass. Conservatively, all 23,000 lbm is assumed to be chloride-bearing.

10. Essentially the only free hanging cable within the DAEC DW is that associated with the loops at the bottom of the vessel and entering junction boxes. This analysis conservatively assumed 50% of the identified cable mass as being "free drop, 11,500 lbm. The other half of the chloride-bearing cable mass (non "free drop" cable mass) is assumed to experience a beta radiation dose equal to 50% of the incident dose due to self and structural shielding.

11. From the DAEC cable information the largest cable radius of 2.22 inches was utilized to conservatively maximize the absorption fraction of the incident gamma and beta energy within the chloride-bearing cables, thus maximizing the Hydrochloric Acid generated.

12. The production of all the strong acids and bases except Nitric Acid are independent of SP volume. The equations utilized for these acids and bases simply divides the production value by the SP volume to obtain the acid or base concentration within the SP. Nitric Acid production is a direct function of the SP volume; however, as pool volume increases, the specific radiation activity decreases proportionally since the total radiation source term does not change. Thus for Nitric Acid production, the specific radiation dose in a liter of the SP drops proportionally with the increase in SP volume. Therefore, the production of all acids and bases is independent of SP volume. The calculation case studies will be performed using the maximum pool volume (and minimum SP initial pH) since the maximum SP volume will be the bounding case when considering the addition of the Sodium Pentaborate buffer (greater dilution of the buffer).

13. When calculating the average to incident beta radiation flux for the production of Hydrochloric Acid in chloride-bearing cable, it was assumed that the jacket thickness of all cables is 0.045 inches. This assumption conservatively bounds DAEC plant configuration and results in greater Hydrochloric Acid production. Also, minimum jacket thickness and the maximum cable radius values were assumed to maximize acid production from these cables. Similarly, when calculating the Hypalon mass to cable mass ratio used in the calculation of Hydrochloric Acid from beta radiation, the jacket thickness value used for all cables is the maximum value. This assumption conservatively maximizes acid production from this term.

Results

60 Isotope LOCA Source Term Inventory

Isotope	Ci/Mwt
Co-58	0.1529E+03
Co-60	0.1830E+03
Kr-85	0.4155E+03
Kr-85m	0.6702E+04
Kr-87	0.1274E+05
Kr-88	0.1792E+05
Rb-86	0.7813E+02
Sr-89	0.2406E+05
Sr-90	0.3331E+04
Sr-91	0.3047E+05
Sr-92	0.3331E+05
Y-90	0.3439E+04
Y-91	0.3133E+05
Y-92	0.3347E+05
Y-93	0.3915E+05
Zr-95	0.4433E+05
Zr-97	0.4493E+05
Nb-95	0.4455E+05
Mo-99	0.5141E+05
Tc-99m	0.4501E+05
Ru-103	0.4341E+05
Ru-105	0.3074E+05
Ru-106	0.1833E+05
Rh-105	0.2894E+05
Sb-127	0.3051E+04
Sb-129	0.8971E+04
Te-127	0.3034E+04
Te-127m	0.4101E+03
Te-129	0.8829E+04
Te-129m	0.1313E+04
Te-131m	0.3985E+04
Te-132	0.3857E+05
I-131	0.2720E+05
I-132	0.3922E+05
I-133	0.5496E+05
I-134	0.6021E+05
I-135	0.5150E+05
Xe-133	0.5279E+05
Xe-135	0.1908E+05
Cs-134	0.8099E+04
Cs-136	0.2443E+04
Cs-137	0.4644E+04
Ba-139	0.4872E+05

Ba-140	0.4703E+05
La-140	0.5060E+05
La-141	0.4437E+05
La-142	0.4272E+05
Ce-141	0.4459E+05
Ce-143	0.4083E+05
Ce-144	0.3701E+05
Pr-143	0.3947E+05
Nd-147	0.1797E+05
Np-239	0.5816E+06
Pu-238	0.1691E+03
Pu-239	0.1368E+02
Pu-240	0.2014E+02
Pu-241	0.5732E+04
Am-241	0.8036E+01
Cm-242	0.2037E+04
Cm-244	0.1530E+03

Atmospheric Dispersion

Table No. 1 – Offsite Ground Level CHI/Q's (PAVAN-PC)

Offsite Ground Level Release CHI/Q's (sec/m^3)		
	EAB (629m ENE)	LPZ (3218 m NE)
0 - 2 hours	5.57-4	1.34-4
0 - 8 hours	3.42-4	6.43-5
8 - 24 hours	2.69-4	4.46-5
1 - 4 days	1.59-4	2.01-5
4 - 30 days	7.43-5	6.43-6

Table No. 2 Offsite Elevated Release CHI/Q's from the Off Gas Stack (sec/m^3)

Offsite Elevated Release CHI/Q's (sec/m^3)		
	EAB (936m NW)	LPZ (3218 m NW)
Fumigation	7.03-5	3.15-5
0 - 2 hours	6.95-6	6.69-6
0 - 8 hours	3.61-6	3.58-6
8 - 24 hours	2.61-6	2.61-6
1 - 4 days	1.28-6	1.32-6
4 - 30 days	4.64-7	4.99-7

Table 3 – Ground Level Release TB Exhaust CHI/Q's (sec/m^3)

Includes Occupancy Adjustment Factors for ARCON96 Values	
Time Period	ARCON96, D = 82 m, A = 1609 m^2, AZ = 153°, Sector = 90°
0 – 2 hours	9.23E-04
2 – 8 hours	7.96E-04
8 – 24 hours	3.57E-04
1 - 4 days	2.47E-04
4 – 30 days	1.88E-04

Table 4 – CR & TSC Ground Level Release – Condenser CHI/Q's (sec/m^3)

Time Period	CR ARCON96 D = 60.8 m, A = 1609 m^2 ZA = 137°, Sector = 90°	TSC ARCON96 D = 52.5 m, A = 1609 m^2 ZA = 168.5°, Sector = 90°
0 – 2 hours	1.48E-03	2.14E-03
2 – 8 hours	1.27E-03	1.86E-03
8 – 24 hours	5.56E-04	8.44E-04
1 – 4 days	3.40E-04	6.10E-04
4 – 30 days	2.65E-04	4.69E-04

Table 5 – CR & TSC Ground Level Release – RB Wall CHI/Q's (sec/m^3)

Time Period	CR ARCON96 D = 15.8 m, A = 1609 m^2 ZA = 180°, Sector = 90°	TSC ARCON96 D = 22.6 m, A = 1609 m^2 ZA = 192°, Sector = 90°
0 – 2 hours	1.33E-02	8.52E-03
2 – 8 hours	1.12E-02	7.09E-03
8 – 24 hours	5.21E-03	3.28E-03
1 – 4 days	3.77E-03	2.36E-03
4 – 30 days	2.87E-03	1.86E-03

Table 6 – CR & TSC Elevated Release – Off-Gas Stack CHI/Q's (sec/m^3)

Time Period	CR PAVAN&ARCON96, D = 210 m, A = 1609 m^2, ZA = 165°, Sector = 90°	TSC PAVAN&ARCON96, D = 214 m, A = 1609 m^2, ZA = 173°, Sector = 90°
0 – 0.5 hours (Fumigation)	2.62E-04	2.38E-04
0 – 2 hours	3.93E-07	2.32E-07
2 – 8 hours	3.75E-07	2.16E-07
8 – 24 hours	1.33E-07	8.00E-08
1 – 4 days	1.04E-07	6.15E-08
4 – 30 days	9.37E-08	5.39E-08

Table No. 7– TSC ELEVATED RELEASE CHI/Q's OFF GAS STACK (sec/m^3)

Time Period	ARCON96 D=214 m A=1609 m^2 AZ=173º Sector=90º	PAVAN D=214 m A=1609 m^2
30 minutes (fumigation)	Not calculated	2.38-4
0 - 8 hours	2.20-7$^{(1)}$	
0 – 2 hours	2.32-7$^{(2)}$	
2 – 8 hours	2.16-7$^{(2)}$	
8 - 24 hours	8.00-8$^{(2)}$	
1 - 4 days	6.15-8	
4 - 30 days	5.39-8	

(1): Based on 0 to 8 hour averaging period.

(2): Based on ARCON96 standard averaging intervals.

Table No. 8– FHA CR/TSC GROUND RELEASE CHI/Q's RB VENT (sec/m^3)

Time Period	CR – RB Vent ARCON96 D=29.2 m A=1609 m^2 AZ=125º Sector=90º	TSC – RB Vent ARCON96 D=24.4 m A=1609 m^2 AZ=196º Sector=90º
0 – 2 hours	2.85-3	2.66-3
2 – 8 hours	2.29-3	2.25-3
8 - 24 hours	1.02-3	1.03-3
1 - 4 days	3.64-4	4.34-4
4 - 30 days	1.80-4	2.30-4

The following table provides a breakdown of the application of the above CHI/Q values to radiological analysis for specific design basis accidents.

Pathways (Applicable DBAs)	**CHI/Q Table**
Ground Release to LPZ (All)	Table No. 1
Ground Release to EAB (All)	Table No. 1
Elevated Release to LPZ (LOCA Post-PPP)	Table No. 2
Elevated Release to EAB (LOCA Post-PPP)	Table No. 2
Reactor Building to CR (LOCA PPP)	Table No. 5
Reactor Building to TSC (LOCA PPP)	Table No. 5
Turbine Building to CR or TSC (MSLB)	Table No. 3
Condenser to CR (LOCA, CRDA)	Table No. 4
Condenser to TSC (LOCA, CRDA)	Table No. 4
Elevated (Stack) to CR (LOCA Post-PPP)	Table No. 6
Elevated (Stack) to TSC (LOCA Post-PPP)	Table No. 7
RB to CR (FHA)	Table No. 8
RB to TSC (FHA)	Table No. 8

Radiological Dose

	Dose Component	**EAB(1) (rem TEDE)**	**LPZ(2) (rem TEDE)**
2017-010	Primary Containment Leakage	0.198	0.391
	ESF Leakage	0.003	0.015
2017-010	Main Steam Pathway Leakage	0.046	0.219
	TOTAL	**0.247**	**0.626**

Notes for EAB, LPZ Results:
1. Worst 2-hour integrated dose.
2. 30-day integrated dose.

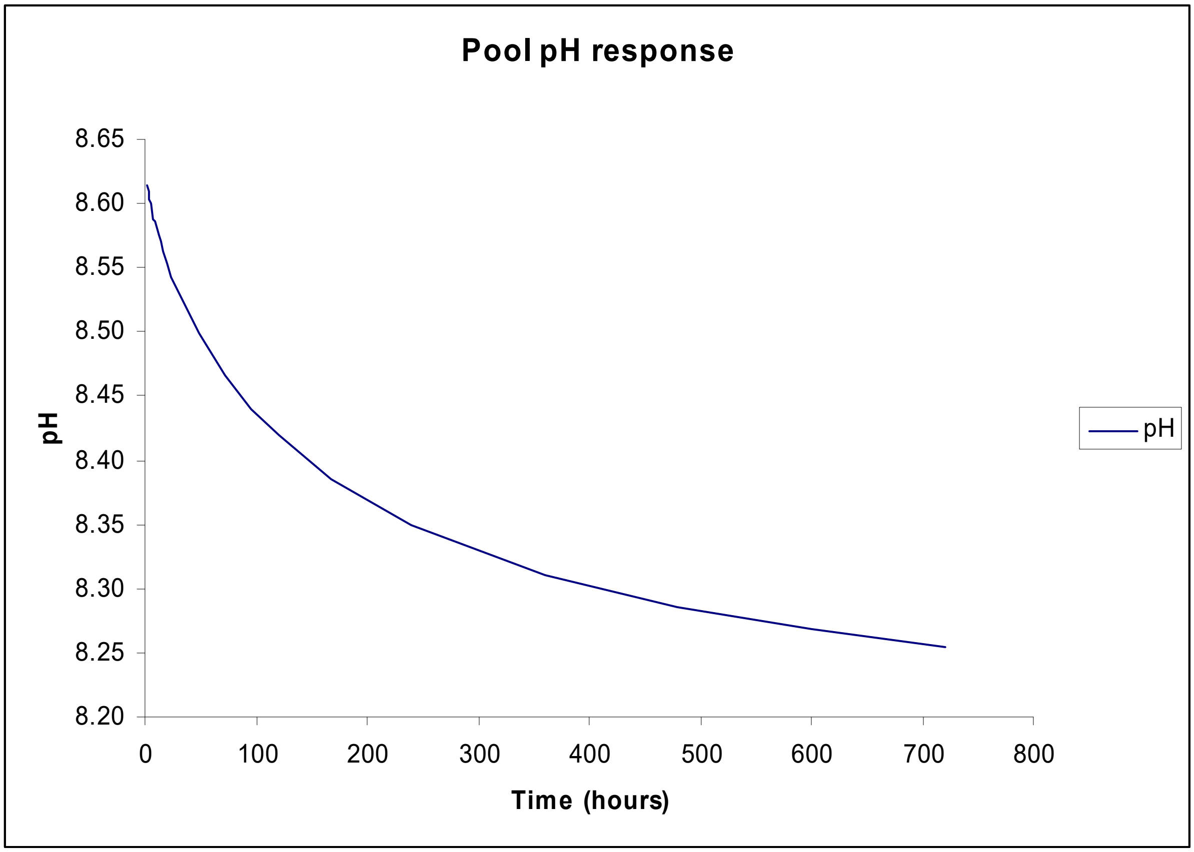
	Dose Component	**CR(1) (rem TEDE)**	**TSC(2) (rem TEDE)**
2017-010	Primary Containment Leakage	2.331	1.108
	ESF Leakage	0.042	0.020
	Main Steam Pathway Leakage	0.905	1.932
	External Cloud	0.003	0.121
	RB Direct Shine	0.094	0.795
	SBGT Filter Direct Shine	0.049	N/A
2017-010	Ingress/Egress	0.670	N/A
	TOTAL	**4.094**	**3.976**

Notes for CR and TSC Results:

2017-010 1. Assumes unfiltered inleakage of 0 cfm.

2. Assumes conservative unfiltered inleakage of 1000 cfm. For TSC dose calculations, higher inleakage values are marginally conservative because the larger unfiltered inleakage results in higher doses during the controlling normal clean-up.

Suppression Pool pH Response



a) Comparison to Acceptance Criteria:

	Dose Component	EAB (rem TEDE)	LPZ (rem TEDE)
2017-010	**Total Calculated Dose from DBA LOCA with Alternative Source Term**	**0.247**	**0.626**
	Regulatory Limit (10 CFR 50.67)	**25**	**25**

	Dose Component	CR (rem TEDE)	TSC (rem TEDE)
2017-010	**TOTAL**	**4.094**	**3.976**
	Regulatory Limit	**5 (10 CFR 50.67)**	**5 (NUREG-0737)**

b) Sensitivities:

Sensitivity analyses of the post-LOCA dose to CR operators were performed at assumed unfiltered CR inleakage rates of 0, 67.5, 500, and 1000 cfm. The limiting dose to control room operators was found to result from the lowest value of unfiltered inleakage. The analysis methodology maximizes calculated dose to the control room by assuming a four minute delay in shifting to emergency ventilation due to high radiation at the control room ventilation intake. This traps a significant source term from the secondary containment positive pressure period in the control room. Once the primary release from the secondary containment shifts to a filtered elevated release from the offgas stack, the concentration of radioisotopes at the control room ventilation intake is significantly reduced. The DAEC control room is not equipped with recirculation filtration, so the only removal mechanisms are from decay and dilution by the filtered intake from the SFU's. Additional unfiltered inleakage would help dilute the control room source term more quickly and would result in lower calculated operator doses.

Sensitivity analyses of the post-LOCA dose to the TSC were performed at assumed unfiltered TSC inleakage rates of 67.5, 500, and 1000 cfm. TSC dose consequences increase with increased unfiltered inleakage.

c) Uncertainties:

Uncertainties in radiological consequences analysis are significant. The single largest uncertainty is the magnitude of the radiological source term. Design basis accidents should not result in significant core damage or radiological releases. Providing systems capable of mitigating beyond-design-basis accidents involving significant fuel damage and radiological releases provides defense-in-depth to assure public health and safety.

To compensate for uncertainties, methodology, assumptions, and inputs are selected to provide significant additional conservatism.

Conclusions

a) Statement of Acceptability
The DAEC radiological consequences analysis demonstrate that the systems designed to limit dose to workers and the public satisfy all acceptance criteria.

b) Conservatisms/Margins
As described above, there is significant uncertainty in estimating radiological consequences from a design basis accident. Accordingly the methodology, inputs, assumptions, and limits include significant conservativism to compensate. Activities that result in changes in dose consequences of 10 percent or less of the available margin to the regulatory limits are not considered to be adverse changes and may be implemented under 10 CFR 50.59.

c) Limiting Event
The LOCA with Design Basis Radiological Releases constitutes the limiting radiological event for onsite personnel. However, the CRDA (Section 15.2.4) is more limiting for dose to the public.

B) Intermediate Breaks

The Intermediate break category is classically those breaks sizes in between Large and Small, but because the upper range of Small breaks varies with the analysis being performed (e.g., fuel versus containment), the definition of this category is somewhat arbitrary. But, as a rule of thumb, Intermediate breaks are between 0.1 ft^2 and 1.0 ft^2.

1) Reactor Response

This discussion centers on the response of the fuel and the RPV to the Large Break LOCA event.

Description of Event

a) Initiator: This event is initiated by an instantaneous, non-mechanistic, partial break of the reactor recirculation pump suction pipe at the nozzle on the RPV – break sizes of 0.1 ft^2 and 0.5 ft^2 have been analyzed (References 15.0-4 and 15.0-44).

b) Sequence of Events: Coincident with the initiation of the break, a complete Loss-of-Offsite Power (LOOP) is assumed to occur, in accordance with GDC 35. Reactor coolant begins to exit the vessel rapidly into the Drywell at the critical mass flux and reactor vessel water level begins to drop, as does the reactor pressure. The reactor is assumed to scram immediately. The Emergency Diesel Generators (EDGs) start on the LOOP condition and all loads are stripped off the Essential AC busses. The non-Essential busses are lost, leading to a loss of Feedwater and a Reactor Recirculation Pump coastdown. As the RPV level reaches the various level setpoints, ECCS systems are actuated (a conservative assumption to delay injection), (Note: High Pressure Coolant Injection (HPCI) has no DC control power and does not operate), Vessel isolation signals are generated (Containment isolations are discussed in the Containment Response below), and Low Pressure Coolant Injection (LPCI) loop select logic actuates to determine which recirculation loop is broken and closes the recirculation pump discharge valve in the non-broken loop (the pump discharge bypass valve is conservatively assumed to remain open). If the plant had previously been operating in single loop recirculation mode, loop select logic would trip the running recirculation pump and effect a short time delay to allow it to coastdown prior to its selecting the “broken” recirculation loop (See Chapter 7.3.1.1.2.4 for a complete explanation of LPCI loop select logic). The reactor level continues to drop and uncovers the fuel, which begins to heat up. Once the EDGs are up to speed, its output breaker closes in on the Essential AC busses, and the low pressure ECCS pumps (and other essential loads) are sequenced onto the busses, the pumps start and their minimum flow bypass valves open. The Automatic Depressurization System (ADS) actuation logic initates on lowering RPV level and ECCS pumps running, but because the RPV depressurizes through the break prior to the ADS 2 minute time delay expiring, the valves never actually open. Once the reactor pressure decreases to their respective permissive setpoints, the injection valves for Core Spray and LPCI (based upon the “chosen” loop by loop select logic), open and allow injection to begin to the RPV. The injection refills the lower vessel plenum area and the water level inside the core shroud rises and terminates the fuel heatup. Water level is maintained at the top of the jet pumps and long-term recovery mode is entered. The analysis of the fuel and RPV response is terminated at this point.

c) Single Failure/Operator Error: The loss of Division II of 125VDC is the limiting single failure for this event. This results in the loss of HPCI, and “B” CS and “B” and “D” RHR (LPCI) pumps.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures): No Operator Actions are assumed in this evaluation. Reactor scrams (high Drywell pressure), EDG starts (either LOOP (undervoltage) or high Drywell pressure) and loads ("dead" buss permissive), Feedwater and Recirculation pumps coastdown on LOOP condition, HPCI actuates on low-low RPV level (conservatively ignore high Drywell pressure), LPCI loop select logic actuates on low-low RPV level and low RPV pressure and chooses the "broken" loop and closes the recirculation discharge valve in the non-broken loop, based upon recirculation loop differential pressure (assuming not in single loop operation), ADS initiation on low and low-low-low RPV levels, with confirmation signal on ECCS pump running, which starts 2 minute time delay, Core Spray and LPCI pumps start signal on low-low-low RPV level and timers sequence the pumps onto the AC busses and the LPCI minimum flow bypass valves open on high pump discharge pressure and close on high flow (dP), the normally-open CS minimum flow valves close on high flow (dP), CS and LPCI injection valves open on low RPV pressure permissive signals.

Event Category & Acceptance Criteria

This is not a Design Basis Accident (non-DBA) for the fuel and ECCS capability.

Fuel shall remain within 10 CFR 50.46 limits as follows:

Peak Cladding Temperature (PCT) shall remain $\leq$ 2200 °F;
Maximum Cladding Oxidation shall not exceed 17% of the total cladding thickness;
Maximum Hydrogen Generation shall not exceed 1% metal-water reaction;
Coolable Geometry shall be maintained;
Long-term Cooling shall be ensured to remove decay heat.

There are no acceptance criteria for the RPV, as this event assumes a breach of the RPV as the initiating event.

Methods

a) Calculation Tools & Computer Codes: SAFER, GESTR-LOCA, LAMB and TASC (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

As part of General Electric's methodology for complying with 10 CFR 50.46 and Appendix K, a statistical approach is used, which relies on the combination of calculations using both "nominal" inputs and assumptions for ECCS performance, fuel parameters, decay heat model, etc. and those meeting the strict requirements of "Appendix K." See References 15.0-4 and 44 for a complete discussion.

b) Inputs:
The primary set of plant inputs used in the LOCA analysis is provided on the OPL-4 and OPL-5 forms (Tables 15.0-4 and 5, respectively).

c) Key Assumptions:
There is a simultaneous LOOP with the LOCA condition.
There is a single active failure. Both a loss of Division II of 125 VDC or LPCI Injection Valve Failure are evaluated to determine which failure gives the limiting response on the fuel.

In addition, to the assumed single failure above, we also assume that the Recirc. discharge bypass valve in the “selected loop” fails to close. This is due to Environmental Qualification issues (See Section 6.3.2.2.4 and Reference 15.0-4).

The reactor scrams immediately, ignoring Control Rod scram time. Only Decay and Sensible Heat are considered.

ECCS initiation is on RPV level. The Drywell Pressure signal is ignored.

Limiting assumptions on fuel exposure, peaking factors, power shape, initial thermal limits are made.

ECCS Injection water is assumed to be at 88 Btu/lb_m (120 °F).

For break sizes smaller than 0.5 ft^2, it is assumed that LPCI Loop Select Logic fails and selects the broken loop for injection.

Results

The plant response to the IBA is less severe than the DBA case discussed above. In general, the PCTs are hundreds of degrees less than the DBA case.

a) Conformance to Acceptance Criteria:

The resulting PCTs are well below the limit. In addition, the local oxidation fraction and metal-water reaction limits are also met.

To show compliance to the long-term core cooling criteria, we need to demonstrate that either: 1) the core is fully reflooded to the Top of Active Fuel (TAF); OR 2) that we are reflooded to a level equal to the top of the jet pumps AND we have at least one Core Spray pump available for cooling. This is to ensure that sufficient cooling is available either by total submergence or by the combination of partial submergence and spray cooling. In the short-term response, there is enough steam cooling from the partially submerged fuel to cool the upper part of the fuel bundle without reliance on sprays. However, as the decay heat dissipates, there may not be enough steam cooling effects to maintain sufficient cooling in the upper part of the bundle to preclude significant cladding oxidation over the long-term, especially if the fuel axial power shape prior to the accident was heavily "top-peaked." Hence, we must rely upon spray cooling to meet the acceptance criteria for local oxidation fraction, coolable geometry and long-term cooling requirements of §50.46.

b) Sensitivities:

A sensitivity case was run to show the impact of Loop Select Logic picking the wrong (i.e., broken) loop for injection for the 0.5ft^2 case. The results show that there is about a 100°F increase in PCT, but the results are still well below the acceptance limits.

c) Uncertainties in Results:

Because this is a non-liming break size and location, the "Upper Bound PCT (UBPCT)," which represents the 95th percentile of the calculation distribution considering the uncertainties, was not calculated for this event. However, the same uncertainties in the modeling and plant parameters exist, they just have a lesser impact on the results than the more-limiting breaks.

2) Containment Response

The response of the Primary Containment to the Intermediate Break LOCA (IBA-LOCA) is only evaluated for the "long-term" response for containment parameters used in the Mark I containment loads evaluation, as the DBA-LOCA case is bounding for peak pressure and temperature response.

The evaluation of the Secondary Containment (Reactor Building) is considered as part of the Radiological response to the DBA-LOCA below.

Mark I Containment Loads Evaluation for IBA-LOCA

Description of Event

a) Initiator: The same as for the Reactor evaluation above.

b) Sequence of Events:

1. The plant is operating at 102% of 120% ORTP (i.e., 1950 MWt) when a 0.1 ft^2 liquid line break occurs. There is also a concurrent loss of offsite power and only minimum diesel power is available. Reactor scrams.

2. HPCI flow starts injecting into the vessel until it trips on high RPV water level at Level 8.

3. For the first 10 minutes (600 seconds) following the accident, two LPCI pumps (in one RHR loop) at a flow rate of 4800 gpm/pump and one CS pump at 3100 gpm inject into the vessel.

4. At 10 minutes (600 seconds), operator activates the RHR heat exchanger in the operating RHR loop. One RHR pump at 4800 gpm is re-aligned so that flow goes through the heat exchanger before returning to the suppression pool. The other RHR pump is shutdown. This configuration is maintained throughout the accident.

5. After 10 minutes (600 seconds), the CS pump is maintained at 3100 gpm.

6. The event ends when the RPV has depressurized to 50 psia.

c) Single Failure/Operator Error:
The same as for the Reactor evaluation above.

d) Key Equipment Response:
Reactor scrams (assumed on High Drywell Pressure), MSIVs close, EDG starts and loads (LOOP), RHR and CS pumps start (High DW pressure), Torus-to-Drywell vacuum breakers open/close, Operators secure one RHR pump and manually load the RHR Service Water (RHRSW) pumps to initiate cooling with the RHR heat exchanger after 10 minutes.

<u>Event Category & Acceptance Criteria</u>

This is not a Design Basis Accident (non-DBA) for the Primary Containment.

The Primary Containment response to the IBA-LOCA shall remain within the criteria for chugging loads, drywell gas temperature (340ºF) and suppression pool temperature (281°F) as defined in the Mark I Plant Unique Load Definition (PULD; Reference 15.0-52) and the Plant Unique Analysis Report (PUAR) (Reference 15.0-56).

<u>Methods</u>

a) Calculation Tools & Computer Codes: SHEX code with the HEM break flow model. (See Table 15.0-2 for complete listing, code versions and NRC acceptance.)

b) Inputs: The primary set of plant inputs used in the containment analysis is provided on the OPL-4a form (Tables 15.0-6).

c) Key Assumptions:
1. The power level for the power/flow point analyzed includes an additional 2% power, consistent with Regulatory Guide 1.49.

2. The shutdown power fractions include fuel relaxation energy, metal-water reaction energy and ANS 5.1 +2sigma decay heat for fuel applicable up to GE14 with 24-month fuel cycle.

3. Initial conditions for drywell pressure, wetwell pressure and suppression pool temperature are based on nominal values, as specified in the PULD.

4. The IBA-LOCA is a 0.1 ft^2 liquid line break.

5. Concurrent with the postulated LOCA, a loss of offsite power occurs.

6. Only minimum diesel power is available. This results in only one RHR loop with one heat exchanger available for containment cooling, starting at 10 minutes (600 seconds).

7. RHR heat exchanger performance is based on one RHR pump (4800 gpm) and two RHRSW pumps (4080 gpm total).

8. All feedwater mass with temperatures higher than 281ºF (saturation temperature at 50 psia) is injected into the vessel, regardless of the availability considerations of feedwater and condensate pumps.

9. The wetwell airspace is in thermal equilibrium with the suppression pool at all times. This is consistent with the PULD.

10. The initial suppression pool water volume corresponds to the TS Low Water Level (LWL) to maximize the suppression pool temperature response.

11. Passive heat sinks in the drywell, wetwell airspace and suppression pool are conservatively neglected to maximize the suppression pool temperature. Heat transfer from the primary containment to the reactor building is also conservatively neglected.

12. Drywell fan coolers are inactive.

13. Operating Core Spray and LPCI/RHR pumps have 100% of their motor horsepower rating converted to pump heat which is added either to the Reactor Pressure Vessel (RPV) liquid or suppression pool water. This assumption is used to maximize the suppression pool temperature response.

14. Main Steam Isolation Valves (MSIVs) start closing at 0.5 seconds and close completely at 3.5 seconds.

15. Only 6 wetwell-to-drywell vacuum breakers are assumed to be active.

Results

a) Conformance to Acceptance Criteria:
The peak drywell gas temperature of 276.7ºF is well below the PULD value of 340ºF.

The peak suppression pool temperature of 173.3°F is well below the PULD value of 281°F.

The onset of chugging and chugging duration for the IBA-LOCA is defined in Table 1-4.1-10 of the PUAR. Per the PUAR, chugging ends when RPV pressure reaches 50 psia, which occurs at 1105 seconds for the IBA-LOCA. By analysis, the RPV has depressurized to approximately 56 psia at 1105 seconds. To assess the effect of the slightly higher RPV pressure on the chugging loads, the vent steam mass flux conditions were evaluated and found to be below the threshold for chugging (i.e., chugging would not be occurring at 1105 seconds). Therefore, the end of chugging time given in the PUAR remains bounding.

b) Sensitivities:
A description of the Mark I containment testing program is contained in References 15.0- 53, 54, and 55.

c) Uncertainties in Results:
The use of the 102% power level and conservative values for the decay heat generation (ANS 5.1-1979 +2sigma) and conservative inputs to the calculation all contribute to compensate for any uncertainties in the calculation methodology.

3) Radiological Response

This event is bounded by the Recirculation Piping LOCA (Section 15.2.1.1) and is not specifically analyzed for dose consequences.

C) Small Breaks

The Small Break spectrum is typically chosen as break sizes such that the RPV does not rapidly depressurize itself and requires the operation of the Automatic Depressurization System (or Operator Action to open the Safety Relief Valves (SRVs)) to lower the reactor pressure to allow the low pressure ECCS to actuate and reflood the RPV. The typical range of break sizes in the Small break spectrum are from 0.01 ft^2 to 0.1 ft^2.

1) Reactor Response

This discussion centers on the response of the fuel and the RPV to the Small Break LOCA (SBLOCA) event.

<u>Description of Event</u>

a) Initiator: This event is initiated by an instantaneous, non-mechanistic, partial break of the reactor recirculation pump suction pipe at the nozzle on the RPV – break sizes of 0.01 ft^2, 0.03 ft^2, 0.04 ft^2, 0.05 ft^2, 0.06 ft^2, and 0.07 ft^2.

b) Sequence of Events: Coincident with the initiation of the break, a complete Loss-of-Offsite Power (LOOP) is assumed to occur, in accordance with GDC 35. Reactor coolant begins to exit the vessel rapidly into the Drywell at the critical mass flux and reactor vessel water level begins to drop, as does the reactor pressure. The reactor is assumed to scram immediately. The Emergency Diesel Generators (EDGs) start on the LOOP condition and all loads are stripped off the Essential AC busses. The non-Essential busses are lost, leading to a loss of Feedwater and a Reactor Recirculation Pump coastdown. As the RPV level reaches the various level setpoints, ECCS systems are actuated (a conservative assumption to delay injection), Vessel isolation signals are generated (Containment isolations are discussed in the Containment Response below), and Low Pressure Coolant Injection (LPCI) loop select logic actuates to determine which recirculation loop is broken and closes the recirculation pump discharge valve in the non-broken loop (the pump discharge bypass valve is conservatively assumed to remain open). If the plant had previously been operating in single loop recirculation mode, loop select logic would trip the running recirculation pump and effect a short time delay to allow it to coastdown prior to its selecting the "broken" recirculation loop (See Chapter 7.3.1.1.2.4 for a complete explanation of LPCI loop select logic). The reactor level continues to drop and uncovers the fuel, which begins to heat up. Once the EDGs are up to speed, its output breaker closes in on the Essential AC busses, and the low pressure ECCS pumps (and other essential loads) are sequenced onto the busses, the pumps start and their minimum flow bypass valves open. The Automatic Depressurization System (ADS) actuation logic initates on lowering RPV level and ECCS pumps running, the ADS 2 minute time delay expires, and the valves open. Once the reactor pressure decreases to their respective permissive setpoints, the injection valves for Core Spray and LPCI (based upon the "chosen" loop by loop select logic), open and allow injection to begin to the RPV. The injection refills the lower vessel plenum area and the water level inside the core shroud rises and terminates the fuel heatup. Water level is maintained at the top of the jet pumps and long-term recovery mode is entered. The analysis of the fuel and RPV response is terminated at this point.

c) Single Failure/Operator Error: The loss of Division II of 125VDC is the limiting single failure for this event. This results in the loss of HPCI, and "B" CS and "B" and "D" RHR (LPCI) pumps.

d) Key Equipment: Responses (trips/actuations) & Operator Actions (successes & failures): No Operator Actions are assumed in this evaluation. Reactor scrams (high Drywell pressure), EDG starts (either LOOP (undervoltage) or high Drywell pressure) and loads ("dead" buss permissive), Feedwater and Recirculation pumps coastdown on LOOP condition, LPCI loop select logic actuates on low-low RPV level and low RPV pressure and chooses the "broken" loop and closes the recirculation discharge valve in the non-broken loop, based upon recirculation loop differential pressure (assuming not in single loop operation), ADS initiation on low and low-low-low RPV levels, with confirmation signal on ECCS pump running, which starts 2 minute time delay, Core Spray and LPCI pumps start signal on low-low-low RPV level and timers sequence the pumps onto the AC busses and the LPCI minimum flow bypass valves open on high pump discharge pressure and close on high flow (dP), the normally-open CS minimum flow valves close on high flow (dP), CS and LPCI injection valves open on low RPV pressure permissive signals.

Event Category & Acceptance Criteria

This is not a Design Basis Accident (non-DBA) for the fuel and ECCS capability.

Fuel shall remain within 10 CFR 50.46 limits as follows:

Peak Cladding Temperature (PCT) shall remain $\leq$ 2200 °F;
Maximum Cladding Oxidation shall not exceed 17% of the total cladding thickness;
Maximum Hydrogen Generation shall not exceed 1% metal-water reaction;
Coolable Geometry shall be maintained;
Long-term Cooling shall be ensured to remove decay heat.

There are no acceptance criteria for the RPV, as this event assumes a breach of the RPV as the initiating event.

Methods

2012-020 a) Calculation Tools & Computer Codes: SAFER, GESTR-LOCA, PRIME-LOCA, LAMB and TASC (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

As part of General Electric's methodology for complying with 10 CFR 50.46 and Appendix K, a statistical approach is used, which relies on the combination of calculations using both "nominal" inputs and assumptions for ECCS performance, fuel parameters, decay heat model, etc. and those meeting the strict requirements
2012-020 of "Appendix K." See References 15.0-4, 44, and 62 for a complete discussion.

2012-020
2015-004
2017-001 The ECCS-LOCA GNF2 analysis is based on the SAFER/PRIME LOCA methodology (Reference 62).

Inputs:

The primary set of plant inputs used in the LOCA analysis is provided on the OPL-4 and OPL-5 forms (Tables 15.0-4 & 5, respectively).

b) Key Assumptions:

There is a simultaneous LOOP with the LOCA condition.
There is a single active failure. Both a loss of Division II of 125 VDC or LPCI Injection Valve Failure are evaluated to determine which failure gives the limiting response on the fuel.

In addition, to the assumed single failure above, we also assume that the Recirc. discharge bypass valve in the "selected loop" fails to close. This is due to Environmental Qualification issues (See Section 6.3.2.2.4 and Reference 15.0-4). The reactor scrams immediately, ignoring Control Rod scram time. Only Decay and Sensible Heat are considered.

ECCS initiation is on RPV level. The Drywell Pressure signal is ignored. Limiting assumptions on fuel exposure, peaking factors, power shape, initial thermal limits are made.

ECCS Injection water is assumed to be at 88 Btu/lb_m (120 °F).
For break sizes smaller than 0.5 ft^2, it is assumed that LPCI Loop Select Logic fails and selects the broken loop for injection.

<u>Results</u>

The PCTs for the SBLOCAs typically run higher than the IBLOCAs, due to the slower depressurization and delayed injection by the low-pressure ECCS pumps. The SBLOCA cases depressurize using ADS (after the 2 minute time delay), whereas the IBLOCA depressurizes faster through the pipe break. However, the SBLOCA PCTs are still less than the LBLOCA cases.

a) Conformance to Acceptance Criteria:

The resulting PCTs are well below the limit. In addition, the local oxidation fraction and metal-water reaction limits are also met.

To show compliance to the long-term core cooling criteria, we need to demonstrate that either: 1) the core is fully reflooded to the Top of Active Fuel (TAF); OR 2) that we are reflooded to a level equal to the top of the jet pumps AND we have at least one Core Spray pump available for cooling. This is to ensure that sufficient cooling is available either by total submergence or by the combination of partial submergence and spray cooling. In the short-term response, there is enough steam cooling from the partially submerged fuel to cool the upper part of the fuel bundle without reliance on sprays. However, as the decay heat dissipates, there may not be enough steam cooling effects to maintain sufficient cooling in the upper part of the bundle to preclude significant cladding oxidation over the long-term, especially if the fuel axial power shape prior to the accident was heavily "top-peaked." Hence, we must rely upon spray cooling to meet the acceptance criteria for local oxidation fraction, coolable geometry and long-term cooling requirements of §50.46.

b) Sensitivities

Small break cases were run with one ADS valve out of service (ADS-OOS). The reduced blowdown capacity causes an increase in PCT relative to the corresponding break size with no valves OOS, due to the delay in reaching the injection pressures of the low pressure ECCS pumps (i.e., delayed injection). The results show that while there is a significant increase in PCT (several hundred degrees) with one less ADS valve, the resulting PCTs are still well below the acceptance criteria.

As part of implementation of Increased Core Flow (ICF) (Ref. 15.0-60), the reactor response to the Small Break-LOCA was evaluated at an initial condition of 105% of rated core flow. The PCT is not sensitive to changes in initial core flow because the fuel bundle remains in nucleate boiling during the small break LOCA until the fuel uncovers due to the inventory loss. Core uncovery during the small break LOCA usually occurs after the ADS valves have opened to depressurize the vessel and much of the remaining inventory flashes into a two-phase mixture. Any effects of the initial increased core flow have dissipated by the time the ADS valves open. Therefore, the effects of ICF on the small break LOCA is negligible.

c) Uncertainties in Results

Because this is a non-liming break size and location, the "Upper Bound PCT (UBPCT)," which represents the 95th percentile of the calculation distribution considering the uncertainties, was not calculated for this event. However, the same uncertainties in the modeling and plant parameters exist, they just have a lesser impact on the results than the more-limiting breaks.

2) Containment Response

The response of the Primary Containment to the Small Break LOCA (SBA-LOCA) is only evaluated for the "long-term" response for containment parameters used in the Mark I containment loads evaluation, as the DBA-LOCA case is bounding for peak pressure and temperature response.

The evaluation of the Secondary Containment (Reactor Building) is considered as part of the Radiological response to the DBA-LOCA below.

Mark I Containment Loads Evaluation for SBA-LOCA

Description of Event

a) Initiator: The same as for the Reactor evaluation above.

b) Sequence of Events:

1. The plant is operating at 102% of 120% ORTP (i.e., 1950 MWt) when a 0.01 ft^2 steam line break occurs. There is also a concurrent loss of offsite power and only minimum diesel power is available. Reactor scrams.

2. HPCI flow starts injecting into the vessel until it trips on high RPV water level at Level 8.

3. For the first 10 minutes (600 seconds) following the accident, two LPCI pumps (in one RHR loop) and one CS pump are operating (on minimum flow), but won't inject into the vessel due to elevated RPV pressure.

4. At 10 minutes (600 seconds), operator initiates a rapid RPV depressurization using four available SRVs. Rapid depressurization at ten minutes is consistent with the PULD.

5. At 10 minutes (600 seconds), operator activates the RHR heat exchanger in the operating RHR loop. One RHR pump at 4800 gpm is re-aligned so that flow goes through the heat exchanger before returning to the suppression pool. The other RHR pump is shutdown. This configuration is maintained throughout the accident.

6. After 10 minutes (600 seconds), the CS pump injection to the vessel is maintained at 3100 gpm.

7. The event ends when the RPV has depressurized to 50 psia.

c) Single Failure/Operator Error:
The same as for the Reactor evaluation above.

d) Key Equipment Response

Reactor scrams (assumed on High Drywell Pressure), MSIVs close, EDG starts and loads (LOOP), RHR and CS pumps start (High DW pressure), Torus-to-Drywell vacuum breakers open/close, Operators secure one RHR pump and manually load the RHR Service Water (RHRSW) pumps to initiate cooling with the RHR heat exchanger after 10 minutes.

Event Category & Acceptance Criteria

This is not a Design Basis Accident (non-DBA) for the Primary Containment.

The Primary Containment response to the SBA-LOCA shall remain within the criteria for chugging loads, drywell gas temperature (340ºF) and suppression pool temperature (281°F) as defined in the Mark I Plant Unique Load Definition (PULD; Reference 15.0-52) and the Plant Unique Analysis Report (PUAR; Reference 15.0-56).

Methods

a) Calculation Tools & Computer Codes: SHEX code with the HEM break flow model. (See Table 15.0-2 for complete listing, code versions and NRC acceptance.)

b) Inputs: The primary set of plant inputs used in the containment analysis is provided on the OPL-4a form (Tables 15.0-6).

c) Key Assumptions:

1. The power level for the power/flow point analyzed includes an additional 2% power, consistent with Regulatory Guide 1.49.

2. The shutdown power fractions include fuel relaxation energy, metal-water reaction energy and ANS 5.1 +2sigma decay heat for fuel applicable up to GE14 with 24-month fuel cycle.

3. Initial conditions for drywell pressure, wetwell pressure and suppression pool temperature are based on nominal values, as specified in the PULD.

4. The SBA-LOCA is a 0.01 ft^2 steam line break.

5. Concurrent with the postulated LOCA, a loss of offsite power occurs.

6. Only minimum diesel power is available. This results in only one RHR loop with one heat exchanger available for containment cooling, starting at 10 minutes (600 seconds).

7. RHR heat exchanger performance is based on one RHR pump (4800 gpm) and two RHRSW pumps (4080 gpm total).

8. All feedwater mass with temperatures higher than 281ºF (saturation temperature at 50 psia) is injected into the vessel, regardless of the availability considerations of feedwater and condensate pumps.

9. The wetwell airspace is in thermal equilibrium with the suppression pool at all times. This is consistent with the PULD.

10. The initial suppression pool water volume corresponds to the TS Low Water Level (LWL) to maximize the suppression pool temperature response.

11. Passive heat sinks in the drywell, wetwell airspace and suppression pool are conservatively neglected to maximize the suppression pool temperature. Heat transfer from the primary containment to the reactor building is also conservatively neglected.

12. Drywell fan coolers are inactive.

13. Operating Core Spray and LPCI/RHR pumps have 100% of their motor horsepower rating converted to pump heat which is added either to the Reactor Pressure Vessel (RPV) liquid or suppression pool water. This assumption is used to maximize the suppression pool temperature response.

14. Main Steam Isolation Valves (MSIVs) start closing at 0.5 seconds and close completely at 3.5 seconds.

15. Only 6 wetwell-to-drywell vacuum breakers are assumed to be active.

<u>Results</u>

a) Conformance to Acceptance Criteria:
With the PULD assumption of no heat sinks, the peak drywell gas temperature is calculated to be 369.9ºF. Since this is higher than the PULD value of 340ºF, a sensitivity analysis was performed with heat sinks to assess the potential for drywell gas temperatures higher than 340ºF during an SBA. The sensitivity analysis calculated 254.8ºF for the peak drywell gas temperature, which is significantly below 340ºF. Therefore, the containment pressure and temperature response given in the PULD remains applicable for the structural evaluations.

The peak suppression pool temperature of 178.9°F is well below the PULD value of 281°F.

The onset of chugging and chugging duration for the SBA-LOCA is defined in Table 1-4.1-10 of the PUAR. Per the PUAR, chugging ends when RPV pressure reaches 50 psia, arbitrarily assumed to be at 1200 seconds for the SBA-LOCA. By analysis (using four SRVs), the RPV does not depressurize to 50 psia until approximately 3600 seconds. To assess the extended period of time with RPV pressure above 50 psia, the conditions at 1200 seconds and 3600 seconds were used to determine the air content for comparison to the upper chugging threshold for air content. The air content at 1200 seconds and 3600 seconds was found to be above the upper chugging threshold (i.e., chugging would not be occurring during this time period). Therefore, the end of chugging time given in the PUAR remains bounding.

b) Sensitivities:

A description of the Mark I containment testing program is contained in References 15.0- 53, 54, and 55.

c) Uncertainties in Results:
The use of the 102% power level and conservative values for the decay heat generation (ANS 5.1-1979 +2sigma) and conservative inputs to the calculation all contribute to compensate for any uncertainties in the calculation methodology.

3) Radiological Response

This event is bounded by the Recirculation Piping LOCA (Section 15.2.1.1) and is not specifically analyzed for dose consequences.

15.2.1.2 – Core Spray Line Break

NOTE: This evaluation was not re-performed at part of the Extended Power Uprate Program. The following evaluation is presented as Historical in nature.

1) Reactor Response

This discussion centers on the response of the fuel and the RPV to the Core Spray Line Break LOCA.

Description of Event

a) Initiator: This event is initiated by an instantaneous, non-mechanistic, double-ended, guillotine break of the Core Spray piping at the nozzle on the RPV (break area is 0.21 ft^2).

b) Sequence of Events: Coincident with the initiation of the break, a complete Loss-of-Offsite Power (LOOP) is assumed to occur and there is a loss of Division II of 125 VDC control power, in accordance with GDC 35. Reactor coolant begins to exit the vessel rapidly into the Drywell at the critical mass flux and reactor vessel water level begins to drop, as does the reactor pressure. The reactor is assumed to scram immediately. The “A” Emergency Diesel Generator (EDG) starts on the

LOOP condition and all loads are stripped off the Essential AC busses. "B" EDG does not start and load due to the loss of 125 VDC control power. The non-Essential busses are lost, leading to a loss of Feedwater and a Reactor Recirculation Pump coastdown. As the RPV level reaches the various level setpoints, ECCS systems are actuated (a conservative assumption to delay injection) (Note: High Pressure Coolant Injection (HPCI) has no DC control power and does not operate), Vessel isolation signals are generated (Containment isolations are discussed in the Containment Response below), and Low Pressure Coolant Injection (LPCI) loop select logic actuates and defaults to the "B" recirculation loop, as there is no break in the recirculation system piping (i.e., no significant dP between the recirculation loops), and closes the recirculation pump discharge valve in the "B" loop. {If the plant had previously been operating in single loop recirculation mode, loop select logic would trip the running recirculation pump and effect a short time delay to allow it to coastdown prior to its selecting the "broken" recirculation loop (See Chapter 7.3.1.1.2.4 for a complete explanation of LPCI loop select logic).} The reactor level continues to drop and uncovers the fuel, which begins to heat up. Once the EDG is up to speed, its output breaker closes in on the Essential AC busses, and the low pressure ECCS pumps (and other essential loads) are sequenced onto the busses, the pumps start and their minimum flow bypass valves open. The Automatic Depressurization System (ADS) actuation logic initates on low RPV level with ECCS pumps running, and after the 2 minute (nominal) time delay expires, the valves open and quickly reduce the vessel pressure. Once the reactor pressure decreases to their respective permissive setpoints, the injection valves for "A" loop of Core Spray and "B" loop of LPCI (based upon the "chosen" loop by loop select logic), open and allow injection to begin to the RPV. Although the "A" CS pump starts and its injection valve opens, it doesn't actually inject to the vessel, as the "A" CS piping is the assumed location for the break. Because the break is in the CS piping, the blowdown flowrate is less than an equivalent break size in the recirculation suction piping, due to the increased pressure drop through the backflow through the CS sparger/nozzles and the coolant flow path being inside the core shroud region. The injection from the 2 RHR pumps ("A" and "C") refills the lower vessel plenum area and the water level inside the core shroud rises and terminates the fuel heatup. Water level is maintained above the top of active fuel (TAF) and long-term recovery mode is entered. The analysis of the fuel and RPV response is terminated at this point.

c) Single Failure/Operator Error: The loss of Division II of 125VDC is the limiting single failure for this event. This results in the loss of HPCI, and "B" CS and "B" and "D" RHR (LPCI) pumps.

In addition, for conservatism, one ADS valve is assumed to fail to open.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures): No Operator Actions are assumed in this evaluation. Reactor scrams (high Drywell pressure), "A" EDG starts (either LOOP (undervoltage) or high

Drywell pressure) and loads ("dead" buss permissive), Feedwater and Recirculation pumps coastdown on LOOP condition, LPCI loop select logic actuates on low-low RPV level and low RPV pressure and chooses the "A" recirculation loop as the "broken" loop and closes the recirculation discharge valve in the "B" loop for injection, (assuming not operating in single loop operation), ADS initiation on low and low-low-low RPV levels, with confirmation signal on ECCS pump running, which starts 2 minute (nominal) time delay, "A" Core Spray and "A" and "C"LPCI pumps start signal on low-low-low RPV level and timers sequence the pumps onto the AC busses and the LPCI minimum flow bypass valve opens on high pump discharge pressure and close on high flow (dP), the normally-open CS minimum flow valve close on high flow (dP), "A" CS and "B" LPCI injection valves open on low RPV pressure permissive signals.

Event Category & Acceptance Criteria:

This is an Accident, due to its very low probability of occurrence.

Fuel shall remain within 10 CFR 50.46 limits as follows:

Peak Cladding Temperature (PCT) shall remain ≤ 2200 °F;
Maximum Cladding Oxidation shall not exceed 17% of the total cladding thickness;
Maximum Hydrogen Generation shall not exceed 1% metal-water reaction;
Coolable Geometry shall be maintained;
Long-term Cooling shall be ensured to remove decay heat.

There are no acceptance criteria for the RPV, as this event assumes a breach of the RPV as the initiating event.

Methods

a) Calculation Tools & Computer Codes: SAFER, GESTR-LOCA, LAMB and TASC (Note: earlier NRC-approved versions of these codes were used to do this evaluation than those currently used.)

b) Inputs:
The primary set of plant inputs used in the LOCA analysis is provided on the OPL-4 and OPL-5 forms (Note: earlier versions of these forms were used to do this evaluation than those currently used).

This evaluation was performed for fuel designs that are no longer in use (P8x8R, BP8x8R and GE6B).

c) Key Assumptions:
There is a simultaneous LOOP with the LOCA condition.
There is a single active failure - a loss of Division II of 125 VDC.
The reactor scrams immediately, ignoring Control Rod scram time. Only Decay and Sensible Heat are considered.
ECCS initiation is on RPV level. The Drywell Pressure signal is ignored.
Limiting assumptions on fuel exposure, peaking factors, power shape, initial thermal limits are made.
ECCS Injection water is assumed to be at 88 Btu/lb_m (120 °F).

Results

Nominal Case
PCT < 610 °F
Oxidation < 0.10%
Metal-water reaction << 0.032%

Appendix K Case
Because this is a non-limiting break for determining the Licensing Basis PCT, only the Nominal case was evaluated.

a) Conformance to Acceptance Criteria:

As can be seen above, the acceptance criteria are all met with significant margin. Thus, a coolable geometry can be maintained.

To show compliance to the long-term core cooling criteria, we need to demonstrate that either: 1) the core is fully reflooded to the Top of Active Fuel (TAF); OR 2) that we are reflooded to a level equal to the top of the jet pumps AND we have at least one Core Spray pump available for cooling. This is to ensure that sufficient cooling is available either by total submergence or by the combination of partial submergence and spray cooling. Because the break location is above TAF, we can satisfy the first criterion and do not need to rely upon CS for long-term cooling.

b) Sensitivities:

As with other LOCAs, the resulting PCT is directly dependent upon two things: the amount of stored energy removed before transition boiling (Initial PLHGR/MCPR and decay heat) and the duration that the core is uncovered before reflood (ECCS capacity and timing). Smaller breaks tend to remove more stored energy before transition boiling and core uncovery, so they tend to have their peak PCT later in the event after core uncovery. Because this break location is above the core, core uncovery tends to be later than the same size break in the recirculation system piping, hence has a lower PCT than those breaks.

c) Uncertainties in Results:

Because this is a non-liming break size and location, the "Upper Bound PCT (UBPCT)," which represents the 95^{th} percentile of the calculation distribution considering the uncertainties, was not calculated for this event. However, the same uncertainties in the modeling and plant parameters exist, they just have a lesser impact on the results than the more-limiting breaks.

2) Containment Response

This is a line break inside the Primary Containment (Drywell), which is bounded by the Recirculation Piping LOCA in Section 15.2.1.1. This event does not impact the Secondary Containment (Reactor Building).

3) Radiological Response

This event is bounded by the Recirculation Piping LOCA (Section 15.2.1.1) and is not specifically analyzed for dose consequences.

15.2.1.3 – Feedwater Line Break

NOTE: This evaluation was not re-performed at part of the Extended Power Uprate Program. The following evaluation is presented as Historical in nature.

1) Reactor Response

This discussion centers on the response of the fuel and the RPV to the Feedwater Line Break LOCA.

Description of Event

a) Initiator: This event is initiated by an instantaneous, non-mechanistic, double-ended, guillotine break of the Feedwater piping at the nozzle on the RPV (break area is 0.51 ft^2).

b) Sequence of Events: Coincident with the initiation of the break, a complete Loss-of-Offsite Power (LOOP) is assumed to occur and there is a loss of Division II of 125 VDC control power, in accordance with GDC 35. Reactor coolant begins to exit the vessel rapidly into the Drywell at the critical mass flux and reactor vessel water level begins to drop, as does the reactor pressure. The reactor is assumed to scram immediately. The "A" Emergency Diesel Generator (EDG) starts on the LOOP condition and all loads are stripped off the Essential AC busses. "B" EDG does not start and load due to the loss of 125 VDC control power. The non-Essential busses are lost, leading to a loss of Feedwater and a Reactor Recirculation Pump coastdown. As the RPV level reaches the various level setpoints, ECCS systems are actuated (a conservative assumption to delay injection) (Note: High Pressure Coolant Injection (HPCI) has no DC control power and does not operate), Vessel isolation signals are generated (Containment isolations are discussed in the Containment Response below), and Low Pressure Coolant Injection (LPCI) loop select logic actuates and defaults to the "B" recirculation loop, as there is no break in the recirculation system piping (i.e., no significant dP between the recirculation loops), and closes the recirculation pump discharge valve in the "B" loop. {If the plant had previously been operating in single loop recirculation mode, loop select logic would trip the running recirculation pump and effect a short time delay to allow it to coastdown prior to its selecting the "broken" recirculation loop (See Chapter 7.3.1.1.2.4 for a complete explanation of LPCI loop select logic).} Once the EDG is up to speed, its output breaker closes in on the Essential AC busses, and the low pressure ECCS pumps (and other essential loads) are sequenced onto the busses, the pumps start and their minimum flow bypass valves open. The Automatic Depressurization System (ADS) actuation logic initiates on lowering RPV level and ECCS pumps running, but because the RPV depressurizes through the break prior to the ADS 2 minute time delay expiring, the valves never actually open. Once the reactor pressure decreases to their respective permissive setpoints, the injection valves for "A" loop of Core Spray and "B" loop of LPCI (based upon the "chosen" loop by loop select logic), open and allow injection to begin to the RPV. Because the break is in the FW piping, the blowdown flowrate is less than an equivalent break size in the recirculation suction piping, due to the break location being higher in the downcomer region, so the break uncovers quickly and is mostly steam flow out of the break. While the pressure drops almost as quickly, there is less inventory loss than the recirculation line break. Thus, while the core does experience some boiling transition, it never becomes uncovered. The injection from the "A" CS and 2 RHR pumps ("A" and "C") enters the lower vessel plenum area and prevents fuel heatup. Water level is maintained above the top of active fuel (TAF) and long-term recovery mode is entered. The analysis of the fuel and RPV response is terminated at this point.

c) Single Failure/Operator Error: The loss of Division II of 125VDC is the limiting single failure for this event. This results in the loss of HPCI, and "B" CS and "B" and "D" RHR (LPCI) pumps.

In addition, for conservatism, one ADS valve is assumed to fail to open.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures): No Operator Actions are assumed in this evaluation. Reactor scrams (high Drywell pressure), "A" EDG starts (either LOOP (undervoltage) or high Drywell pressure) and loads ("dead" buss permissive), Feedwater and Recirculation pumps coastdown on LOOP condition, LPCI loop select logic actuates on low-low RPV level and low RPV pressure and chooses the "A" recirculation loop as the "broken" loop and closes the recirculation discharge valve in the "B" loop for injection, (assuming not operating in single loop operation), ADS initiation on low and low-low-low RPV levels, with confirmation signal on ECCS pump running, which starts 2 minute (nominal) time delay, “A” Core Spray and “A” and “C”LPCI pumps start signal on low-low-low RPV level and timers sequence the pumps onto the AC busses and the LPCI minimum flow bypass valve opens on high pump discharge pressure and close on high flow (dP), the normally-open CS minimum flow valve close on high flow (dP), “A” CS and “B” LPCI injection valves open on low RPV pressure permissive signals.

Event Category & Acceptance Criteria

This is an Accident, due to its very low probability of occurrence.

Fuel shall remain within 10 CFR 50.46 limits as follows:

Peak Cladding Temperature (PCT) shall remain $\leq$ 2200 °F;
Maximum Cladding Oxidation shall not exceed 17% of the total cladding thickness;
Maximum Hydrogen Generation shall not exceed 1% metal-water reaction;
Coolable Geometry shall be maintained;
Long-term Cooling shall be ensured to remove decay heat.

There are no acceptance criteria for the RPV, as this event assumes a breach of the RPV as the initiating event.

Methods

a) Calculation Tools & Computer Codes: SAFER, GESTR-LOCA, LAMB and TASC (Note: earlier NRC-approved versions of these codes were used to do this evaluation than those currently used.)

b) Inputs:
The primary set of plant inputs used in the LOCA analysis is provided on the OPL-4 and OPL-5 forms (Note: earlier versions of these forms were used to do this evaluation than those currently used).

This evaluation was performed for fuel designs that are no longer in use (P8x8R, BP8x8R and GE6B).

c) Key Assumptions:
There is a simultaneous LOOP with the LOCA condition.
There is a single active failure - a loss of Division II of 125 VDC.
The reactor scrams immediately, ignoring Control Rod scram time. Only Decay and Sensible Heat are considered.
ECCS initiation is on RPV level. The Drywell Pressure signal is ignored.

Limiting assumptions on fuel exposure, peaking factors, power shape, initial thermal limits are made.
ECCS Injection water is assumed to be at 88 Btu/lb_m (120 °F).

Results

Nominal Case
PCT < 585 °F (Note: there is no fuel heatup above the initial fuel temperature.)
Oxidation < 0.10%
Metal-water reaction << 0.032%

Appendix K Case
Because this is a non-limiting break for determining the Licensing Basis PCT, only the Nominal case was evaluated.

a) Conformance to Acceptance Criteria:
As can be seen above, the acceptance criteria are all met with significant margin. Thus, a coolable geometry can be maintained.

To show compliance to the long-term core cooling criteria, we need to demonstrate that either: 1) the core is fully reflooded to the Top of Active Fuel (TAF); OR 2) that we are reflooded to a level equal to the top of the jet pumps AND we have at least one Core Spray pump available for cooling. This is to ensure that sufficient cooling is available either by total submergence or by the combination of partial submergence and spray cooling. Because the break location is above TAF and we have one loop of CS available, we satisfy both criteria for long-term cooling.

b) Sensitivities:
As with other LOCAs, the resulting PCT is directly dependent upon two things: the amount of stored energy removed before transition boiling (Initial PLHGR/MCPR and decay heat) and the duration that the core is uncovered before reflood (ECCS capacity and timing). Smaller breaks tend to remove more stored energy before transition boiling and core uncovery, so they tend to have their peak PCT later in the event after core uncovery. Because this break location is well above the core, it behaves similar to a small steam line break and there is no core uncovery, hence has no PCT increase.

c) Uncertainties in Results:
Because this is a non-limiting break size and location, the “Upper Bound PCT (UBPCT),” which represents the 95th percentile of the calculation distribution considering the uncertainties, was not calculated for this event. However, the same uncertainties in the modeling and plant parameters exist, they just have a lesser impact on the results than the more-limiting breaks.

2) Containment Response

This is a line break inside the Primary Containment (Drywell), which is bounded by the Recirculation Piping LOCA in Section 15.2.1.1. This event does not impact the Secondary Containment (Reactor Building).

3) Radiological Response

This event is bounded by the Recirculation Piping LOCA (Section 15.2.1.1) and is not specifically analyzed for dose consequences.

15.2.1.4 – Main Steam Line Break – Inside Containment

1) Reactor Response

2014-003 NOTE: This portion of the evaluation was NOT re-performed at part of the Extended Power Uprate Program. The following evaluation is presented as Historical in nature.

This discussion centers on the response of the fuel and the RPV to the Main Steam Line (MSL) Break LOCA – Inside Primary Containment.

Description of Event

a) Initiator: This event is initiated by an instantaneous, non-mechanistic, double-ended, guillotine break of the MSL piping at the nozzle on the RPV (break area is 1.77 ft^2).

b) Sequence of Events: Coincident with the initiation of the break, a complete Loss-of-Offsite Power (LOOP) is assumed to occur and there is a loss of Division II of 125 VDC control power, in accordance with GDC 35. Reactor coolant begins to exit the vessel rapidly into the Drywell at the critical mass flux and reactor vessel water level begins to drop, as does the reactor pressure. The reactor is assumed to scram immediately. The “A” Emergency Diesel Generator (EDG) starts on the LOOP condition and all loads are stripped off the Essential AC busses. “B” EDG does not start and load due to the loss of 125 VDC control power. The non-Essential busses are lost, leading to a loss of Feedwater and a Reactor Recirculation Pump coastdown. As the RPV level reaches the various level setpoints, ECCS systems are actuated (a conservative assumption to delay injection) (Note: High Pressure Coolant Injection (HPCI) has no DC control power and does not operate), Vessel isolation signals are generated (Containment isolations are discussed in the Containment Response below), and Low Pressure Coolant Injection (LPCI) loop select logic actuates and defaults to the “B” recirculation loop, as there is no break in the recirculation system piping (i.e., no significant dP between the recirculation loops), and closes the recirculation pump discharge valve in the “B” loop. {If the plant had previously been operating in single loop recirculation mode, loop select logic would trip the running recirculation pump and effect a short time delay to allow it to coastdown prior to its selecting the "broken" recirculation loop (See Chapter 7.3.1.1.2.4 for a complete explanation of LPCI loop select logic).} Once the EDG is up to speed, its output breaker closes in on the Essential AC busses, and the low pressure ECCS pumps (and other essential loads) are sequenced onto the busses, the pumps start and their minimum flow bypass valves open. The Automatic Depressurization System (ADS) actuation logic initiates on lowering RPV level and ECCS pumps running, but because the RPV depressurizes through the break prior to the ADS 2 minute time delay expiring, the valves never actually open. Once the reactor pressure decreases to their respective permissive setpoints, the injection valves for "A" loop of Core Spray and "B" loop of LPCI (based upon the "chosen" loop by loop select logic), open and allow injection to begin to the RPV. Because the break is in the MSL piping, the blowdown flowrate is less than an equivalent break size in the recirculation suction piping, due to the break location being higher in the steam dome region, so the break flow is all steam flow out of the break. While the pressure drops more quickly, there is less inventory loss than the recirculation line break. The reactor level continues to drop and uncovers the fuel, which begins to heat up. The injection from the "A" CS and 2 RHR pumps ("A" and "C") refills the lower vessel plenum area and the water level inside the core shroud rises and terminates the fuel heatup before it reaches the initial temperature. Water level is maintained above the top of active fuel (TAF) and long-term recovery mode is entered. The analysis of the fuel and RPV response is terminated at this point.

c) Single Failure/Operator Error: The loss of Division II of 125VDC is the limiting single failure for this event. This results in the loss of HPCI, and "B" CS and "B" and "D" RHR (LPCI) pumps.

In addition, for conservatism, one ADS valve is assumed to fail to open.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures): No Operator Actions are assumed in this evaluation. Reactor scrams (high Drywell pressure), "A" EDG starts (either LOOP (undervoltage) or high Drywell pressure) and loads ("dead" buss permissive), Feedwater and Recirculation pumps coastdown on LOOP condition, LPCI loop select logic actuates on low-low RPV level and low RPV pressure and chooses the "A" recirculation loop as the "broken" loop and closes the recirculation discharge valve in the "B" loop for injection, (assuming not operating in single loop operation), ADS initiation on low and low-low-low RPV levels, with confirmation signal on ECCS pump running, which starts 2 minute (nominal) time delay, "A" Core Spray and "A" and "C"LPCI pumps start signal on low-low-low RPV level and timers sequence the pumps onto the AC busses and the LPCI minimum flow bypass valve opens on high pump discharge pressure and close on high flow (dP), the normally-open CS minimum flow valve close on high flow (dP), “A” CS and “B” LPCI injection valves open on low RPV pressure permissive signals.

Event Category & Acceptance Criteria

This is an Accident, due to its very low probability of occurrence.

Fuel shall remain within 10 CFR 50.46 limits as follows:

Peak Cladding Temperature (PCT) shall remain $\leq$ 2200 °F;
Maximum Cladding Oxidation shall not exceed 17% of the total cladding thickness;
Maximum Hydrogen Generation shall not exceed 1% metal-water reaction;
Coolable Geometry shall be maintained;
Long-term Cooling shall be ensured to remove decay heat.

There are no acceptance criteria for the RPV, as this event assumes a breach of the RPV as the initiating event.

Methods

a) Calculation Tools & Computer Codes: SAFER, GESTR-LOCA, LAMB and TASC (Note: earlier NRC-approved versions of these codes were used to do this evaluation than those currently used.)

b) Inputs:
The primary set of plant inputs used in the LOCA analysis is provided on the OPL-4 and OPL-5 forms (Note: earlier versions of these forms were used to do this evaluation than those currently used).

This evaluation was performed for fuel designs that are no longer in use (P8x8R, BP8x8R and GE6B).

c) Key Assumptions:
There is a simultaneous LOOP with the LOCA condition.
There is a single active failure - a loss of Division II of 125 VDC.
The reactor scrams immediately, ignoring Control Rod scram time. Only Decay and Sensible Heat are considered.
ECCS initiation is on RPV level. The Drywell Pressure signal is ignored.
Limiting assumptions on fuel exposure, peaking factors, power shape, initial thermal limits are made.
ECCS Injection water is assumed to be at 88 Btu/lb_m (120 °F).

Results

Nominal Case
PCT = 584 °F (Note: there is no fuel heatup above the initial fuel temperature.)
Oxidation < 0.10%
Metal-water reaction << 0.032%

Appendix K Case
Because this is a non-limiting break for determining the Licensing Basis PCT, only the Nominal case was evaluated.

a) Conformance to Acceptance Criteria:
As can be seen above, the acceptance criteria are all met with significant margin. Thus, a coolable geometry can be maintained.

To show compliance to the long-term core cooling criteria, we need to demonstrate that either: 1) the core is fully reflooded to the Top of Active Fuel (TAF); OR 2) that we are reflooded to a level equal to the top of the jet pumps AND we have at least one Core Spray pump available for cooling. This is to ensure that sufficient cooling is available either by total submergence or by the combination of partial submergence and spray cooling. Because the break location is above TAF and we have one loop of CS available, we satisfy both criteria for long-term cooling.

b) Sensitivities:
As with other LOCAs, the resulting PCT is directly dependent upon two things: the amount of stored energy removed before transition boiling (Initial PLHGR/MCPR and decay heat) and the duration that the core is uncovered before reflood (ECCS capacity and timing). Steamline breaks maintain nucleate boiling longer than recirculation line breaks, thus they remove more energy before core uncovery. So, when the core finally uncovers, there is less heatup than a recirculation line break of equivalent size.

c) Uncertainties in Results:
Because this is a non-liming break size and location, the “Upper Bound PCT (UBPCT),” which represents the 95th percentile of the calculation distribution considering the uncertainties, was not calculated for this event. However, the same uncertainties in the modeling and plant parameters exist, they just have a lesser impact on the results than the more-limiting breaks.

2) Containment Response

NOTE: This portion of the evaluation was done for Extended Power Uprate and is considered to be part of the current licensing basis for DAEC.

The response of the Primary Containment to various steam line break sizes is analyzed for the “long-term” to calculate peak drywell gas temperature, peak shell temperature, peak suppression pool temperature, and to obtain data for a composite EQ envelope of drywell (i.e., gas) temperature. The steam line breaks are the most limiting events for drywell temperature response since steam has higher energy content than liquid. Additionally, leakage between the drywell and wetwell airspace (i.e., bypassing the suppression pool) is evaluated.

This event does not directly impact the Secondary Containment (Reactor Building) and its response is not analyzed.

Description of Event

a) Initiator: The same as for the Reactor evaluation above, except the steam line break location is at the HPCI steam line.

b) Sequence of Events:

1. The plant is operating at 102% of 120% ORTP (i.e., 1950 MWt) when a steam line break occurs. There is also a concurrent loss of offsite power and only minimum diesel power is available. Reactor scrams.

2. For the first 10 minutes (600 seconds) following the accident, two LPCI pumps (in one RHR loop) and one CS pump are operating, but may not inject into the vessel due to elevated RPV pressure.

3. At 10 minutes (600 seconds), operator activates the RHR heat exchanger in the operating RHR loop. One RHR pump at 4800 gpm is re-aligned so that flow goes through the heat exchanger before returning to the containment. The other RHR pump is shutdown. This pump and heat exchanger configuration is maintained throughout the accident.

4. After 10 minutes (600 seconds), the CS pump injection to the vessel is maintained at 3100 gpm.

5. When the suppression pool temperature reaches 120ºF, operator initiates controlled vessel depressurization at 100ºF/hr using the SRVs. For those breaks that depressurize the vessel faster than 100ºF/hr, this operator action is not required.

c) Single Failure/Operator Error:
Assumed LOOP and loss of Division II 125 VDC.

d) Key Equipment Response:
Reactor scrams (assumed on High Drywell Pressure), MSIVs close, EDG starts and loads (LOOP), RHR and CS pumps start (High DW pressure), Torus-to-Drywell vacuum breakers open/close, Operators secure one RHR pump and manually load the RHR Service Water (RHRSW) pumps to initiate cooling with the RHR heat exchanger after 10 minutes, Operators initiate containment sprays, Operators performed a controlled depressurization once the suppression pool teaches the TS limit of 120ºF.

Event Category & Acceptance Criteria

This is not a Design Basis Accident (non-DBA) for the Primary Containment. Its primary purpose is to obtain data for a composite envelope of drywell (i.e., gas) temperature to be used in the Environmental Qualification (EQ) program for qualification of safety-related electrical equipment inside Primary Containment. Hence, inputs and assumptions are selected to provide conservative results for the EQ program, but may be different than those used in the DBA-LOCA case (e.g., use of heat sinks).

The Primary Containment response to the steam line breaks shall remain within the design criteria for pressure (56 psig), drywell gas temperature (340ºF), shell temperature (281°F) and suppression pool temperature (281°F). The Primary Containment is designed for 100% humidity.

Methods

a) Calculation Tools & Computer Codes: Long-Term Response: SHEX code with the HEM break flow model. (See Table 15.0-2 for complete listing, code versions and NRC acceptance.) Extended Long-Term Response (beyond one day): Simplified heat and mass transfer model.

b) Inputs: The primary set of plant inputs used in the containment analysis is provided on the OPL-4a form (Tables 15.0-6).

c) Key Assumptions:
1. The power level for the power/flow point analyzed includes an additional 2% power, consistent with Regulatory Guide 1.49.

2. The shutdown power fractions include fuel relaxation energy, metal-water reaction energy and ANS 5.1 +2sigma decay heat for fuel applicable up to GE14 with 24-month fuel cycle.

3. The steam line break sizes analyzed are 0.01 ft^2, 0.10 ft^2, 0.25 ft^2 and 1.00 ft^2.

4. HPCI is not operational (disabled by the steam line break).

5. RCIC is operational.

6. Concurrent with the postulated steam line break, a loss of offsite power occurs.

7. Only minimum diesel power is available. This results in only one RHR loop with one heat exchanger available for containment cooling, starting at 10 minutes (600 seconds).

8. RHR heat exchanger performance is based on one RHR pump (4800 gpm) and two RHRSW pumps (4080 gpm total).

9. Wetwell sprays (5% of 4800 gpm) are initiated at 10 minutes (600 seconds).

10. Drywell sprays (95% of 4800 gpm) are initiated at 10 minutes (600 seconds) for all break sizes except for the 0.01 ft^2 break size. For the 0.01 ft^2 break size, drywell sprays are initiated at 30 minutes (1800 seconds).

11. Feedwater flow to the vessel stops at 7 seconds (at the end of the pump/motor coastdown period).

12. Heat and mass transfer from the suppression pool to the wetwell airspace is determined mechanistically.

13. The initial suppression pool water volume corresponds to the TS Low Water Level (LWL) to maximize the suppression pool temperature response.

14. Initial conditions for drywell pressure, wetwell pressure and suppression pool temperature are based on limiting (e.g., analytical, TS) values.

15. Passive heat sinks in the drywell and wetwell airspace are modeled.

16. Heat transfer from the primary containment to the reactor building is conservatively neglected.

17. Drywell fan coolers are inactive.

18. Operating Core Spray and LPCI/RHR pumps have 100% of their motor horsepower rating converted to pump heat which is added either to the Reactor Pressure Vessel (RPV) liquid or suppression pool water. This assumption is used to maximize the suppression pool temperature response.

19. Main Steam Isolation Valves (MSIVs) start closing at 0.5 seconds and close completely at 3.5 seconds.

20. Only 6 wetwell-to-drywell vacuum breakers are assumed to be active.

Results

a) Conformance to Acceptance Criteria:
The peak drywell gas temperature of 330.5ºF is below the design criteria of 340ºF.

The peak shell temperature of 275.7ºF is below the design criteria of 281ºF.

The peak suppression pool temperature of 208.8°F is well below the design criteria of 281°F.

b) Sensitivities:
Drywell Bypass Leakage
Same event sequence and key assumptions as the steam line breaks above, with the following exceptions:

a. The steam line break size of 0.01 ft^2 is limiting for evaluating tolerable leakage (based on highest wetwell airspace pressure when containment sprays are activated).
b. A leakage path with an effective flow area ($A/\sqrt{K}$) of 0.11 ft^2 exists between the drywell and wetwell airspace, thus pressurizing the wetwell airspace more rapidly than the corresponding cases without bypass leakage.
c. When the wetwell airspace pressure reaches 35 psig (based on operator surveillance of plant parameters), operator is alerted to the existence of a bypass leakage path and prepares to take action.
d. Ten minutes (600 seconds) after the wetwell airspace pressure reaches 35 psig, operator activates the drywell and wetwell sprays to terminate the pressure rise.

The peak wetwell pressure of 41.9 psig is well below the design criteria of 56 psig.

The peak drywell pressure of 43.2 psig is well below the design criteria of 56 psig.

c) Uncertainties in Results:
The use of the 102% power level and conservative values for the decay heat generation (ANS 5.1-1979 +2sigma) and conservative inputs to the calculation all contribute to compensate for any uncertainties in the calculation methodology.

3) Radiological Response

This event is bounded by the Recirculation Piping LOCA (Section 15.2.1.1) and is not specifically analyzed for dose consequences.

15.2.1.5 – Main Steam Line Break – Outside Containment

2014-003

1) Reactor Response

2014-003

NOTE: This portion of the evaluation was not re-performed at part of the Extended Power Uprate Program. The following evaluation is presented as Historical in nature.

This discussion centers on the response of the fuel and the RPV to the Main Steam Line (MSL) Break LOCA – Outside Primary Containment.

Description of Event

a) Initiator: This event is initiated by an instantaneous, non-mechanistic, double-ended, guillotine break of the MSL piping inside the Turbine Building, i.e., downstream of the outboard Main Steamline Isolation Valve (MSIV) (break area is 1.77 ft^2).

b) Sequence of Events: Coincident with the initiation of the break, a complete Loss-of-Offsite Power (LOOP) is assumed to occur and there is a loss of Division II of 125 VDC control power, in accordance with GDC 35. The reactor is assumed to scram immediately. The non-Essential busses are lost, leading to a loss of Feedwater and a Reactor Recirculation Pump coastdown. Reactor coolant begins to exit the vessel rapidly at the critical mass flux through the venturi in the broken MSL. The MSIVs received a trip signal on the high steamflow condition in the broken MSL and close at the maximum stroke time (5 seconds). The initial pressure decrease is terminated due to the vessel isolation and the decay heat causes the reactor pressure to increase and the Safety/Relief Valves (SRVs) lift and arm Low-Low Set (LLS) Logic, which cycles to control reactor pressure. The High Pressure Coolant Injection (HPCI) has no DC control power, it does not operate; and, because this is an accident condition, credit is not allowed for the Reactor Core Isolation Cooling (RCIC) system. Thus, there is no high pressure coolant makeup available and vessel inventory slowly goes down with each cycling of LLS valves. Eventually the RPV level reaches the various level setpoints, ECCS systems are actuated (a conservative assumption to delay injection), Vessel isolation signals are generated (Containment isolations are discussed in the Containment Response below), and Low Pressure Coolant Injection (LPCI) loop select logic actuates and defaults to the “B” recirculation loop, as there is no break in the recirculation system piping (i.e., no significant dP between the recirculation loops), and closes the recirculation pump discharge valve in the “B” loop. {If the plant had previously been operating in single loop recirculation mode, loop select logic would trip the running recirculation pump and effect a short time delay to allow it to coastdown prior to its selecting the "broken" recirculation loop (See Chapter 7.3.1.1.2.4 for a complete explanation of LPCI loop select logic).} The "A" Emergency Diesel Generator (EDG) has started earlier on the LOOP condition and all loads are stripped off the Essential AC busses. "B" EDG does not start and load due to the loss of 125 VDC control power. Once the EDG is up to speed, its output breaker closes in on the Essential AC busses, and the low pressure ECCS pumps (and other essential loads) are sequenced onto the busses, the pumps start and their minimum flow bypass valves open. The Automatic Depressurization System (ADS) actuation logic initiates on lowering RPV level and ECCS pumps running, when the ADS 2 minute time delay expires, the valves open and depressurize the vessel. Once the reactor pressure decreases to their respective permissive setpoints, the injection valves for "A" loop of Core Spray and "B" loop of LPCI (based upon the "chosen" loop by loop select logic), open and allow injection to begin to the RPV. Because the break is in the MSL piping outside containment and is quickly isolated by the high steamflow trip, the de-pressurization transient is mild and the core does not experience any significant boiling transition until ADS blowdown, which occurs quickly and there is no significant fuel heatup. The injection from the "A" CS and 2 RHR pumps ("A" and "C") enters the lower vessel plenum area and water level is promptly restored above the top of active fuel (TAF) and long-term recovery mode is entered. The analysis of the fuel and RPV response is terminated at this point.

c) Single Failure/Operator Error: The loss of Division II of 125VDC is the limiting single failure for this event. This results in the loss of HPCI, and "B" CS and "B" and "D" RHR (LPCI) pumps.

In addition, for conservatism, one ADS valve is assumed to fail to open.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures): No Operator Actions are assumed in this evaluation. PCIS Group I isolation (High Steamline flow), MSIV closure (5 second stroke time), Reactor scram (MSIV closure), S/RVs open/close, LLS logic activates and cycles the LLS valves, "A" EDG starts (LOOP (undervoltage)) and loads ("dead" buss permissive), Feedwater and Recirculation pumps coastdown on LOOP condition, LPCI loop select logic actuates on low-low RPV level and low RPV pressure and chooses the "A" recirculation loop as the "broken" loop and closes the recirculation discharge valve in the "B" loop for injection, (assuming not operating in single loop operation), ADS initiation on low and low-low-low RPV levels, with confirmation signal on ECCS pump running, which starts 2 minute (nominal) time delay, "A" Core Spray and "A" and "C"LPCI pumps start signal on low-low-low RPV level and timers sequence the pumps onto the AC busses and the LPCI minimum flow bypass valve opens on high pump discharge pressure and close on high flow (dP), the normally-open CS minimum flow valve close on high flow (dP), "A" CS and "B" LPCI injection valves open on low RPV pressure permissive signals.

Event Category & Acceptance Criteria:

This is an Accident, due to its very low probability of occurrence.

Fuel shall remain within 10 CFR 50.46 limits as follows:

Peak Cladding Temperature (PCT) shall remain $\leq$ 2200 °F;
Maximum Cladding Oxidation shall not exceed 17% of the total cladding thickness;
Maximum Hydrogen Generation shall not exceed 1% metal-water reaction;
Coolable Geometry shall be maintained;
Long-term Cooling shall be ensured to remove decay heat.

There are no acceptance criteria for the RPV, as this event assumes a breach of the RPV as the initiating event.

Methods

a) Calculation Tools & Computer Codes: SAFER, GESTR-LOCA, LAMB and TASC (Note: earlier NRC-approved versions of these codes were used to do this evaluation than those currently used.)

b) Inputs:
The primary set of plant inputs used in the LOCA analysis is provided on the OPL-4 and OPL-5 forms (Note: earlier versions of these forms were used to do this evaluation than those currently used).

This evaluation was performed for fuel designs that are no longer in use (P8x8R, BP8x8R and GE6B).

c) Key Assumptions:
There is a simultaneous LOOP with the LOCA condition.
There is a single active failure - a loss of Division II of 125 VDC.
The reactor scrams immediately, ignoring Control Rod scram time. Only Decay and Sensible Heat are considered.
ECCS initiation is on RPV level. The Drywell Pressure signal is ignored.
Limiting assumptions on fuel exposure, peaking factors, power shape, initial thermal limits are made.
ECCS Injection water is assumed to be at 88 Btu/lb_m (120 °F).

Results

Nominal Case
PCT = 584 °F (Note: there is no fuel heatup above the initial fuel temperature.)
Oxidation < 0.10%

Metal-water reaction << 0.032%

<u>Appendix K Case</u>
Because this is a non-limiting break for determining the Licensing Basis PCT, only the Nominal case was evaluated.

a) Conformance to Acceptance Criteria:
As can be seen above, the acceptance criteria are all met with significant margin. Thus, a coolable geometry can be maintained.

To show compliance to the long-term core cooling criteria, we need to demonstrate that either: 1) the core is fully reflooded to the Top of Active Fuel (TAF); OR 2) that we are reflooded to a level equal to the top of the jet pumps AND we have at least one Core Spray pump available for cooling. This is to ensure that sufficient cooling is available either by total submergence or by the combination of partial submergence and spray cooling. Because the break location is above TAF and we have one loop of CS available, we satisfy both criteria for long-term cooling.

b) Sensitivities:
As with other LOCAs, the resulting PCT is directly dependent upon two things: the amount of stored energy removed before transition boiling (Initial PLHGR/MCPR and decay heat) and the duration that the core is uncovered before reflood (ECCS capacity and timing). Steamline breaks maintain nucleate boiling longer than recirculation line breaks, thus they remove more energy before core uncovery. So, when the core finally uncovers, there is less heatup than a recirculation line break of equivalent size.

c) Uncertainties in Results:
Because this is a non-liming break size and location, the “Upper Bound PCT (UBPCT),” which represents the 95^{th} percentile of the calculation distribution considering the uncertainties, was not calculated for this event. However, the same uncertainties in the modeling and plant parameters exist, they just have a lesser impact on the results than the more-limiting breaks.

2) <u>Containment Response</u>

The pipe break is outside Primary Containment, so there is no initial impact on the Drywell. Once the MSIVs close, causing the SRVs to open and LLS valves to cycle, there will be a gradual heatup of the Suppression Pool. The Operators will put RHR into Suppression Pool Cooling mode after 10 minutes into the event, which is prior to the ADS blowdown. However, this event does not challenge the Primary Containment. Thus, it is not analyzed.

This event does not directly impact the Secondary Containment (Reactor Building), as the break location is assumed to be in the Turbine Building.

3) Radiological Response

2014-003 NOTE: This portion of the evaluation was done for the Extended Power Uprate and is considered to be part of the current licensing basis for DAEC.

Description of Event

a) Initiator:
The postulated accident assumes a double ended break (DEB) of one main steam line, with the reactor operating at 1950 MWt (102% of 1912 MWt), outside the secondary containment with displacement of the pipe ends that permits maximum blowdown rates.

b) Sequence of Events:

Time	Event
0	Accident Begins (Coolant Release)
10.5 sec	Break isolated by MSIV closure CR shifts to Emergency Ventilation Mode
30 days	End of analyzed event scenario

c) Single Failure/Operator Error:
Safety-related, systems and components that provide active functions to prevent or mitigate radiological releases are designed with to be single-failure proof. Operator manual actions are not credited during the initial 10 minutes of an accident.

d) Key Equipment Response:
Main Steam Isolation terminates the break.

No other structures, systems, or components have been credited for mitigation functions for the MSLB design basis accident.

Event Category and Acceptance Criteria

With regard to radiological consequences, a Main Steamline Break Outside Containment is a design basis accident. Since no fuel damage is expected to occur, the radiological consequences of the MSLB are limited to the effects of coolant radiation sources. Due to the large mass and energy release of a MSLB it is a bounding analysis for the radiological effects of other High Energy Line Breaks (HELB) involving steam releases.

The acceptance criteria from 10 CFR 50.67 are:

- An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

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

The radiological consequences to personnel in the Technical Support Center are evaluated using the acceptance criterion issued in Section 8.2. of Generic Letter 83-11 "Supplement 1 to NUREG-0737 – Requirements for Emergency Response Capability," dated December 17, 1982.

- Adequate radiation protection is provided to assure that radiation exposure to any person working in the Technical Support Center would not exceed 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Per RG 1.183, for a main steam line break with an assumed pre-accident iodine spike corresponding to the maximum concentration stated in the Technical Specifications (2 μCi/gm), the calculated dose should not exceed the guideline values of 10CFR50.67 (i.e., 25 rem TEDE at the EAB and LPZ and 5 rem TEDE at the CR).

Per RG 1.183, for a main steam line break with an assumed iodine concentration corresponding to the equilibrium value for continued full power operation stated in the Technical Specifications (0.2 μCi/gm), the doses should not exceed a small fraction (i.e., 10 percent) of the 10CFR50.67 guideline values (i.e., 2.5 rem TEDE at the EAB and LPZ and 0.5 rem TEDE at the CR).

Methods

The radiological consequences of design basis accidents were analyzed using the methods and guidelines of RG 1.183 “Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors.”

a. Calculation Tools and Computer Codes:

Atmospheric Dispersion

Atmospheric dispersion factors (CHI/Q’s) were calculated with the ARCON96 computer code. The ARCON96 code calculates relative concentrations in plumes from nuclear power plants at the control room and Technical Support Center air intakes and accounts for the effects of building wakes. The ARCON96 code was verified and validated in accordance with DAEC Software Quality Assurance Program. See the discussion of atmospheric dispersion analysis inputs and results in Section 15.2.1.1.

2014-008 Atmospheric dispersion factors for offsite dose consequences were calculated with the PAVAN code. The PAVAN code was verified and validated in accordance with the DAEC Software Quality Assurance Program.

Radiological Dose

2014-008 The RADTRAD computer code is a radiological consequence analysis code used to estimate radiological source transport, removal, decay, and post-accident doses
2014-008 at plant offsite locations, the control room, and Technical Support Center. The code was verified and validated in accordance with DAEC Software Quality
2014-008 Assurance Program and/or an approved vendor program.

b) Key Assumptions and Inputs:

1. The break mass released includes the line inventory plus the system mass released through the break prior to isolation.

2. Break isolation was assumed in 10.5 seconds. This assumption is consistent with the isolation time used in evaluation of the HELB pressure, temperature, pipe whip and jet impingement for MSLBs. It is a conservatively longer isolation time than the expected 3 to 5 second isolation based on MSIV technical specifications and testing results. This also resulted in a conservatively large radiological release for analysis.

3. The coolant activity is released over the 10.5 second MSIV closure time to the turbine building.

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4. It is assumed that all of the activity is released to the environment at a turbine building leakage rate of 2.4E7 %/day. This effectively sweeps all radioactivity from the building as quickly as it is released and is consistent with a puff release.

5. The release to the environment is unfiltered, and the iodine plateout is not credited.

6. Reactor coolant specific activities of 2 μCi/gm for a pre-accident Iodine spike and 0.2 μCi/gm for the technical specification equilibrium iodine concentration were analyzed.

7. The iodine released from the main steam line is assumed to consist of 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic.

8. Iodine carryover fraction from reactor water to steam is assumed to be 8% (conservative upper bound).

9. No fuel damage (perforations and/or failures) is postulated.

10. All the cesium activity remains in the reactor coolant, none is released to the steam.

11. The Emergency Mode outside air makeup to the CR will be modeled as 900 cfm (1000 cfm ± 10%) to maximize calculated operator dose. The minimum outside air intake value is more conservative because it reduces the filtered intake that dilutes the source term within the CR.

12. The CR unfiltered air inleakage values that are conservatively considered for the Supplemental MSLBA Cases are bounding values of 1000 cfm and 0 cfm. The minimum unfiltered air inleakage value is more conservative because it minimizes the source term dilution rate in the CR by outside air that has a lower radioactivity concentration following break isolation.

13. Prior to isolation, the activity is assumed to enter the control room at the normal ventilation rate of 3150 cfm.

14. Emergency Mode Ventilation flow rate was assumed at 900 cfm per the low limit in Technical Specification 3.7.4.

15. CR isolation time of 10.5 seconds was assumed to maximize dose to the CR operators. Longer or shorter isolation times or crediting manual isolation would reduce control room dose.

Results

	Case	EAB[(1)] (rem TEDE)	LPZ[(2)] (rem TEDE)	CR Dose[(3,4)] (rem TEDE)
2014-008	2 μCi/gm dose equivalent I-131	0.79	0.19	2.61
2014-008	0.2 μCi/gm dose equivalent I-131	7.9E-02	1.9E-02	0.26

1. Worst 2-hour integrated dose.
2. 30-day integrated dose.
3. Assumes a conservative unfiltered inleakage of 0 cfm. For the CR MSLBA calculations, a lower inleakage is conservative because the source is a limited release over 10.5 seconds.
4. Assumes CR Emergency Mode Ventilation rate of 900 cfm

a) Comparison to Acceptance Criteria:

		Case	EAB (rem TEDE)	LPZ (rem TEDE)	CR Dose (rem TEDE)	TSC Dose
2014-008	2 μCi/gm dose equivalent I-131	Analysis Results	0.79	0.19	2.61	Not Calculated
		Regulatory Limit (10 CFR 50.67)	25	25	5	5
2014-008	0.2 μCi/gm dose equivalent I-131	Analysis Results	7.9E-02	1.9E-02	0.26	Not Calculated
		Regulatory Guideline (RG 1.183)	2.5	2.5	0.5	0.5

TSC dose analysis was not performed based on the results of the control room dose analysis and comparison to limiting event analysis for the LOCA. (See 15.2.1.1.) The radiation release and transport paths are similar. No credit for ventilation isolation is needed. Therefore it was concluded that TSC doses would also be bounded by the TSC dose consequences of the LOCA event.

b) Sensitivities:

2014-008 Sensitivity analyses of the post-MSLB dose to CR operators were performed at assumed unfiltered CR inleakage rates of 0 and 1000 cfm. These analyses confirmed that the assumption of control room isolation in 10.5 seconds maximizes calculated dose consequences.

c) Uncertainties:
The single largest source of uncertainty is the value of coolant activity concentration assumed in the analysis. The assumed levels are significantly greater than expected levels from DAEC operating history, including operation with minor fuel leakers.

Conclusions

a) Statement of Acceptability
The DAEC radiological consequences for the MSLB Design Basis Accident analysis demonstrate that the isolation of the break is sufficient to limit dose to workers and the public satisfy all acceptance criteria.

b) Conservatisms/Margins
Main Steam Isolation Valve normal closure times are from 3 to 5 seconds. The assumption that isolation is delayed until 10.5 seconds introduces significant margin into the radiological consequences analysis.

Assumed coolant source term activities assumed in the analysis (2 μCi/gm iodine spike and 0.2 μCi/gm) significantly overstate the concentration of coolant radioiodine based on DAEC operating history.

c) Limiting Event
The radiological consequences of a Main Steamline Break Outside Containment are non-limiting compared to the consequences of the DBA LOCA and CRDA.

15.2.2 – INSTRUMENT LINE BREAKS

NOTE: This evaluation was not re-performed at part of the Extended Power Uprate Program. The following evaluation is presented as Historical in nature.

1) Reactor Response

The impact of this event on the Reactor and Fuel is bounded by the evaluation of Small Breaks in the Recirculation System piping in Section 15.2.1.1.

2) Containment Response

This is a line break outside the Primary Containment (Drywell); thus, this event only impacts the Secondary Containment (Reactor Building).

Description of Event

a) Initiator: A non-mechanistically caused piping break in a reactor vessel instrument line outside the Primary Containment.

b) Sequence of Events: A reactor vessel instrument line breaks in the Reactor Building. The resulting blowdown causes the temperature and pressure in the building to increase. The Standby Gas Treatment System (SGTS) starts on a Reactor Building Exhaust Shaft - High Radiation Signal. The Operators detect the break and bring the reactor to Cold Shutdown at 3.5 hours into the event.

c) Single Failure/Operator Error: None assumed.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures): SGTS starts on a Reactor Building Exhaust Shaft - High Radiation Signal. The Operators detect the break and bring the reactor to Cold Shutdown.

Event Category & Acceptance Criteria:

This is an Accident.

Resulting pressure within the Reactor Building shall be within the design values of 7 inches H_2O pressure.

Methods

a) Calculation Tools & Computer Codes: MIMIC, used to solve simultaneous, non-linear differential equations (Note: although this computer code was utilized in this analysis, there is no record of its review and acceptance by the NRC.)

The model used to calculate the pressure and temperature response consisted of a volume, assumed to be the total free volume of the reactor building, into which reactor water is blown down from reactor temperature and pressure to atmospheric pressure. Mass and energy are removed from the volume by the standby gas treatment system and by leakage. Mass balance equations were written for the mass of vapor and mass of air in the building atmosphere. A heat balance equation was written for the atmosphere to calculate temperature, and the pressure was calculated from the mass inventory, leakage, temperature, and volume.

b) Inputs:

Instrument lines have 0.25 in. orifices.

Mass flow rate of vapor from blowdown (constant).	62.5 lbm/min
Mass of air in the building	1.285 x 105 lbm
Mass of vapor in the building at 50% r.h.	1938 lbm
Atmospheric (exterior) pressure (constant)	2120 lb/ft^2
SGTS flow rate (constant)	1263 ft^3/min
Total building free volume	1.82 x 106 ft^3
Temperature of air in building	90°F
Leakage at 0.25 in. H2O	1263 ft^3/min
Blowdown flow rate (vapor + water)	164 lbm/min
Quality of blowdown (constant)	0.38

c) Key Assumptions:

The standby gas treatment system starts automatically on high reactor building ventilation activity. Normal building ventilation is assumed to not be running.

Leakage from the building is proportional to the square root of pressure differential (1263 cfm at 1/4 in. H_2O).

No heat transfer to building.

No friction losses in instrument lines.

Blowdown flow rate is the maximum for a two-phase mixture according to Moody (8000 lbm water/sec-ft^2 at 1050 psia).

No credit is taken for the Excess Flow Check Valves in the instrument lines.

Reactor pressure is constant at 1050 psia throughout the event.

Building pressure is atmospheric at the beginning of the event.

Quality of blowdown is assumed constant at 0.38, calculated from an energy balance.

<u>Results</u>

a) Conformance to Acceptance Criteria:

The results of this analysis are shown in Figure 15.2-9, which shows that after 3.5 hr (the duration of the detection and cooldown sequence) of continuous blowdown, the temperature and pressure in the reactor building are 110°F and 0.94 in. H_2O, respectively.

b) Sensitivities:

Unknown

c) Uncertainties in Results:

Unknown

Conclusion

a) Statement of Acceptability:
The structural integrity of the building is ensured as the resulting pressure is well within the design value.

b) Known Conservatisms/Margins:
If normal building ventilation is considered to be operating, the equilibrium reactor building pressure would be lower because of the greater steam removal rate.

SGTS flowrate is approximately 30% of rated for one train. If rated flow were used, the pressure would be lower because of the higher removal rate.
Reactor pressure is assumed constant in this event, which is conservative, as the pressure would eventually begin to decrease with the loss of vessel inventory. This, in turn, would reduce the blowdown proportionally.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
This is a non-limiting Accident. It is a non-limiting event for Environmental Qualification (EQ) conditions within the Reactor Building, as it is bounded by other, larger High Energy Line Breaks (HELBs), such as RWCU.

3) Radiological Response

This event is bounded by the Main Steamline Break – Outside Containment (Section 15.2.1.5).

The potential offsite radiological exposure attributable to postulated rupture of an instrument line has been investigated. It was conservatively assumed that for fission product concentrations equivalent to a 100,000 µCi/sec offgas rate at 30-min delay, 100% of noble gases associated with the water and 30% of the iodines (the fraction associated with water flashing to steam) are immediately available to the atmosphere via the normal ventilation system, although secondary containment isolation and standby gas treatment system operation may be expected. Therefore, no credit for filtration, plateout, or other deposition of iodine was assumed. Releases were assumed to occur at ground level to the reactor building wake under Pasquill Type F diffusion conditions and a wind speed of 1.0 m/sec. The resulting whole-body and thyroid doses at the site boundary are 2 mrem and 0.27 rem, respectively. These doses are well below the guideline values of 10 CFR 100.

15.2.3 – RECIRCULATION PUMP SEIZURE ACCIDENT

1) Reactor Response

Description of Event

a) Initiator: The recirculation pump seizure assumes instantaneous stoppage of the pump motor shaft of the operating recirculation pump while operating in Single Loop Operation (SLO).

b) Sequence of Events (NOT a time line): The plant is operating in SLO, at the analyzed* maximum power/flow point on the MELLLA boundary (66.8% core power/53% core flow), when the operating recirculation pump seizes instantaneously. The drive flow in the active loop jet pumps quickly drops to zero and the reverse flow in the inactive loop jet pumps turns over and becomes forward flow again, with the two loops equilibrating at the natural circulation point (~30% core flow). The drop in core flow causes the core void fraction to increase, which in turn, causes a rapid decrease in core power.

   *The analyzed power/flow point was set as part of EPU. This evaluation was not re-performed as part of ICF. Administrative limits are in place to ensure operation remains within the analyzed domain.

c) Single Failure/Operator Error (as applicable): None, beyond the initial pump seizure.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures): Feedwater Control system reacts to the loss of recirculation flow and maintains vessel water level. Pressure Control system reacts to the lower steamflow/turbine inlet pressure and stabilizes reactor pressure at a lower condition.

Event Category & Acceptance Criteria

While this event is considered to be an Accident, based upon its probability of occurrence, it uses the more conservative acceptance criterion of the Safety Limit MCPR (SLMCPR) that is used in the evaluation of Abnormal Operational Transients.

In order to reduce the impact of potential cycle-dependent SLMCPR variations, the following bounding approach can be adopted:

2012-020
2017-001

- For a SLO SLMCPR of ≤ 1.12, the corresponding Two-Loop Operation (TLO) Operating Limit MCPR (OLMCPR) is 1.43 for GNF2.
- If SLO SLMCPR is > 1.12, or, TLO OLMCPR is < 1.43 for GNF2, then a cycle-specific adjustment will be needed.

Note: MOP and TOP are not evaluated, as this event does not produce an overpower condition.

Methods

a) Calculation Tools & Computer Codes: Primary Codes: PANACEA (GEMINI methods) for the steady state initial conditions; ODYN for the transient response of the pump seizure event; and; TASC for the calculation of the time dependent, single hot channel critical power response (see Table 15.0-1 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
Core and Fuel Designs per the FRED form (See Section 15.0.7)
BOC, MOC and EOC core exposure points are evaluated.
OPL-3 Form (See Table 15.0-3)

c) Key Assumptions:
Maximum operating condition while in Single Loop Operation is 66.8% rated thermal power and 53% rated core flow, on the MELLLA boundary. Note: The analyzed maximum power/flow point was set as part of EPU. This evaluation was not re-performed as part of ICF. Administrative limits are in place to ensure operation remains within the analyzed domain.

2012-020 A multiplier of 0.85 is applied to the calculated void coefficient at each exposure point.
No Scram occurs.

Results

a) Comparison to Acceptance Criteria:

See current cycle's SRLR for evaluation against the above conformance check.

b) Known Sensitivities:
The higher the initial core flow at the beginning of the event, the greater the change in core flow and subsequent reduction in MCPR. The event is less sensitive to the initial power level.
The results of this event are most sensitive to the core average void fraction, which is direct result of the axial power shape. The more bottom peaked the power shape, the larger the change in MCPR.

c) Uncertainties in Results:
The plant performance is analyzed to 95%/95% confidence levels using GEMINI methods.

Conclusion

a) Statement of Acceptability:

Either the acceptance criterion is satisfied or the SLO OLMCPR is adjusted.

b) Known Conservatisms/Margins:
Use of the multiplier on the void coefficient adds conservatism to the evaluation.
Use of the $MCPR_f$ curve for SLO is conservative, as the maximum increase in runout flow in SLO is less than half of the potential flow increase in TLO.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
This is a non-limiting accident. It is only re-analyzed when a significant change in
2012-020 fuel design occurs (e.g., GE14 to GNF2). However, a cycle-specific conformance check is made to ensure that the bounding evaluation remains valid for the specific operating cycle.

2) Containment Response
None. This is a fuel response event only.

3) Radiological Response
None. Because the fuel never violates the Safety Limit MCPR, there is no assumed fuel failure or radiological release.

15.2.4 – CONTROL ROD DROP ACCIDENT

1) Reactor Response

Description of Event

a) Initiator: A control blade becomes decoupled from its control rod drive mechanism, and sticks inside the core at the fully-inserted position, such that it does not follow its drive when withdrawn from the core. Later in the startup sequence, when this stuck control rod is at its maximum possible control rod worth, breaks free and drops at its maximum velocity, causing a prompt, supercritical reactivity event.

b) Sequence of Events (NOT a time line): The CRDA scenario postulates the following:

(a) Reactor is at a control rod pattern corresponding to maximum incremental rod worth.

(b) Rod pattern control systems (Rod Worth Minimizer, Rod Sequence Control System or Rod Pattern Controller) or operators are functioning within the constraints of the Banked Position Withdrawal Sequence (BPWS). The control rod that results in the maximum incremental reactivity worth addition at any time in core life under any operating condition while employing the BPWS becomes decoupled from the control rod drive.

(c) Operator selects and withdraws the drive of the decoupled rod along with the other required control rods assigned to the Banked–position group such that the proper core geometry for the maximum incremental rod worth exists.

(d) Decoupled control rod sticks in the fully inserted position.

(e) Control rod becomes unstuck and drops at the maximum velocity determined from experimental data (3.11 feet per second).

(f) Reactor goes on a positive period and initial power burst is terminated by the Doppler reactivity feedback.

(g) APRM 120% power signal scrams reactor (conservative; in startup mode APRM scram would be operative + IRM).

(h) Scram terminates accident.

c) Single Failure/Operator Error (as applicable): Operator does not acknowledge the overtravel alarm on the fully-withdrawn control rod that is de-coupled.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures): APRM High Flux Scram (120%)

Event Category & Acceptance Criteria:

This is a Design Basis Accident.

Peak Fuel Enthalpy < 280 cal/gm

Methods

a. Calculation Tools & Computer Codes: Primary Code: PANACEA (See Table 15.0-1 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
Core and Fuel Designs per the FRED form (See Section 15.0)
Control Rod withdrawal pattern follows the BPWS sequence.

c) Key Assumptions:
No credit for void reactivity feedback.
No credit is taken for either the APRM Flux Scram in Startup (15%) or the IRM Flux Scram.

Results

a) Comparison to Acceptance Criteria:
Peak Fuel Enthalpy = 162 cal/gm

b) Known Sensitivities:
Control Rod Worth and Scram Reactivity are the key parameters of interest that directly affect the results.

c) Uncertainties in Results:
Control Rod Worth is calculated to give results at the 95% probability and 95% Confidence Level (95%/95% Statistical Confidence).

Conclusion

a) Statement of Acceptability:
The peak fuel enthalpy is well within the acceptance limits.

b) Known Conservatisms/Margins:
Credit for void feedback would significantly reduce the severity of the event.
Use of the Tech Spec Scram times is also conservative.
Credit for Scram by either the APRM Flux Scram in Startup or the IRM Flux Scram would terminate this event would not significantly impact the results, as the flux transient is very quick and the primary mechanism for terminating the event is Doppler Feedback, not Scram Reactivity.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
This is a Design Basis Accident. It is not re-analyzed as part of the reload licensing process.

2) Containment Response

Because this event occurs during the startup sequence with the Main Steamline Isolation Valves open. There is no impact on either the Primary or Secondary Containment.

3) Radiological Response

Description of Event

a) Initiator: The plant design basis Control Rod Drop Accident (CRDA) involves the rapid removal of a high worth control rod resulting in a reactivity excursion that encompasses the consequences of any other postulated CRDA.

b) Sequence of Events:

Time	Event
0	Accident Begins (Control Rod drops)
5 seconds	Fuel damage and fission product release. 1200 rods gap failure, 0.77% of these also undergo fuel melt. Fission products are instantaneously transported to the Main Condenser via the Main Steam Lines.
10 minutes	Operators manually secure the Mechanical Vacuum Pump after receipt of alarm on high radiation at the offgas stack.
24 hours	Release ends

c) Single Failure/Operator Error:
Operator action to trip the mechanical vacuum pump, if operating, is sufficient to keep radiological consequences within regulatory guidelines.

d) Key Equipment Response:
Offgas Stack high radiation alarm.

Condenser provides holdup and plateout of a portion of the radiological source term.

Event Category and Acceptance Criteria

A Control Rod Drop Accident (CRDA) is a design basis accident. Radiological consequences analysis is required.

The acceptance criteria from 10 CFR 50.67 are:

- An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

NOTE: Per the guidelines of RG 1.183 for a Control Rod Drop Accident, the doses should not exceed 25% of the above regulatory limits (i.e., 6.3 rem TEDE) at the EAB or LPZ in 24 hours.

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

The radiological consequences to personnel in the Technical Support Center are evaluated using the acceptance criterion issued in Section 8.2. of Generic Letter 83-11 "Supplement 1 to NUREG-0737 – Requirements for Emergency Response Capability," dated December 17, 1982.

- Adequate radiation protection is provided to assure that radiation exposure to any person working in the Technical Support Center would not exceed 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

<u>Methods</u>

a. Calculation Tools and Computer Codes:

<u>Source Term Inventory</u>
The ORIGEN2 code (Reference Table 15.0-2), which is a widely used Oak Ridge National Laboratory code used in the production and decay of radioactive material, was used in the calculation of plant-specific fission product inventories which bound the effect of two year fuel cycles, power operation at 1950 MWt (102% of 1912 MWt), and anticipated fuel designs. This analysis was performed by General Electric (GE) under the GE software quality assurance program.

<u>Atmospheric Dispersion</u>

Atmospheric dispersion factors (CHI/Q's) were calculated with the ARCON96 computer code. The ARCON96 code calculates relative concentrations in plumes from nuclear power plants at the control room and Technical Support Center air intakes and accounts for the effects of building wakes. The ARCON96 code was verified and validated in accordance with DAEC Software Quality Assurance Program.

Atmospheric dispersion factors for offsite dose consequences were calculated with the PAVAN code. The PAVAN code was verified and validated in accordance with DAEC Software Quality Assurance Program.

Radiological Dose

2014-008 The RADTRAD computer code is a radiological consequence analysis code used to estimate radiological source transport, removal, decay, and post-accident doses 2014-008 at plant offsite locations, the control room, and Technical Support Center. The code was verified and validated in accordance with DAEC Software Quality 2014-008 Assurance Program and/or an approved vendor program.

b) Key Assumptions and Inputs:

1. Assumptions for core inventory and the release of radioactivity from the fuel is per USNRC Regulatory Guide No. 1.183 Revision 0, “Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors”, July 2000, Appendix C.1. The release from the breeched fuel clad is based on 10% of the core noble gas and halogen inventory being in the gap and the estimate of clad damage. The release from melted fuel is based on release of 100% of the core noble gasses and 50% of the core radioiodine and the percent of fuel that melts. Release of other fission products from the fuel is per RG 1.183 Table 3 for the gap release and from RG 1.183 Table 1 (Early in Vessel) for the pellet release.
2. 1200 fuel rods were assumed damaged.
3. 0.77 % of the damaged rods experience clad melting.
4. The inventory of fission products in the reactor core and available for release to the containment is based on the maximum full power operation of the core times 1.02 the current licensed rated thermal power (1950 MWt).
5. 2014-008 Fission product inventory is adjusted for the radial peaking factor (1.55).
6. The activity released from the fuel gap and from fuel melting is assumed to be instantaneously mixed in the reactor coolant within the pressure vessel.
7. Credit is not assumed for partitioning in the pressure vessel or by removal by the steam separators.
8. Of the activity released from the reactor coolant within the pressure vessel, 100% of the noble gases, 10% of the iodines, and 1% of the remaining radionuclides are assumed to reach the turbine and the condensers.
9. Of the activity that reaches the turbine and the condenser, 100% of the noble gases, 10% of the iodines, and 1% of the particulate radionuclides are available for release to the environment. The turbine and condensers

leak to the atmosphere as a ground level release at a rate of 1% per day for a period of 24 hours, at which time the leakage is assumed to terminate.

10. No credit is assumed for dilution or holdup in the Turbine Building.

11. Radioactive decay during holdup in the turbine and the condenser is assumed.

12. The release from the reactor coolant within the pressure vessel is assumed to consist of 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic.

13. The release from the turbine and condenser is assumed to be 97% elemental and 3% organic.

14. No automatic or manual isolation of the Control Room or TSC is assumed.

15. As a conservative assumption, the MSL drain flowpath to the condenser modeled and all the fission product release is transported directly to the turbine and condenser through the larger main steam piping.

16. The release path from the Recirculation Sample Valves is also not specifically modeled; assuming all the fission products are released via the Condenser - Mechanical Vacuum Pump (MVP) pathway.

17. The following volumes and flowrates were used:
    - Condenser volume of 55,000 cu. ft.
    - Control Building Volume of 155,000 cu. ft.
    - TSC Volume of 68,300 cu. ft.
    - Control Building Intake flow rate of 3150 cfm.
    - TSC Intake flow rate of 900 cfm.
    - Mechanical Vacuum Pump operation is modeled at a constant 1800 cfm (design flow capacity).

18. When the CRDA occurs, the source term from the damaged fuel is transferred to the reactor coolant over a 5 second period and is immediately transported to the main condenser. This assumption conservatively ignores transport time and assumes the full source term is transported to the condenser during the 5 second release period.

19. The MVP is assumed to be operating to draw a vacuum in the condenser until manually isolated by the Operators at 10 minutes after the event initiation.

20. The MVP pumps the contents of the Main Condenser through a 1.75-minute delay line to the Offgas Stack where it is released to the environment.

21. Once the MVP is secure, the release continues due to leakage from the condenser at 1 % volume change per day for the remainder of the 24 hour release duration. All condenser leakage is immediately released to the

environment via direct leakage out of the turbine building without holdup, plateout, or dilution.

22. No credit is taken for isolation or filtration systems for the CR or TSC. Normal ventilation is assumed for the duration of the event. Control room ventilation also assumes 1000 cfm of unfiltered in-leakage.

23. The following are the X/Q's for the various release paths and dose receptor locations:

DAEC EAB
x/Q Values (sec/m^3)

Time Period	Ground Level	Elevated
0-0.5 hrs	$5.57x10^{-4}$	$7.03x10^{-5}$ (fumigation)
0.5-2 hrs	$5.57x\ 10^{-4}$	$6.95x10^{-6}$

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DAEC LPZ
x/Q Values (sec/m^3)

Time Period	Ground Level	Elevated
0-0.5 hrs	$1.34x10^{-4}$	$3.15x10^{-5}$ (fumigation)
0.5-2 hrs	$1.34x10^{-4}$	$6.69x10^{-6}$
2-8 hrs	$6.43x10^{-5}$ *	$3.58x10^{-6}$ *
8-24 hrs	$4.46x10^{-5}$	$2.61x10^{-6}$

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DAEC CR and TSC
x/Q Values (sec/m^3)

	CR		TSC	
Time Period	Ground Level	Elevated	Ground Level	Elevated
0-2 hrs	$1.48x10^{-3}$	$1.68x10^{-5}$	$2.14x10^{-3}$	$1.37x10^{-5}$
2-8 hrs	$1.27x10^{-3}$	$3.75x10^{-7}$	$1.86x10^{-3}$	$2.16x10^{-7}$
8-24 hrs	$5.56x10^{-4}$	$1.33x10^{-7}$	$8.44x10^{-4}$	$8.00x10^{-7}$

As this is a short-term release of fission products (i.e., the fuel failure is prompt and abrupt), the entire fission product inventory is assumed to be transported to the main condenser within 5 seconds of event initiation. Over the release duration, there is both an elevated release (during MVP operation), and a subsequent ground level release, via condenser leakage and turbine building leakage (after MVP operation is secured).

The results from this analysis are as follows:

Dose Receiver Location (REM TEDE)			
EAB	LPZ	CR	TSC
3.03	1.36	0.51	0.58

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a) Comparison to Acceptance Criteria:

Accident Type	**Exclusion Area Boundary (2 hr)**	**Low Population Zone (24 hr)**	**Control Room (24 hr)**	**TSC (24 hr)**
	TEDE (rem)			
Total Dose	3.03	1.36	0.51	0.58
CRDA Regulatory Limits	6.25	6.25	5	5

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b) Sensitivities:

Sensitivity analysis was performed for control room ventilation rates, unfiltered inleakage rates and control room isolation times. The analysis described here uses assumptions for these sensitivities that maximize calculated dose consequences.

Several scenarios were analyzed with varying MVP isolation times. As expected, the offsite dose results were most sensitive to the length of time the MVP operates; the longer the MVP operates, the longer the elevated release period. Because of the limited nature of the release (quantity and duration), sustained MVP operation evacuates the condenser of fission products to the point where almost all, ~99.9%, of the offsite dose occurs within the first 2 hours of MVP operation.

To bound this event, specifically to demonstrate the scenario where the Operator fails to take the assumed actions to manually isolate the MVP, a case was performed assuming 24-hour MVP operation. The 24-hour duration is consistent with the guidelines of App. C of RG 1.183. The results of that case follow:

Dose (REM TEDE)			
EAB	LPZ	CR	TSC
7.85	3.74	0.94	0.73

2014-008

The dose rates to the LPZ, Control Room, and Technical Support Center remain within regulatory guidelines. The limiting 2-hour EAB dose does exceed the
2014-008 regulatory guidelines of RG 1.183, but by only 25.6%. However, this value remains well below the regulatory limit of 25 REM TEDE in 10 CFR 50.67. This is considered to meet the “well within the exposure guideline values” criterion
2014-008 stated in SRP Chapter 15.4.9, App. A. A value exceeding the regulatory guidance was deemed to be acceptable as documented in Reference 59 (“NRC staff considers such a deviation as insignificant from the point of view of the intent of the regulatory guidance.”).

More importantly, this case demonstrates that the assumption of the Operator response time of 10 minutes is not critical to achieving acceptable results (i.e., not “time critical”) and the associated instrumentation used to detect the offsite release need not be upgraded to a Type A variable, per RG 1.97.

c) Uncertainties:
The risk potential of a CRDA is extremely low. The risk of a CRD during MVP operation is even more unlikely as MVP operation is limited to short periods during startup and shutdown. Velocity limiters, Rod Worth Minimizer, Banked Position Withdrawal Sequence, and Reduced Notch Worth procedure (GE SIL-316) limit the potential for fuel damage from a CRDA. Isolation of main steam lines could reduce radiation release.

Conclusions

a) Statement of Acceptability:

The DAEC radiological consequences for the CRDA Design Basis Accident analysis demonstrates that all acceptance criteria are met.

b) Conservatisms/Margins:

The following lists some of the key conservative assumptions in this analysis:

The assumption that the entire source term is released to the condenser volume in the first 5 seconds is conservative both for duration and quantity of radioisotopes transported.

Assuming that all the fission product release is transported directly to the turbine and condenser through the larger main steam piping (ignoring the MSL drains and Recirc. Sample Valves pathways) maximizes the release, both in magnitude and timing, as the holdup in these drain lines is ignored and this is not a filtered release path. Any small system leakage from the Recirc. Sample System would be into the Secondary Containment, which is a filtered release path via SGTS.

The assumption that MVP flow rate is constant at 1800 cfm for up to 24 hours is conservative. The CRDA will not result in damage to MSL piping or to the condenser walls and seals. The MVP will be drawing a vacuum on the condenser and, consequently its flow rate will decrease accordingly over time, thus reducing the release rate through the offgas stack.

No holdup time or transport delay is considered for condenser leakage into the turbine building.

No holdup time, plateout, or transport delay is considered for leakage from the turbine building to the environment.

1000 scfm of unfiltered in-leakage into the Control Room is very conservative relative to measured in-leakage (<100 scfm).

Calculated radiological consequences for the CRDA are only a small fraction of regulatory limits providing ample margin.

c) Limiting Event:

The radiological consequences of a Control Rod Drop Accident are non-limiting compared to the consequences of the DBA LOCA for on-site personnel. However, the CRDA is the most limiting event for off-site dose consequences to the public.

15.2.5 – FUEL HANDLING ACCIDENT

1) Reactor Response

Description of Event

a) Initiator: An irradiated fuel assembly and the last segment of the refueling mast becomes separated from the rest of the refueling mast during in-vessel fuel movement and drop onto the top of the core from the maximum possible height.

b) Sequence of Events (NOT a time line): During a refueling operation a fuel assembly is moved over the top of the core. While the fuel grapple is in the overhoist condition with the bottom of the assembly 30 feet above the top of the core (the maximum height allowed by the fuel handling equipment), a main hoist cable fails allowing the assembly, the fuel grapple mast and head to fall on top of the core impacting a group of four assemblies. The grapple head and mast are fixed vertically to the dropped assembly such that all the kinetic energy is transferred through the dropped assembly to the group of impacted assemblies. The dropped assembly impacts the core at a slight angle and the rods in this assembly are subjected to bending. After the assembly impacts the core, the assembly, grapple head and mast fall onto the core horizontally without contacting the side of the pressure vessel.

c) Single Failure/Operator Error (as applicable):
Beyond the initial failure of the refueling mast, there are no other equipment failures assumed in the Reactor Response.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
None.

Event Category & Acceptance Criteria

This is a Design Basis Accident. It represents the event that releases the largest quantity of radioactive material directly to the Secondary Containment.

The acceptance criterion for this event is radiological. There are no specific fuel or reactor parameters of significance for this event.

Methods

a) Calculation Tools & Computer Codes:
No computer codes are used to perform this evaluation. It is a generic mechanical stress calculation.
Because of the complex nature of the impact and the resulting damage to fuel assembly components, a rigorous prediction of the number of failed rods is not possible. For this reason a simplified energy approach is taken and numerous

conservative assumptions are made to assure a conservative estimate of the number of failed rods.

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
Core and Fuel Designs per the FRED form (See Section 15.0.7)

c) Key Assumptions:
Fuel Assembly is dropped from the maximum height of 30 feet above the core.
One-half of the impact energy is absorbed by the dropped assembly and the remaining half by the four impacted fuel assemblies.
None of the impact energy is assumed to be absorbed by the fuel pellets (i.e., all the energy is transferred to the fuel cladding.)
All the fuel rods in the dropped assembly are assumed to fail due to bending moments (1% clad strain).

<u>Results</u>

a) Comparison to Acceptance Criteria:
The results of the analysis predict that 151 fuel rods will fail.

b) Known Sensitivities:
The number of full and partial length fuel rods in the assembly and the weight of the assembly and mast section assumed to drop are the critical parameters for the number of rods assumed to fail.

c) Uncertainties in Results:
Use of conservative assumptions (e.g., all the impact energy is absorbed by the cladding) in the evaluation are intended to bound the uncertainties in the final results.

<u>Conclusion</u>

a) Statement of Acceptability:
See the Radiological Response Section below.

b) Known Conservatisms/Margins:
Accounting for the energy that could be absorbed by the fuel pellets would significantly reduce the number of rods assumed to fail.
It is highly likely that the dropped bundle/mast assembly would fall over after initial impact and lean against the core shroud wall and not further impact the fuel in the core.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
This is a Design Basis Accident. It is not re-analyzed as part of the reload licensing process.

2) <u>Containment Response</u>

Because this event occurs during the refueling sequence with the Drywell open, There is no impact on the Primary Containment. Secondary Containment is not challenged for pressure or temperature integrity as a result of this event. See Radiological Section below for Secondary Containment response.

3) <u>Radiological Response</u>

<u>Description of Event</u>

a) Initiator: The FHA is initiated when a fuel bundle and refueling mast detach from the refueling bridge and drop onto the reactor vessel core. The drop over the reactor core is more limiting than the drop over the spent fuel pool because the kinetic energy for the drop of thirty feet over the vessel produces a much greater number of damaged fuel pins on impact than the shorter drops that could occur
2010-019 over the fuel pool. However, there is one special case where special controls must be in place to ensure a drop in the spent fuel pool is bounded by a drop over the reactor vessel. (see Sensitivity Studies below)

b) Sequence of Events:

Time	Event
-60 hours0	Reactor Shutdown for Refueling
0	Accident Begins (Fuel Bundle and Refueling Bridge Mast fall onto Reactor Core)
10 minutes	Manual initiation of Control Building Standby Filter Units and Emergency Mode Ventilation.
62 hours	Completion of Radiological Release
30 days	End of analyzed event

2010-019

c) Single Failure/Operator Error:
Not applicable. No safety systems or operator actions are required to mitigate a FHA.

d) Key Equipment Response:
Control Building Envelop is credited as an enclosed, shielded volume for reduction of operator dose. Standby Filter Operation and Emergency mode operation of the Control Building HVAC system are credited for reduction of operator doses.

The DAEC Secondary containment and Standby Gas treatment Systems were originally designed with a safety function to mitigate the effects of a FHA. Reanalysis has concluded that these systems are not needed to maintain dose consequences within regulatory limits and they are no longer credited for mitigation of fuel handling accidents greater than 60 hours following reactor shutdown. The ability to use these systems to mitigate a FHA has been retained as a defense in depth measure, but this function is no longer considered safety-related, nor required by Technical Specifications.

Event Category and Acceptance Criteria

A FHA is a design basis accident. Radiological consequences analysis is required.

The acceptance criteria from 10 CFR 50.67 are:

- An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).
- Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

The radiological consequences to personnel in the Technical Support Center are evaluated using the acceptance criterion issued in Section 8.2. of Generic Letter 83-11 “Supplement 1 to NUREG-0737 – Requirements for Emergency Response Capability,” dated December 17, 1982.

- Adequate radiation protection is provided to assure that radiation exposure to any person working in the Technical Support Center would not exceed 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Per RG 1.183, for a FHA the doses should not exceed 6.3 rem TEDE at the EAB or LPZ in 2 hours.

Methods

The radiological consequences of design basis accidents were analyzed using the methods and guidelines of RG 1.183 “Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors.”

a. Calculation Tools and Computer Codes:

Source Term Inventory
The ORIGEN2 code (Reference 1), which is a widely used Oak Ridge National Laboratory code used in the production and decay of radioactive material, was used in the calculation of plant-specific fission product inventories which bound the effect of two year fuel cycles, power operation at 1950 MWt (102% of 1912

MWt), and anticipated fuel designs. This analysis was performed by General Electric (GE) under the GE software quality assurance program.

<u>Atmospheric Dispersion</u>
Atmospheric dispersion factors (CHI/Q's) were calculated with the ARCON96 computer code. The ARCON96 code calculates relative concentrations in plumes from nuclear power plants at the control room and Technical Support Center air intakes and accounts for the effects of building wakes. The ARCON96 code was verified and validated in accordance with DAEC Software Quality Assurance Program.

Atmospheric dispersion factors for offsite dose consequences were calculated with the PAVAN code. The PAVAN code was verified and validated in accordance with DAEC Software Quality Assurance Program.

<u>Radiological Dose</u>
2014-008 The RADTRAD computer code is a radiological consequence analysis code used
to estimate radiological source transport, removal, decay, and post-accident doses
2014-008 at plant offsite locations, the control room, and Technical Support Center. The
code was verified and validated in accordance with DAEC Software Quality
2014-008 Assurance Program and/or an approved vendor program.

b) Key Assumptions and Inputs:

1. Consistent with USNRC Regulatory Guide No. 1.183 Revision 0, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors", July 2000, Appendix B Section 1.1, the number of fuel rods damaged in a postulated FHA are based are based on a conservative analysis that considers the most limiting case. Information concerning the FHA for GE 14 fuel is specified in Section 2.13 of NEDC 32868P. A total of 151 rods were assumed damaged.

2. The gap activity fractions of Table 3 in Regulatory Position 3 of Regulatory Guide 1.183.

2014-008 3. A radial peaking factor of 1.55 was assumed.

4. Per plant refueling procedures, a post-shutdown 60-hour decay period was used to determine the release activity inventory.

5. All gap activity in the damaged fuel rods is assumed to be instantaneously released.

6. Radionuclides considered include the xenons, kryptons, halogens, cesiums, and rubidiums. However, all particulate radionuclides species (some halogens, cesiums, and rubidiums) are assumed to be retained in the

fuel pool or reactor cavity (infinite decontamination factor) consistent with RG 1.183 Appendix B Section 3.

7. Consistent with RG 1.183 Appendix B Section 3, all noble gases (xenons & kryptons) escape to the environment.

8. Of the radioiodine released from the damaged fuel rods, 99.85% of the released iodine is assumed to be in the form of elemental iodine and 0.15% of the released iodine is assumed to be in the organic species.

9. Consistent with RG 1.183 Appendix B Section 4.1 and 5.3, all radionuclide releases from the pool to the environment are assumed to occur over a 2-hour period.

10. Conservatively, no credit is taken for any dilution, holdup, or ESF filtration within secondary containment for the radionuclides escaping the pool.

11. Consistent with RG 1.183 Appendix B Section 2, the decontamination factor for organic iodine is assumed to be 1. The pool decontamination factor for elemental iodine is assumed such that the overall pool decontamination factor is 200 for all iodine species (500 for elemental iodine) with a 23-foot water level above the postulated damaged fuel assembly.)

12. For the FHA analysis, the depth of water over the damaged fuel assembly is assumed to be 23 feet.

13. Fuel pin pressure for the DAEC AEP GE 14 spent fuel is assumed to be less than 1200 psig.

14. For the FHA event over the reactor core, CR isolation is assumed to occur at 10-minutes post-accident based on manual operator action.

15. Per RG 1.183 Appendix B, no Loss-of Offsite-Power is postulated for the FHA event. Therefore, any activity release during the event may be postulated to be released via the normal HVAC; (i.e., via the RB exhaust vent). This pathway would realistically result in the quickest release of radiation due to higher ventilation flow rates. The release is treated as a ground release with no credit for elevated release from the exhaust stack. No credit is taken for automatic isolation of the reactor building ventilation on high radiation.

16. Control room emergency ventilation mode was assumed to be manually initiated by operators 10 minutes after the FHA begins.

17. Although acceptance criteria are based on doses received within a 2 hour period, DAEC analysis conservatively calculated doses for the LPZ, CR and TSC for a full 30 days exposure.

Results

Fuel Handling Accident CR Inleakage	EAB(1) (rem TEDE)	LPZ(2) (rem TEDE)	CR Operator (30 day) (rem TEDE)	TSC Operator (30 Day) (rem TEDE)
2014-008 FHA 1000 CFM	1.00	0.24	3.34	3.00
2014-008 0 CFM	1.00	0.24	2.45	1.82

1. Worst 2-hour integrated dose.
2. 30-day integrated dose.

a) Comparison to Acceptance Criteria:

Fuel Handling Accident CR Inleakage	EAB(1) (rem TEDE)	LPZ(2) (rem TEDE)	CR Operator (30 day) (rem TEDE)	TSC Operator (30 Day) (rem TEDE)
2014-008 FHA 1000 CFM	1.00	0.24	3.34	3.00
2014-008 0 CFM	1.00	0.24	2.45	1.82
Regulatory Limit	6.25	6.25	5.00	5.00

1. Worst 2-hour integrated dose.
2. 30-day integrated dose.

b) Sensitivities:
Sensitivity analysis was performed for unfiltered ventilation flow into the control room envelope at 0 cfm and 1000 cfm. The results confirmed that operator doses increase with assumed inleakage rates. Since the nominal flow rate of emergency ventilation flow to the control room is 1000 cfm, this inleakage rate is considered to bound actual inleakage rates based on engineering judgment that. as the building began to pressurize inleakage would become outleakage, it would be difficult to achieve control building pressurization if leakage paths exceeded this capacity.

2010-019 Sensitivity studies were performed by the BWR Owners' Group for various bundle drop scenarios in the spent fuel pool (Reference 15.0-61). In order to assure that the bundle drop in the vessel remains the bounding case, an unchanneled bundle may not be moved in the spent fuel pool unless it has decayed for at least 45 days. Plant procedures will assure this assumption is met.

c) Uncertainties:
The single largest uncertainty is the magnitude of the radioiodine source term. The analysis is based on a conservative estimate of the number of fuel pins that would be damaged. The radiological source term is maximized based on time after shutdown and use of radial peaking factors.

Conclusions

a) Statement of Acceptability:
The DAEC radiological consequences for the FHA Design Basis Accident analysis demonstrates that all acceptance criteria are met.

b) Conservatisms/Margins:
The capacity for fuel pool and reactor cavity water to scrub radioiodine is based upon a lower water level than would be present over the core during fuel movement. Calculated radiological consequences for the FHA are well within regulatory limits. Operability of secondary containment and the SBGT system would significantly reduce dose consequences

c) Limiting Event:
The radiological consequences of a FHA are non-limiting compared to the consequences of the DBA LOCA and CRDA.

Table 15.2-1
Loss-of-Coolant Accident Analysis Results for DAEC

2012-020
2017-001

Parameter	GNF2 Fuel[1]	Acceptance Criteria
1. Limiting Break	DBA Suction	
2. Limiting Failure (Nominal Case)	Battery	
3. Limiting Failure (Appendix K Case)	Battery	
4. Peak Cladding Temperature (Licensing Basis)[2]	< 1730°F	≤ 2200°F
5. Estimated Upper Bound PCT (95% Probability PCT)[3]	< 1610°F	≤ LBPCT[4]
6. Maximum Local Oxidation	< 2%	≤ 17%
7. Core-Wide Metal-Water Reaction	< 0.1%	≤ 1.0%
8. Coolable Geometry	Criteria 4 and 6 satisfied.	Maintain coolable geometry; which is satisfied by meeting Criteria 4 and 6.
9. Long Term Cooling	Satisfied by either: core reflooded above TAF; or, core reflooded to top of the jet pumps and 1 Core Spray system in operation.	Core temperature acceptably low and long-term decay heat removed

2012-020
2015-004
2017-001

[1] The ECCS-LOCA GNF2 analysis is based on the SAFER/PRIME-LOCA methodology.
[2] This is the base analysis. Subsequently, errors have been reported pursuant to 10 CFR 50.46.
[3] This is the base analysis. Subsequently, errors have been reported pursuant to 10 CFR 50.46.
[4] Per NRC SER, transmitted with letter, Richards (USNRC) to Klapproth (GENE), dated 02/01/02, GE is no longer required to do plant-specific UBPCT calculations and the previous 1600 °F limit has been removed.

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Table 15.2-2

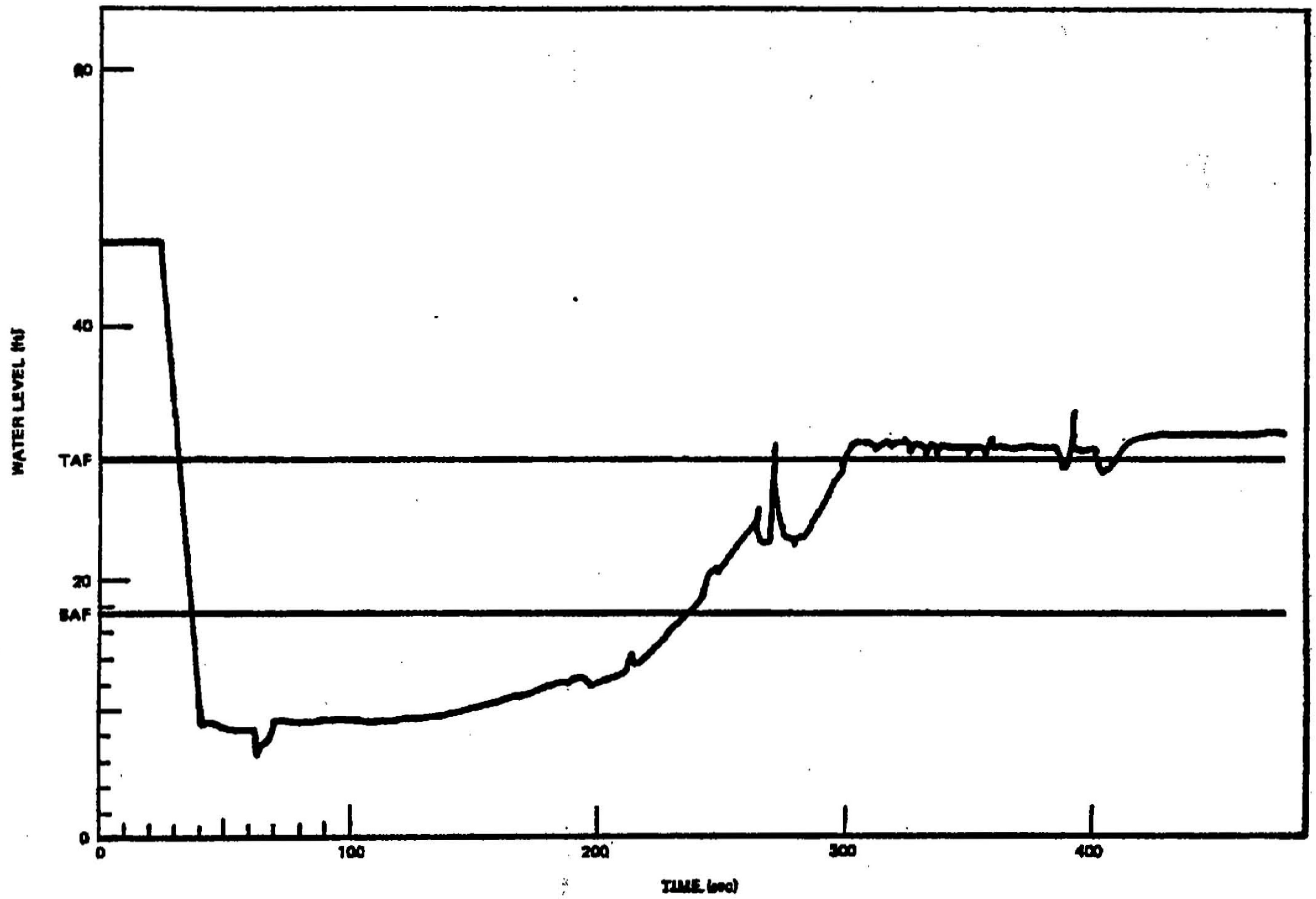
2010-019 FUEL HANDLING ACCIDENT DESIGN BASIS ANALYSIS ASSUMPTIONS

Reactor Power (102%), MWt,	1950
2014-008 Radial Peaking Factor	1.55
Fuel Decay Period, hours	60
Number of Assemblies in Core	368
Number of Fuel Rods in an Assembly (equivalent full and part length rods)	87.3
Number of Damaged Rods	151
Fraction of Gap Activity Released from Damaged Rods	1.0
Fraction of Core Activity in Gap	
I-131	0.08
Kr-85	0.10
Other Iodine and Noble Gases	0.05
Pool Decontamination Factor, Effective	200
Iodine Species fraction Above Pool Water	
Elemental	0.57
Organic	0.43
Release Duration, hours	
From Fuel and Pool	Instantaneous
From Secondary Containment	2
Release Rate to Environment, %/day	2.47E7
Collection and Filtration by SBGT	None
Assumed Release Point - Reactor Building Vent (Ground Release)	
Atmospheric Dispersion, 0-2 hours, sec/m^3	
EAB	5.57E-4
LPZ	1.34E-4
Control Room	2.85E-3
TSC	2.66E-3
Control Room Volume, ft^3	155,000
Control Room Normal Makeup, cfm	3150
Control Room Emergency Flow, cfm	1000
Control Room and TSC Filter Efficiency, %	
Elemental	90
Organic	30
Aerosol	99
Control Room Unfiltered In-Leakage, cfm	1000
Control Room Isolation Delay, Minutes	10
TSC Volume, ft^3	68,300
TSC Normal Makeup, cfm	900
TSC Emergency Flow, cfm	200
TSC Recirculation Flow	800
TSC Emergency Ventilation Actuation, minutes	30
TSC Unfiltered In-Leakage	1000

Table 15.2-3

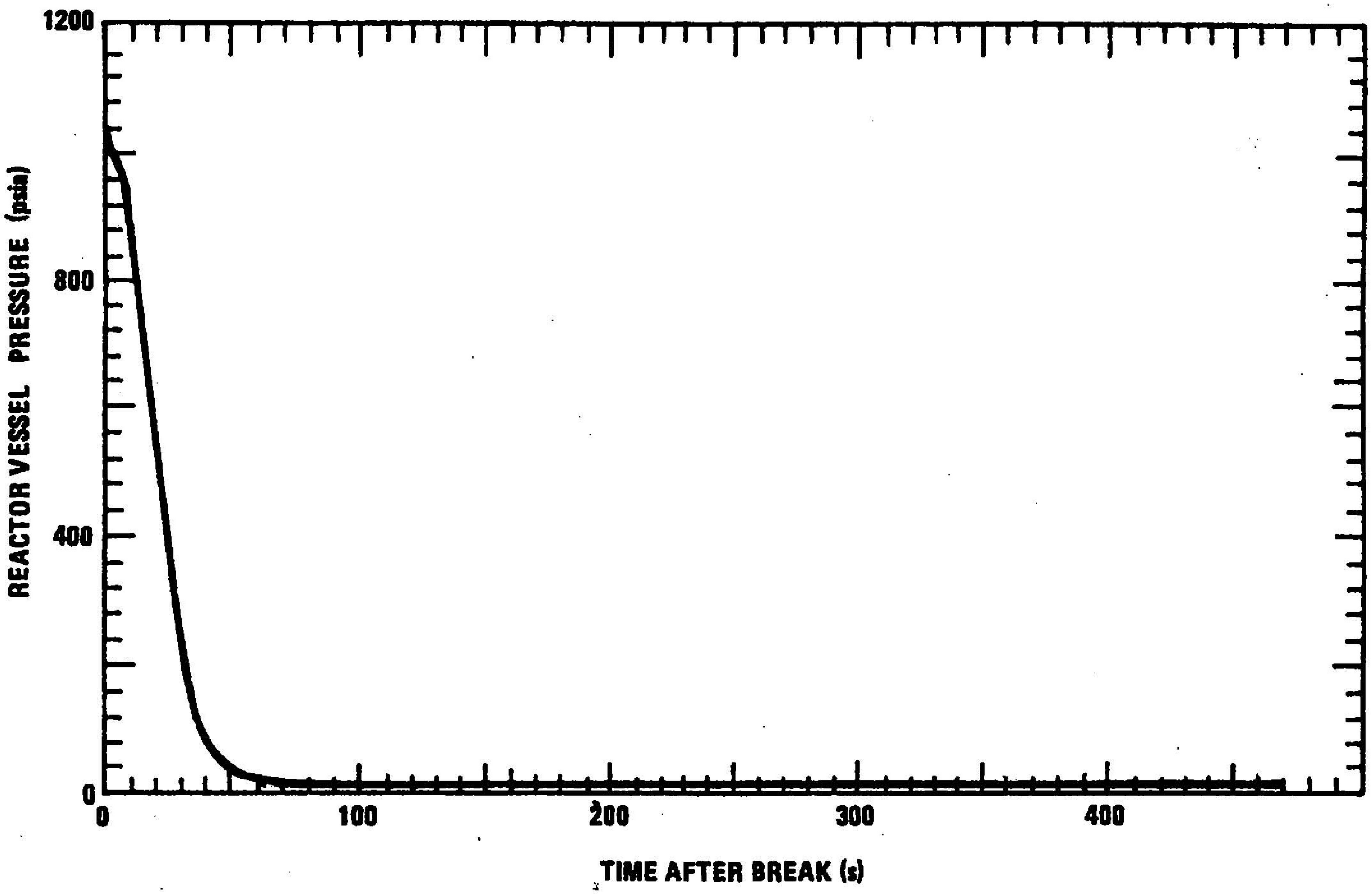
REFUELING ACCIDENT
FISSION PRODUCT RELEASE

	Isotope	Activity Released (Ci)
2014-008	I-131	1.562E+02
	I-132	1.403E+02
	I-133	1.952E+02
	I-134	2.137E+02
	I-135	1.829E+02
2014-008	Kr-85	6.393E+02
	Kr-85m	4.759E+03
	Kr-87	9.047E+03
	Kr-88	1.273E+04
2014-008	Xe-133	3.748E+04
	Xe-135	1.355E+04



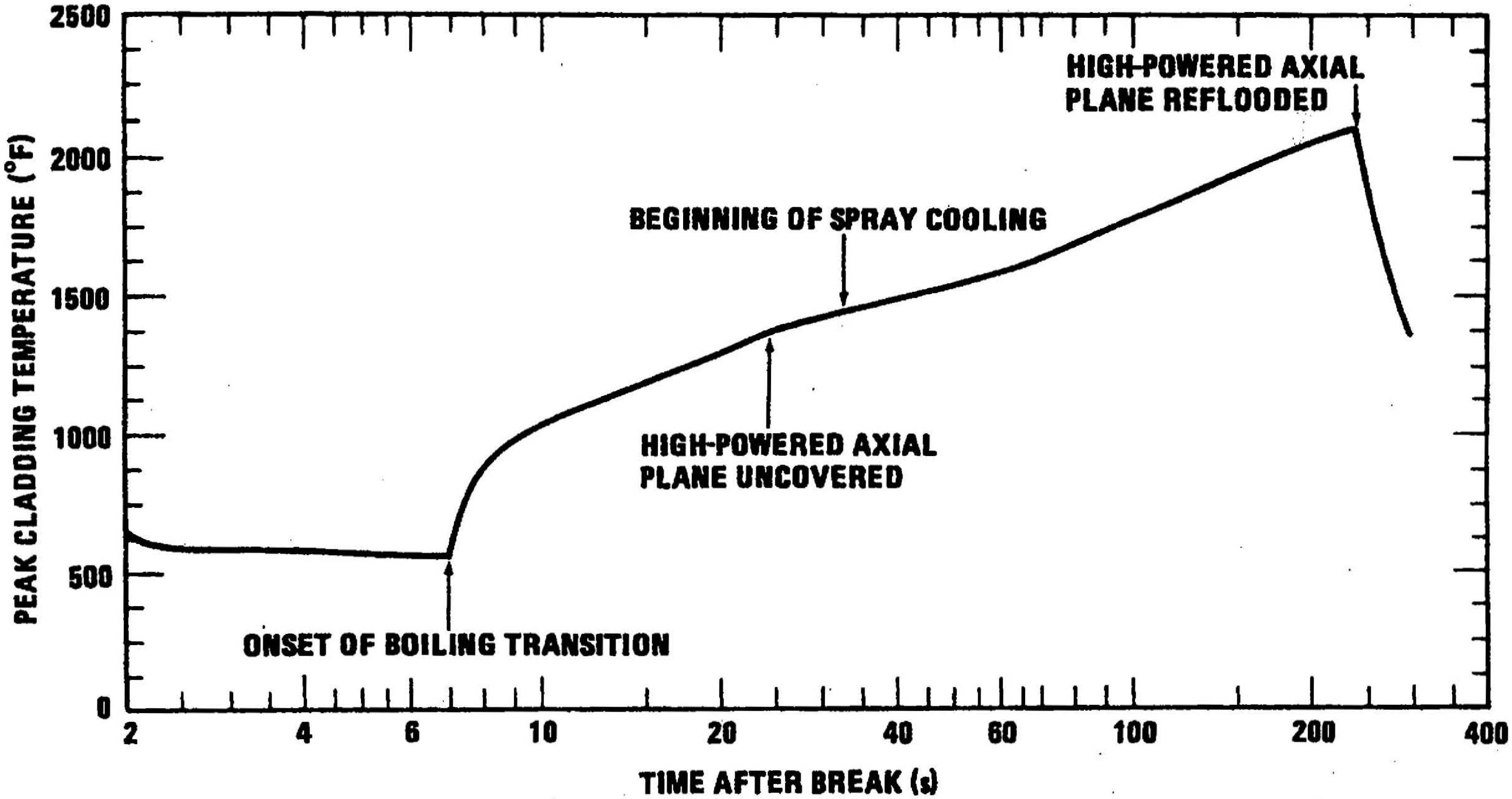
Design Basis Accident, Recirculation Suction Line Break, LPCI Injection Valve Failure

DUANE ARNOLD ENERGY CENTER IES UTILITIES, INC. UPDATED FINAL SAFETY ANALYSIS REPORT
Water Level Inside Shroud vs Time Large Break LOCA Figure 15.2-1



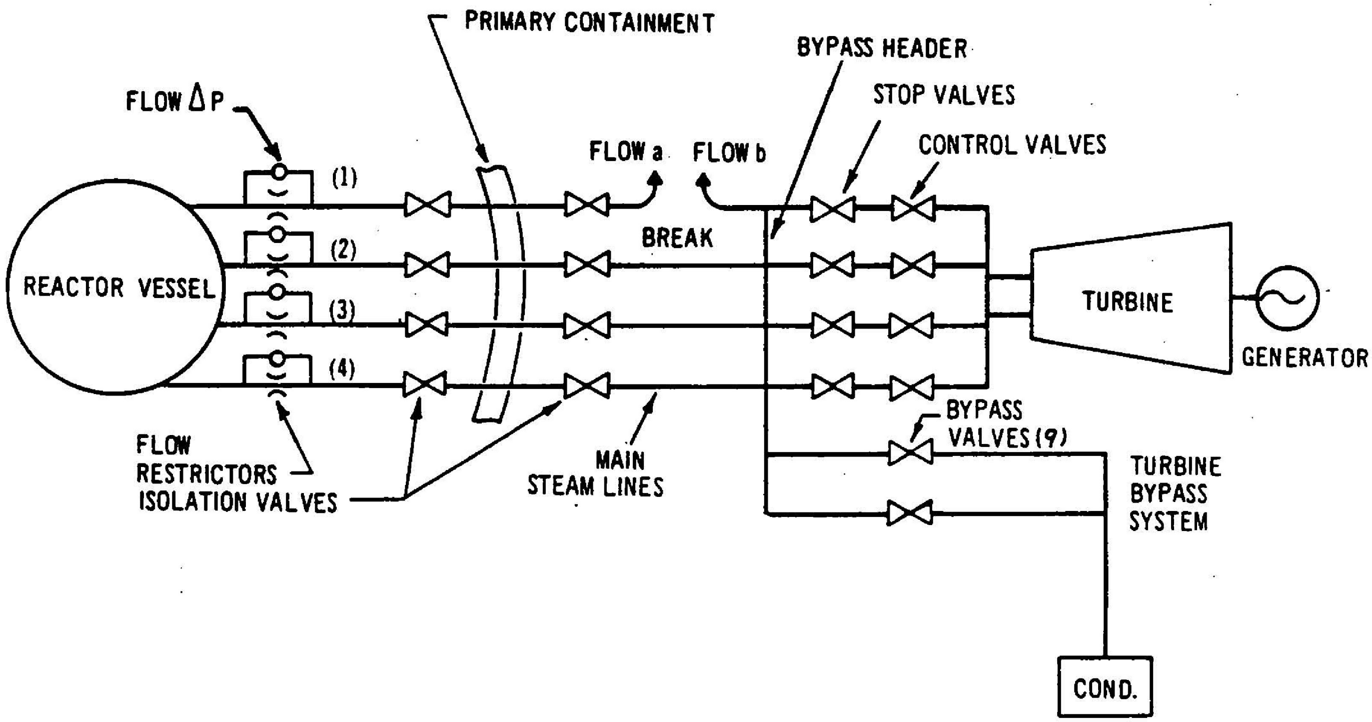
Design Basis Accident, Recirculation Suction Line Break, LPCI Injection Valve Failure

DUANE ARNOLD ENERGY CENTER IES UTILITIES, INC. UPDATED FINAL SAFETY ANALYSIS REPORT
Reactor Pressure vs Time Large Break LOCA Figure 15.2-2



Design Basis Accident, Recirculation Suction Line Break, LPCI Injection Valve Failure

DUANE ARNOLD ENERGY CENTER IES UTILITIES, INC. UPDATED FINAL SAFETY ANALYSIS REPORT
Peak Cladding Temperature vs Time Large Break LOCA Figure 15.2-3



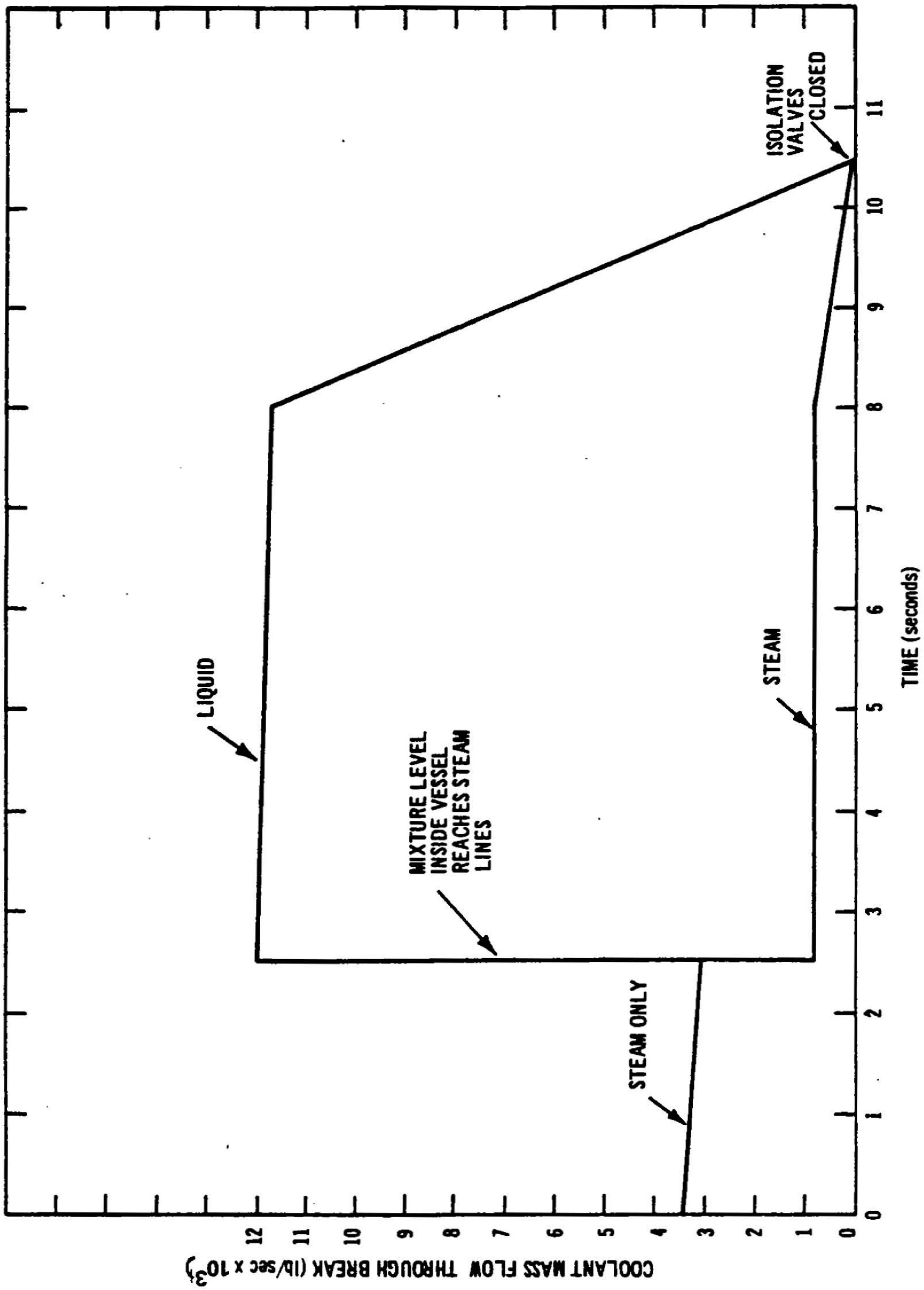
DUANE ARNOLD ENERGY CENTER

IES UTILITIES, INC.

UPDATED FINAL SAFETY ANALYSIS REPORT

Break Location for
Main Steam Line Break Accident

Figure 15.2-7



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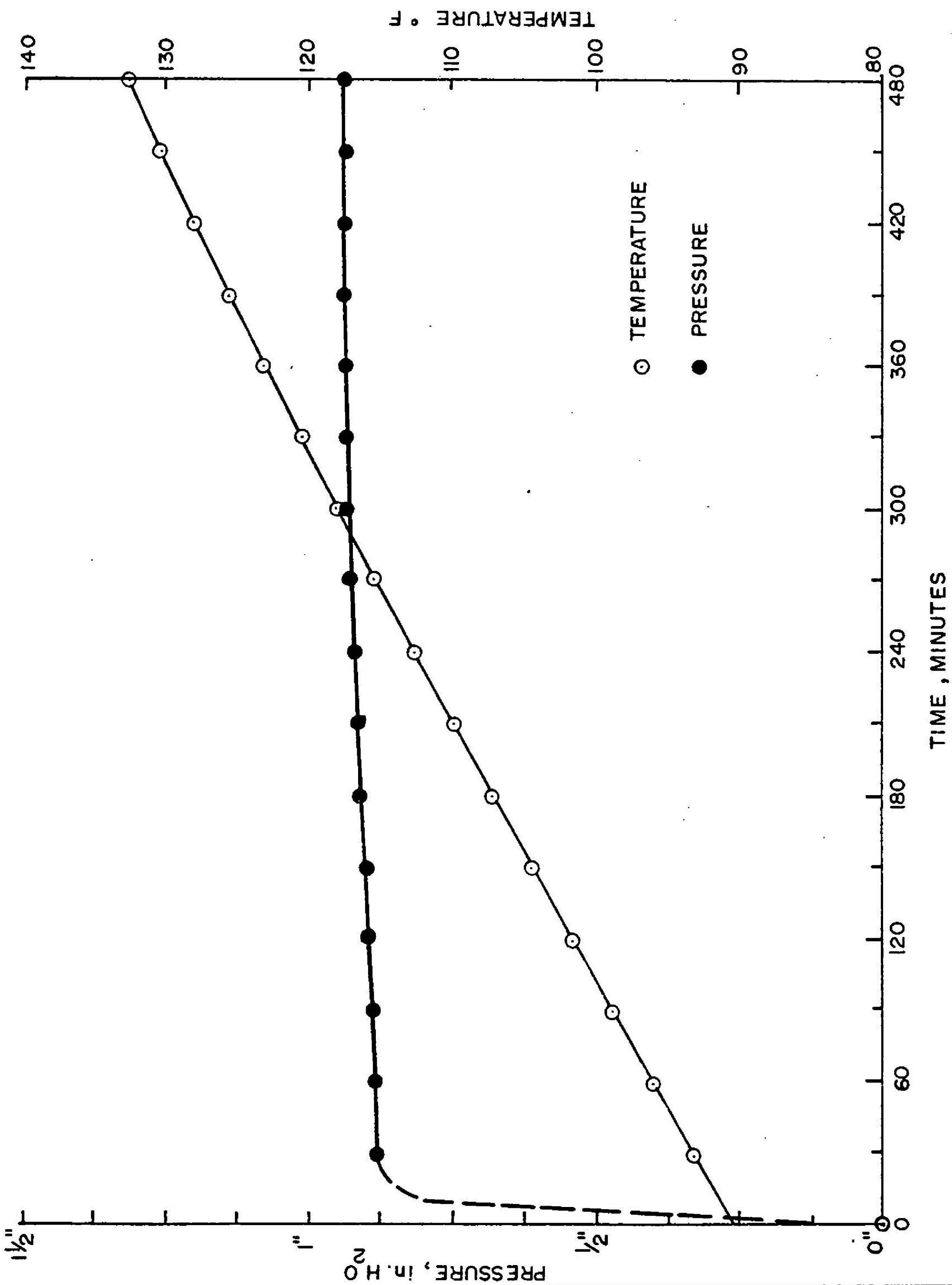
IES UTILITIES, INC.

UPDATED FINAL SAFETY ANALYSIS REPORT

Main Steam Line Break Accident,
Mass of Coolant Lost through Break

Figure 15.2-8



DUANE ARNOLD ENERGY CENTER

IES UTILITIES, INC.

UPDATED FINAL SAFETY ANALYSIS REPORT

Secondary Containment Pressure - Temperature Response to Instrument Line Break

Figure 15.2-9

15.3 SPECIAL EVENTS

Events in this category are those where the plant has demonstrated its ability to respond to specific events that were postulated after the plant was initially designed and licensed. While these events are now part of the current licensing basis for the DAEC, they are commonly referred to as "beyond design basis" events. Some events were added to the licensing basis by changes to the Code of Federal Regulations (CFR) (i.e., by rule changes), since the initial licensing of the plant.

The following are the identified events in this category:

a) Anticipated Transients Without Scram (ATWS) – 10 CFR 50.62

b) Station Blackout (SBO) – 10 CFR 50.63

2013-013 c) NFPA 805 Fire

In addition, two special evaluations are included in this Section, which demonstrate the plant's ability to respond to specific transients and accidents, but with unique evaluation assumptions, methods and acceptance criteria, which are different from those in the Transient and Accident Sections of this Chapter. They are:

a) Thermal-hydraulic Stability – 10 CFR 50, App. A - General Design Criterion (GDC) 12

b) Reactor Internals Pressure Differentials (RIPD)

Other postulated events, primarily those dealing with the radwaste systems and control of heavy loads (i.e., the dropping of a fuel shipping cast), were originally part of this Section. Those event descriptions have been moved to their respective system description sections of the UFSAR.

15.3.1 ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

This is a Special Event for demonstrating acceptable plant response to a specified set of Anticipated Operational Occurrences, but with the added requirement that the Reactor Protection System and/or Control Rod Drive system fails in such a way that the scram function is disabled, so-called "Anticipated Transients without Scram (ATWS)." It should be noted that no single failure of equipment or Operator error can lead to a complete failure to Scram. Because this ability to cope with such an event was promulgated by rule change (10 CFR 50.62), after the initial licensing of the facility, the methods (inputs and assumptions) used and acceptance criteria applied are less stringent than for design basis events. Also noteworthy, is that credit may be taken for non-safety-related equipment and Operator actions (within the first 10 minutes) in responding to these events, provided there are written procedures and training to implement them.

The following are the specified events in this category, and are a subset of all events generically evaluated, which represent the bounding cases:

a) Closure of All Main Steamline Isolation Valves (MSIVC)

b) Pressure Regulator Failure – Maximum Demand (PRFO)

c) Loss-of-Offsite Power (LOOP)

d) Inadvertent Opening of one Safety-Relief Valve (IORV)

Systems required at DAEC to meet NRC requirements of the ATWS Rule (10CFR 50.62) are the Standby Liquid Control (SLC) system and the Alternate Rod Injection-Recirculation Pump Trip (ARI-RPT) system, which are described in Sections 9.3 and 7.2, respectively.

15.3.1.1 ATWS - MSIVC

<u>Description of Event</u>

a) Initiator:
A spurious trip that causes all the MSIVs to rapidly close with failure of the control rods to Scram.

b) Sequence of Events (NOT a time line):
The plant is operating at 100% power, when a non-mechanistic failure causes the MSIVs to begin to fast close. However, there is a failure in the Reactor Protection System (RPS) and/or Control Rod Drive (CRD) System that prevents the control rods from inserting (i.e., does not Scram). The power increases quickly from the collapsing of the voids in the core from the pressure increase due to the closure of the MSIVs. Reactor pressure increases and SRVs lift to relieve the pressure. The pressure reaches the ATWS-RPT setpoint, which trips the reactor recirculation pumps and they begin to coastdown. The pressure rise is sufficiently large that the Spring Safety Valves (SSVs) also open momentarily. Low-low set logic cycles the SRVs to control reactor pressure. The SRVs discharge steam, corresponding to the power level, to the suppression pool. The added heat causes the pool temperature to increase. The Operators respond to the event, using the Emergency Operating Procedures (EOPs). Operator actions include: starting suppression pool cooling to control pool temperatures; adjusting vessel level to control power; and starting the Standby Liquid Control System (SLCS) pumps to inject boron into the vessel.

Long-term response (beyond the explicit analyzed period): Operators guide the plant to a cold shutdown condition and assure that sufficient boron has been injected to maintain the core in a subcritical state until the control rods can be inserted into the core.

c) Single Failure/Operator Error (as applicable):
Failure of the RPS and/or CRD systems to trip and insert control rods
In addition, no credit is taken for the Alternate Rod Insertion (ARI) System.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
ATWS-RPT (< 1150 psig Reactor Dome Pressure); SRVs & SSVs open/close;
Recirculation Pumps coastdown, and LLS actuates.
Operators initially inhibit vessel injection by HPCI and RCIC, per EOPs.
Operators initiate feedwater runback at 90 seconds from the beginning of the event or at Boron Injection Initiation Temperature (BIIT), whichever comes later, to lower vessel level, per EOPs.
Operators initiate SLCS at either 120 seconds after the ATWS trip point (low water level or high pressure), or the time at which the suppression pool temperature reaches the BIIT, whichever occurs later, per EOPs.
Operators initiate and maximize SPC using 2 RHR pumps and 2 RHRSW pumps per loop.

Event Category & Acceptance Criteria

This is a Special Event – an Anticipated Operational Transient (Occurrence) with more than a single failure and/or Operator error - a complete failure to Scram the control rods.

Peak Fuel Cladding Temperature < 2200 °F (10 CFR 50.46)
Peak Fuel Local Cladding Oxidation Fraction < 17% (10 CFR 50.46)
Peak Reactor Vessel Pressure < 1500 psig (ASME Service Level C)
Peak Suppression Pool Temperature < 281 °F (design value)
Peak Containment Pressure < 62 psig (110% of the design value)

Methods

a) Calculation Tools & Computer Codes:
Fuel Response: Primary Code – TASC, with input from Secondary Code - ISCOR (see Table 15.0-2 for complete listing, code versions and NRC acceptance).
Reactor Response: Primary Code – ODYN with approved boron mixing model (see Table 15.0-2 for complete listing, code versions and NRC acceptance).
Containment Response: Primary Code – STEMP, with input from Secondary Code - ODYN (above), for the suppression pool temperature response. Hand calculations are performed assuming thermodynamic equilibrium between the suppression pool and containment airspace to determine the peak wetwell pressure, neglecting the heat sinks of the structure.

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-3, with specific changes as outlined in Table 15.3-1.

c) Key Assumptions:
Plant is initially at rated thermal power and 99% rated core flow (MELLLA).

MSIVs close in a linear ramp in 4.0 seconds.
The RCIC enthalpy (based upon CST water temperature of 100 °F) is used for all injection flow through the feedwater sparger.
Operators initiate feedwater runback at 90 seconds from the beginning of the event or at Boron Injection Initiation Time (BIIT), whichever comes later.
Operators initiate SLCS at either 120 seconds after the ATWS trip point (low water level or high pressure), or the time at which the suppression pool temperature reaches the BIIT, whichever occurs later.
Hot Shutdown is defined as neutron flux < 0.1% of rated for > 100 seconds.
May-Witt Decay Heat curve is used in the containment analysis once the plant is in Hot Shutdown.

Results

2012-020 The GNF2 Amendment 22 Compliance to GESTAR II (Reference 15.0-63) requires a plant specific demonstration that the limiting ATWS event response is within the ATWS acceptance criteria. For DAEC, the previously calculated ATWS results, given below, have sufficient margin to the overpressure and suppression pool temperature limit to not require explicit analysis as allowed per Reference 15.0-63. Therefore, while it is recognized that the GNF2 NFI will affect the ATWS results, DAEC is able to demonstrate compliance to the ATWS acceptance criteria for an ATWS event with GNF2 fuel.

a) Comparison to Acceptance Criteria:

Peak Fuel Cladding Temperature = 1302 °F
Peak Fuel Local Cladding Oxidation Fraction = (Note 1)
Peak Reactor Vessel Pressure = 1340 psig
Peak Suppression Pool Temperature = 215.6 °F (Note 2)
Peak Containment Pressure =18.3 psig

Note 1: Peak Cladding Temperature is below the 1800 °F threshold temperature at which significant cladding oxidation could occur, so the cladding oxidation was not explicitly calculated.

Note 2: A generic study was conducted for a limiting plant to investigate the degradation of the drywell environment due to the steam discharge from the unpiped SSVs during the ATWS event. Compared to the limiting plant, DAEC has only 2 unpiped SSVs versus 8 SSVs in the limiting plant. DAEC also has 6 SRVs versus 5 RV/SRVs in the limiting plant. Even though DAEC has a drywell volume only about 80% of that in the limiting plant, the total amount of steam discharged into the drywell from unpiped SSVs during ATWS events is significantly lower than that of the limiting plant. Therefore, the drywell temperature during ATWS for DAEC is bounded by the result in the generic study.

b) Known Sensitivities:
Both Beginning of Cycle (BOC) and End of Cycle (EOC) fuel conditions were analyzed to determine the limiting case. The BOC conditions have a higher core void fraction and a more negative void reactivity coefficient than EOC. Upon pressurization, the BOC conditions generate more steam even though the initial power surge may not surpass that from the EOC conditions. This leads to a higher peak vessel pressure. For the long-term results, i.e., pool temperature calculations, the less negative void reactivity coefficient makes the strategy of lowering water level in controlling power slightly less effective for EOC than the BOC exposure. This results in a higher power level during the water level control period and more steam being discharged into the suppression pool for EOC conditions. In the case of PCT, the axial power shape for EOC is generally flatter than that of BOC. In the top portion of the fuel rod where the PCT occurs, the EOC conditions generate more power and slip into boiling transition more often. Hence, the PCT is higher for the EOC exposure.

Peak Vessel Pressure is most sensitive to the capacity of the SRVs and SSVs. For the acceptance criterion to be met, the SSVs must open.
A lower opening setpoint of the SSVs at the nominal value of 1240 psig would produce more steam discharge into the drywell since the valves open sooner and remain opening longer. A sensitivity study with the same event and conditions except using the nominal SSV opening setpoint is conducted. The resulting integrated SSV flow is still significantly lower than the limiting value in the generic study.

Peak Fuel Cladding Temperature is most affected by the initial power level and axial power shape at EOC.

c) Uncertainties in Results:
Emergency reactor depressurization when EOP criteria require it (e.g., the Heat Capacity Temperature Limit (HCTL) for the suppression pool is exceeded) cannot be evaluated with ODYN.

<u>Conclusion</u>

a) Statement of Acceptability:
This event satisfies all the acceptance criteria for this Special Event.

b) Known Conservatisms/Margins:
No credit is taken for the ARI system, which would mitigate the event before the BIIT point is reached, requiring SLCS injection.
For the SRV setpoints, an upper limit of the opening setpoint is established with a 44 psi bias added to the nominal opening setpoint for the Target Rock SRVs. A statistical spread is performed based on this upper limit.
The BIIT time for this event is 55 seconds. Thus, the Operators would be taking actions sooner than credited in this analysis.

Because of limitations in the ODYN model, all ATWS analysis with ODYN is performed with water level lowered to Top of Active Fuel (TAF) + 5 feet and with the conservative bias applied to the user input boron mixing efficiency tables.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
Within the Special Event category of ATWS events, the MSIVC event is most-limiting for the acceptance criteria of peak suppression pool temperature and containment pressure.

15.3.1.2 – ATWS – PRFO

Description of Event

a) Initiator:
A failure of either the primary or back-up pressure regulator occurs at rated conditions, sending a signal to the turbine control and turbine bypass valves to open to the maximum combined flow limit setpoint, with failure of the control rods to Scram.

b) Sequence of Events (NOT a time line):
The plant is operating at 100% power, when a non-mechanistic failure causes either the primary or backup pressure regulator to fail to the full open position. This causes the turbine control valves to open to full flow and turbine bypass valve to partially open. However, the maximum combined flow limiter setpoint in the EHC system will limit the turbine valves opening to the equivalent of 125% of rated steamflow. This sudden increase in steamflow causes the vessel pressure to drop. The decrease in pressure causes a sudden swell in water level due to increased voiding in the core. Thus, reactor power initially goes down with the increased voiding. The vessel swell is not sufficient to reach the high level trip setpoint, so feedwater remains available. However, the pressure drop is sufficient to reach the Group I isolation on low steamline pressure and the MSIVs quickly go closed. But, there is a failure in the Reactor Protection System (RPS) and/or Control Rod Drive (CRD) System that prevents the control rods from inserting (i.e., does not Scram). The power increases quickly from the collapsing of the voids in the core from the pressure increase due to the closure of the MSIVs. Reactor pressure increases and SRVs lift to relieve the pressure. The pressure reaches the ATWS-RPT setpoint, which trips the reactor recirculation pumps and they begin to coastdown. Low-low set logic cycles the SRVs to control reactor pressure. The SRVs discharge steam, corresponding to the power level, to the suppression pool. The added heat causes the pool temperature to increase. The Operators respond to the event, using the Emergency Operating Procedures (EOPs). Operator actions include: starting suppression pool cooling to control pool temperatures; adjusting vessel level to control power; and, starting the Standby Liquid Control System (SLCS) pumps to inject boron into the vessel.

Long-term response (beyond the explicit analyzed period): Operators guide the plant to a cold shutdown condition and assure that sufficient boron has been injected to maintain the core in a subcritical state until the control rods can be inserted into the core.

c) Single Failure/Operator Error (as applicable):
Failure of the RPS and/or CRD systems to trip and insert control rods. In addition, no credit is taken for the Alternate Rod Insertion (ARI) System.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
ATWS-RPT (< 1150 psig Reactor Dome Pressure); SRVs open/close; Recirculation Pumps coastdown, and LLS actuates.
Operators initially inhibit vessel injection by HPCI and RCIC, per EOPs.
Operators initiate feedwater runback at 90 seconds from the beginning of the event <u>or</u> at Boron Injection Initiation Temperature (BIIT), whichever comes later, to lower vessel level, per EOPs.
Operators initiate SLCS at either 120 seconds after the ATWS trip point (low water level or high pressure), <u>or</u> the time at which the suppression pool temperature reaches the BIIT, whichever occurs later, per EOPs.
Operators initiate and maximize SPC using 2 RHR pumps and 2 RHRSW pumps per loop.

<u>Event Category & Acceptance Criteria</u>

This is a Special Event – an Anticipated Operational Transient (Occurrence) with more than a single failure and/or Operator error - a complete failure to Scram the control rods.

Peak Fuel Cladding Temperature < 2200 °F (10 CFR 50.46)
Peak Fuel Local Cladding Oxidation Fraction < 17% (10 CFR 50.46)
Peak Reactor Vessel Pressure < 1500 psig (ASME Service Level C)
Peak Suppression Pool Temperature < 281 °F (design value)
Peak Containment Pressure < 62 psig (110% of the design value)

<u>Methods</u>

a) Calculation Tools & Computer Codes:
Fuel Response: Primary Code – TASC, with input from Secondary Code - ISCOR (see Table 15.0-2 for complete listing, code versions and NRC acceptance).
Reactor Response: Primary Code – ODYN with approved boron mixing model (see Table 15.0-2 for complete listing, code versions and NRC acceptance).
Containment Response: Primary Code – STEMP, with input from Secondary Code - ODYN (above), for the suppression pool temperature response. Hand calculations are performed assuming thermodynamic equilibrium between the suppression pool and containment airspace to determine the peak wetwell pressure, neglecting the heat sinks of the structure.

b) Inputs (Reference common list in 15.0, and/or include event-specific items): OPL-3, with specific changes as outlined in Table 15.3-1.

c) Key Assumptions:
Plant is initially at rated thermal power and 99% rated core flow (MELLLA).
MSIVs close in a linear ramp in 4.0 seconds.
The RCIC enthalpy (based upon CST water temperature of 100 °F) is used for all injection flow through the feedwater sparger.
Operators initiate feedwater runback at 90 seconds from the beginning of the event or at Boron Injection Initiation Time (BIIT), whichever comes later.
Operators initiate SLCS at either 120 seconds after the ATWS trip point (low water level or high pressure), or the time at which the suppression pool temperature reaches the BIIT, whichever occurs later.
Hot Shutdown is defined as neutron flux < 0.1% of rated for > 100 seconds.
May-Witt Decay Heat curve is used in the containment analysis once the plant is in Hot Shutdown.

Results

2012-020 The GNF2 Amendment 22 Compliance to GESTAR II (Reference 15.0-63) requires a plant specific demonstration that the limiting ATWS event response is within the ATWS acceptance criteria. For DAEC, the previously calculated ATWS results, given below, have sufficient margin to the overpressure and suppression pool temperature limit to not require explicit analysis as allowed per Reference 15.0-63. Therefore, while it is recognized that the GNF2 NFI will affect the ATWS results, DAEC is able to demonstrate compliance to the ATWS acceptance criteria for an ATWS event with GNF2 fuel.

a) Comparison to Acceptance Criteria:

Peak Fuel Cladding Temperature = 1380 °F
Peak Fuel Local Cladding Oxidation Fraction = (Note 1)
Peak Reactor Vessel Pressure = 1343 psig
Peak Suppression Pool Temperature = 214.4 °F (Note 2)
Peak Containment Pressure =17.9 psig

Note 1: Peak Cladding Temperature is below the 1800 °F threshold temperature at which significant cladding oxidation could occur, so the cladding oxidation was not explicitly calculated.

Note 2: A generic study was conducted for a limiting plant to investigate the degradation of the drywell environment due to the steam discharge from the unpiped SSVs during the ATWS event. Compared to the limiting plant, DAEC has only 2 unpiped SSVs versus 8 SSVs in the limiting plant. DAEC also has 6 SRVs versus 5 RV/SRVs in the limiting plant. Even though DAEC has a drywell volume only about 80% of that in the limiting plant, the total amount of steam discharged into the drywell from unpiped SSVs during ATWS events is significantly lower than that of the limiting plant. Therefore, the drywell temperature during ATWS for DAEC is bounded by the result in the generic study.

b) Known Sensitivities:
Both Beginning of Cycle (BOC) and End of Cycle (EOC) fuel conditions were analyzed to determine the limiting case. The BOC conditions have a higher core void fraction and a more negative void reactivity coefficient than EOC. Upon pressurization, the BOC conditions generate more steam even though the initial power surge may not surpass that from the EOC conditions. This leads to a higher peak vessel pressure. For the long-term results, i.e., pool temperature calculations, the less negative void reactivity coefficient makes the strategy of lowering water level in controlling power slightly less effective for EOC than the BOC exposure. This results in a higher power level during the water level control period and more steam being discharged into the suppression pool for EOC conditions. In the case of PCT, the axial power shape for EOC is generally flatter than that of BOC. In the top portion of the fuel rod where the PCT occurs, the EOC conditions generate more power and slip into boiling transition more often. Hence, the PCT is higher for the EOC exposure.

Peak Vessel Pressure is most sensitive to the capacity of the SRVs and SSVs. For the acceptance criterion to be met, the SSVs must be available to open.
A lower opening setpoint of the SSVs at the nominal value of 1240 psig would produce more steam discharge into the drywell since the valves open sooner and remain opening longer. A sensitivity study with the same event and conditions except using the nominal SSV opening setpoint is conducted. The resulting integrated SSV flow is still significantly lower than the limiting value in the generic study.

Peak Fuel Cladding Temperature is most affected by the initial power level and axial power shape at EOC.

c) Uncertainties in Results:
Emergency reactor depressurization when EOP criteria require it (e.g., the Heat Capacity Temperature Limit (HCTL) for the suppression pool is exceeded) cannot be evaluated with ODYN.

Conclusion

a) Statement of Acceptability:
This event satisfies all the acceptance criteria for this Special Event.

b) Known Conservatisms/Margins:
No credit is taken for the ARI system, which would mitigate the event before the BIIT point is reached, requiring SLCS injection.
For the SRV setpoints, an upper limit of the opening setpoint is established with a 44 psi bias added to the nominal opening setpoint for the Target Rock SRVs. A statistical spread is performed based on this upper limit.
The BIIT time for this event is 67 seconds. Thus, the Operators would be taking actions sooner than credited in this analysis.
Because of limitations in the ODYN model, all ATWS analysis with ODYN is performed with water level lowered to Top of Active Fuel (TAF) + 5 feet and with the conservative bias applied to the user input boron mixing efficiency tables.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
Within the Special Event category of ATWS events, the PRFO event is most-limiting for the acceptance criteria of peak cladding temperature and vessel pressure.

15.3.1.3 – ATWS - LOOP

Description of Event

a) Initiator:
A Loss-of-Offsite Power (LOOP) occurs with failure of the control rods to Scram.

b) Sequence of Events (NOT a time line):
The plant is operating at 100% power, when all offsite power is lost. There is a resulting turbine-generator trip. However, there is a failure in the Reactor Protection System (RPS) and/or Control Rod Drive (CRD) System that prevents the control rods from inserting (i.e., does not Scram). The power increases quickly from the collapsing of the voids in the core from the pressure increase due to the closure of the Turbine Stop Valves (TSVs). The Turbine Bypass Valves (TBVs) open quickly to arrest the pressure and power increase. However, their capacity is limited in assisting due to the failure to Scram (i.e., continued steam production is greater than the TBV capacity.) The power increase is mitigated by the lowering of the water level as a result of the tripping of both the Recirculation and Feedpumps with the LOOP, which begin to coastdown immediately. SRVs lift to relieve the pressure. The pressure reaches the ATWS-RPT setpoint. (Note: this is only important for the analytical aspect of determining the SLCS injection time.) The MSIVs will close on loss of power to the RPS M/G sets, resulting in another reactor pressure increase, as the TBVs are now isolated. Low-low set logic cycles the SRVs to control reactor pressure. The SRVs discharge steam, corresponding to the power level, to the suppression pool. The added heat causes the pool temperature to increase. The Operators respond to the event, using the Emergency Operating Procedures (EOPs). Operator actions include starting suppression pool cooling to control pool temperatures, because the RHR and RHRSW pumps are

being powered by the Emergency Diesel Generators (EDGs), only one RHR pump can be loaded onto the buss. Hence, the amount of cooling for the suppression pool is reduced. The Operators also inhibit vessel injection to maintain a low vessel water level in order to control power. The Operators then start the Standby Liquid Control System (SLCS) pumps and inject boron into the vessel.

Long-term response (beyond the explicit analyzed period): Operators guide the plant to a cold shutdown condition and assure that sufficient boron has been injected to maintain the core in a subcritical state until the control rods can be inserted into the core.

c) Single Failure/Operator Error (as applicable):
Failure of the RPS and/or CRD systems to trip and insert control rods
In addition, no credit is taken for the Alternate Rod Insertion (ARI) System.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):

TSVs close; TBVs open; Recirculation and Feedwater Pumps coastdown;
SRVs open/close; LLS actuates; and EDGs start
Operators initially inhibit vessel injection by HPCI and RCIC, per EOPs.
Operators initiate SLCS at either 120 seconds after the ATWS trip point (low water level or high pressure), or the time at which the suppression pool temperature reaches the Boron Injection Initiation Temperature (BIIT), whichever occurs later, per EOPs.
Operators initiate and maximize SPC using one RHR pump and 2 RHRSW pumps per loop.

Event Category & Acceptance Criteria

This is a Special Event – an Anticipated Operational Transient (Occurrence) with more than a single failure and/or Operator error - a complete failure to Scram the control rods.

Peak Fuel Cladding Temperature < 2200 °F (10 CFR 50.46)
Peak Fuel Local Cladding Oxidation Fraction < 17% (10 CFR 50.46)
Peak Reactor Vessel Pressure < 1500 psig (ASME Service Level C)
Peak Suppression Pool Temperature < 281 °F (design value)
Peak Containment Pressure < 62 psig (110% of the design value)

Methods

a) Calculation Tools & Computer Codes:
Fuel Response: Primary Code – TASC, with input from Secondary Code - ISCOR (see Table 15.0-2 for complete listing, code versions and NRC acceptance).
Reactor Response: Primary Code – ODYN with approved boron mixing model (see Table 15.0-2 for complete listing, code versions and NRC acceptance).
Containment Response: Primary Code – STEMP, with input from Secondary Code - ODYN (above), for the suppression pool temperature response. Hand calculations are performed assuming thermodynamic equilibrium between the suppression pool and containment airspace to determine the peak wetwell pressure, neglecting the heat sinks of the structure.

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-3, with specific changes as outlined in Table 15.3-1.

c) Key Assumptions:
Plant is initially at rated thermal power and 99% rated core flow (MELLLA).
Turbine Stop Valves close in a linear ramp in 0.1 seconds.
MSIVs close in a linear ramp in 4.0 seconds.
The RCIC enthalpy (based upon CST water temperature of 100 °F) is used for all injection flow through the feedwater sparger.
Operators initiate SLCS at either 120 seconds after the ATWS trip point (low water level or high pressure), or the time at which the suppression pool temperature reaches the BIIT, whichever occurs later.
Hot Shutdown is defined as neutron flux < 0.1% of rated for > 100 seconds.
May-Witt Decay Heat curve is used in the containment analysis once the plant is in Hot Shutdown.

Results

2012-020 The GNF2 Amendment 22 Compliance to GESTAR II (Reference 15.0-63) requires a plant specific demonstration that the limiting ATWS event response is within the ATWS acceptance criteria. For DAEC, the previously calculated ATWS results, given below, have sufficient margin to the overpressure and suppression pool temperature limit to not require explicit analysis as allowed per Reference 15.0-63. Therefore, while it is recognized that the GNF2 NFI will affect the ATWS results, DAEC is able to demonstrate compliance to the ATWS acceptance criteria for an ATWS event with GNF2 fuel.

a) Comparison to Acceptance Criteria:

Peak Fuel Cladding Temperature = 708 °F
Peak Fuel Local Cladding Oxidation Fraction = (Note 1)
Peak Reactor Vessel Pressure = 1261 psig
Peak Suppression Pool Temperature = 202.8 °F

Peak Containment Pressure =14.4 psig

Note 1: Peak Cladding Temperature is below the 1800 °F threshold temperature at which significant cladding oxidation could occur, so the cladding oxidation was not explicitly calculated.

b) Known Sensitivities:
Peak Vessel Pressure is most sensitive to the capacity of the SRVs. Despite the turbine-generator trip and reactor isolation, the effect of the pressurization is less pronounced than other ATWS events because of the initial fast opening of the TBVs.

The initiation of Recirculation pump coastdown and water level reduction due to the Feedpump coastdown at the beginning of the transient reduces the severity of this event.

Peak Fuel Cladding Temperature is most affected by the initial power level and axial power shape at EOC.

Even though the RHR heat exchanger effectiveness for this event is less than maximum, due only loading one RHR pump on its Emergency Diesel Generator to allow both the RHR Service Water pumps to be put into service, the early water level reduction due to feedwater pump coastdown at time zero, and subsequent lockout of HPCI and RCIC injection, limits the amount of the steam discharged into the suppression pool.

c) Uncertainties in Results:
Emergency reactor depressurization when EOP criteria require it (e.g., the Heat Capacity Temperature Limit (HCTL) for the suppression pool is exceeded) cannot be evaluated with ODYN.

Conclusion

a) Statement of Acceptability:
This event satisfies all the acceptance criteria for this Special Event.

b) Known Conservatisms/Margins:
For the SRV setpoints, an upper limit of the opening setpoint is established with a 44 psi bias added to the nominal opening setpoint for the Target Rock SRVs. A statistical spread is performed based on this upper limit.
The BIIT time for this event is 64 seconds. Thus, the Operators would be taking actions sooner than credited in this analysis.
It is possible that additional suppression pool cooling could be realised, as there should be load margin available on the EDG for the second RHR pump, as there is no need for the Core Spray pumps to be loaded onto the EDGs for this event.

Because of limitations in the ODYN model, all ATWS analysis with ODYN is performed with water level lowered to Top of Active Fuel (TAF) + 5 feet and with the conservative bias applied to the user input boron mixing efficiency tables.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
Within the Special Event category of ATWS events, the LOOP event is non-limiting for all acceptance criteria.

15.3.1.4 – ATWS - IORV

<u>Description of Event</u>

a) Initiator:
An spurious and spontaneous opening of a single Safety/Relief Valve (S/RV) occurs with failure of the control rods to Scram.

b) Sequence of Events (NOT a time line):
The plant is operating at 100% power, when one S/RV spuriously opens and will not close. Initially, the vessel pressure drops and a minor level swell. The pressure regulator compensates by closing down on the Turbine Control Valves (TCVs). Vessel parameters stabilize at a slightly lower operating state. The open SRV discharges steam to the suppression pool causing the pool temperature to increase and the pool temperature quickly reaches the Technical Specification limit, whereupon the Operators insert a manual Scram. However, there is a failure in the Reactor Protection System (RPS) and/or Control Rod Drive (CRD) System that prevents the control rods from inserting (i.e., does not Scram). The Operators respond to the event, using the Emergency Operating Procedures (EOPs). Operator actions include: starting suppression pool cooling to control pool temperatures; power is mitigated by lowering the water level and inhibiting further vessel injection (i.e., HPCI/RCIC) to maintain a low vessel level in order to control power; and, starting the Standby Liquid Control System (SLCS) pumps to inject boron into the vessel.

Long-term response (beyond the explicit analyzed period): Operators guide the plant to a cold shutdown condition and assure that sufficient boron has been injected to maintain the core in a subcritical state until the control rods can be inserted into the core.

c) Single Failure/Operator Error (as applicable):
Failure of the RPS and/or CRD systems to trip and insert control rods
In addition, no credit is taken for the Alternate Rod Insertion (ARI) System.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
Other than the initial response of the pressure regulator system, no other automatic actions are assumed in this event.

Operators initially inhibit vessel injection by HPCI and RCIC, per EOPs. Operators initiate feedwater runback at 90 seconds from the beginning of the event or at Boron Injection Initiation Temperature (BIIT), whichever comes later, to lower vessel level, per EOPs. Operators trip the Recirculation pumps when suppression pool temperature reaches the BIIT. Operators initiate SLCS at either 120 seconds after the ATWS trip point (low water level or high pressure), or the time at which the suppression pool temperature reaches the BIIT, whichever occurs later, per EOPs. Operators initiate and maximize SPC using both RHR pumps and 2 RHRSW pumps per loop.

Event Category & Acceptance Criteria

This is a Special Event – an Anticipated Operational Transient (Occurrence) with more than a single failure and/or Operator error - a complete failure to Scram the control rods.

Peak Fuel Cladding Temperature < 2200 °F (10 CFR 50.46)
Peak Fuel Local Cladding Oxidation Fraction < 17% (10 CFR 50.46)
Peak Reactor Vessel Pressure < 1500 psig (ASME Service Level C)
Peak Suppression Pool Temperature < 281 °F (design value)
Peak Containment Pressure < 62 psig (110% of the design value)

Methods

a) Calculation Tools & Computer Codes:
Fuel Response: Primary Code – No calculation performed, see Note 1 in the Results section.

Reactor Response: Primary Code – ODYN with approved boron mixing model (see Table 15.0-2 for complete listing, code versions and NRC acceptance).
Containment Response: Primary Code – STEMP, with input from Secondary Code - ODYN (above), for the suppression pool temperature response. Hand calculations are performed assuming thermodynamic equilibrium between the suppression pool and containment airspace to determine the peak wetwell pressure, neglecting the heat sinks of the structure.

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-3, with specific changes as outlined in Table 15.3-1.

c) Key Assumptions:
Plant is initially at rated thermal power and 99% rated core flow (MELLLA).
The S/RV with the lowest opening setpoint is assumed to spuriously open.
The RCIC enthalpy (based upon CST water temperature of 100 °F) is used for all injection flow through the feedwater sparger.
Operators initiate feedwater runback at 90 seconds from the beginning of the event or at BIIT, whichever comes later.

Operators trip the Recirculation pumps upon reaching the BIIT, as the ATWS-RPT trip setpoints are not reached.
Operators initiate SLCS at either 120 seconds after the ATWS trip point (low water level or high pressure), or the time at which the suppression pool temperature reaches the BIIT, whichever occurs later.
Hot Shutdown is defined as neutron flux < 0.1% of rated for > 100 seconds.
May-Witt Decay Heat curve is used in the containment analysis once the plant is in Hot Shutdown.

Results

2012-020 The GNF2 Amendment 22 Compliance to GESTAR II (Reference 15.0-63) requires a plant specific demonstration that the limiting ATWS event response is within the ATWS acceptance criteria. For DAEC, the previously calculated ATWS results, given below, have sufficient margin to the overpressure and suppression pool temperature limit to not require explicit analysis as allowed per Reference 15.0-63. Therefore, while it is recognized that the GNF2 NFI will affect the ATWS results, DAEC is able to demonstrate compliance to the ATWS acceptance criteria for an ATWS event with GNF2 fuel.

a) Comparison to Acceptance Criteria:

Peak Fuel Cladding Temperature = (Note 1)
Peak Fuel Local Cladding Oxidation Fraction = (Note 2)
Peak Reactor Vessel Pressure = 1068 psig (Note 3)
Peak Suppression Pool Temperature = 180.5 °F
Peak Containment Pressure = 9.2 psig

Note 1: There is no increase in cladding temperature from the initial temperature, because a reactor power excursion does not occur and the fuel does not experience boiling transition.
Note 2: Peak Cladding Temperature is below the 1800 °F threshold temperature at which significant cladding oxidation could occur, so the cladding oxidation was not explicitly calculated.
Note 3: The absence of reactor vessel isolation avoids vessel pressurization; therefore, there is no increase in vessel pressure from the initial value.

b) Known Sensitivities:
The time to reach the BIIT, which drives all the Operator actions in this event, is determined by the initial power level, vessel pressure and the capacity of the S/RV that is assumed to spuriously open.

c) Uncertainties in Results:
Emergency reactor depressurization when EOP criteria require it (e.g., the Heat Capacity Temperature Limit (HCTL) for the suppression pool is exceeded) cannot be evaluated with ODYN.

Conclusion

a) Statement of Acceptability:
This event satisfies all the acceptance criteria for this Special Event.

b) Known Conservatisms/Margins:
For this event, the ODYN transient run is terminated before the Hot Shutdown criterion is met (due to a code stability problem). The STEMP code extrapolates a constant steam flow beyond the time the operator restores the water level before imposing the May-Witt decay heat correlation. The amount of steam discharged to the suppression pool was conservatively over-estimated.
Because of limitations in the ODYN model, all ATWS analysis with ODYN is performed with water level lowered to Top of Active Fuel (TAF) + 5 feet and with the conservative bias applied to the user input boron mixing efficiency tables.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
Within the Special Event category of ATWS events, the IORV event is non-limiting for all acceptance criteria.

15.3.2 STATION BLACKOUT (SBO)

This is a Special Event for demonstrating acceptable plant response to a loss of all offsite AC power, coupled with the failure of the on-site AC power sources (i.e., the Emergency Diesel Generators (EDGs)) to start and/or load, a so-called "Station Blackout" event. It should be noted that no single failure of equipment or Operator error can lead to a complete failure of both the on-site and off-site AC sources. Because this ability to cope with such an event was promulgated by rule change (10 CFR 50.63), after the initial licensing of the facility, the methods (inputs and assumptions) used and acceptance criteria applied are less stringent than for design basis events. Also noteworthy, is that credit may be taken for non-safety-related equipment and Operator actions (within the first 10 minutes) in responding to these events, provided there are written procedures and training to implement them.

10 CFR 50.63 requires that the plant be capable of maintaining core cooling and appropriate containment integrity during a Station Blackout and identifies the factors that must be considered. The NRC issued Regulatory Guide 1.155, "Station Blackout," which describes a method acceptable to the NRC staff for meeting the requirements of 10 CFR 50.63. RG 1.155 states that NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," also provides guidance that is essential to RG 1.155 and is acceptable for meeting 10 CFR 50.63 requirements. The assessment and supporting documentation provides assurance that DAEC can successfully cope with a four hour Station Blackout (SBO) event, relying on only AC-independent systems, i.e., DC-powered/controlled, pneumatically-actuated/controlled, or steam-driven (References 15.0-37 and 15.0-38).

Specific areas that are required to be evaluated are:

- Core Cooling*
- Condensate Storage inventory*
- Battery loads and Emergency Diesel Generator (EDG) reliability
- Compressed Air Capacity
- Effects of Loss of Ventilation/Cooling
    - Suppression Pool temperature and wetwell pressure*
    - Drywell temperature and pressure*
    - Spent Fuel Pool cooling
    - High Pressure Cooling Injection (HPCI) Room Temperature
    - Reactor Core Cooling Isolation Cooling (RCIC) Room Temperature
    - Control Room Temperature
    - Battery Room Temperature
- Containment Isolation Capability

Section 15.3.2.1 presents the results of the thermal-hydraulic evaluations (*). The remainder of the Safe Shutdown systems and safety actions (Reference Table 15.3-2), are discussed as follows:

Battery loads and Emergency Diesel Generator Reliability

2011-017 The number of RCIC cycles is insignificant with respect to challenging battery capacity compared to RCIC operation. RCIC will be used when needed, but use will be minimized and RCIC operation will be discontinued whenever possible. Battery capacity can support the required loads under SBO conditions at EPU.

The bounding target reliability of 0.975 is established for a coping duration of four hours, per Regulatory Guide 1.155. This does not change at EPU.

Compressed Air Capacity

The new SBO containment analysis determined that only one Low-Low Set valve operation, in combination with HPCI and RCIC use, occurs prior to the assumed manual depressurization. Some additional manual SRV actuations are needed during the depressurization. Based on the containment analysis for EPU, about 19 SRV lifts are estimated during the remainder of the coping period. Therefore, reasonable assurance of air capacity and operability of the ADS accumulators is established for the SBO event at EPU.

Effects of Loss of Ventilation/Cooling

Spent Fuel Pool (SFP) cooling: The time-to-boil, time to provide makeup water and boiling water loss calculations show that there is still adequate time and capacity of makeup systems to establish the required flow of cooling water prior to spent fuel being uncovered in the SFP. While boiling in the SFP may occur during the four hour coping period, restoration of forced cooling for the SFP can be accomplished prior to a large change in SFP level. Therefore, adequate cooling for the SFP and the radiation shielding for the spent fuel is ensured (Reference Section 9.1.2.3.2.1).

HPCI Equipment Area: The EPU evaluation of HPCI room heatup during SBO shows that area temperature remains within a range that is acceptable for operation of equipment within the room, assuming Operator actions (e.g., opening doors). The evaluation is conservative because the HPCI system is assumed to operate throughout the SBO coping period, when in fact, it is required to operate for only a portion of the period. Other environmental conditions (pressure, humidity, and radiation) remain within normal operational parameters, since no other failures are assumed. Thus, there is reasonable assurance that equipment in the HPCI room is Operable during the SBO coping period.

RCIC Equipment Area: The EPU evaluation of RCIC room heatup during SBO shows that area temperature remains within a range that is acceptable for operation of equipment within the room. The evaluation is conservative because the RCIC system is assumed to operate throughout the SBO coping period, when in fact, it is required to operate for only a portion of the period. Other environmental conditions (pressure, humidity, and radiation) remain within normal operational parameters, since no other failures are assumed. Thus, there is reasonable assurance that equipment in the RCIC room is Operable during the SBO coping period.

Control Room: During the SBO event, heat is added to the control room as a result of operating personnel activities and equipment powered by DC sources. It is conservatively assumed that all energy expended within the control room is converted to heat and contributes to the heat rise. The EPU evaluation of control room heatup shows that all control room equipment necessary to cope with an SBO remain Operable and that all dominant areas of concern remain within acceptable temperature ranges, assuming Operator actions (e.g., opening cabinet doors).

Battery Room: During the SBO event, heat is added to the battery room as a result of supplying equipment powered DC sources. It is conservatively assumed that all energy expended within the battery room is converted to heat and contributes to the heat rise. The pre-EPU analysis determined that all battery room equipment necessary to cope with an SBO remained operable and that all dominant areas of concern remain within acceptable temperature ranges, assuming Operator actions (e.g., opening doors). Since the type of equipment and sequence of equipment operation do not change significantly at EPU, the results of the pre-EPU evaluation are still valid at EPU.

Containment Isolation Capability

The pre-EPU assessment of the effects of SBO on containment isolation concluded that the isolation capability meets the requirements of NRC Regulatory Guide 1.155. This assessment is not affected by the EPU because the containment conditions at the beginning of the SBO event are not changed; therefore, adequate isolation capability is assured during the SBO coping period.

15.3.2.1 – Station Blackout (SBO)

Description of Event

a) Initiator:
A loss of offsite power (LOOP) resulting from a switchyard-related random fault, or other external event, such as a grid disturbance or weather condition, that affects the offsite power system throughout the grid and at the plant. In addition, the on-site AC power sources (i.e., the Emergency Diesel Generators (EDGs)) fail to start and/or load onto the essential busses.

b) Sequence of Events (NOT a time line):
A LOOP occurs, which also causes the main turbine to trip, resulting in a loss of AC power to both the essential and non-essential plant busses. The on-site AC power sources (EDGs) fail to start and/or load onto the essential busses. The plant Scrams on the LOOP/turbine trip condition and Primary Containment Isolation System (PCIS) Group 1 – 5 isolations occur on the loss of RPS power. The MSIVs close quickly and there is an immediate pressure increase, lifting the S/RVs. Low-low set (LLS) initially controls the pressure. The pressure increase causes void collapse resulting in an initial power spike and vessel level shrinks. As feedwater is assumed instantaneously lost at the beginning of the event, water level continues to drop and HPCI and RCIC start on low water level and eventually trip on high level. The Operators inhibit HPCI from further starts, as RCIC is capable of maintaining vessel level; also, to minimize suppression pool heatup and to conserve Condensate Storage Tank (CST) inventory. The injection of the cooler water, coupled with the steam usage by HPCI and RCIC, lower reactor pressure sufficiently to close the LLS valves. Due to the loss of Drywell cooling, Drywell temperature and pressure increase slowly throughout the event. Due to the steam exhaust from both HPCI and RCIC operation, coupled with the S/RV discharges, the suppression pool temperature and wetwell pressure also increase. At 30 minutes into the event, it is assumed that the Operators begin a controlled reactor vessel depressurization, using the S/RVs, at a cooldown rate of 80 °F/hour to limit the temperature rise in the primary containment. They stop the depressurization prior to reaching the RCIC low pressure operating range to maintain it available. RCIC cycles on and off between low and high level trips maintaining vessel water level. The Torus-to-Drywell vacuum breakers open to equalize the pressure between the wetwell airspace and the Drywell. The Operators manually open S/RVs, as needed, to maintain vessel pressure. Near the end of the 4-hour coping period, suppression pool water temperature reaches the Heat Capacity Temperature Limit (HCTL) in the Emergency Operating
2014-009 Procedures (EOPs) and the Operators will depressurize the RPV using SRVs while maintaining adequate pressure in the RPV to ensure the availability of steam driven systems to inject to maintain adequate core cooling. Due to the lower initial starting pressure, the rate of depressurization is slow enough that the end of the 4-hour coping period is reached before RCIC isolates on low pressure. Thus, vessel level is maintained throughout the coping period. At the end of the coping period, AC power is restored and the Operators initiate both suppression pool cooling and shutdown cooling to bring the plant to a controlled, cold shutdown condition.

During the SBO event, the Operators will use their procedures to assure that adequate room cooling is maintained throughout the necessary areas of the plant by opening doors/cabinets, using auxiliary cooling equipment, etc., to ensure necessary equipment is maintained in a functional capability during the room heatup due to loss of forced cooling and ventilation in the SBO.

c) Single Failure/Operator Error (as applicable):
Loss of the on-site AC power sources (EDGs) to start and/or load onto the essential busses.

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
Reactor Scram on LOOP/turbine trip; PCIS Groups 1-5 isolations; MSIV closure; Feedpump trip; S/RVs and LLS valves open/close; HPCI and RCIC start/stop on vessel low and high water levels, Torus-to-Drywell vacuum breakers open/close.
Operators inhibit HPCI operation after its initial cycle; Operators begin a controlled vessel depressurization at 30 minutes into the event at 80 °F/hour, stopping at the lower operating range of RCIC; Operators make use of procedures and auxiliary cooling equipment to maintain room temperatures within the maximum operating ranges of the equipment; Upon restoration of AC power at the end of the coping period, Operators will bring the plant to a cold shutdown condition by operating suppression pool cooling and shutdown cooling modes of RHR.

Event Category & Acceptance Criteria

This is a Special Event – to demonstrate the ability of the plant to cope with a loss of all (on and off-site) AC power sources for a 4 hour period.

Adequate Core Cooling – Vessel Level > Top of Active Fuel (TAF)
Peak Containment (Drywell and Wetwell) Pressure < 56 psig (design value)
Peak Suppression Pool Temperature < 281 °F (design value)
Peak Drywell Air Temperature < 340 °F (design value)
Peak Drywell Shell (Metal) Temperature < 281 °F (design value)
CST Inventory Usage < 75,000 gals (minimum reserved volume)
Suppression Pool Level < 13.5 ft (EOPs – Vacuum Breakers)

Equipment outside of containment needs to be evaluated for Operability only if located in a dominant area of concern, where ambient temperatures exceed 120 °F during the event. Otherwise, there is a reasonable assurance of equipment Operability.

Methods

a) Calculation Tools & Computer Codes:
Reactor Vessel and Containment Response: Primary Code – SHEX (see Table 15.0-2 for complete listing, code versions and NRC acceptance). Some minor hand calculations/spreadsheet are performed to convert the units of the computer code output.

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-4a inputs (Table 15.0-6) are used, with the following changes:
Initial reactor power and core flow are at rated conditions for 100 days prior to the SBO event.

Decay Heat Curve - ANS 5.1-1979, nominal (i.e., no adders for uncertainty)
MSIVs close in 3.5 seconds (0.5 seconds for logic and 3.0 second stroke time)

CST is at an initial capacity of 75,000 gallons (620,482 lbm) at 100°F.
Initial drywell and wetwell airspace pressures are 15.4 psia.
Initial suppression pool level is at the Technical Specification minimum limit of 58,900 ft^3 (10.1 ft).

Initial heat loads are based on the initial drywell temperature of 135°F and an RPV temperature of 550°F.
Drywell heatup calculations take credit for heat sinks of the structures within the Drywell.

Initial temperatures in various locations:
85 °F in the Control Room and Essential Switchgear Room,
90 °F in the Cable Spreading Room and Administrative Building,
91 °F for Outside ambient conditions,
104 °F in the Turbine and Control Buildings and the RCIC Room,
120 °F in the HPCI Room and,
190 °F in the Steam Tunnel.

c) Key Assumptions:
The analyzed sequence is a hybrid scenario of an automatic response and an operator-controlled response. An automatic response is assumed for the first 30-minutes and then the Operator-controlled response is assumed thereafter.
Reactor Scram is instantaneous at the beginning of the event. (Reactor Scram is AC-power independent, as RPS is designed to "fail safe" on loss of power.)
Feedwater injection goes to zero instantaneously (no coastdown or runout).

The CST is the sole suction source for HPCI and RCIC (no transfer to suppression pool)
Operators start manually depressurizing the vessel at 30 minutes at 80°F/hr using SRVs and are assumed to be manually operated to maintain RPV pressure within 50 psi of the desired RPV pressure or temperature.
For the CST inventory analysis only, a total drywell leakage of 61 gpm (25 gpm for the primary system leakage and 36 gpm for the recirculation pump seal leakage) is required to be assumed.
For the suppression pool heat-up analysis the minimum torus level is assumed throughout the event, because suppression pool temperature is the limiting parameter (i.e., Heat Capacity Temperature Limit (HCTL)).

Results

a) Comparison to Acceptance Criteria:

Minimum Vessel Level - > 50 inches TAF
Peak Drywell Pressure = 8.7 psig
Peak Wetwell Pressure = 9.1 psig
Peak Suppression Pool Temperature = 193.2 °F
Peak Drywell Air Temperature = 307.5 °F
Peak Drywell Shell (Metal) Temperature = (Note)
CST Inventory Usage = 66,734 gallons (552,100 lbm)
Maximum Suppression Pool Level = 11.1 ft

Note: The analysis shows that the Drywell shell temperature limit is reached after 3.7 hours. However, the duration above the temperature limit is short and the Drywell pressure is low. Therefore, adequate drywell integrity and recovery capability is assured for the SBO event.

The pre-EPU analysis determined that all equipment necessary to cope with an SBO remained Operable and that all dominant areas of concern remain within acceptable temperature ranges, assuming Operator actions (e.g., opening doors). Since the type of equipment and sequence of equipment operation do not change significantly at EPU and reactor steam temperature and pressure does not change with EPU, the results of the pre-EPU evaluation are still valid at EPU.

b) Known Sensitivities:
The torus levels, based on using the maximum Technical Specification level of 61,500 ft^3, are approximately 0.3 feet higher than those using the minimum level. This torus water level response does not increase to the point during the 4 hour SBO period such that pressure suppression capability or containment integrity is challenged and the resulting suppression pool level remains below EOP limits of concern for high suppression pool level.

c) Uncertainties in Results:
The SHEX code does not model vessel level accurately, as it is based upon a simple mass and energy balance.
The torus water level analysis does not account for the toroidal shape of the DAEC containment, but treats it like a rectangular structure. However, the difference between the torus level at a given time and the initial torus level closely approximates the actual values.

Conclusion

a) Statement of Acceptability:
The results of this analysis support the conclusion that the plant is able to cope with the SBO event for the required duration of four hours.

b) Known Conservatisms/Margins:
The CST inventory evaluation is based upon an assumed 61 gpm primary coolant leakage rate, which is much greater than would realistically be expected.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
This is a unique Special Event.

2013-013
15.3.3 Fire – SAFE SHUTDOWN ANALYSIS

2013-013 In September of 2013, DAEC received a license amendment to transition the fire protection program from 10 CFR 50.48(b) and 10 CFR 50 Appendix R to a risk-informed, performance-based program per 10 CFR 50.48(c) which incorporates by reference NFPA 805.

The nuclear safety goal of NFPA 805 is to provide reasonable assurance that a fire during any operational mode and plant configuration will not prevent the plant from achieving and maintaining the fuel in a safe and stable condition. A safe and stable condition is defined as the ability to maintain $K_{eff} < 0.99$, with a reactor coolant temperature at or below the requirements for hot shutdown.

To meet this nuclear safety goal, Fire Protection features shall be capable of providing reasonable assurance that, in the event of a fire, the plant is not placed in an unrecoverable condition. The licensee must demonstrate the ability to maintain one success path of required equipment effectively free of fire damage. This was accomplished by developing and analyzing a comprehensive list of systems and equipment to identify those critical components required to achieve and maintain hot shutdown conditions following any fire occurring with the reactor operating at power, shutdown prior to aligning the RHR system for shutdown cooling, or in transition between these two operational phases.

The NFPA 805 analysis demonstrates the assured success paths for each fire area. The location and severity of the fire dictates whether the bounding (i.e., worst case, complete room burnout) compliance strategy or a limited response (i.e., addressing partial area scenarios with specific ignition sources and zones of influence) is implemented. To meet the nuclear safety goal of NFPA 805, DAEC relies upon redundant and independent trains of safe shutdown systems controlled from the main control room. For fire scenarios that result in main control room abandonment, DAEC relies on an Alternative Shutdown Capability controlled from [redacted] (Alternate Shutdown Panel).

Although the analysis described here was originally developed to support 10 CFR 50 Appendix R compliance, NFPA 805 also relies on this analysis to demonstrate achievement of the NFPA 805 Nuclear Safety Performance Criteria.

With the exception of Operator actions taken in the Control Room prior to abandonment, this analysis only credits the designated Remote Shutdown System (RSS). The RSS is composed of the two pumps in the B loop of the RHR system in the three modes of operation: the Low Pressure Coolant Injection (LPCI) mode, the Shutdown Cooling (SDC) mode, and the Suppression Pool Cooling (SPC) mode. The RSS also includes one Core Spray (CS) system, three Safety/Relief Valves (S/RVs), two RHR Service Water (RHRSW) pumps, an Emergency Diesel Generator (EDG) and its associated Emergency Service Water (ESW) pump and the necessary monitoring and auxiliary equipment to support these functions.

2013-013 To demonstrate that the plant can achieve the NFPA 805 Nuclear Safety Performance Criteria, the following 4 scenarios are evaluated:

Event 1: No spurious operation of any plant equipment occurs and the Operator initiates 2 SRVs at the RSS panel to depressurize the Reactor Pressure Vessel (RPV) at 30 minutes into the event.

Event 2: One SRV spuriously opens during Hot Shutdown. The Operator closes the spuriously-opened SRV within 20 minutes at the RSS panel and begins to depressurize the reactor at 20 minutes into the event using 2 SRVs.

Event 3: A SRV spuriously opens for the first 10 minutes and the Operator starts to depressurize the RPV at 20 minutes using 2 SRVs at the RSS panel.

Event 4: The isolation valves on a one-inch liquid line representing a high-low pressure interface spuriously open at event initiation. The Operator
2012-020 depressurizes at 20 minutes utilizing 3 S/RVs at the RSS panel. The Operator is not able to close these valves until the reactor is close to achieving Cold Shutdown.

2013-013

2013-013

15.3.3.1 – Fire – No Spurious Operations

Description of Event

a) Initiator:
A spontaneous fire breaks out in the Control Room, necessitating that the Operators evacuate the Control Room and bring the plant to a safe shutdown condition from outside the Control Room.

b) Sequence of Events (NOT a time line):
A fire breaks out in the Control Room necessitating that the Operators evacuate the Control Room and manage the event from outside the Control Room. Simultaneously, a Loss-of-Offsite Power (LOOP) is assumed to occur. The plant trips immediately (Scram) and the MSIVs go closed - either from the LOOP or by manual action by the Operators. The Feedwater pumps quickly coast down after the LOOP. Reactor pressure increases quickly and the S/RVs open to control the pressure for the first 30 minutes of the event. Reactor water level steadily decreases with each cycling open/closed of the S/RVs. HPCI and RCIC are assumed to not be available, due to the fire. At 30 minutes, it is assumed that the Operators take manual control of the plant from the Remote Shutdown Panels (i.e., Remote Shutdown System). From the RSS, the Operators begin vessel depressurization using 2 S/RVs, start the "B" EDG and its associated ESW pump, and start the "B" CS pump (and/or RHR pumps). Water level inside the shroud drops below the Top of Active Fuel (TAF) and fuel cladding temperatures begin to rise. When reactor pressure reaches the low-pressure permissives, either the CS or RHR inject valves are opened manually and the pump(s) will begin to inject into the vessel, quickly restoring level to normal and terminating the rise in fuel cladding temperature. (Note: credit is only taken for either the CS pump OR the 2 RHR pumps for vessel makeup.) After reactor inventory is sufficiently recovered and reactor pressure has decreased below the SDC interlock pressure, the Operators place one RHR pump into Shutdown Cooling mode, along with the requisite 2 RHR Service Water (RHRSW) pumps to begin cooling the reactor down. Once reactor pressure is below their backpressure limit, the 2 S/RVs close and suppression pool temperature and pressure stabilize. The plant reaches Cold Shutdown approximately 7.5 hours after shutdown cooling is initiated.

c) Single Failure/Operator Error (as applicable):
None

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
Reactor Scram (LOOP or Manual); MSIV closure (LOOP or Manual); S/RVs cycle open/closed;

Operators begin manual vessel depressurization using 2 S/RVs, at 30 minutes into the event; Operators start the "B" EDG and ESW pump, CS (or RHR) pump(s), and open the CS (or RHR) inject valves; Operators secure vessel injection after restoring level to normal; Operators initiate Shutdown Cooling mode, using 1 RHR pump and 2 RHRSW pumps, after vessel injection and depressurization below the SDC interlock pressure.

Event Category & Acceptance Criteria

This is a Special Event – to demonstrate the plant's ability to cope with a fire in the Control Room that requires the Operators to evacuate and bring the plant to a hot shutdown condition from outside the Control Room.

Fuel Cladding Integrity – Vessel Level > Top of Active Fuel (TAF) (Note)
Peak Reactor Pressure < 1375 psig (ASME Code Level C)
Peak Containment (Drywell and Wetwell) Pressure < 56 psig (design value)
Peak Suppression Pool Temperature < 281 °F (design value)
Peak Drywell Air Temperature < 281 °F (design value)
Suppression Pool Heat Capacity < Heat Capacity Temperature Limit Curve (EOPs)
Adequate NPSH for ECCS pumps
Time to reach Cold Shutdown < 72 hours from beginning of event.

2013-013

Methods

a) Calculation Tools & Computer Codes:
Reactor Vessel Response: Primary Code – SAFER (see Table 15.0-2 for complete listing, code versions and NRC acceptance).
Fuel Temperature Response: Primary Code – GESTR-LOCA (see Table 15.0-2 for complete listing, code versions and NRC acceptance).
Primary Containment Response: Primary Code – SHEX (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-4 (Table 15.0-4) for Fuel Temperature Response.
OPL-4a (Table 15.0-6) are used for Containment Response.
The following changes to the above input sources are made for this event:
Initial reactor conditions (power, core flow, dome pressure, vessel level) are at nominal, rated (1912 MWt) conditions.
Decay Heat Curve - ANS 5.1-1979, nominal (i.e., no adders for uncertainty)

GE14 fuel design
MSIVs close in 5.0 seconds

Initial drywell and wetwell airspace pressures are 17.0 psia (2.3 psig).
Initial suppression pool level is at the Technical Specification minimum limit of 58,900 ft^3 (10.1 ft).
Initial Drywell temperature of 135 ºF.
Initial Suppression Pool Temperature of 95 ºF.
RHR Heat Exchanger Effectiveness of 135 Btu/sec-ºF.

c) Key Assumptions:
Concurrent LOOP with the initiation of the Control Room fire.
Reactor Scram is instantaneous at the beginning of the event.
Feedwater injection ramps to zero in 5 seconds.
Either a single division of RHR in the LPCI mode (2 pumps) <u>OR</u> a single CS pump is assumed for vessel inventory makeup.
HPCI and RCIC are not available for this event.
SRV opening nominal trip setpoint for Group 1 is 1110 psig. Applying a 4% blowdown range, the closing nominal trip setpoint is 1066 psig. (The blowdown range is the ratio of the difference between the opening and closing pressure setpoints to the opening pressure setpoints.)
Low-low Set Logic/Valves are not available for this event.
2012-020 Operators begin vessel depressurization, using 2 S/RVs, at 20 minutes into the event.

<u>Results</u>

a) Comparison to Acceptance Criteria:

Peak Cladding Temperature = 810 ºF (CS) / 989 ºF (LPCI)
Peak Vessel Pressure = 1145 psig
Peak Drywell Pressure = 21 psia
Peak Drywell Temperature = 140 ºF
Peak Suppression Pool Temperature = 173.5 ºF (CS) / 176.9 ºF (LPCI)
Peak Wetwell Pressure = 20.8 psia (CS) / 21.1 psia (LPCI)
Heat Capacity Temperature Limit Curve = satisfied (Suppression Pool temperature before reactor depressurization is approximately 130 ºF)
NPSH for ECCS pumps = Adequate (low Suppression Pool pressure and temperature)
Time to reach Cold Shutdown ≈ 8.5 hours from beginning of event.

b) Known Sensitivities:
As can be seen from the above results, the Suppression Pool responses are not very sensitive to the injection source (CS or RHR) used. However, the PCT result is mildly sensitive to the injection source, as core reflood occurs sooner using CS than RHR.

Long-term decay heat removal can to be accomplished by SDC mode, as assumed in the case above, or by using Suppression Pool Cooling mode. A sensitivity case was performed to determine the limiting time for the initiation of decay heat removal, based upon suppression pool heatup and the associated effect on CS (or RHR) pump net positive suction head (NPSH) requirements (Reference CAL-M03-002, Rev. 1). The results of this analysis determined that adequate NPSH margin can be maintained provided decay heat removal is initiated within 2.5 hours from the beginning of the event.

c) Uncertainties in Results:
Use of conservative assumptions (e.g., simultaneous LOOP) in the evaluation are intended to bound the uncertainties in the final results.

Conclusion

a) Statement of Acceptability:
The results of this analysis support the conclusion that the plant is able to cope with the Control Room fire and bring the plant to a Hot Shutdown condition initially and a Cold Shutdown condition within 72 hours following the event.

b) Known Conservatisms/Margins:
For fuel analysis with the SAFER code, the part length fuel rods in the GE14 assemblies are treated as full-length rods, which conservatively overestimates the hot bundle power.
The LLS function was designed to extend the time between S/RV subsequent actuation to ensure sufficient time to allow the water leg inside the S/RV discharge line to return the normal level before each subsequent actuation. The time between S/RV subsequent actuation without LLS is approximately 18 seconds which far exceeds the minimum 3.7 seconds requirement established as the time limit for subsequent openings of S/RVs. Therefore, based on this evaluation, the automatic function of LLS is not required for a fire event that results in Control Room evacuation.
The spurious operation of the RHR pool suction valve was considered and it was concluded that it would not affect the safe shutdown capability because only one RHR pump is sufficient to achieve and maintain Cold Shutdown within 72 hours.
The spurious operation of the RHR minimum flow valve was considered and it was concluded that it is not detrimental to the RHR pump and would not affect the safe shutdown capability.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
2013-013 This is a unique Special Event. Within the category of Fire events, this event is limiting for Suppression Pool temperature and NPSH.

2013-013

15.3.3.2 – Fire – Spurious Opening of One S/RV (20 mins)

Description of Event

a) Initiator:
A spontaneous fire breaks out in the Control Room, necessitating that the Operators evacuate the Control Room and bring the plant to a safe shutdown condition from outside the Control Room.

b) Sequence of Events (NOT a time line):
A fire breaks out in the Control Room necessitating that the Operators evacuate the Control Room and manage the event from outside the Control Room. The fire is assumed to cause one Safety/Relief Valve (S/RV) to spuriously open at the beginning of the event. Simultaneously, a Loss-of-Offsite Power (LOOP) is assumed to occur. The plant trips immediately (Scram) and the MSIVs go closed - either from the LOOP or by manual action by the Operators. The Feedwater pumps quickly coast down after the LOOP. Reactor pressure spikes initially with the closing of the MSIVs, but the spuriously-opened S/RV turns around the pressure increase and slowly depressurizes the vessel for the first 20 minutes of the event. Reactor water level drops initially on the void collapse from the MSIV closure and then steadily decreases due to the open S/RV. HPCI and RCIC are assumed to not be available, due to the fire. Water level inside the shroud drops below the Top of Active Fuel (TAF) and fuel cladding temperatures begin to rise. At 20 minutes, it is assumed that the Operators take manual control of the plant from the Remote Shutdown Panels (i.e., Remote Shutdown System) and are successful in closing the open S/RV. From the RSS, the Operators begin vessel drepressurization using 2 S/RVs, start the “B” EDG and its associated ESW pump, and start the “B” CS pump (and/or RHR pumps). When reactor pressure reaches the low-pressure permissives, either the CS <u>or</u> RHR inject valves are opened manually and the pump(s) will begin to inject into the vessel, quickly restoring level to normal and terminating the rise in fuel cladding temperature. (Note: credit is only taken for either the CS pump <u>OR</u> the 2 RHR pumps for vessel makeup.) After reactor inventory is sufficiently recovered and reactor pressure has decreased below the SDC interlock pressure, the Operators place one RHR pump into Shutdown Cooling mode, along with the requisite 2 RHR Service Water (RHRSW) pumps to begin cooling the reactor down. Once reactor pressure is below their backpressure limit, the 2 S/RVs close and suppression pool temperature and pressure stabilize. The plant reaches Cold Shutdown approximately 7.5 hours after shutdown cooling is initiated.

c) Single Failure/Operator Error (as applicable):
None

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures): Reactor Scram (LOOP or Manual); MSIV closure (LOOP or Manual); S/RVs cycle open/closed; Operators close the spuriously-open S/RV and begin manual vessel depressurization, using 2 other S/RVs, at 20 minutes into the event; Operators start the "B" EDG and ESW pump, CS (or RHR) pump(s), and open the CS (or RHR) inject valves; Operators secure vessel injection after restoring level to normal; Operators initiate Shutdown Cooling mode, using 1 RHR pump and 2 RHRSW pumps, after vessel injection and depressurization below the SDC interlock pressure.

Event Category & Acceptance Criteria

This is a Special Event – to demonstrate the plant's ability to cope with a fire in the Control Room that requires the Operators to evacuate and bring the plant to a hot shutdown condition from outside the Control Room.

Fuel Cladding Integrity – Vessel Level > Top of Active Fuel (TAF) (Note)
Peak Reactor Pressure < 1375 psig (ASME Code Level C)
Peak Containment (Drywell and Wetwell) Pressure < 56 psig (design value)
Peak Suppression Pool Temperature < 281 °F (design value)
Peak Drywell Air Temperature < 281 °F (design value)
Suppression Pool Heat Capacity < Heat Capacity Temperature Limit Curve (EOPs)
Adequate NPSH for ECCS pumps
Time to reach Cold Shutdown < 72 hours from beginning of event.

2013-013

Methods

a) Calculation Tools & Computer Codes:
Reactor Vessel Response: Primary Code – SAFER (see Table 15.0-2 for complete listing, code versions and NRC acceptance).
Fuel Temperature Response: Primary Code – GESTR-LOCA (see Table 15.0-2 for complete listing, code versions and NRC acceptance).
Primary Containment Response: Primary Code – SHEX (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-4 (Table 15.0-4) for Fuel Temperature Response.
OPL-4a (Table 15.0-6) are used for Containment Response.
The following changes to the above input sources are made for this event:
Initial reactor conditions (power, core flow, dome pressure, vessel level) are at nominal, rated (1912 MWt) conditions.
Decay Heat Curve - ANS 5.1-1979, nominal (i.e., no adders for uncertainty)
GE14 fuel design
MSIVs close in 5.0 seconds
Initial drywell and wetwell airspace pressures are 17.0 psia (2.3 psig).
Initial suppression pool level is at the Technical Specification minimum limit of 58,900 ft^3 (10.1 ft).

Initial Drywell temperature of 135 ºF.
Initial Suppression Pool Temperature of 95 ºF.
RHR Heat Exchanger Effectiveness of 135 Btu/sec-ºF.

c) Key Assumptions:

Concurrent LOOP with the initiation of the Control Room fire.
Reactor Scram is instantaneous at the beginning of the event.
Feedwater injection ramps to zero in 5 seconds.
Either a single division of RHR in the LPCI mode (2 pumps) OR a single CS pump is assumed for vessel inventory makeup.
HPCI and RCIC are not available for this event.
SRV opening nominal trip setpoint for Group 1 is 1110 psig. Applying a 4% blowdown range, the closing nominal trip setpoint is 1066 psig. (The blowdown range is the ratio of the difference between the opening and closing pressure setpoints to the opening pressure setpoints.)
2012-020 Operators begin vessel depressurization, using 2 S/RVs, at 20 minutes into the event.

Results

a) Comparison to Acceptance Criteria:

Peak Cladding Temperature = 591 ºF (CS) / 591 ºF (LPCI) (Note)
Peak Vessel Pressure < 1145 psig
Peak Drywell Pressure = 21 psia
Peak Drywell Temperature = 140 ºF
Peak Suppression Pool Temperature = 169.1 ºF (CS) / 172.4 ºF (LPCI)
Peak Wetwell Pressure = 20.5 psia (CS) / 20.7 psia (LPCI)
Heat Capacity Temperature Limit Curve = satisfied (Suppression Pool temperature before reactor depressurization is approximately 140 ºF)
NPSH for ECCS pumps = Adequate (low Suppression Pool pressure and temperature)
Time to reach Cold Shutdown ≈ 8.0 hours from beginning of event.

Note: The PCT is the initial cladding temperature, as the open S/RV causes a steady pressure drop in the core and increased core flow, improving the heat transfer and PCT initially decreases until the water level drops below TAF. However, the core uncovery time is short and the corresponding PCT increase does not exceed the initial cladding temperature.

b) Known Sensitivities:
As can be seen from the above results, neither the PCT nor Suppression Pool responses are very sensitive to the injection source (CS or RHR) used.

c) Uncertainties in Results:
Use of conservative assumptions (e.g., simultaneous LOOP) in the evaluation are intended to bound the uncertainties in the final results.

Conclusion

a) Statement of Acceptability:
The results of this analysis support the conclusion that the plant is able to cope with the Control Room fire and bring the plant to a Hot Shutdown condition initially and a Cold Shutdown condition within 72 hours following the event.

b) Known Conservatisms/Margins:
The spurious operation of the RHR pool suction valve was considered and it was concluded that it would not affect the safe shutdown capability because only one RHR pump is sufficient to achieve and maintain Cold Shutdown within 72 hours. The spurious operation of the RHR minimum flow valve was considered and it was concluded that it is not detrimental to the RHR pump and would not affect the safe shutdown capability.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
2013-013 This is a unique Special Event. Within the category of Fire events, this event is non-limiting for any key parameter.

2013-013 **15.3.3.3 – Fire – Spurious Opening of One S/RV (10 mins)**

Description of Event

a) Initiator:
A spontaneous fire breaks out in the Control Room, necessitating that the Operators evacuate the Control Room and bring the plant to a safe shutdown condition from outside the Control Room.

b) Sequence of Events (NOT a time line):
A fire breaks out in the Control Room necessitating that the Operators evacuate the Control Room and manage the event from outside the Control Room. The fire is assumed to cause one Safety/Relief Valve (S/RV) to spuriously open at the beginning of the event. Simultaneously, a Loss-of-Offsite Power (LOOP) is assumed to occur. The plant trips immediately (Scram) and the MSIVs go closed - either from the LOOP or by manual action by the Operators. The Feedwater pumps quickly coast down after the LOOP. Reactor pressure spikes initially with the closing of the MSIVs, but the spuriously-opened S/RV turns around the pressure increase and slowly depressurizes the vessel for the first 10 minutes of the event, at which time it is assumed that the S/RV closes. Reactor water level drops initially on the void collapse from the MSIV closure and then steadily decreases due to the open S/RV. HPCI and RCIC are assumed to not be available, due to the fire. Water level inside the shroud drops below the Top of Active Fuel

(TAF) and fuel cladding temperatures begin to rise. At 20 minutes, it is assumed that the Operators take manual control of the plant from the Remote Shutdown Panels (i.e., Remote Shutdown System). From the RSS, the Operators begin vessel depressurization using 2 S/RVs, start the “B” EDG and its associated ESW pump, and start the “B” CS pump (and/or RHR pumps). When reactor pressure reaches the low-pressure permissives, either the CS or RHR inject valves are opened manually and the pump(s) will begin to inject into the vessel, quickly restoring level to normal and terminating the rise in fuel cladding temperature. (Note: credit is only taken for either the CS pump OR the 2 RHR pumps for vessel makeup.) After reactor inventory is sufficiently recovered and reactor pressure has decreased below the SDC interlock pressure, the Operators then place one RHR pump into Shutdown Cooling mode, along with the requisite 2 RHR Service Water (RHRSW) pumps to begin cooling the reactor down. Once reactor pressure is below their backpressure limit, the 2 S/RVs close and suppression pool temperature and pressure stabilize. The plant reaches Cold Shutdown approximately 7.5 hours after Shutdown Cooling is initiated.

c) Single Failure/Operator Error (as applicable):
None

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
Reactor Scram (LOOP or Manual); MSIV closure (LOOP or Manual); S/RVs cycle open/closed;

Operators begin manual vessel depressurization, using 2 S/RVs, at 20 minutes into the event; Operators start the “B” EDG and ESW pump, CS (or RHR) pumps(s), and open the CS (or RHR) inject valves; Operators secure vessel injection after restoring level to normal; Operators initiate Shutdown Cooling mode, using 1 RHR pump and 2 RHRSW pumps, after vessel injection and depressurization below the SDC interlock pressure.

Event Category & Acceptance Criteria

This is a Special Event – to demonstrate the plant’s ability to cope with a fire in the Control Room that requires the Operators to evacuate and bring the plant to a hot shutdown condition from outside the Control Room.

Fuel Cladding Integrity – Vessel Level > Top of Active Fuel (TAF) (Note)
Peak Reactor Pressure < 1375 psig (ASME Code Level C)
Peak Containment (Drywell and Wetwell) Pressure < 56 psig (design value)
Peak Suppression Pool Temperature < 281 °F (design value)
Peak Drywell Air Temperature < 281 °F (design value)
Suppression Pool Heat Capacity < Heat Capacity Temperature Limit Curve (EOPs)
Adequate NPSH for ECCS pumps
Time to reach Cold Shutdown < 72 hours from beginning of event.

2013-013

Methods

a) Calculation Tools & Computer Codes:
Reactor Vessel Response: Primary Code – SAFER (see Table 15.0-2 for complete listing, code versions and NRC acceptance).
Fuel Temperature Response: Primary Code – GESTR-LOCA (see Table 15.0-2 for complete listing, code versions and NRC acceptance).
Primary Containment Response: Primary Code – SHEX (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-4 (Table 15.0-4) for Fuel Temperature Response.
OPL-4a (Table 15.0-6) are used for Containment Response.
The following changes to the above input sources are made for this event:
Initial reactor conditions (power, core flow, dome pressure, vessel level) are at nominal, rated (1912 MWt) conditions.
Decay Heat Curve - ANS 5.1-1979, nominal (i.e., no adders for uncertainty)
GE14 fuel design
MSIVs close in 5.0 seconds
Initial drywell and wetwell airspace pressures are 17.0 psia (2.3 psig).
Initial suppression pool level is at the Technical Specification minimum limit of 58,900 ft^3 (10.1 ft).
Initial Drywell temperature of 135 ºF.
Initial Suppression Pool Temperature of 95 ºF.
RHR Heat Exchanger Effectiveness of 135 Btu/sec-ºF.

c) Key Assumptions:
Concurrent LOOP with the initiation of the Control Room fire.
Reactor Scram is instantaneous at the beginning of the event.
Feedwater injection ramps to zero in 5 seconds.
Either a single division of RHR in the LPCI mode (2 pumps) OR a single CS pump is assumed for vessel inventory makeup.
HPCI and RCIC are not available for this event.
SRV opening nominal trip setpoint for Group 1 is 1110 psig. Applying a 4% blowdown range, the closing nominal trip setpoint is 1066 psig. (The blowdown range is the ratio of the difference between the opening and closing pressure setpoints to the opening pressure setpoints.)
Low-low Set Logic/Valves are not available for this event.
2012-020 Operators begin vessel depressurization, using 2 S/RVs, at 20 minutes into the event.

Results

a) Comparison to Acceptance Criteria:

Peak Cladding Temperature = 591 ºF (CS) / 591 ºF (LPCI) (Note)
Peak Vessel Pressure < 1145 psig
Peak Drywell Pressure = 21 psia
Peak Drywell Temperature = 140 ºF
Peak Suppression Pool Temperature = 169.1 ºF (CS) / 172.4 ºF (LPCI)
Peak Wetwell Pressure = 20.5 psia (CS) / 20.7 psia (LPCI)
Heat Capacity Temperature Limit Curve = satisfied (Suppression Pool temperature before reactor depressurization is approximately 140 ºF)
NPSH for ECCS pumps = Adequate (low Suppression Pool pressure and temperature)
Time to reach Cold Shutdown ≈ 8.1 hours from beginning of event.

Note: The PCT is the initial cladding temperature, as the open S/RV causes a steady pressure drop in the core and increased core flow, improving the heat transfer and PCT initially decreases until the water level drops below TAF. However, the core uncovery time is short and the corresponding PCT increase does not exceed the initial cladding temperature.

b) Known Sensitivities:
As can be seen from the above results, neither the PCT nor Suppression Pool responses are very sensitive to the injection source (CS or RHR) used.

c) Uncertainties in Results:
Use of conservative assumptions (e.g., simultaneous LOOP) in the evaluation are intended to bound the uncertainties in the final results.

Conclusion

a) Statement of Acceptability:
The results of this analysis support the conclusion that the plant is able to cope with the Control Room fire and bring the plant to a Hot Shutdown condition initially and a Cold Shutdown condition within 72 hours following the event.

b) Known Conservatisms/Margins:
The spurious operation of the RHR pool suction valve was considered and it was concluded that it would not affect the safe shutdown capability because only one RHR pump is sufficient to achieve and maintain Cold Shutdown within 72 hours. The spurious operation of the RHR minimum flow valve was considered and it was concluded that it is not detrimental to the RHR pump and would not affect the safe shutdown capability.

2013-013 c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
This is a unique Special Event. Within the category of Fire events, this event is non-limiting for any key parameter.

2013-013 15.3.3.4 – Fire – With Leakage from a One Inch Line

Description of Event

a) Initiator:
A spontaneous fire breaks out in the Control Room, necessitating that the Operators evacuate the Control Room and bring the plant to a safe shutdown condition from outside the Control Room.

b) Sequence of Events (NOT a time line):
A fire breaks out in the Control Room necessitating that the Operators evacuate the Control Room and manage the event from outside the Control Room. The fire is assumed to cause spurious operation of the vessel isolation valve(s) on a one-inch liquid line that is also a high-low pressure interface at the beginning of the event. Simultaneously, a Loss-of-Offsite Power (LOOP) is assumed to occur. The plant trips immediately (Scram) and the MSIVs go closed - either from the LOOP or by manual action by the Operators. The Feedwater pumps quickly coast down after the LOOP. Reactor pressure spikes initially with the closing of the MSIVs and the S/RVs open to control the pressure. Reactor water level drops initially on the void collapse from the MSIV closure and then steadily decreases due to the cycling open/closed of the S/RVs. HPCI and RCIC are assumed to not be available, due to the fire. At about 20 minutes, water level inside the shroud drops below the Top of Active Fuel (TAF) and it is assumed that the Operators take manual control of the plant from the Remote Shutdown Panels (i.e., Remote Shutdown System). From
2012-020 the RSS, the Operators begin vessel depressurization using 3 S/RVs, start the “B” EDG and its associated ESW pump, and start the “B” CS pump (and/or RHR pumps). Because of the earlier loss of inventory due to the cycling-open SR/Vs, the manual depressurization further drops water level inside the shroud significantly below (TAF) and fuel cladding temperatures begin to rise sharply. When reactor pressure reaches the low-pressure permissives, either the CS <u>or</u> RHR inject valves are opened manually and the pump(s) will begin to inject into the vessel, quickly restoring level to normal and terminating the rise in fuel cladding temperature. (Note: credit is only taken for either the CS pump <u>OR</u> the 2 RHR pumps for vessel makeup.) After reactor inventory is sufficiently recovered and reactor pressure has decreased below the SDC interlock pressure, the Operators then place one RHR pump into Shutdown Cooling mode, along with the requisite 2 RHR Service Water (RHRSW) pumps to begin cooling the reactor down. Once reactor pressure is below
2012-020 their backpressure limit, the 3 S/RVs close and suppression pool temperature and pressure stabilize. The plant reaches Cold Shutdown approximately 7.5 hours after Shutdown Cooling is initiated.

c) Single Failure/Operator Error (as applicable):
None

d) Key Equipment Responses (trips/actuations) & Operator Actions (successes & failures):
Reactor Scram (LOOP or Manual); MSIV closure (LOOP or Manual); S/RVs cycle
2012-020 open/closed; Operators begin manual vessel depressurization, using 3 other S/RVs, at 20 minutes into the event; Operators start the "B" EDG and ESW pump, CS (or RHR) pump(s), and open the CS (or RHR) inject valves; Operators secure vessel injection after restoring level to normal; Operators initiate Shutdown Cooling mode, using 1 RHR pump and 2 RHRSW pumps, after vessel injection and depressurization below the SDC interlock pressure.

Event Category & Acceptance Criteria

This is a Special Event – to demonstrate the plant's ability to cope with a fire in the Control Room that requires the Operators to evacuate and bring the plant to a hot shutdown condition from outside the Control Room. Fuel Cladding Integrity –
Vessel Level > Top of Active Fuel (TAF) (Note)
Peak Reactor Pressure < 1375 psig (ASME Code Level C)
Peak Containment (Drywell and Wetwell) Pressure < 56 psig (design value)
Peak Suppression Pool Temperature < 281 °F (design value)
Peak Drywell Air Temperature < 281 °F (design value)
Suppression Pool Heat Capacity < Heat Capacity Temperature Limit Curve (EOPs)
Adequate NPSH for ECCS pumps
Time to reach Cold Shutdown < 72 hours from beginning of event.

2013-013

Methods

a) Calculation Tools & Computer Codes:
2012-020 Reactor Vessel Response: Primary Code – SAFER (see Table 15.0-2 for complete listing, code versions and NRC acceptance).
2012-020 Fuel Temperature Response: Primary Code – PRIME-LOCA (see Table 15.0-2 for complete listing, code versions and NRC acceptance).
2012-020 Primary Containment Response: Primary Code – SHEX (see Table 15.0-2 for complete listing, code versions and NRC acceptance).
Inventory Analysis: determined by manual calculation, using the kinetic energy per unit volume, $\Delta p = \rho \frac{v^2}{2g_c}$, and volumetric flow rate, $\dot{V} = vA$.

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
OPL-4 (Table 15.0-4) for Fuel Temperature Response.
OPL-4a (Table 15.0-6) are used for Containment Response.
The following changes to the above input sources are made for this event:
Initial reactor conditions (power, core flow, dome pressure, vessel level) are at nominal, rated (1912 MWt) conditions.
Decay Heat Curve - ANS 5.1-1979, nominal (i.e., no adders for uncertainty)
2012-020 GNF2 fuel design
MSIVs close in 5.5 seconds
Initial drywell and wetwell airspace pressures are 17.0 psia (2.3 psig).
Initial suppression pool level is at the Technical Specification minimum limit of 58,900 ft^3 (10.1 ft).
Initial Drywell temperature of 135 °F.
Initial Suppression Pool Temperature of 95 °F.
RHR Heat Exchanger Effectiveness of 135 Btu/sec-°F.

c) Key Assumptions:
Concurrent LOOP with the initiation of the Control Room fire.
Reactor Scram is instantaneous at the beginning of the event.
Feedwater injection ramps to zero in 5 seconds.
Either a single division of RHR in the LPCI mode (2 pumps) <u>OR</u> a single CS pump is assumed for vessel inventory makeup.
HPCI and RCIC are not available for this event.
The spuriously-opened vessel isolation valve is assumed to leak into the Drywell as part of the Containment Response analysis, but is treated as a vessel inventory loss for determining long-term makeup capability (i.e., water does not return to the suppression pool.)
SRV opening nominal trip setpoint for Group 1 is 1110 psig. Applying a 4% blowdown range, the closing nominal trip setpoint is 1066 psig. (The blowdown range is the ratio of the difference between the opening and closing pressure setpoints to the opening pressure setpoints.)
Low-low Set Logic/Valves are not available for this event.
2012-020 Operators begin vessel depressurization, using 3 S/RVs, at 20 minutes into the event.

<u>Results</u>

a) Comparison to Acceptance Criteri

2012-020 Peak Cladding Temperature = 1362 °F (CS) / 1275 °F (LPCI)
Peak Vessel Pressure = 1145 psig
Peak Drywell Pressure = 34 psia (Note 1)
Peak Drywell Temperature = 241 °F (Note 1)
Peak Suppression Pool Temperature = 159.8 °F (CS) / 163.6 °F (LPCI)
Peak Wetwell Pressure = 33.8 psia (CS) / 34.5 psia (LPCI)

Heat Capacity Temperature Limit Curve = satisfied (Suppression Pool temperature before reactor depressurization is approximately 120 ºF)
NPSH for ECCS pumps = Adequate (low Suppression Pool pressure and temperature)
Time to reach Cold Shutdown ≈ 8.1 hours from beginning of event. (Note 2)

Note 1: the leakage through the one-inch valve is assumed to be into the Drywell for this analysis.
Note 2: The leakage flow rate is calculated to be about 440 gpm. If the spuriously open valve(s) are not closed within 17 hours, the suppression pool is completely drained, assuming no additional water is available to replenish the pool. Thus, it could have an impact on the capability of the RSS to maintain the plant in Cold Shutdown for 72 hours. However, this long term concern is resolved by the operators beginning to check the position of the identified valves on all high-low pressure interface of one-inch or less liquid lines at 8 hours after event initiation, or when all necessary Cold Shutdown functions and equipment are secured.

b) Known Sensitivities:
As can be seen from the above results, the Suppression Pool responses are mildly sensitive to the injection source (CS or RHR) used. However, the PCT result is
2012-020 sensitive to the injection source, as a counter-current flow limitation in the upper plenum prevents CS from reflooding the core sooner than RHR.

c) Uncertainties in Results:
Use of conservative assumptions (e.g., simultaneous LOOP) in the evaluation are intended to bound the uncertainties in the final results.

<u>Conclusion</u>

a) Statement of Acceptability:
The results of this analysis support the conclusion that the plant is able to cope with the Control Room fire and bring the plant to a Hot Shutdown condition initially and a Cold Shutdown condition within 72 hours following the event.

b) Known Conservatisms/Margins:
2012-020 For fuel analysis with the SAFER code, the part length fuel rods in the GNF2 assemblies are treated as full-length rods, which conservatively overestimates the hot bundle power.

The LLS function was designed to extend the time between S/RV subsequent actuation to ensure sufficient time to allow the water leg inside the S/RV discharge line to return the normal level before each subsequent actuation. The time between S/RV subsequent actuation without LLS is approximately 18 seconds which far exceeds the minimum 3.7 seconds requirement established as the time limit for subsequent openings of S/RVs. Therefore, based on this evaluation, the automatic function of LLS is not required for a fire event that results in Control Room evacuation.

The Drywell temperature and pressure response are conservative, as the leak is through a pipe/line attached to the vessel that is not discharging to the containment (i.e., not a pipe break, but a draindown.)
The spurious operation of the RHR pool suction valve was considered and it was concluded that it would not affect the safe shutdown capability because only one RHR pump is sufficient to achieve and maintain Cold Shutdown within 72 hours. The spurious operation of the RHR minimum flow valve was considered and it was concluded that it is not detrimental to the RHR pump and would not affect the safe shutdown capability.

c) Limiting or Non-Limiting Event (Reload – transients; DBA - accidents):
2013-013 This is a unique Special Event. Within the category of Fire events, this event is most limiting in term of PCT and Drywell Temperature and Pressure. It also gives the same peak Reactor Pressure as the No Spurious Operation event.

15.3.4 THERMAL –HYDRAULIC STABILITY

This is a Special Event for demonstrating acceptable plant response to a plant trip of a recirculation pump(s), while operating on a high loadline, that results in an unstable condition in the reactor where the power and flow exhibit an oscillatory behavior, a so-called “thermal-hydraulic instability” event. It should be noted that no specific single failure of equipment or Operator error has been identified that causes the thermal-hydrualic instability. It is a very complex phenonmenon that is very difficult to model analytically. The ability to cope with such an event is required by regulation (10 CFR 50, App. A – General Design Criterion (GDC) 12):

> *Criterion 12—Suppression of reactor power oscillations.*
>
> The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Because there is actual operating experience that demonstrates that BWR can experience such unstable power oscillations, the conformance strategy is to “readily detect and suppress” such oscillations prior to them exceeding the specified acceptable fuel design limits (SAFDLs), i.e., fuel thermal limits such as Safety Limit MCPR (SLMCPR). The DAEC utilizes the so-called “Option I-D” detect and suppress strategy. This is accomplished by providing an administrative boundary for normal operations around the region on the power/flow map where instability could be expected to occur, the Exclusion Zone. The Exclusion Region boundary is defined where the core oscillation decay ratio (DR) is equal to 0.80. Decay ratios are calculated on the power/flow map to determine the intersection of the Exclusion Region boundary with the Natural Circulation Line (NCL) and the MELLLA boundary. Steady state operation is not permitted within the Exclusion Zone, i.e., where the DR is > 0.80. This is a preventive measure to preclude the likelihood of an instability event occurring. A “Buffer Zone” is established around the Exclusion Zone to provide a region of awareness, a defense-in-depth feature of the Option I-D solution. Operation is permitted within the Buffer Zone, provided that the on-line stability monitor (SOLOMON) is available.

This solution methodology takes credit for unique plant characteristics which make regional (out-of-phase) mode oscillations very unlikely, in conjunction with a demonstration that existing plant instrumentation is adequate to detect and suppress core-wide (in-phase) mode oscillations. The Option I-D stability solution is thus available to those plants, such as DAEC, which can demonstrate that core-wide mode power oscillations are the dominant type for their design. Smaller size cores (about 560 bundles or less) with a large eigenvalue separation (difference between the fundamental mode and the first harmonic mode eigenvalues of the neutron flux diffusion equation) and those with relatively tight inlet core orifices are unlikely to experience regional mode oscillations. Should core-wide oscillations occur, they will be automatically detected and suppressed by the flow-biased APRM neutron flux scram. The Option I-D analysis demonstrates SLMCPR protection is provided on the rated flow control line by the flow-biased APRM scram. This is the mitigation feature, should an instability event occur.

The stability Option I-D application analysis has two parts: (1) the Exclusion Region determination and (2) the detect and suppress calculation. The detect and suppress calculation has three sub-parts: (a) determination of the Delta CPR over Initial MCPR
2012-020 versus the Oscillation Magnitude (DIVOM) curve, (b) determination of Hot Channel Oscillation Magnitude (HCOM) and (c) determination of the stability based MCPR
2012-020 operating limit. The reload review table (15.3-4) is used to determine if the HCOM evaluation needs to be re-preformed due to plant changes. Cycle 21 is the first reload in which the plant- and cycle-specific DIVOM curve was determined (previously, a generic curve was used). The detect and suppress stability-based OLMCPR (Item 2c) is re-
2019-001 performed for every reload since it is sensitive to the core design. Starting from Cycle 27, the GEH Simplified Stability Solution (GS3) methodology may be used instead of TRACG DIVOM methodology. GS3 is a methodology improvement rather than a new Long Term Solution; it doesn’t affect the Backup Stability Protection or Exclusion Region determination.

15.3.4.1 – Themal-Hydraulic Stability – Exclusion/Buffer Zone Determination

Methods

a) Calculation Tools & Computer Codes:
Primary Code – ODYSY (see Table 15.0-2 for complete listing, code versions and
2012-020 NRC acceptance). Secondary codes - PANAC, CRNC, ODYN and ISCOR (see Table 15.0-2 for complete listing, code versions and NRC acceptance). These codes provide inputs to ODYSY.
The boundaries for both the Exclusion Region and Buffer Region are defined using either the Generic Shape Function (GSF):

$$P = P_B \left(\frac{P_A}{P_B} \right)^{\frac{1}{2}\left[\frac{W - W_B}{W_A - W_B} + \left(\frac{W - W_B}{W_A - W_B} \right)^2 \right]}$$

or the Modified Shape Function (MSF):

$$P = P_B \left(\frac{P_A}{P_B} \right)^{\left[\frac{W - W_B}{W_A - W_B} \right]}$$

where:
P = a core thermal power value on the region boundary (% of rated),
W = the core flow rate corresponding to power, P, on the region boundary (% of rated),
P_A = core thermal power at point A (% of rated),
P_B = core thermal power at point B (% of rated),
W_A = core flow rate at point A (% of rated), and
W_B = core flow rate at point B (% of rated).

Point A is the point on the MELLLA boundary on the Power/Flow map where the core Decay Ratio (DR) is equal to 0.80.

Point B is the point on the Natural Circulation Line (NCL) on the Power/Flow map where the core DR is equal to 0.80.

b) Inputs (Reference common list in 15.0, and/or include event-specific items):
Current cycle-specific core loading pattern in the Supplemental Reload Licensing Report (SRLR).
Rated thermal power heat balance (Figure 15.0-3).
2012-020 The MELLLA upper boundary line on the Power/Flow map (Figure 15.0-7) has been used instead of the highest licensed flow control line in the Exclusion Region determination.

c) Key Assumptions:
The Generic Shape Function (GSF) or Modified Shape Function (MSF) are used to define the Exclusion Region boundary between the endpoints (Points A and B, above).
The Buffer Region is defined as the more conservative of either:
1) Increasing W_A by 5% and decreasing P_B by 5%, or
2) Points on MELLLA and Natural Circulation Line boundaries with core decay ratios of 0.65.

d) Acceptance Criteria:

The Exclusion Region boundary is defined where the core decay ratio (DR) is equal to 0.80.
To provide assurance that core-wide mode is the predominate oscillation mode, the channel DR must be <0.56, at the least stable core conditions.

<u>Results</u>

a) Analysis Results:
The Exclusion Zone and Buffer Zone, plotted on a Power/Flow map, can be found in the current cycle-dependent COLR. A "typical" representation can be found in Figure 15.0-7.

An additional calculation at the intersection of the NCL and the MELLLA boundary, the least stable point on the Power/Flow map, is performed to demonstrate that regional mode reactor instability is not anticipated to occur throughout the entire Exclusion Region. Even though the calculated core decay ratio could demonstrate that the core can be made unstable, (i.e., DR > 1.0), the channel decay ratio is still low (< 0.56). This demonstrates that core-wide is the predominant mode of oscillation throughout the entire Exclusion Region.

b) Known Sensitivities:
High reactor power/loadline (especially at the high power/low core flow area of the Power/Flow map) tends to increase the core average void fraction and this tends to be destabilizing (i.e., increases the DR).

2012-020 The smaller fuel thermal time constant in 10×10 fuel tends to increase the DR. The Exclusion Region analysis shows that the large single-phase pressure drop induced by the relatively tight inlet orifices of DAEC creates a preference for core-wide mode oscillations, which dominate the regional-mode oscillations.

c) Uncertainties in Results:
Use of conservative acceptance limits (e.g., DR of 0.80) in the evaluation are intended to bound the uncertainties in the final results.

d) Known Conservatisms/Margins:
The ODYSY stability application procedure calculates a best-estimate core DR.

15.3.4.2 – Thermal-Hydraulic Stability – Safety Limit MCPR Protection

Methods

a) Calculation Tools & Computer Codes:
This calculation has three distinct parts: the first part is the determination of the
Delta CPR over Initial MCPR versus the Oscillation Magnitude (DIVOM) curve,
2012-020 to determine the relationship between MCPR and HCOM; the second is the
calculation of the Hot Bundle Oscillation Magnitude (HCOM), which determines
how high the bundle power grows during the oscillation before the APRM flow-
biased Scram occurs; and, the third part is to calculate the change in bundle
2012-020 critical power (ΔCPR) at natural circulation flow to determine the necessary
Operating Limit MCPR for stability to ensure that the SLMCPR is protected.
2019-001 Starting from Cycle 27, the GS3 methodology may be used to confirm safety limit
MCPR protection by using the APRM setpoints.

2012-020 Primary Code – OPRM, that calculates the HCOM (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

2012-020 Primary code - PANAC for the OLMCPR calculations, with Secondary Code: ISCOR, providing heat balance input to PANAC (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

Primary Code – TRACG, to establish the plant- and cycle-specific DIVOM curve (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

b). Inputs (Reference common list in 15.0, and/or include event-specific items)
Current cycle-specific core loading pattern in the Supplemental Reload Licensing Report (SRLR).

<u>Statistical Inputs</u>
Growth Rate: A review of actual instability events indicates that most BWR oscillations would be expected to have a growth rate only slightly above 1.00. For Option I-D application, the growth rate is randomly selected from the probability density function with a $\chi 2$ distribution.

Overshoot: The trip setpoint overshoot is a measure of how much an oscillation exceeds the trip setpoint. The overshoot is the fraction of the peak-to-peak difference between two consecutive oscillation cycles that are above the setpoint, when a trip occurs. Thus, $0.0 \leq \delta \leq 1.0$; and the value of δ can be considered to be essentially random. For Option I-D application, the overshoot is randomly selected from a uniform distribution.
Oscillation Period: The statistical methodology considers a range of oscillation periods. Studies of actual instability events indicate that the expected value for the period is approximately 1.8 to 2.0 seconds. However, it is desirable to consider an oscillation frequency range between 0.7 Hz and 0.3 Hz. This corresponds to a desired period range of 1.4 sec < T < 3.3 sec. For Option I-D application, the oscillation period is randomly selected from the probability density function with a $\chi 2$ distribution.
LPRM Failures: The statistical model provides options for considering an input LPRM failure probability distribution, a fixed failure percentage, or no LPRM failures in the calculation of hot bundle oscillation magnitude. For Option I-D application, a random number of LPRM failures are selected from the distribution, which is representative of plant data on LPRM failure rates. The specific LPRMs, which are defined to fail for a given trial, are then randomly selected from the total DAEC LPRM population.
Oscillation Contours: The statistical model randomly selects from the specified set of oscillation contours. DAEC application uses contours developed for core-wide mode oscillations for the DAEC 368-bundle plant.

<u>Deterministic Inputs</u>
LPRM Assignments: Option 1-D relies on the APRM flow-biased trip to terminate core-wide mode reactor instability. LPRMs are assigned to their respective APRM channels according to the plant configuration. All non-failed LPRM signals in an APRM are used to produce an averaged power signal for comparison to the trip setpoint. DAEC is designed with 80 LPRMs, in 6 APRM channels, though, for conservatism, only 4 were used in the calculations. One LPRM is permanently disabled. In the DAEC design, the LPRMs assigned to Channel A are also assigned to Channel B, and the LPRMs assigned to Channel C are also assigned to Channel D. The LPRMs in Channel E and F are not assigned to any other channels. Since there are channel pairs with identical LPRMs, DAEC normally operates with either Channels A & D bypassed, or with Channels B & C bypassed.

Trip Setpoint: The nominal APRM trip setpoint is input as a percentage of rated power. For DAEC, at natural circulation, the flow-biased APRM trip has a bounding value of 68% of rated reactor power.

Radial Peaking Factor: Since only the fundamental (core-wide) oscillation mode is used to calculate the relative LPRM signal averages, there is only one hot bundle in the core-wide mode oscillation. This bundle is also the "true" hot bundle with the highest radial peaking factor. Its normalized oscillation magnitude, Δh, is the same as any other location in the core. To account for reasonable variations in the radial peaking factor as the result of normal operation, a multiplier is used on the radial peaking factor to conservatively determine the DIVOM curve.

RPS Trip Logic: DAEC has a one-out-of-two, taken twice trip logic. Therefore, at least one channel from Division I, and at least one channel from Division II, must reach the APRM trip setpoint for the trip signal to be generated.

APRM Channel Failure: In addition to the failure of individual LPRMs, the failure of one APRM channel is considered. The model provides several options: no APRM channel failure, failure of a specified channel, failure of a randomly selected channel, and failure of the most responsive channel. For conservatism, the failure of the most responsive channel (i.e., the first channel to reach the trip setpoint) is used for Option 1-D analysis.

Delay Time: The delay time for control rod insertion to terminate oscillation growth is input to the model. The time at which the reactor trip setpoint is reached plus the delay time determines the time window in which the peak hot bundle oscillation magnitude can occur. The delay time is defined to be a plant-specific input consisting of the APRM response time, the RPS processing time, the control rod drive delay time before rod motion begins and the time for control rods to insert two (2) feet into the core, assuming control rods insert at the minimum scram speed allowed by the plant Technical Specifications. Even though control rod insertion two feet into the core will not shut the reactor down, it is judged to be adequate to prevent further growth of the hot bundle oscillation. The total delay time for DAEC EPU is 885 msec.

c). Key Assumptions:
A flow-biased APRM trip power level (nominal value) at natural circulation is used. Using the Nominal Trip Setpoint (NTSP) value has been standard practice
2012-020 for Option I-D applications.

Beginning with Cycle 21, the plant- and cycle-specific DIVOM curve was used in determining the plant's response to a thermal-hydraulic instability. The generic DIVOM curve methodology previously approved by the NRC used the TRACG02 code. Adoption of the cycle-specific DIVOM methodology uses the TRACG04 code. DAEC 10 CFR 50.59 Evaluation 07-002 showed that the results from the TRACG04 code are "essentially the same" as those from the TRACG02 code as referenced in the NRC-approved Licensing Topical Report (LTR) NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications, (August 1996)."

d). Acceptance Criteria
The OLMCPR for stability should not set the actual cycle-dependent operating limit in the COLR. The stability-based OLMCPR is compared to the non-stability based OLMCPR at a minimum of two conditions: (1) the rated power/rated flow operating limit, OLMCPR 100%P/100%F, is compared to the stability-based operating limit from a two recirculation pump trip, OLMCPR (2PT): (2) the off-rated operating limit on the rated flow control line at 45% core flow, OLMCPR 100%RL/45%F, is compared to a stability based operating limit for limiting steady-state operations at the same condition, OLMCPR $(SS)_{45}$. Additional points may be added as needed by the cycle-specific analysis. For all scenarios considered, the criteria are met if the stability based operating limits are the same or lower.

<u>Results</u>

a) Analysis Results:

For the key results from the Option I-D stability evaluations, see the current cycle's Supplemental Reload Licensing Report (SRLR) for actual values.

The statistical methodology consists of a 1000-trial Monte Carlo analysis. Based on non-parametric tolerance limits, the methodology rank orders the 1000 trials and selects the 39^{th} trial from the highest values of hot bundle oscillation magnitude (Δh) as the 95% probability/95% confidence level value, per standard statistical methods.

For the current evaluation of the OLMCPR, see the current cycle's Supplemental Reload Licensing Report (SRLR) for actual values.

b) Known Sensitivities:
High initial reactor power/loadline at the beginning of the oscillation tends to produce more limiting results, when compared to oscillations that begin at a low power/flow condition. Although an oscillation originating at the lower power/flow condition has more margin to grow to reach the flow-biased APRM Scram setpoint, it originates at a higher ICPR than one that originates at the MELLLA loadline; and thus, produces a lower ΔCPR/ICPR than the MELLLA case. In the DAEC design, the LPRMs assigned to Channel A are also assigned to Channel B, and the LPRMs assigned to Channel C are also assigned to Channel D. The LPRMs in Channel E and F are not assigned to any other channels. Since there are channel pairs with identical LPRMs, DAEC normally operates with either

Channels A & D bypassed, or with Channels B & C bypassed. For stability trip applications, there is no difference between the two operational configurations.

c) Uncertainties in Results:
2012-020 The statistical model calculates the HCOM (Δh), which is dependent on a combination of statistical inputs and deterministic plant-specific factors. The statistical model results in selection of a conservative value of the hot bundle oscillation magnitude, $\Delta h_{95/95}$, at the 95% probability with a 95% confidence.
2012-020 The OLMCPR is analyzed to 95%/95% confidence levels using PANAC.

d) Known Conservatisms/Margins:
The number of individual LPRMs assumed to be failed, and the assumption of the most-responsive APRM channel failure, give a conservative result.

15.3.5 ANALYSIS OF REACTOR INTERNALS PRESSURE DIFFERENTIALS (RIPD)

This is a Special Event for developing the plant response to various events to be used as input to the structural evaluations of the vessel internals to the requirements of ASME Boiler and Pressure Vessel Code (including any approved relief thereto.) The purpose of the Reactor Internal Pressure Differences (RIPD) analysis is to determine the differential pressures for the reactor internal components and fuel bundle lift margins during Normal, Upset, Emergency, and Faulted conditions. The acoustic and flow-induced loads on the core shroud, shroud support and jet pump as a result of a postulated recirculation line break, are also determined as part of the RIPD analysis.

To develop that the required plant loading conditions, the following scenarios are evaluated:

Normal: 100% power/105% core flow, steady-state conditions.

Upset: Anticipated Operational Transients (moderate frequency), which are expected to occur during the operational plant's lifetime.

Emergency: Infrequent events, which are postulated to occur once during the operational plant's lifetime. The limiting event within this category is an inadvertent actuation of the Automatic Depressurization System (ADS).

Faulted: Accidents or limiting faults, which are postulated as part of the plant's design basis. The limiting event is an instantaneous circumferential break of a main steam line (MSL). Both MSL breaks inside and outside the containment are evaluated. For both MSLBs, the reactor is analyzed at both the high power and the cavitation interlock (low power/high flow) conditions. For the determination of the Faulted condition pressure drop across the RPV internal components and the fuel lift margin, the limiting case is the Main Steam Line Break (MSLB) inside the containment, between the vessel and the Main Steam Isolation Valves (MSIVs). The pressure difference for the steam dryer is determined for the MSLB outside the containment. This is because failure of the steam dryer assembly could prevent closure of the MSIVs, the only consequence of concern. The design condition for the steam dryer pressure difference is the MSLB outside containment at Hot Standby.

The acoustic and flow-induced loads are determined for the postulated instantaneous recirculation line break (i.e., the Design Basis Accident – Loss-of-Coolant Accident (DBA-LOCA).) The DBA-LOCA results in the maximum loads on the components located near the breaks. The acoustic loads are imposed on the core shroud, shroud support and jet pump as a result of the propagation of the decompression wave created by the instantaneous break. The flow-induced loads on the shroud and jet pump are due to the asymmetric blowdown force in the vessel following a break of one of the recirculation lines, and the momentum changes as the downcomer fluid exits the reactor vessel through the recirculation line break.

The RIPDs are evaluated for the following components considering the GE10, GE12 and
2017-002 GE14 core configurations (See Figure 15.3-1 for definitions of the various ΔP locations):

- Core Plate and Guide Tube
- Shroud Support Ring and Lower Shroud
- Upper Shroud
- Shroud Head
- Channel Wall - Core Average Power Bundle
- Channel Wall - Maximum Power Bundle
- Channel Wall - Central Average Power Bundle (normal & upset conditions only)
- Top Guide
- Steam Dryer

The fuel bundle lift margins are evaluated for the following components considering the GE10, GE12 and GE14 core configurations:

- Average Channel (Core Average Power Bundle)
- Hot Channel (Maximum Power Bundle)

The flow-induced loads are evaluated for the following components:

- Core Shroud
- Jet Pump

The acoustic loads are evaluated for the following components:

- Core Shroud
- Shroud Support
  Jet Pump

2012-020 The RIPDs are also evaluated considering the GNF2 core configuration. The impact of GNF2 on Reactor Internal Pressure Differences (RIPDs) is determined by calculating a core pressure difference (DP) for the Normal, steady state operating condition. Because of higher core flow, Increased Core Flow (ICF) will result in higher DPs. Analyses of Normal operating conditions for a full core of GNF2 are performed with the steady-state thermal hydraulic model at 100% rated thermal power / 105% core flow (ICF).

15.3.5.1 – RIPD – Normal Limits

Methods

a) Calculation Tools & Computer Codes:
Primary Code – ISCOR (see Table 15.0-2 for complete listing, code versions and NRC acceptance). Note: for the steam dryer calculation, an additional term for elevation (static head) loss has been added.

b) Inputs (Reference common list in 15.0, and/or include event-specific items):

Parameter - Units	Normal Conditions	Increased Core Flow Conditions
Core Power, MWt % of rated	1912 100.0	1912 100.0
Steam Flow, Mlb/hr % of rated	8.352 100.0	8.356 100.05
Core Flow, Mlb/hr % of rated	49.0 100.0	51.5 105.0
Dome Pressure, psia	1040	1040
FW Temperature, °F	431.4	431.4
Core Exit Quality	0.171	0.162

Fuel Types: GE14 bundle weight including channel is 645 lbm.

c) Key Assumptions:
Based on a full core of GE14.

Results

a) Analysis Results:

Reactor Internals	Reactor Internal Pressure Drops (psid)	
	Normal 100P/100F GE14	Normal 100P/105F GE14
Core Plate and Guide Tube	24.96	27.1
Shroud Support Ring and Lower Shroud	32.57	34.9
Upper Shroud	7.61	7.9
Shroud Head	7.92	8.6
Shroud Head to Water Level, Irreversible	10.67	11.3
Shroud Head to Water Level, Elevation	0.85	0.7
Core Average Power Bundle	8.83	9.4
Central Average Power Bundle	10.14	10.7
Maximum Power Bundle	11.71	12.6
Top Guide	0.64	0.7
Steam Dryer	0.43	0.6

Fuel Type	Bundle Type	Fuel Bundle Lift Margin[(1)] (lbf)
GE14	Hot	339.2
	Average	358.2

(1) The fuel bundle lift margin is determined by the difference between the downward forces due to the bundle weight, weight of fluid in the bundle and bundle outlet flow, and the lift forces due to the bundle pressure drop, bundle bypass pressure drop and bundle inlet flow. If the bundle downward force balanced the uplift force, the fuel bundle lift margin would be zero.

2012-020

The effect of the GNF2 fuel introduction on RIPDs is determined by calculating core, core plate, and channel wall DPs for the Normal condition, i.e. steady state. These DPs are compared to equivalently derived values for GE14. If the comparison indicates that GE14 under similar conditions has higher DPs, then the GE14 fuel design bounds GNF2 and no further analysis is required. The following table illustrates the analysis conditions.

Parameter	**Unit**	**GE14 Normal**	**GNF2 Normal**
Core Power	MWt	1912	1912
Rated Core Power	%	100	100
Core Flow	Mlbm/hr	51.45	51.45
Rated Core Flow	%	105	105
Vessel Steam Flow	Mlbm/hr	8.356	8.356
Rated Steam Flow	%	100.05	100.05
Dome Pressure	psia	1040	1040
Rated FW Temperature	ºF	431.4	431.4

The RIPD analysis results for Normal condition are shown in the following table.

Reactor Internal Components	**GE14 Normal (psid)**	**GNF2 Normal (psid)**
Core Pressure Drop	31.57	30.90
Core Plate and Guide Tube	27.1	26.5
Upper Shroud	7.9	7.9
Shroud Head	8.6	8.6
Shroud Head to Water Level, Irreversible	11.3	11.3
Shroud Head to Water Level, Elevation	0.7	0.7
Shroud Support Ring and Lower Shroud	34.9	34.2
Core Average Power Bundle	9.4	8.7
Central Average Power Bundle	10.7	9.8
Maximum Power Bundle	12.6	11.7
Top Guide	0.7	0.7
Steam Dryer	0.6	0.6

As shown in the above table, the GNF2 RIPDs at Normal condition are bounded by or equivalent to GE14 RIPDs at the same operating conditions. Therefore, the GNF2 RIPDs are bounded by GE14 RIPDs for all operating conditions. Because the fuel type that has the highest DPs at Normal condition governs the DP response for all design analysis states, i.e. Upset, Emergency and Faulted conditions, the application of the bounding GE14 RIPDs to GNF2 is acceptable for these states.

The minimum fuel lift margin (FLM) is the force required to lift the bundle off its fuel support. Increased differential pressure loadings act vertically on the fuel

2012-020 assemblies. Such loadings, combined with the vertical component of seismic loads, may cause displacement of the fuel assembly. The FLM is determined by the difference between the downward forces due to the bundle weight, weight of fluid in the bundle and bundle outlet flow, and the lift forces due to the bundle pressure drop, bundle bypass pressure drop and bundle inlet flow. GNF2 has a lower core DP than GE14 and thus lower fuel assembly DP. Also, the GNF2 bundle (655 lbm) is slightly heavier than GE14 (645 lbm). Thus, the application of the bounding GE14 minimum FLM to GNF2 is acceptable at Normal, Upset, Emergency and Faulted conditions.

GNF2 has no effect on acoustic/flow-induced loads on jet pumps, core shroud and shroud support resulting from a postulated recirculation line break. The magnitudes of these loads are not dependent on the fuel type; they are dependent on the pressure vessel geometry outside the core shroud, initial annular pressures, and temperatures, which remain unchanged for GNF2.

In conclusion, the RIPD inputs to the reactor internal structural integrity evaluation do not increase for GNF2 fuel.

b) Known Sensitivities
This evaluation is most sensitive to changes in core exit steam quality and core pressure drop.
Top guide pressure drop is sensitive to the bypass flow (leakage flow).

c) Uncertainties in Results:
Use of conservative assumptions in the evaluation are intended to bound the uncertainties in the final results.

d) Known Conservatisms/Margins:
Use of a single fuel type is a bounding evaluation compared with a mixed core.

15.3.5.2 – RIPD – Upset Limits

Methods

a) Calculation Tools & Computer Codes:
Primary Code – None. Calculation is based upon generic BWR and plant-specific adders and multipliers, derived from transient analysis methods over a spectrum of transient events, that are applied to the normal condition values.

b) Inputs (Reference common list in 15.0, and/or include event-specific items):

Parameter, Units	Upset Conditions Extended Power Uprate	Upset Conditions Increased Core Flow
Core Power, MWt % of rated	1950.2 102.0	1950.2 102.0
Steam Flow, Mlb/hr % of rated	8.554 102.42	8.558 102.47
Core Flow, Mlb/hr % of rated	49.0 100.0	51.5 105.0
Dome Pressure, psia	1055	1055
FW Temperature, °F	433.8	433.8
Core Exit Quality	0.175	0.166
MSL Nozzle ID (in)	18.0	18.0
MSL Nozzle Safe End ID (in)	18.165	18.165
MSL Flow Limiter ID (in)	9.015	9.015
MSL Nozzle Elevation (in)	620.344	620.344
Initial Water Level - Level 4 (inches above Top-of-Active Fuel)	530.5	530.5
# of ADS Valves Capacity (lbm/hr) at Referenced Pressure of 1080 psig	4 829,000	4 829,000

Fuel Types: GE14 (w/debris filters)
GE14 bundle weight including channel is 645 lbm.

The pressure differences for the core plate, guide tube, shroud support ring and lower shroud at Upset conditions are calculated by applying a DAEC plant-specific adder of 2.1 to the values at Normal conditions. The top guide differential pressure at Upset conditions is calculated from the evaluation loss at Normal condition and a bounding generic value of 1.25 for the friction loss at Normal condition. The value of 1.25 is the square of the ratio of the maximum transient flow of 112% rated core flow over 100% rated core flow, which is based on the largest delta over initial flow in the limiting transient event of Generator Load Reject w/o Bypass for BWR/2 through BWR/6. The pressure differences for the upper shroud, shroud head, and shroud head to water level at Upset conditions, are calculated by applying a generic multiplier of 1.5 to the values at Normal conditions. The pressure differences for the channel walls are calculated by applying a generic adder of 2.9 to the values at Normal conditions. For the Upset condition, the Normal steam dryer differential pressure (irreversible + elevation) is multiplied by a plant-specific multiplier of 2.08, which is based upon the one stuck open relief valve (SORV) transient event. For this Fuel Bundle lift margin evaluation, the Upset value is determined by applying a generic adder of -31.3 lbf to the Normal result.

c) Key Assumptions:
The reactor is initially at 102% of rated power, per Reg. Guide 1.49.
Based on a full core of GE14.

Results

a) Analysis Results:

Reactor Internals	**Reactor Internal Pressure Drops (psid)**	
	Upset 102P/100F GE14	**Upset 102P/105F GE14**
Core Plate and Guide Tube	27.06	29.2
Shroud Support Ring and Lower Shroud	34.67	37.0
Upper Shroud	11.41	11.8
Shroud Head	11.88	12.9
Shroud Head to Water Level, Irreversible	16.0	<16.5
Shroud Head to Water Level, Elevation	1.28	1.1
Core Average Power Bundle	11.73	<10.7
Central Average Power Bundle	13.04	13.6
Maximum Power Bundle	14.61	<14.1
Top Guide	< 1.0	0.8
Steam Dryer	0.64	1.1

Fuel Type	Bundle Type	Fuel Bundle Lift Margin(1) (lbf)
GE14	Hot	307.9
	Average	326.9

(1) The fuel bundle lift margin is determined by the difference between the downward forces due to the bundle weight, weight of fluid in the bundle and bundle outlet flow, and the lift forces due to the bundle pressure drop, bundle bypass pressure drop and bundle inlet flow. If the bundle downward force balanced the uplift force, the fuel bundle lift margin would be zero.

b) Known Sensitivities:
This evaluation is most sensitive to changes in core exit steam quality and core pressure drop.
Top guide pressure drop is sensitive to the bypass flow (leakage flow).

c) Uncertainties in Results:
Use of conservative assumptions in the evaluation are intended to bound the uncertainties in the final results.

d) Known Conservatisms/Margins:
Use of a single fuel type is a bounding evaluation compared with a mixed core.

15.3.5.3 – RIPD – Emergency Limits

Methods

a) Calculation Tools & Computer Codes:
Primary Code – LAMB (see Table 15.0-2 for complete listing, code versions and NRC acceptance).

b) Inputs (Reference common list in 15.0, and/or include event-specific items):

Parameter, Units	Emergency Conditions Extended Power Uprate	Emergency Conditions Increased Core Flow
Core Power, MWt % of rated	1950.2 102.0	1950.2 102.0
Steam Flow, Mlb/hr % of rated	8.554 102.42	8.558 102.47

Core Flow, Mlb/hr % of rated	49.0 100.0	51.5 105.0
Dome Pressure, psia	1055	1055
FW Temperature, °F	433.8	433.8
Core Exit Quality	0.175	0.166
MSL Nozzle ID (in)	18.0	18.0
MSL Nozzle Safe End ID (in)	18.165	18.165
MSL Flow Limiter ID (in)	9.015	9.015
MSL Nozzle Elevation (in)	620.344	620.344
Initial Water Level - Level 4 (inches above Top-of-Active Fuel)	530.5	530.5
# of ADS Valves Capacity (lbm/hr) at Referenced Pressure of 1080 psig	4 829,000	4 829,000

Fuel Types: GE14 (w/debris filters)
GE14 bundle weight including channel is 645 lbm.

c) Key Assumptions:
The reactor is initially at 102% of rated power, per Reg. Guide 1.49.
Initial vessel water level is at the Low level alarm point (Level 4).
Based on a full core of GE14.

The inadvertent actuation of all the Automatic Depressurization (ADS) valves is modeled by assuming a steamline break of equivalent size to the total flow area of the ADS valves.
A turbine trip is assumed to occur when the water level reaches the bottom of the steamlines (above the expected Level 8 trip point).
Feedwater flow is ramped to zero in 4 seconds.

Results

a) Analysis Results:

Reactor Internals	Reactor Internal Pressure Drops (psid)	
	Emergency 102P/100F GE14	**Emergency 102/105F GE14**
Core Plate and Guide Tube	28.5	30.0
Shroud Support Ring and Lower Shroud	39	41.0
Upper Shroud	13.5	14.2
Shroud Head	13.9	14.4
Shroud Head to Water Level, Irreversible	15.9	16.5
Shroud Head to Water Level, Elevation	1.2	1.2
Core Average Power Bundle	11.1	10.7
Maximum Power Bundle	14.2	14.1
Top Guide	0.5	0.6
Steam Dryer(1)	< 3.9	<5.2

(1) The pressure difference for the steam dryer is not calculated for Emergency conditions since the LAMB model for the steam dryer is qualified only for the Main Steam Line Break (MSLB) outside the containment. The pressure difference for the steam dryer at Emergency conditions is bounded by Faulted conditions because of a slower depressurization rate at Emergency conditions.

Fuel Type	Bundle Type	Fuel Bundle Lift Margin(1) (lbf)
GE14	Hot	325.6
	Average	334.7

(1) The fuel bundle lift margin is determined by the difference between the downward forces due to the bundle weight, weight of fluid in the bundle and bundle outlet flow, and the lift forces due to the bundle pressure drop, bundle bypass pressure drop and bundle inlet flow. If the bundle downward force balanced the uplift force, the fuel bundle lift margin would be zero.

b) Known Sensitivities:
For a postulated inadvertent, simultaneous actuation of all the ADS valves, the reactor depressurizes suddenly, although this is less severe than for a Loss of Coolant Accident (LOCA). As a result, there is an upward load on the shroud head due to the mismatch between the steam generated in the core and the steam

leaving the reactor vessel through the ADS valves and to the steam turbine. The upper shroud and shroud support experience peak differential pressure shortly after the actuation of the ADS. The core plate does not experience the effect of the depressurization until later, when the depressurization of the vessel causes the liquid in the lower plenum to flash.

c) Uncertainties in Results:
Use of conservative assumptions in the evaluation are intended to bound the uncertainties in the final results.

d) Known Conservatisms/Margins:
Reactor Power is 102% of rated.
Initial water level is at the low level alarm point (Level 4) versus being at "normal" level.

Use of a single fuel type is a bounding evaluation compared with a mixed core. Turbine trip would occur sooner than modeled, at the High Level trip point (Level 8) versus at the bottom of the steamlines. This reduces the ΔP created between inside the vessel and outside steam exit path.

15.3.5.4 – RIPD – Faulted Limits

Methods

a) Calculation Tools & Computer Codes:
Primary Code – LAMB (see Table 15.0-2 for complete listing, code versions and NRC acceptance). The Moody homogeneous equilibrium mixture (HEM) break flow model is used in this evaluation for the Main Steamline (MSL) break - outside containment. Since the steam dryer is a non-safety component, the HEM break flow model is only assumed for the steam dryer evaluation.

b) Inputs (Reference common list in 15.0, and/or include event-specific items):

Parameter, Units	Faulted Conditions			
	Extended Power Uprate High Power	Extended Power Uprate Low-Power (Interlock)	Increased Core Flow High Power	Increased Core Flow Low Power (Interlock)
Core Power, MWt % of rated	1950.2 102.0	365.2 19.1	1950.2 102.0	367.1 19.2
Steam Flow, Mlb/hr % of rated	8.554 102.42	1.33 15.93	8.558 102.47	1.332 15.94
Core Flow, Mlb/hr % of rated	49.0 100.0	53.9 110.0	51.5 105.0	51.5 105.0
Dome Pressure, psia	1055	1040	1055	1040
FW Temperature, °F	433.8	278.6	433.8	278.7
Core Exit Quality	0.175	0.026	0.166	0.027
MSL Nozzle ID (in)	18.0	18.0	18.0	18.0
MSL Nozzle Safe End ID (in)	18.165	18.165	18.165	18.165
MSL Flow Limiter ID (in)	9.015	9.015	9.015	9.015
MSL Nozzle Elevation (in)	620.344	620.344	620.344	620.344
Initial Water Level - Level 4 (inches above Top-of-Active Fuel)	530.5	530.5	530.5	530.5
# of ADS Valves Capacity (lbm/hr) at Referenced Pressure of 1080 psig	4 829,000	4 829,000	4 829,000	4 829,000

Fuel Types: GE14 (w/debris filters)
GE14 bundle weight including channel is 645 lbm.

c) Key Assumptions:
For the High-power case, the reactor is initially at 102% of rated power, per Reg. Guide 1.49.
Initial vessel water level is at the Low level alarm point (Level 4).
Feedwater flow is ramped to zero in 4 seconds.
Main Steamline Isolation Valves (MSIVs) instantaneously close at 5.5 seconds (0.5 logic response time and 5.0 second stroke time.)
Based on a full core of GE14.

Two MSL breaks are evaluated, Inside and Outside Containment. Each has a set of unique assumptions, as follows:

Inside
A turbine trip occurs at 1 second. The turbine stop valves close in 0.1 second and the turbine bypass valves open in 0.1 second, beginning 1.0 second after the break.

Outside
The MSLB is modeled by assuming a steamline break of equivalent size to the total flow area of the 4 main steamline flow limiters.
All turbine steamflow is assumed to stop at time zero and is diverted out the break.

Results

a) Analysis Results:

Reactor Internals	**Reactor Internal Pressure Drops (psid)**			
	High-Power 102P/100F GE14	**Interlock 19.1P/110F GE14**	**High Power 102P/105F GE14**	**Interlock 19.2P/105F GE14**
Core Plate and Guide Tube	30	36	32.0	33.0
Shroud Support Ring and Lower Shroud	50	53	53.0	50.0
Upper Shroud	27	32	28.0	31.0
Shroud Head	27.5	32	28.5	31.0
Shroud Head to Water Level, Irreversible	29.5	33	30.0	32.0
Shroud Head to Water Level, Elevation	1.2	2.3	1.2	2.3
Core Average Power Bundle	12.4	9.8	12.3	9.3
Maximum Power Bundle	14.4	10.4	14.8	9.8
Top Guide	1.4	2.1	1.5	2.2
Steam Dryer(1)	3.9	5.0	4.9	5.2

(1) The pressure difference for the steam dryer at the Faulted condition is still bounded by the Hot Standby condition (7.5 psi), per APED-A61-080, Revision 0 (DAEC Reactor Internal Pressure Differences Data Book, 257HA737, Revision 4, May 1978.)

Fuel Type	**Bundle Type**	**Fuel Bundle Lift Margin[(1)] (lbf)**		
		Extended Power Uprate		**Increased Core Flow**
		High-Power Case	Low-Power (Interlock) Case	High-Power Case (2)
GE14	Hot	271.7	323.3	272.7
	Average	288.4	324.2	277.9

(1) The fuel bundle lift margin is determined by the difference between the downward forces due to the bundle weight, weight of fluid in the bundle and bundle outlet flow, and the lift forces due to the bundle pressure drop, bundle bypass pressure drop and bundle inlet flow. If the bundle downward force balanced the uplift force, the fuel bundle lift margin would be zero.

2). The High Power Case for Increased Core Flow is bounding over the Low-Power (Interlock) Case.

b) Known Sensitivities:
The high-power (rated) condition is limiting for certain components because the maximum loads occur at the maximum core flow and maximum void formation in the bundles. The low-power (interlock) condition is limiting for certain components because it results in a higher mismatch between the steam flow from the break and the steam generated in the core during a postulated steamline break. At the interlock point with lower thermal power, the core steam flow is much lower than the high power case resulting in a greater difference between the core generated steam flow and the steam exiting through the break.
The depressurization rate is proportional to the mass flow rate and the excess of enthalpy of the escaping fluid above saturated water enthalpy, h_f. Mass flow rate is inversely proportional to the enthalpy of the escaping fluid, h_e, and the depressurization rate is approximately proportional to $1-h_f/h_e$. Consequently, depressurization rate decreases as h_e decreases, that is, the depressurization rate is less for mixed flow than for steam flow. Therefore, the steam-line break is the design-basis accident for internal pressure differentials.

For the MSLB-Inside, the reactor depressurizes rapidly**.** As a result, there is an upward load on the shroud head due to the flow mismatch between the steam generated in the core and the steam leaving the reactor vessel through the break. The upper shroud and shroud support experience peak differential pressure shortly

after the break. The core plate does not experience the effect of the break until later, when the rapid depressurization of the vessel causes the liquid in the lower plenum to flash. Because of the higher depressurization rate, the RIPD results at Faulted conditions are much higher than at Emergency conditions.
For the MSLB-Inside, the RIPD results at the cavitation Interlock point bound the RIPDs results for the High-power point, except for channel wall pressure drops. At the Interlock point, the core flow is higher than the core flow at the High-power point. Higher core flow and less downcomer and inlet subcooling at the Interlock condition result in higher core pressure drops across the reactor internals. The initial steam flow at the High-power condition is higher than that at the Interlock condition; this would result in a lower flow mismatch (the steam flow generated in the core vs. the steam flow leaving the vessel or break flow) and thus lower pressure difference. However, the steam dome pressure at the High-power condition is higher than that at the Interlock condition, which would result in higher critical flow leaving the reactor vessel and thus a slightly higher flow mismatch. The mismatch increases as the initial power decreases (e.g., initial steam flow decreases) and the core flow increases (e.g., the steam flow leaving the vessel increases). Thus, the pressure differences for most components (except the fuel channel wall) at the Interlock condition are higher than those at the High-power condition.

The effect of varying the time of the turbine trip on the MSL-Inside break has little impact on the blowdown rate, which determines the ΔP.

c) Uncertainties in Results:
Use of conservative assumptions in the evaluation are intended to bound the uncertainties in the final results.

d) Known Conservatisms/Margins:
Reactor Power is 102% of rated (High-power case).
Initial water level is at the low level alarm point (Level 4) versus being at "normal" level.
Use of a single fuel type is a bounding evaluation compared with a mixed core.
Turbine trip would occur sooner than modeled, at the High Level trip point (Level 8), but 1 second is conservatively assumed to simplify the analysis (0.1 second is a typical value.)

15.3.5.5 – RIPD – Flow-induced Loads

Methods

a) Calculation Tools & Computer Codes:
Primary Code – TRACG (see Table 15.0-2 for complete listing, code versions and NRC acceptance). The methodology is based on the TRACG result for a representative BWR plant (reference plant). The TRACG results are then scaled for other BWR plants by applying the scaling factors that account for the plant geometry differences (such as the size of the core shroud, vessel and recirculation line) and thermal-hydraulic condition differences (such as the downcomer subcooling) between the reference plant and the plant of interest.

b) Inputs (Reference common list in 15.0, and/or include event-specific items):

Thermal-hydraulic Input

Parameter, Units	High-Power Condition	MELLLA Condition(1)	Natural Circulation Condition
Core Power, MWt % of rated	1950.2 102.0	1950.2 102.0	908.2 47.5
Steam Flow, Mlb/hr % of rated	8.548 102.34	8.547 102.33	3.559 42.61
Core Flow, Mlb/hr % of rated	49.0 100	48.51 99	14.21 29
Dome Pressure, psia	1043	1043	980
FW Temperature,°F	433.7	433.7	354.1
Downcomer Enthalpy, Btu/lbm	527.1	526.8	486.8
Downcomer Subcooling, Btu/lbm	23.7	24.0	54.5
Downcomer Fluid Density, lbm/ft^3	47.18	47.19	49.16

(1) Maximum Extended Load Line Limit Analysis

Plant Geometry Input

Parameter, Units	Reference Plant	DAEC
Core Shroud O.D., (inch)	207	145(1)
Core Shroud Height, (inch)	277	297
Recirculation Pipe Outlet Diameter, (inch)	28	22
RPV I.D., (inch)	251	183
Jet Pump Height , (inch)	254	233.5

(1) Based on the outside diameter of the core shroud (middle portion).

c) Key Assumptions:
The reactor is initially at 102% of rated power, per Reg. Guide 1.49.
Evaluation is independent of fuel type.

Results

a) Analysis Results:

<table>
<tr><th colspan="6">Maximum Integrated Flow-induced Loads for a Recirculation Line Break</th></tr>
<tr><th rowspan="2">Internal Component</th><th rowspan="2">Baseline Force (lbf)</th><th rowspan="2">Baseline Moment (in-lbf)</th><th colspan="3">Load Multiplier</th></tr>
<tr><th>102P/100F</th><th>102P/99F</th><th>47.5P/29F</th></tr>
<tr><td>Shroud</td><td>149,921</td><td>8,208,000</td><td rowspan="2">1.0</td><td rowspan="2">1.0051</td><td rowspan="2">1.4366</td></tr>
<tr><td>Jet Pump</td><td>12,408</td><td>652,000</td></tr>
</table>

b) Known Sensitivities:
The flow-induced loads are dependent on two primary factors: (1) the plant vessel and internal geometry and, (2) the fluid pressure and temperature conditions in the downcomer region. Downcomer subcooling increases with decreasing core flow (such as MELLLA). Conversely, higher core flow (such as Increased Core Flow – 105% of rated) result in lower subcooling in the downcomer region. Higher subcooling in the downcomer results in higher break flow and higher loads. The low power/low flow point along with the highest flow rod line represents the limiting condition for the flow-induced loads, e.g., natural recirculation power/flow point.

The flow-induced loads are the result of the momentum change of the downcomer fluid as it leaves the vessel through the recirculation line break. The flow-induced loads are significantly smaller than the acoustic loads but are of greater duration.

They last a few seconds. The loads affect primarily the shroud and jet pumps. The shroud support is not significantly impacted by the loads because it does not impede coolant flow to the break.

c) Uncertainties in Results:
Use of conservative assumptions in the evaluation are intended to bound the uncertainties in the final results.

d) Known Conservatisms/Margins:
Reactor Power is 102% of rated.

15.3.5.6 – RIPD – Acoustic Loads

<u>Methods</u>

a) Calculation Tools & Computer Codes:
Primary Code – TRACG (see Table 15.0-2 for complete listing, code versions and NRC acceptance). The methodology is based on the TRACG result for a representative BWR plant (reference plant). The TRACG results are then scaled for other BWR plants by applying the scaling factors that account for the plant geometry differences on the core shroud (the largest shroud and the widest shroud support), the limiting acoustic wave frequency (the smallest plant with the highest subcooling) and the maximum subcooling (up to 65 Btu/lbm subcooling) for all BWR plants. (Note: DAEC is not the smallest plant with the highest subcooling as DAEC does not have the alternate operating mode for feedwater temperature reduction, which would have a higher subcooling.) The generic BWR bounding acoustic loads on the jet pump considers the acoustic acceleration drag load, which is the predominant component in the acoustic loads on the jet pump. The method assumes that the discrete acoustic wave only causes acceleration along a wave front at any particular time, that a very small surface area of the jet pump is exposed to the acceleration fluid, and that the location around the jet pump of the acceleration fluid moves as the wave passes by.

b) Inputs (Reference common list in 15.0, and/or include event-specific items):

Plant Geometry Input

Core Shroud and Shroud Support		
Parameter, Units	**Bounding BWR Plant**	**DAEC**
Core Shroud O.D., (inch)	217	145$^{(1)}$
Core Shroud Height, (inch)	316	297
Recirculation Pipe Outlet Diameter, (inch)	28	22
RPV I.D., (inch)	251	183
Shroud Plate Width, (inch)	22	19
Downcomer Subcooling (Btu/lbm)	65	54.5

Jet Pump		
Parameter, Units	**Reference Plant**	**DAEC**
Core Shroud O.D., (inch)	207	145$^{(1)}$
Recirculation Pipe Outlet Diameter, (inch)	28	22
RPV I.D., (inch)	251	183
Jet Pump Height , (inch)	238.64	233.5
Jet Pump Base Diameter, (inch)	20.75	14.8
Annulus Gap (inch)	22	19
Downcomer Subcooling (Btu/lbm)	65	54.5

(1) Based on the outside diameter of the core shroud (middle portion).

c) Key Assumptions:
The reactor is initially at 102% of rated power, per Reg. Guide 1.49.
Evaluation is independent of fuel type.
The broken recirculation pipe is realistically modeled as an area of a 28-inch diameter pipe.

Results

a) Analysis Results:

Maximum Integrated Acoustic Loads for a Recirculation Line Break(1)			
Internal Component	**Force (lbf)**	**Moment (in-lbf)**	**Duration (sec)**
Shroud	6,887,000	896,900,000	0.0156
Shroud Support	2,202,000	323,600,000	0.037
Jet Pump	1,180,600	57,379,000	0.00025

(1) The loads are applicable for 102% of rated power (1950.2 MWt), with the operating domain, including MELLLA, Increased Core Flow (105% of rated), and natural recirculation point.

b) Known Sensitivities:
The acoustic loads are dependent on two primary factors: (1) the plant vessel and internal geometry and, (2) the fluid pressure and temperature conditions in the downcomer region. Downcomer subcooling increases with decreasing core flow (such as MELLLA). Conversely, higher core flow (such as Increased Core Flow – 105% of rated) result in lower subcooling in the downcomer region. Higher subcooling in the downcomer results in higher break flow and higher loads. The low power/low flow point along with the highest flow rod line represents the limiting condition for the acoustic loads, e.g., natural recirculation power/flow point.

The acoustic load is a time-dependent load due to a recirculation line break. The initial decompression wave associated with the instantaneous break impinges upon the shroud, shroud support, and jet pump. The acoustic load is large in magnitude and of short duration, lasting less than a millisecond for the jet pump and in a few milliseconds range for the shroud and shroud support.

c) Uncertainties in Results:
Use of conservative assumptions in the evaluation are intended to bound the uncertainties in the final results.

d) Known Conservatisms/Margins:
Reactor Power is 102% of rated

Table 15.3-1
Inputs to ATWS Analysis

Parameter	Value
Dome Pressure	1025 psig
Rated Core flow	49.0 Mlbm/hr
Minimum Core flow at Rated Power	48.5 Mlbm / 99% of rated
Rated Power	1912 MWt
Rated Steam Flow	8.35 Mlbm/hr
Feedwater temperature	431°F
Fuel Exposure	200 MWD/T (BOC) 14,800 MWD/T (EOC)
Initial Suppression Pool Liquid Volume	58900 ft^3
Initial Suppression Pool Temperature	90 °F
Nominal Closure Time of MSIV	4.0 sec
Relief Valve System Capacity No. of Valves	59.6% NBR Steam Flow at 1080 psig 6
Relief Valve Opening Analytic Setpoint Range	1154/1162/1164/1172/1187/1196 psig
Relief Valve Closing Setpoint	1107/1115/1117/1124/1138/1148 psig
Relief Valve Time Delay On Opening Signal	0.2 sec
Relief Valve Opening Duration	0.2 sec
Relief Valve Closure Time Delay	0.2 sec
Relief Valve Closure Duration	0.2 sec
Safety Valve System Capacity No. of Valves	15.4% NBR Steam Flow at 1240 psig 2
Safety Valve Opening Analytic Setpoint	1277 psig
Safety Valve Closing Setpoint	1226 psig
Relief Valve Opening Duration	0.2 sec
Relief Valve Closure Duration	0.2 sec
Vessel Pressure Pump Trip Sensor Time Constant	0.0 sec
Total SRV and SSV Capacity	73% NBR Steam Flow at 1080 psig

Table 15.3-1
Inputs to ATWS Analysis

Parameter	Value
Low-Low Set Setpoint Open	1025/1030 psig*
Low-Low Set Setpoint Close	915/920 psig*
Recirc Pump Trip Delay	0.175 sec
SLCS Injection Location	Lower Plenum Standpipe
Number of SLCS Pumps	2
SLCS Injection Rate per Pump	26.2 gpm
Nominal Boron-10 Enrichment	19.8 %
Sodium Pentaborate Concentration	11.8 wt %
Boron Injection Initiation Temperature (BIIT)	110 °F
SLCS Liquid Transport Time	60 sec
SLCS Liquid Solution Enthalpy	48.09 Btu/lbm
RCIC Flow Rate	400 gpm
Enthalpy of the RCIC Flow	68.04 Btu/lbm
HPCI Flow Rate	3000 gpm
Enthalpy of the HPCI Flow	68.04 Btu/lbm
ATWS High Pressure Setpoint (Analytical Limit)	1168.6 psig
Low Pressure Isolation Setpoint	850 psig
Number of RHR Loops	2
RHR Service Water Temperature	85 °F
RHR Heat Exchanger K-Factor per Loop in Containment Cooling Mode	142 Btu/sec-°F
RHR Heat Exchanger K-Factor per Loop during the Loss of Offsite Power Event	135 Btu/sec-°F

*Subsequent to the EPU analysis, the LLS NTSP values were revised as follows:

LLS SRV	Opening min / max (NTSP)	Closing min / max (NTSP)
PSV-4401	1030	910
PSV-4407	1035	915

There is no impact on any of the transient or accident analysis results, as those analyses use the Analytical Limits, which were not revised. See footnote to Table 15.3-1 for evaluation of impact on the ATWS analyses.

Table 15.3-2

SBO Safe Shutdown Systems and Safety Actions

SCRAM Equipment
1. RPS logic (fail safe action)
2. CRD Drives
3. HCU solenoids
4. HCU accumulators
5. N2 supply (recovery only)

Pressure Relief Equipment
1. ADS including SRV
2. LLS including SRV
3. N2 accumulators
4. N2 supply (recovery only)
5. 125vdc Div I or II (valve controls,sensors,logic,valve solenoids, low-low set logic)

Core Cooling Equipment
1. HPCI
2. 250vdc (valves, motors)
3. 125vdc Div II (valve controls, sensors, logic, valve solenoids)
4. Condensate inventory and condensate storage tank
5. Torus water (extended event only)
6. Auxiliary 480vac power (recovery only)
7. Emergency service water (recovery only)
8. RCIC
9. 125vdc Div I (valves, valve controls, sensors, logic, valve solenoids)
10. Low pressure systems (recovery only)

Primary Nuclear System Isolation Equipment
1. MSIV logic (fail safe action)
2. 125vdc Div I and II (valve controls, sensors, logic, valve solenoids)

Auxiliary AC Power Equipment (Recovery only)
1. Offsite & Onsite AC Generator System (battery charging, N2 supply, equipment cooling, area cooling, decay heat removal)
2. 125vdc Div I and II (breaker control)

125vdc and 250vdc Power Equipment
1. 125vdc Div I & II (scram, LLS, HPCI, RCIC, PCIS, Auxiliary AC power - valves, valve controls, sensors, logic, valve solenoids, breaker control, inverters)
2. 250vdc (valves, motors, inverters)

Table 15.3-2

SBO Safe Shutdown Systems and Safety Actions

Primary Containment Integrity Equipment

1. PCIS logic (fail safe action)
2. 125vdc Div I and II (valve controls, sensors, logic, valve solenoids)

Fuel Pool Cooling Equipment (Recovery only)

1. 480vac auxiliary power

Equipment Room Cooling Equipment (Recovery only)

1. 480vac auxiliary power

Table 15.3 – 3

2012-020 This Table has been deleted.

Table 15.3 - 4

Parameters for Hot Bundle Oscillation Magnitude Reload Review Evaluation

2012-020
2015-004

Parameter	Description	Acceptance Criterion (Range)	Base ValueC21	Value for Current Cycle	Data Source	Disposition (OK/Not OK)
#LPRMs	Number of installed LPRMs	No change from base value	80	80	FRED	Ok
APRM assignment	LPRM assignment to APRMs in 6 channels, etc.	No APRM design change	See EPU OPL-3	Same as C21	FRED	Ok
APRM trip @ NC	Flow-biased APRM trip power level (nominal value) at natural circulation	≤ base value	65.4% rated power	65.4% rated power	FRED	OK
Reactor Power @ NC	Average power level on the rated licensing procedure flow-control line at natural circulation	≥ base value	47.2% rated power	47.2% rated power	FRED	Ok
APRM trip @ 45% Rated Core Flow	Flow-biased APRM trip power level (nominal value) at 45% Rated Core Flow	≤ base value	88.5% rated power	88.5% rated power	FRED	OK
Reactor Power @ 45% Rated Core Flow	Average power level on the rated licensing procedure flow-control line at 45% Rated Core Flow	≥ base value	60.2% rated power	60.2% rated power	FRED	Ok
T_{delay}	Total delay time (APRM response time, RPS processing time, delay before start of control rod motion, plus time for 2 feet of control rod insertion)	≤ base value	885 msec	885 msec	FRED	Ok

2012-020