CHAPTER 15.0 - ACCIDENT ANALYSES

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DRAWINGS CITED IN THIS CHAPTER*

DRAWING*	SUBJECT
A-17	Service Building Roof Plan West Area
A-186	Auxiliary Building Mezzanine Floor Plan
M-15	General Arrangement - Section “C-C”
S-326	Reactor Containment Liner Sections & Details

* The listed drawings are included as "General References" only; i.e., refer to the drawings to obtain additional detail or to obtain background information. These drawings are not part of the UFSAR. They are controlled by the Controlled Documents Program.

CHAPTER 15.0 - ACCIDENT ANALYSES

15.0.0 Introduction

The following UFSAR sections evaluate the capability of the plant to control or accommodate postulated failures and events. Prior to the initial startup on both units, Chapter 15 was written to provide a description of the analyses performed that showed the plant was capable of withstanding all credible events. The remainder of Section 15.0 describes the approach and methodology that was used in analyzing the events described in sections 15.1 through 15.10. Sections 15.1 through 15.10 describe various events and their causes, initial conditions, sequence of events, probable consequences and plant performance for a number of nuclear system transients/accidents which were supposed to pose potential challenges to the Nuclear Steam Supply System. These analyses were performed for the initial cores on each unit. However, each analysis is not required to be performed on subsequent cycles as only a few of the events could cause a decrease in the margin to the Minimum Critical Power Ratio (MCPR) Safety Limit. For each core reload, a cycle-specific safety analysis is performed utilizing the methods described in GESTAR (Reference 1). The safety analysis reevaluates the limiting transient events to establish the required core thermal operating limits. Those events that are limiting, or near limiting, are identified as such in the following sections. The results of these safety analyses are documented in the cycle-specific Supplemental Reload Licensing Report (SRLR). Information from the SRLR is used in the development of the cycle-specific Core Operating Limits Report (COLR). A description of each of the limiting transients analyzed each cycle can be found in Section 15.A. "Typical" descriptions of the event can be found in the appropriate subsections of Sections 15.1 through 15.10.

The 1999 LaSalle County Station Power Uprate Project included re-evaluating a broad set of most limiting transient events at the power uprate conditions. Transient analyses to support MUR power uprate were performed for the first reload for MUR operation. The transient events which are re-analyzed with power uprate conditions form 3323 MWth to 3489 MWth core thermal power are documented in Section 15.B.

15.0.1 Approach to Safety Analysis

The safety analysis described in Sections 15.0.0 through 15.10 evaluates the ability of the plant with the initial core to operate within the regulatory guidelines without undue risk to the public health and safety. The analysis investigates the categories of events by type and expected frequency to delineate the limiting cases where the radiological consequences are significant. This approach ensures that a broad spectrum of initiating events is considered. It also enables the focusing of more detailed treatment of the radiologically important cases, while subordinating trivial and nondominant cases to lesser relative importance. A hypothetical ATWS event is also included at the request of the NRC. It has an extremely low probability of occurrence at LSCS.

In the treatment of specific safety cases initiated by typical plant events, the concept of expected frequency was mutually considered with the mechanisms of radiological release, to scope the safety risk associated with that particular event.

The safety analysis presents two categories of events: transients and accidents. Transients are subdivided into two subsets: moderate frequency events and infrequent incidents. For the purpose of simplifying a summary of results, however, all transient events are tabulated, independently of frequency, with regard to their Critical Power Ratio (CPR) operating limit. The accident results are also tabulated in the same summary (Table 15.0-2). The initiating events were assigned one of the following expected frequencies based upon practical Exelon operating experiences with six nuclear power stations:

a. Transients

Moderate frequency - Events which may occur during a calendar year to once per 20 years for a particular plant. Anticipated operational transients are in this frequency class.
Infrequent incidents - Events which are expected to occur once during the lifetime of the plant (including those that may occur once every 20 to 100 years). Unexpected or abnormal operational transients are included in this frequency class.

b. Accidents

Limiting fault frequency - refers to those incidents that are never expected to happen but for which safety analyses are arbitrarily made to represent upper bounds on the radiological consequences. The design-basis accident is included in this frequency class.

It should be noted, for example, that the frequency of an initiating event may be described by the term limiting fault frequency and not be characterized as the limiting fault per se. For the LaSalle County Station (LSCS), no limiting fault was found, hence this UFSAR does not attempt to describe a limiting fault, but treats it only as a concept.

In the treatment of particular events, the product of the expected frequency and the scale of the radiological consequence was used to categorize the safety significance of the event. For example, an initiating incident of moderate frequency that has no radiological consequence (because no radioactivity was released beyond the primary pressure boundary) was subordinated in safety importance to an infrequent incident which allowed radioactivity to be released beyond the primary pressure boundary.

The plant design takes into account the fact that the integrity of this pressure boundary, the primary containment and the secondary containment constitute significant safety barriers. Indirectly then, the conceptual probability for breaches to all of these barriers aids not only in the analytical treatment for the UFSAR appraisals but also in the delineation of physical processes important to knowledgeable safety design.

15.0.2 Categories of Safety Events

Transient and accident events are categorized by their initiating cause via the process variable whose change may have a deleterious effect on the nuclear reactor fuel. Each postulated initiating incident is assigned to one of the following categories:

a. Nuclear system pressure increase threatens to overstress the reactor coolant pressure boundary from internal pressure. A pressure increase collapses the voids in the moderator thereby increasing reactivity and power which threaten fuel cladding due to overheating.

b. Reactor vessel water (moderator) temperature reduction results in an increase in core reactivity as density increases. Positive reactivity increases have the effect described in item a.

c. Reactivity and power distribution anomalies may reduce the void content of the moderator thus resulting in increased reactivity and power levels. Such transient anomalies may affect the fuel cladding.

d. Reactor vessel coolant inventory decrease could threaten the fuel as the coolant becomes less able to remove the heat generated in the core.

e. Reactor core coolant flow decrease could result in overheating of the cladding as the coolant becomes unable to adequately remove the heat generated in the fuel.

f. Excess of coolant inventory could result in damage resulting from excessive moisture carryover to the main turbine.

g. Postulated radioactive release from a subsystem or component due to loss of integrity.

h. Postulated (anticipated) transients without scram which results from multisystem maloperation plus active component failures. This is a hypothetical situation with an extremely low probability of occurrence.

i. Postulated thermal-hydraulic instabilities in certain portions of the core and flow operating domain, which could threaten MCPR limits.

These nine categories include all of the effects on the nuclear system caused by abnormal operational transients which might lead to degradation of the reactor fuel barrier or reactor coolant pressure boundary. The variation of any one of these parameters may affect another. For purposes of analysis in the UFSAR, events are analyzed in groups, according to the initiating incident or event. For example, positive reactivity insertions resulting from sudden pressure increases are evaluated in the increase in reactor pressure classification.

The input parameters and initial conditions used for the initial core transient and accident analyses are listed in Table 15.0-1.

15.0.3 Judgment of Nonacceptable Safety Results

For all transients of moderate and infrequent frequencies, the following are considered to be unacceptable safety results:

a. Release of radioactive material to environs in excess of 10 CFR 20 limits.

b. Reactor operation induced fuel clad failures.

c. Nuclear system stresses in excess of those allowed for in the transient classification by applicable industry codes.

d. Containment stresses in excess of those allowed for in the transient classification by applicable industry codes.

For the design-basis accidents (limiting faults), the following are considered to be unacceptable safety results:

a. Release of radioactivity resulting in dose consequences in excess of 10 CFR 100 values.

b. Failure of fuel cladding sufficient to cause changes in core geometry such that core cooling would be inhibited.

c. Nuclear system stresses in excess of those allowed for the accident (faulted) classification by applicable industry codes.

d. Containment stresses in excess of those allowed for the accident classification by applicable industry codes when containment is required as a barrier.

e. Radiation exposure to plant operations personnel in the main control room in excess of 5 rem whole body, 30 rem inhalation dose and 75 rem skin dose.

15.0.4 Method of Analysis

Each transient or accident analyzed for the initial core is discussed and evaluated in terms of a sequence of events from the initiating condition to final stabilized state. The normal operation of unfailed equipment and controls is assumed. Credit is taken for plant systems and reactor protection systems in their normal functioning mode. The operation of unfailed engineered safety features (ESF) is also included. The effect of a single operator error or a single failure of active equipment is also included in certain analyses; however, this is done on an or basis for transient evaluations. In the evaluation of these postulated events, the plant damage allowances or limits are the same as those for normal operation. The evaluation presented herein interprets the accidents and transients in consonance with the historical frequency classification for the initiating events, i.e., the margin or limit was reported on the initiating event frequency rather than upon contingent or conditional frequencies of multiple events involved in certain sequences.

It is important to recognize that certain arbitrary accident scenarios require the application of single failures and operator errors. Others, such as ATWS, require multiple failures for the postulated end condition. In these accidents the event

frequency has a much lower probability. Credible frequency classes for these multifailure, multierror scenarios are not currently recognized, hence recourse was made to the former more simple classification for the convenience of cataloging these very low probability transients.

Most events postulated for consideration are already the results of single equipment failures or single operator errors hypothesized during normal or planned plant operations. Typical operational equipment failures or operator errors that can initiate events important to safety are as follows:

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

b. Undesired starting or stopping of a single component (a change of state is not assumed without an assignable cause).

c. Malfunction or maloperation of any single control device.

d. Single failure of any electrical component.

e. Single operator error event by one person.

In general, the analyzed events have numerical input parameters and initial (state) conditions as specified in Table 15.0-1. Note that these are analytical values. Analyses that assume data inputs different from these values are designated accordingly in their discussion and the specific parameters are defined therein for such cases.

Initial Power/Flow Operating Constraints

The analytical basis for most of the initial core transient safety analysis is the thermal power at rated core flow (100%) corresponding to approximately 105% original Nuclear Boiler Rated steam flow. For LaSalle, the initial cycle analyzed power was 3454 MWt. At the time, this operating point was the apex of a bounded operating power/flow map which, in response to any abnormal operational transients, will yield the minimum pressure and thermal margins of any operating point within the bounded map. The initial core power/flow condition is now bounded by the current operating power/flow map shown in Figure 15.0-1. Referring to Figure 15.0-1, the upper bound is the MELLLA flow control line (rod line D-B-A), the lower bound is the cavitation protection line E-F-G, the right bound is the ICF line D-E, and the left bound is either the two pump minimum flow line A-G or the natural circulation line H-G .

The power/flow map, D-B-A-H-G-F-E, represents the acceptable operational constraints based on abnormal operational transient evaluations.

Any other constraint which may truncate the bounded power/flow map must be observed, such as the licensed power limit and other restrictions based on pressure and thermal margin criteria. Constraints from the recirculation valve and pump cavitation regions have been built into the power/flow map. Reactor operation must be confined within the boundaries described above. For a derated operating power level, such as point C' which is applicable to satisfy a pressure margin criteria (for example), the upper constraint on power/flow is correspondingly reduced to an appropriate rod line (not depicted in Figure 15.0-1), which intersects the power/flow coordinate of the new operating basis. For this example, the operating bounds would eliminate the MELLLA boundary and be restricted based on the new rod line. Operation would not be allowed at any point along line C'-B', left of point C', at the derated power but at reduced flow. On the other hand, if derated operation is restricted to point C' by some MCPR limitation, operation at point B' (or right of it) would be allowed provided the MCPR safety limit is not violated. Consequently, the upper operating power/flow limit of the reactor is predicated on the operating constraint of the analysis (i.e., flow or MCPR) and the corresponding constant rod pattern line. Consequently, the upper operating power/flow limit of a reactor is predicated on the operating basis of the analysis and the corresponding constant, rod pattern line.

Certain localized events are evaluated at other than the above mentioned conditions. When applicable, such conditions are discussed specifically for that appropriate event.

The power and flow used in the reload analyses are given in the Technical Requirements Manual. The GE models used are given in Reference 1 and Reference 2.

Core and System Performance

Section 4.2 describes the various fuel failure mechanisms. An acceptable criterion was determined to be that 99.9% of the fuel rods in the core would not be expected to experience boiling transition (Reference 1). This criterion is met by demonstrating that transients and accidents do not result in a minimal critical power ratio (MCPR) less than 1.06 for the initial core, or the value given in Technical Specification for reload cores which is defined as the safety limit MCPR for LaSalle 1&2.

The steady-state reactor operating limit is determined as follows:

a. The change in the critical power ratio (ΔCPR) which would result in the safety limit CPR being reached, is calculated for each event. These ΔCPR values are shown in Table 15.0-2 for

the initial cores and in the Technical Requirements Manual for reload cores.

b. For GE the ΔCPR value for each event is then added to the Safety Limit CPR value to yield the event-based MCPR, FANP develops an operating limited based on a delta CPR and safety limit that bounds the limiting events.

c. The exception to b are those events whose ΔCPRs are calculated using ODYN. For events whose ΔCPR is determined by ODYN (GE only) (all rapid pressurization events), the event-based MCPR is determined in conjunction with NRC-additive correction factors, the ΔCPR, and the Safety Limit CPR. These correction factors are listed in Table 15.0-1 for the initial core and in the Reload Licensing Onsite Review Package for reload cores.

d. An additional exception to b are those events whose OLMCPR is calculated with the TRACG AOO process. The OLMCPR calculation is described in References 19 and 20.

These results are given graphically in Figure 15.0-2 for limiting initial core transients and accidents. The operating limit MCPR is the maximum locus of values from these event MCPRs calculated with the above method. The maximum calculated MCPR for the initial core is depicted by the solid line in Figure 15.0-2. Maintaining the CPR operating limit at or above the operating limit assures that 99.9% of the fuel rods in the core would not be expected to experience boiling transition. The MCPR operating limit for reload cores can be found in the LaSalle Administrative Technical Requirements.

In addition to MCPR, MAPLHGR and LHGR are also fuel design limits.

For situations in which fuel damage is sustained, the extent of damage is determined by correlating fuel energy content, cladding temperature, fuel rod internal pressure, and cladding mechanical characteristics. The bases for these correlations are the fuel rod failure tests discussed in Section 4.4, and in Section 6.3.

Barrier Performance

If there is no cladding damage, fission products are constrained to the fuel and only activation products are present in the reactor coolant. The performance of the Reactor Coolant Pressure Boundary (RCPB) and the containment system during transients and accidents is the primary evaluation of this section.

During transients that occur with no release of coolant to the containment only RCPB performance is considered. If release to the containment occurs as in the case of limiting faults, then challenges to the containment are evaluated as well.

Releases from containment are evaluated for those specific cases which involve source terms outside the primary containment boundary. The normal operation of the SGTS and the single point stack is covered in such cases, as applicable.

Reactor Coolant Pressure Boundary Damage

The only significant areas of interest for internal pressure damage are the high-pressure portions of the reactor coolant pressure boundary (the reactor vessel and the high-pressure pipelines attached to the reactor vessel). The overpressure below which no damage can occur is defined as the pressure increase over design pressure allowed by the applicable ASME Boiler and Pressure Vessel Code, Section III, Class 1, for the reactor vessel and the high-pressure nuclear system piping.

Because the ASME Boiler and Pressure Vessel Code, Section III, Class 1, permits pressure transients up to 10% over design pressure, the design pressure portion of the reactor coolant pressure boundary meets the design requirement if peak nuclear system pressure remains below 1375 psig (110% x 1250 psig).

Peak fuel enthalpy (discussed in Subsection 4.3) is used to evaluate whether reactor coolant pressure boundary damage occurs as a result of reactivity accidents. If peak fuel enthalpy remains below 280 cal/g, no reactor coolant pressure boundary (clad) damage results from nuclear excursion accidents and therefore no other barriers are challenged to retain concentrated fission products.

Radiological Consequences

In this section, the consequences of radioactivity release during both types of events: (1) operational transients, and (2) limiting faults or design-basis accidents are considered. For all events whose consequences are limiting, a detailed quantitative evaluation is presented. For non-limiting events, a qualitative evaluation is presented or results are referenced from a more limiting or enveloping case or event.

For limiting faults or design-basis accidents, two quantitative analyses are considered:

a. The first is based on conservative assumptions considered to be acceptable to the NRC for the purposes of the worst case bounding event which determines the adequacy of the plant design to meet 10 CFR Part 100 guidelines. This analysis is referred to as the "design-basis analysis".

b. The second is based on realistic assumptions considered to reflect expected radiological consequences, i.e., what could be

measured as an average value. This analysis is referred to as the "realistic analysis."

Results for both are shown to be within NRC guidelines.

Results

The results of analytical evaluations are provided for each initial core event. In addition the result summary is shown in Table 15.0-2. From that table, a comparison can be made of the limiting event for any particular safety event category. Radiological analyses are not performed for reloads as the UFSAR Section 15.6.5 analyses are bounding.

15.0.5 Meteorological Parameters (Other than Alternative Source Terms)

Atmospheric dilution factors (X/Q's sec/m^3) are summarized here for use in this chapter. The atmospheric dilution factors for the conservative analyses are based on the diffusion models presented in U.S. NRC Regulatory Guide 1.3, Revision 2 (June 1974) and U.S. NRC Regulatory Guide 1.145, Revision 1 (November 1982).

X/Q values determined per RG 1.145 methodology are determined for an elevated release and for a ground level release. The X/Q values for an elevated release are based on 1978 through 1987 historical meteorology at 375 feet above grade and represent a release via the plant exhaust stack. The X/Q values for a ground level release are based on 1982 through 1987 historical meteorology at 33 feet above grade and represent a release via the turbine building.

The atmospheric dilution factors at the 50th percentile for the realistic analyses have been derived from 2 years of onsite meteorological data.

Estimates of atmospheric dispersion for effluents released through the standby gas treatment system (SGTS) vent are based on values given in Table 2.3-48, 2.3-50, and 2.3-58. Estimates of atmospheric dispersion for effluents released through the plant common stack are based on values given in Tables 2.3-33, 2.3-35, and 2.3-37. These data reflect a "realistic" average value estimate of expected consequences.

The atmospheric dilution factors are:

a. Conservative NRC Regulatory Guide 1.3 Values

	Time Periods-hrs	X/Q sec/m^3
1. Exclusive Area Boundary (509 meters)	0-0.5	1.8×10^{-4}
	0.5-2.0	1.5×10^{-5}

2. Low Population Zone (6400 meters)	0-0.5	$1.8x10^{-5}$
	0.5-8	$4.4x10^{-6}$
	8-24	$1.7x10^{-6}$
	24-96	$5.0x10^{-7}$
	96-720	$1.7x10^{-7}$

b. Conservative NRC Regulatory Guide 1.145 Values

b.1 Elevated Release Out the Plant Exhaust Stack

		Time Periods-hrs	X/Q sec/m^3
1.	Exclusion Area Boundary (2800 meters)***	0-0.5	$8.4x10^{-5}$ (fumigation)
		0.5-2.0	$2.6x10^{-6}$
2.	Low Population Zone (6400 meters)	0-0.5	$8.9x10^{-6}$
		0.5-2.0	$1.6x10^{-6}$
		2.0-8.0	$9.2x10^{-7}$
		8.0-24	$5.5x10^{-7}$
		24-96	$2.5x10^{-7}$
		96-720	$8.2x10^{-8}$

b.2 Ground Level Release via the Turbine Building

		Time Periods-hrs	X/Q sec/m^3
1.	Exclusion Area Boundary (423 meters) ****	0-2	$5.1x10^{-4}$
2.	Low Population Zone (6400 meters)	0-8	$1.0x10^{-5}$
		8-24	$6.7x10^{-6}$
		24-96	$2.6x10^{-6}$

96-720	6.5×10^{-7}

c. Realistic Values – Standby Gas Treatment System Vent

	Time Periods-hrs	X/Q sec/m^3
1. Exclusion Area Boundary (509 meters)	0-2*	4.0×10^{-7}**
2. Low Population Zone (6400 meters)		
	0-8	1.47×10^{-7}
	8-24	3.29×10^{-7}
	24-96	1.37×10^{-8}
	96-720	1.14×10^{-8}

d. Realistic Values - Plant Common Stack

	Time Periods hrs	X/Q (sec/m^3)
1. Exclusion Area Boundary (509 meters)	0-2*	1.71×10^{-7}**
2. Low Population Zone (6400 meters)	0-8	5.75×10^{-8}
	8-24	1.33×10^{-8}
	24-96	5.62×10^{-9}
	96-720	4.96×10^{-9}

* A predicated fumigation condition at the onset of an accident is not considered realistic for observed meteorology at LSCS.

** The "realistic boundary" for maximum dose is not the EAB; the releases are elevated, and therefore not monotonic with distance. The "realistic boundaries" for the SGTS vent and the plant common stack are 4500 and 6400 meters, respectively. The EAB 50th percentile X/Q's for the SGTS vent and the plant common stack are 5.03×10^{-13} and 6.13×10^{-25} sec/m^3, respectively.

*** The distance of 2800 meters is in the SW downwind direction. This distance is greater than the Exclusion Area Boundary distance in the SW downwind direction. For elevated releases, the maximum sector X/Q value can occur at a distance

greater than the EAB boundary. In accordance with Regulatory Guide 1.145, the maximum sector X/Q value is used.

**** The distance of 423 meters is the shortest distance between the turbine building and the Exclusion Area Boundary (EAB) within a 45° sector centered on the WNW downwind direction. This is the methodology for determining sector distances to the EAB stipulated in Regulatory Guide 1.145.

Comparison between the realistic values with those obtained from Regulatory Guide 1.3 and Regulatory Guide 1.145 shows that the realistic values are consistently lower.

15.0.5a Meteorological Parameters (Alternative Source Terms)

Alternative Source Terms (AST) Atmospheric dilution factors (X/Q's sec/m^3) for the EAB and LPZ were calculated with the model PAVAN which implements the guidance provided in Regulatory Guide 1.145.

X/Q values were calculated for an elevated release via the stack and ground-level release via the turbine building utilizing the 1999-2003 meteorological tower data. This tower data consists of wind speed and direction measurements at 33 ft, 200 ft and 375 ft and delta temperature measurements at 375 – 33 ft and 200 – 33 ft. The meteorological data and the X/Q values are contained in Chapter 2.3.4a.

15.0.6 Nuclear Safety Operational Analysis (NSOA) Relationship

The main objectives of the operational analyses are to identify all essential protection sequences and to identify the detailed hardware conditions essential to satisfying the nuclear safety operational criteria. The main objective of the analyses of Chapter 15.0 is to provide detailed analyses of the "worst cases."

15.0.7 MSIV Closure Change from Reactor Water Level 2 to Water Level 1

The sequence of events for the cases analyzed in Chapter 15 indicate the MSIV isolation occurs at reactor water Level 2. The MSIV Isolation was changed from Level 2 to Level 1 and as stated Reference 2 in this design change has been included in the current analyzed licensing accident events described in Chapters 15 and 6.3.3.

15.0.8 Deleted

15.0.9 Impact of Increased Initial Suppression Pool Temperature

The initial conditions for the cases evaluated in Chapter 15 indicate an initial suppression pool temperature of 100°F. The maximum suppression pool temperature limit (for normal operation) was changed to 105°F as stated in Section 6.2.1.8. This temperature limit change was verified to have an insignificant impact on the accident events described in Chapter 15.

15.0.10 Reduction in the Total Number of SRVs

An evaluation and analysis has been performed based on the removal of five (5) SRVs, for a total of 13 installed SRVs (References 12 and 13). See Table 5.2-9 for a summary of the remaining valves. None of the ADS or Low-Low Setpoint Valves are affected, thus postulated accidents involving ADS valves are not impacted.

Reference 14 documents an evaluation of the effect on MCPR for the reduction in number of SRVs. This effect has been determined to be negligible. Based on this analysis it can be concluded that there is no significant effect on the consequences associated with any of the accidents or transients associated with the SRVs. The cycle specific safety analysis process (see Section 15.A), including the effect on ASME overpressure, is based on 13 installed SRVs and is documented in the SRLR. The effect on ATWS was evaluated separately and is discussed in Section 15.8.

15.0.11 References

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, (Unit 1: Rev 22, Unit 2: Rev 20).

2. Letter Dated March 6, 1987, from C.M. Allen (CECO NLA) to H.R. Denton (NRC) concerning MSIV level setpoint change from Level 2 to Level 1.

3. GE document NEDC-31455, "Extended Operating Domain and Equipment Out of Service for LaSalle County Station Units 1 and 2," dated March 1990, with addenda.

4. Deleted

5. Deleted

6. GE Letter LS-2209, lt. R. Peffer (GE) to G. R. Crane (ComEd), Subject: LaSalle County Station Unit 1 & 2 Technical Specifications on MSIV Closure Scram, dated March 15, 1982.

7. Deleted

8. Deleted

9. Deleted

10. Deleted

11. Deleted

12. "Safety Review for LaSalle County Station Units 1 and 2 Safety Relief Valves Reduction and Setpoint Tolerance Relaxation Analyses", Rev. 2, GE Report, GE-NE-B13-01760, by H. X. Hoang, dated February 1996. (On Site Review No. 96-020)

13. Safety Evaluation Report (SER) by NRC, dated 06-03-99, for Amendment Nos. 133 and 118 for LaSalle County Station Units 1 & 2, respectively.

14. Letter No. NFS:BSA:96-084, dated 8-07-96, from R. W. Tsai to P. Antonopoulos, "MCPR and LOCA Impact for Safety Relief Valve Setpoint Tolerance Relaxation and Removal.

15. Design Analysis L-003560, Revision 0, "T200 Series – Operating Power/Flow Map," July 2010.

16. Power Uprate Project Task 900, "Transient Analysis," GE-NE-A1300384-08 Revision 1, September 1999.

17. Deleted

18. Deleted

19. "Migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients," NEDE-32906P Supplement 3-A, Revision 1, April 2010.

20. "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," NEDE-32906P-A, Revision 3, September 2006.

TABLE 15.0-1
(SHEET 1 OF 3)

INPUT PARAMETERS AND INITIAL CONDITIONS FOR
ANALYSIS OF INITIAL CORE TRANSIENTS AND ACCIDENT

1. Thermal power level, MWt Analysis value	3454 (104.8% NBR)
2. Steam flow, lb per hr	14.81×10^6 (104% NBR)
3. Core flow, lb per hr	108.36×10^6
4. Feedwater flow rate, lb per sec	4115
5. Feedwater temperature, °F	420
6. Vessel dome pressure, psig	1020
7. Vessel core pressure, psig	1031
8. Turbine bypass capacity, %NBR	25
9. Core coolant inlet enthalpy, Btu per lb	529
10. Turbine inlet pressure, psig	962
11. Fuel lattice	8 x 8
12. Core average gap conductance, Btu/sec-ft^2 - °F	0.1662
13. Core leakage flow, %	12
14. Required MCPR operating limit	See Figure 15.0-2
15. MCPR Safety Limit	1.06
16. Doppler coefficient (-)¢/°F	
Nominal EOC-1	0.221
Analysis data	0.221

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17. Void coefficient (-)¢/% rated voids	
Nominal EOC-1	7.479
Analysis data for power Increase events	12.63
Analysis data for power Decrease events	7.01
18. Core average rated void fraction,%	41.03
19. Scram reactivity, Analysis data	FSAR Figure 15.0-2
20. Control rod drive speed, position versus time	FSAR Figure 15.0-2
21. Jet pump ratio, M	2.28
22. Safety/Relief valve capacity, % NBR	
at 1165 psig	111.5
Manufacturer	Crosby
Quantity Installed	18
23. Relief function delay, seconds	0.1
24. Relief function response, seconds	0.1
25. Analytical setpoints for safety/relief valves	
Safety function, psig	1150, 1175, 1185, 1195, 1205
Relief function, psig	1076, 1086, 1096, 1106, 1116
26. Number of valve groupings simulated	
Safety function, No.	5
Relief function, No.	5
27. High flux trip, % NBR	
Analysis setpoint (120 x 1.038), % NBR	124.6

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28. High-pressure scram setpoint, psig	1071
29. Vessel level, inches above steam dryer skirt bottom (instrument zero 527.5)	
Level 8 - (L8) Analytical Value	57.1*
Level 3 - (L3) Actual Setpoint	12.5
Level 2 - (L2) Actual Setpoint	-50
30. APRM thermal trip	
Setpoint, % NBR	121.54
31. RPT delay, seconds	0.190
32. RPT inertia time constant, sec**	6

33. ΔCPR/ICPR Adjustment Factor to be applied to LaSalle ODYN deterministic results to establish Option 95/95 pressure transient CPR operating limits.

TIME	LR/TT w/o BP	FW CONTROL FAILURE
BOC	-0.004	+0.029
MOC	-0.021	+0.016

* NRC reference level is 60.0 inches. GE sensitivity analyses for the 2.9 inch difference showed that a ΔCPR of 0.003 would exist; however, this small increment is dropped in rounding off the MCPR's in the ODYN solution. Actual setpoints are not used at L3 or L2 in the transient/ accident analyses.

$$**t = \frac{2\pi J_0 n}{g T_0}$$

where t = inertia time constant; J_0 = pump motor inertia,
n = rated pump speed, g = gravitational constant;
T_0 = pump electrical torque

TABLE 15.0-2
(SHEET 1 OF 4)

SUMMARY OF EVENTS RESULTS FOR INITIAL CORES

PARA-GRAPH+	FIGURE	DESCRIPTION	MAXIMUM NEUTRON FLUX % NBR	MAXIMUM DOME PRESSURE psig	MAXIMUM VESSEL PRESSURE psig	MAXIMUM STEAM LINE PRESSURE psig	MAXIMUM CORE AVERAGE SURFACE HEAT FLUX % OF INITIAL	ΔCPR	FREQUENCY CATEGORY ††	DURATION OF BLOWDOWN: NO. OF VALVES 1st BLOW-DOWN	DURATION OF BLOWDOWN: DURATION OF BLOW-DOWN SEC †††
15.1		DECREASE IN CORE COOLANT TEMPERATURE									
15.1.1	15.1.1-1	Loss of Feedwater Heater, AFC	111.4	1020	1058	994	106.1	0.06	a	0	0
15.1.3	15.1.3-1	Pressure Regulator Fail Open 115% Flow	103.9	1068	1083	1066	100.0	0.05*	a	0	0
15.2		INCREASE IN REACTOR PRESSURE									
15.2.2A	15.2.2-1	Generator Load Rejection, Bypass-On, RPT-On	214	1140	1166	1131	106.4	0.07*	a	18	
15.2.2A	15.2.2-2	Generator Load Rejection, Bypass-Off, RPT-On	350	1166	1192	1163	113.6	0.15*	b	19	

TABLE 15.0-2
(SHEET 2 OF 4)

PARA-GRAPH+	FIGURE	DESCRIPTION	MAXIMUM NEUTRON FLUX % NBR	MAXIMUM DOME PRESSURE psig	MAXIMUM VESSEL PRESSURE psig	MAXIMUM STEAM LINE PRESSURE psig	MAXIMUM CORE AVERAGE SURFACE HEAT FLUX % OF INITIAL	ΔCPR	FREQUENCY CATEGORY ††	DURATION OF BLOWDOWN: NO. OF VALVES 1st BLOW-DOWN	DURATION OF BLOWDOWN: DURATION OF BLOW-DOWN SEC †††
15.2.3	15.2.3-1	Turbine Trip, Bypass-On, RPT-On	165	1138	1164	1123	103.1	0.08	a	18	5.4
15.2.4	15.2.4-1	Main Steam Line Isolation, Position Scram	269	1163	1199	1152	108.6	<0.04**	a	18	6.4
15.2.5	15.2.5-1	Loss of Condenser Vacuum at 2 inches per sec	151	1134	1159	1120	104	<0.08**	a	14	6.0

TABLE 15.0-2
(SHEET 3 OF 4)

PARA-GRAPH+	FIGURE	DESCRIPTION	MAXIMUM NEUTRON FLUX % NBR	MAXIMUM DOME PRESSURE psig	MAXIMUM VESSEL PRESSURE psig	MAXIMUM STEAM LINE PRESSURE psig	MAXIMUM CORE AVERAGE SURFACE HEAT FLUX % OF INITIAL	ΔCPR	FREQUENCY CATEGORY ††	DURATION OF BLOWDOWN: NO. OF VALVES 1st BLOW-DOWN	DURATION OF BLOWDOWN: DURATION OF BLOW-DOWN SEC †††
15.2.6	15.2.6-1	Loss of Auxiliary Power Transformer	103.9	1092	1103	1092	100.0	~0.0	a	2	5.6
15.2.6	15.2.6-2	Loss of All Grid Connections	150.4	1135	1161	1121	101.8	<0.08**	a	18	6.4
15.2.7	15.2.7-1	Loss of All Feedwater Flow	103.9	1094	1105	1094	100.0	~0.0	a	2	5.5
15.3		DECREASE IN REACTOR COOLANT SYSTEM FLOWRATE									
15.3.1	15.3.1-1	Trip of One Recirculation Pump Motor	104.0	1020	1058	994	100.0	~0.0	a	0	0
15.3.1	15.3.1-2	Trip of Both Recirculation Pump Motors	103.9	1094	1107	1092	100.0	~0.0	a	2	5.3
15.3.2	15.3.2-1	Fast Closure of One Main Recirc Valve - 30% /sec	103.9	1095	1108	1093	100.0	~0.0	a	2	5.4
15.3.2	15.3.2-2	Fast Closure of Two Main Recirc Valves - 11% /sec	103.9	1095	1108	1099	100.0	~0.0	a	6	5.3
15.3.3	15.3.3-1	Seizure of One Recirculation Pump	103.9	1107	1119	1101	100.2	~0.0	c	6	5.6
15.4		REACTIVITY AND POWER DISTRIBUTION ANOMALIES									
15.4.4	15.4.4-1	Startup of Idle Recriculation Loop	110.2 100.2	982 982	995 985	971	79.2	<0.18***	a	0	0

TABLE 15.0-2
(SHEET 4 OF 4)

PARA-GRAPH+	FIGURE	DESCRIPTION	MAXIMUM NEUTRON FLUX % NBR	MAXIMUM DOME PRESSURE psig	MAXIMUM VESSEL PRESSURE psig	MAXIMUM STEAM LINE PRESSURE psig	MAXIMUM CORE AVERAGE SURFACE HEAT FLUX % OF INITIAL	ΔCPR	FREQUENCY CATEGORY ††	DURATION OF BLOWDOWN: NO. OF VALVES 1st BLOW-DOWN	DURATION OF BLOWDOWN: DURATION OF BLOW-DOWN SEC †††
15.4.5	15.4.5-1	Fast Opening of One Main Recirc Valve - 30% /sec	281.8	980	1000	971	76.2	<0.18***	a	0	0
15.4.5	15.4.5-2	Fast Opening of Both Main Recirc Valves - 11% /sec	193.8	973	980	961	72.0	<0.18***	a	0	0
15.5		INCREASE IN REACTOR COOLANT INVENTORY									
15.5.1	15.5.1-1	Inadvertent HPCS Pump Start	103.9	1020	1058	994	100.0	~0.0	a	0	0

†† = Incidents of moderate freq; b = infrequent incidents; c = limiting faults.
††† = Estimated value.
+ = Paragraphs denoted with suffix A indicate Reanalyses with ODYN Code (see Reference 2 of Subsection 15.1.2.6).
* = ODYN results without adders.
** = Events obviously bounded by more severe transients, hence unique MCPR were not calculated, but a limiting calculation was used because there is no threat to the MCPR safety limit when the event is initiated from less than 100% power.
*** = Events initiated from low power and therefore resulting MCPR is well above the MCPR safety limit.

15.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

The events listed under the increase in heat removal by the secondary system include loss of feedwater heater, feedwater controller failure maximum demand, pressure regulator failure in the open position, and inadvertent RHR shutdown cooling operation. The limiting events in terms of minimum critical power ratio were evaluated for the 1999 LaSalle County Station Power Uprate Project and documented in Appendix 15.B and Reference 12.

Events described in this section that result in decreased feedwater temperature may also result in a core thermal hydraulic instability transient. Refer to Section 15.10 for an overview of this event.

15.1.1 Loss of Feedwater Heater

The loss of feedwater heaters event is analyzed each cycle. This reload analysis is a steady – state analysis that assumes a 100°F temperature drop. A cycle specific methodology in Reference 2 is used for GE reload analyses.

15.1.1.1 Identification of Causes and Frequency Classification

A feedwater heater can be lost in at least two ways:

a. steam extraction line to heater is closed, and

b. feedwater is bypassed around heater.

The first case produces a gradual cooling of the feedwater. In the second case, the feedwater bypasses the heater and no heating of that feedwater occurs. In either case the reactor vessel receives cooler feedwater. The maximum number of feedwater heaters which can be tripped or bypassed by a single event represents the most severe transient for loss of feedwater heater analysis considerations. This event incurs a loss of up to 100° F off the feedwater heating capability of the plant

and causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient. The event can occur with the reactor in either the automatic or manual control mode. In automatic control mode, some compensation of core power is realized by modulation of core flow, so the event is less severe than for the manual control mode. Only the manual control mode is analyzed for each reload because it is more severe.

A feedwater heater loss is considered low enough to warrant being an infrequent incident. However, because of the lack of historical data, this incident is analyzed as a disturbance of moderate frequency.

15.1.1.2 Sequence of Events and Systems Operation

In the automatic flux/flow control mode, the reactor settles out at a lower recirculation flow with no change in steam output. An average power range monitor (APRM) neutron flux or thermal power alarm will alert the operator that he must insert control rods to get back down to the rated flow control line, or that he must reduce flow if in the manual mode. The operator must determine from

existing tables the maximum allowable turbine-generator (TG) output with feedwater heaters out of service. If reactor scram occurs, as it does in manual flow control mode, the operator must monitor the reactor water level and the pressure controls and the TG auxiliaries during coastdown.

The thermal power monitor (TPM) is the primary protection system trip in mitigating the consequences of this incident. Operation of Engineered Safeguards is not needed for either of these transients.

In establishing the sequence of events and simulating the plant performance, it was assumed that normal functioning occurred in the plant instrumentation and controls, plant protection system, and the reactor protection system. The sequence of events for the loss of feedwater heater while in automatic flow control mode (Figure 15.1-1) is as follows:

TIME (sec)	EVENT
0	Initiate a 100° F temperature reduction in the feedwater system.
5 (est)	Initial effect of unheated feedwater starts to raise core power level but AFC (Recirculation System) system automatically reduces core flow to maintain initial steam flow.
154.1	APRM initiates reactor scram on high thermal power.
173.3	L2 trip initiates RCIC (not simulated). L2 trip initiates HPCS (not simulated). L2 trip initiates MSIV closure (see subsection 15.0.7) (not simulated).
200.0	Reactor variables settle into new steady-state.

A typical sequence of events for the loss of feedwater heater while on manual flow control (Figure 15.1-2) is as follows. This sequence of events is representative of the sequence of events for the 145°F temperature reduction analyzed for past reloads.

TIME (sec)	EVENT
0	Initiate a 150°F temperature reduction in the feedwater system.
4.50 (est)	Initial effect of unheated feedwater starts to raise core power level and steam flow.

59.53	APRM initiates reactor scram on high thermal power.
65.70	L8 trip initiates a turbine trip (not simulated). L8 trip initiates a feedwater trip (not simulated).
65.70	Recirculation pump trip (RPT) actuated by turbine stop valve position switch (not simulated).
80.00(est)	L2 trip initiates MSIV closure (not simulated). L2 trip initiates RCIC and HPCS (not simulated).
82.00(est)	Bypass valves closed (not simulated).

These two transients generally lead to an increase in reactor power level. Because the thermal power monitor (TPM) is designed to be single failure proof, single failures are not expected to result in a more severe transient than analyzed here.

15.1.1.3 Core and System Performance

Mathematical Model

The detailed, nonlinear dynamic model described in Reference 1 was used to simulate this event for the initial core automatic flow control analysis.

For reload analyses, the event is treated as a quasi-steady-state transient due to its slow progression, and is analyzed with a steady-state, three dimensional BWR core simulator. The core simulator is used to evaluate the differences, including thermal margins, between the initial conditions of the event and the final equilibrium state point after the loss of feedwater heating.

Input Parameters and Initial Conditions

Cycle specific reload analyses are performed at initial conditions described in Section 15.A. Only the manual mode of flow control is considered for reload analysis.

Results

In the automatic flux/flow control mode, the recirculation flow control system responds to the power increase by reducing core flow so that steam flow from the reactor vessel to the turbine remains essentially constant. The reactor thermal power increases above the initial value and settles below the flow-referenced APRM thermal power scram setting and core flow is reduced.

The increased core inlet subcooling aids thermal margins, and smaller power increase makes this event less severe than the manual flow control case given below. Nuclear system pressure does not change and consequently the reactor coolant pressure boundary is not threatened. If scram occurs, the results become very similar to the manual flow control case. The initial core transient is illustrated in Figure 15.1-1.

In manual mode, no compensation is provided by core flow and thus the power increase is greater than in the automatic mode. A scram on high APRM thermal power occurs for this transient, but is not typically simulated in cycle-specific reload analysis. Vessel steam flow increases and the initial system pressure increase is slightly larger. The increased core inlet subcooling aids core thermal margins. System pressure does not reach the safety/relief valve setpoints.

A typical transient response of the key plant variables for this mode of operation are shown in Figure 15.1-2.

This transient is less severe from lower initial power levels for two main reasons: (l) lower initial power levels will have initial values greater than the limiting initial value assumed, and (2) the magnitude of the power rise decreases with lower initial power conditions. Therefore, transients from lower power levels will be less severe.

Important factors (such as reactivity coefficient, scram characteristics, magnitude of the feedwater temperature change) are assumed to be at the worst configuration so that any deviations seen in the actual plant operation reduce the severity of the event. Operation of the RCIC or HPCS systems and MSIV closure (see Subsection 15.0.7) on low water level (L2) are not included in the simulation since they occur in the latter part of the transient and therefore have no significant effect on the results.

15.1.1.4 Barrier Performance

As noted above and shown in Figures 15.1-1 and 15.1-2, the consequences of this event do not result in temperature or pressure values in excess of the criteria for which the clad, pressure vessel, or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.1.1.5 Radiological Consequences

Because this event does not result in any fuel failures nor any release of primary coolant to either secondary containment or to the environment, there are no radiological consequences associated with this event.

15.1.2A Feedwater Controller Failure Maximum Demand

15.1.2A.1 Identification of Causes and Frequency Classification

A postulated event which results in excess coolant inventory is one in which feedwater flow is increased without allowing changes in other reactor parameters. The postulated event is a feedwater controller failure to maximum flow demand. The feedwater controller is forced to its upper limit at the beginning of the event.

This event is considered to be an incident of moderate frequency except that when turbine bypass failure is added, the frequency is properly classified as an infrequent event.

This event is analyzed for every reload since this transient may be the most limiting for determination of core thermal limits (see Section 15.A for a discussion of core thermal limits). For GE analyses, methodology outlined in References 2, 3, 15, and 16 may be used for this event. GE utilizes the ODYN plant transient code or the TRACG transient code. Below is a description of the Feedwater Controller Failure Maximum Demand event. This description outlines the general trends for this event. The results of the reload analysis can be found in the Reload Licensing Onsite Review Package.

15.1.2A.2 Sequence of Events and System Operation

A typical initial cycle sequence of events for this transient at the normal operating condition (Figure 15.1-3) is as follows:

TIME (second)	EVENT (ODYN)
0	Initiate simulated failure of 135% of upper limit of feedwater flow.
10.93	L8 vessel level setpoint trips main turbine and feedwater pumps.
10.94	Reactor scram trip actuated from main turbine stop valve position switches.
10.94	Recirculation pump downshift (EOC-RPT) actuated by stop valve position switches.
11.04	Main turbine stop valves closed and main turbine bypass valves start to open.
12.17, 12.32 12.50, 12.66, and 12.90	Relief valves actuated sequentially by groups: 1, 2, 3, 4, 5.
16.0 (Est)	Relief valves close sequentially by groups: 5, 4, 3, 2, and 1.

The normal functioning of plant instrumentation and controls, plant protection systems, and the reactor protection system is included in the transient simulation. Important operational actions for safety considerations are the high-level tripping of the main turbine, the trip of the feedwater turbine(s), turbine stop valve scram-trip initiation, Recirculation pump downshift, and eventually HPCS actuation to maintain long-term water level control following trip of the feedwater pumps. Because multiple level sensors detect when the water level reaches the L8 setpoint, a single failure will not prevent nor initiate a turbine trip signal. The turbine trip signal transmission, however, is not built to single failure criterion, thus failure here would have the effect of delaying the pressurization "signature" in time. Excessive moisture entering the turbine will cause vibration sufficient to trip the turbine. The scram trip signals from the turbine are designed redundantly so that a single failure does not initiate nor impede a reactor scram initiation.

The operator actions expected during the course of these events, assuming no immediate restart of the reactor following the high-level (L8) trip, are as follows: Observe that the L8 trip of the feedwater pump terminates this event; switch the feedwater controller from auto to manual control to regain a correct output signal,

assure that an orderly reactor shutdown occurs via observation and that adequate core cooling capability is activated to protect the fuel cladding integrity.

Recovery of the feedwater controller is a power generation objective to be pursued independently of the safety objective of a safe shutdown and assured core cooling.

15.1.2A.3 Core and System Performance

Mathematical Model

For reload analyses, the model described in References 2, 9, 15, and 16 may be used.

Input Parameters and Initial Conditions

Cycle specific reload analyses are performed at nominal or rated conditions (see Section 15.A).

Results

A typical simulated feedwater controller failure at maximum demand is shown in Figure 15.1-3 for the case with turbine bypass. The high water level trip (L8) provides turbine trip and feedwater trip. Simultaneously, turbine stop valve closure causes scram and recirculation pump trip (RPT). This limits the neutron flux peak and fuel thermal transients so that no fuel damage occurs.

The reactor coolant pressure boundary pressure limit is not endangered. Blowdown continues to where the turbine bypass valves subsequently close to reestablish pressure control in the vessel during shutdown. Level gradually drops to the low-level isolation reference point, where HPCS or RCIC are initiated for long-term level control or if ADS is employed, the low pressure systems LPCS/LPCI are initiated for water inventory control.

The results of the reload analysis are given in the Technical Requirements Manual.

Effect of Single Failures and/or Operator Errors

The first sensed parameter to initiate corrective action to the rising water level transient is the vessel high water level trip (L8). Multiple level sensors are used to detect when the water level reaches the L8 setpoint. At this point in the logic a single failure will not initiate or prevent a turbine trip signal. The turbine trip signal transmission, however, is not built to single failure criteria. The result of a failure at this point would have the effect of delaying the pressurization "signature" in time. The high moisture levels entering the turbine will be detected by high levels in the moisture separators preceding the turbine; thus resulting in a turbine trip. If that sensing fails, then the excessive moisture entering the turbine will cause vibration to the point where mechanical vibration will trip the turbine.

Scram trip signals from the turbine are designated such that a single failure will neither initiate nor impede the initiation of a reactor scram (trip).

The effect of a failure of the bypass valves to open when the turbine stop valves have closed has been analyzed. With this additional failure present, a slightly higher pressurization rate would occur and a slightly higher peak pressure would result. A typical response for this simulation is shown in Figure 15.1-4. Again, however, the reactor vessel pressure limit would not be exceeded.

All systems utilized for protection in this event were assumed to have the poorest allowable response (e.g., relief setpoints, scram stroke time, and work characteristics). Plant behavior is, therefore, expected to lead to a less severe transient.

In the case of scram speeds, analyses are performed with Technical Specification Scram speeds, but may be performed with quicker speeds and the resulting thermal limits utilized, provided the actual control rod scram speeds remain faster than those assumed.

15.1.2A.4 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel clad, pressure vessels, or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.1.2A.5 Radiological Consequences

While the consequence of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under defined meteorological and controlled release conditions. If purging of the containment is chosen, the release will have to be in accordance with established technical specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.1.3 Pressure Regulator Failure-Open

The Pressure Regulator Failure - Open event is not analyzed for reload cores. The analysis and results presented are for the initial core.

15.1.3.1 Identification of Causes and Frequency Classification

If one of the three processors in the pressure controller failed low, an alarm will be generated and the pressure controller would maintain control of the turbine valves with no change in pressure. If either one or two of the three pressure transmitters providing input to the pressure controller failed low, the turbine control valves would adjust to the pressure sensed by the functioning transmitter and a small controller change in pressure could occur. The Digital Electro-Hydraulic Control (DEHC) Pressure and Turbine control system will continue to regulate pressure and will maintain control of the bypass valves and the turbine valves.

If a second processor or the third transmitter failed low, the turbine control valves will close resulting in an increase in reactor pressure.

A single controller failure has a very minor effect on operation due to the small change in valve position as the new median setpoint is determined by the remaining two controllers. There is no other effect.

The total steam flow rate resulting from a pressure regulator malfunction is limited by a maximum flow limiter imposed at the turbine controls.

This limiter is set to at least 115% (120%, Reference 5) of NB rated flow.

A pressure regulator failure (open) is estimated to occur with a moderate frequency.

15.1.3.2 Sequence of Events and Systems Operation

The sequence of events for this transient resulting from pressure regulator failure in the open position (Figure 15.1-5) is as follows:

TIME (sec)	EVENT
0	Simulate the maximum limit on steam flow to maintain turbine.
0.6(est)	Turbine control valves wide open. Turbine bypass valves open to about 9% NBR steam flow.
10.03	Vessel water level L8 trips the main turbine off and also trips the feedwater turbine. Turbine trip initiates bypass operation to full flow.
10.04	Main turbine stop valves reach 90% open position and actuates reactor scram and recirculation pump trip (RPT).
10.13	Turbine stop valves closed. Turbine bypass valves opening to full flow.
10.17	Recirculation pump motor circuit breakers open thus causing decreases in core flow to natural circulation level.
46.6	Vessel water level (L2) trip initiates MSIV's (see Subsection 15.0.7). Vessel water level (L2) trip initiates RCIC and HPCS systems (not simulated).

In order to accurately simulate this sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection system, and the reactor protection system except as follows. Initiation of the HPCS and RCIC functions will occur when the vessel water level reaches the L2 setpoint. Normal startup and actuation of these systems can take up to 30 seconds before effects are realized. The HPCS and RCIC functions, therefore, follow later in time after the primary concerns for fuel thermal margin and overpressure effects have passed. In the real world, actuation of these systems at L2 is expected to result in

less severe effects than those already experienced by the equipment during this transient.

This transient leads to a loss of pressure control such that the increased steam flow demand causes a depressurization, a high vessel water level turbine trip, and a subsequent turbine trip scram. Instrumentation for pressure sensing of the turbine inlet pressure is designed to be single failure proof for initiation of MSIV closure. Reactor scram sensing, originating from limit switches on the turbine stop valves or on the MSIV's themselves, is designed to be single failure proof. It is, therefore, concluded that the basic phenomenon of pressure decay is adequately terminated.

The TMR controller alarms on loss of any one signal and remains stable at the median of the two remaining inputs.

Expected Response If Sequence Of Events Results In MSIV Closure

If the reactor scrams as a result of the isolation caused by the low pressure at the turbine inlet (840 psig) in the run mode, the following is the sequence of operator actions expected during the course of the event. Once isolation occurs the pressure will increase to a point where the relief valves open. The operator should assume that an orderly reactor shutdown occurs and that adequate core cooling capability is activated to protect the fuel cladding integrity. No unique safety actions are required beyond verification that required automatic actions occur at their proper setpoints for example: that the reactor pressure relief valves open at their setpoint and that RCIC and HPCS initiates on low-water level.

In accordance with Reference 11, the analytical limit at which a reactor scram is initiated upon MSIV closure is 85%. This is conservative because the nominal setpoint is 92% open and the Technical Specification limit is 88% open. The original FSAR analysis presented in this section shows that a reactor scram is initiated from Stop Valve closure due to a reactor water level 8 turbine trip. The change in analytical limit of the MSIV position scram does not invalidate the previous discussion regarding the evaluation of the sequence of operator events should an MSIV closure terminate the event. Additionally, since the analysis presented in this section is terminated by a turbine trip, the change in the MSIV position scram analytical limit does not invalidate the analysis of this event.

15.1.3.3 Core and System Performance

Mathematical Model

The nonlinear dynamic model described in Reference 1 is used to simulate this event.

Input Parameters and Initial Conditions

The pressure regulator failure-open is no longer a postulated failure scenario because of the use of TMR controllers and sensors; no MKVI EHC pressure regulation systems failure scenarios are postulated that could result in all valves open.

A 5-second isolation valve closure instead of a 3-second closure is assumed when the turbine pressure decreases below 840 psig, and a reactor scram is initiated when the isolation valves reach the 85% open position; this is conservative for LaSalle where the minimum allowable setpoint is 88% open. The recirculation flow is run back approximately 10 seconds. The simulation is well within the specification limits of the valves and this tends to aggravate the results of the analysis. All other input conditions are those listed in Subsection 15.0-1.

Results

Figure 15.1-5 shows graphically the resulting transient for the cases where pressure regulator failure occurs with subsequent isolation and reactor scram.

Depressurization results in formation of voids in the reactor coolant and causes a decrease in reactor power almost immediately. In this simulation the depressurization rate is large enough such that water level swells to the sensed level trip setpoint (L8), initiating main turbine and feedwater turbine trips. Position switches on the turbine stop valves initiate reactor scram and recirculation pump trip (RPT) and shut down the reactor. After the turbine trip, the complete turbine control TMR failure, now signals the turbine bypass valves to open to fully bypass 25% NBR steam flow. After the pressurization resulting from the turbine stop valves closure, pressure again drops and continues to drop until turbine inlet pressure is below the low turbine pressure isolation setpoint, then main steamline isolation finally terminates the depressurization. The turbine trip and MSIV isolation limit the duration and severity of the depressurization so that no significant thermal stresses are imposed on the reactor coolant pressure boundary. A short duration neutron flux increase from about 55% to a maximum of 70% of NBR results after the turbine trip. No significant reductions in fuel thermal margins occur.

A consideration of uncertainties indicates that if the maximum flow limiter were set higher or lower than normal, there would result a faster or slower loss in nuclear steam pressure. The rate of depressurization then may be limited by the bypass capacity, but it is unlikely. For example, the turbine control valves will open to the valves-wide-open state admitting slightly more than the initial turbine steam flow and with the limiter in this analysis set to fail at 115% (130%, Reference 5) it is expected that something less than 25% NBR steam flow will be bypassed. Bypass capacity is, therefore, not a limiting factor on this plant. If the rate of

depressurization does change, it will still be terminated by the low turbine inlet pressure trip setpoint. The reactor scram will still result from main turbine stop valve position switches set at > 90% open position.

Depressurization rate has a proportional effect upon the voiding action in the core and the water flashing to steam in the vessel bulkwater regions. If the rate is low enough, the water level will swell quicker to the high water level trip setpoint, thus the isolation will occur earlier. The transient, after isolation valve closure and reactor scram, would not exceed the analyzed case, and no significant difference in vessel stresses would occur.

15.1.3.4 Barrier Performance

The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which fuel clad, pressure vessel, or containment are designed; therefore, these barriers maintain their integrity and function as designed.

Peak pressure in the bottom of the vessel reaches 1083 psig which is below the ASME code limit of 1375 psig for the reactor coolant pressure boundary.

Vessel dome pressure reaches 1068 psig, just slightly below the setpoint of the first pressure relief group.

Minimum vessel dome pressure of 831 psig occurs at about 50 seconds.

15.1.3.5 Radiological Consequences

Because this event does not result in any fuel failure nor any release of primary coolant to either the secondary containment or to the environment, there are no radiological consequences associated with this event.

15.1.4 Inadvertent RHR Shutdown Cooling Operation

The Inadvertent RHR Shutdown Cooling Operation event is not analyzed for reload cores. The analysis and results presented are for the initial core.

15.1.4.1 Identification of Causes and Frequency Classification

At design power conditions no conceivable malfunction in the shutdown cooling system could cause temperature reduction.

If the reactor were critical or near critical, a very slow increase in reactor power could result. A shutdown cooling malfunction leading to a moderator temperature decrease could result from misoperation of the cooling water controls for the RHR heat exchangers. The resulting temperature decrease would cause a slow increase of positive reactivity in the core. If the operator did not act to control the power level, a high neutron flux reactor scram would terminate the effects without violating fuel thermal limits and without any measurable increase in nuclear system pressure.

Although no single failure could cause this event, it is conservatively categorized as an event of moderate frequency.

15.1.4.2 Sequence of Events and Systems Operation

Approximate time (min)	Event
0	Reactor not shutdown with vessel head on or off when RHR shutdown cooling is erroneously activated.
0-10	Slow rise in reactor power.
>10	Operator may take action to limit power rise, flux scram eventually occurs.

A shutdown cooling malfunction causing a moderator temperature decrease can be considered for all operating states. However, this malfunction is not considered while at power operation because the nuclear system pressure is too high to permit operation of the shutdown cooling (RHR).

In startup or cooldown operation, where the reactor is at or near critical, the slow power increase resulting from the cooler moderator temperature would be controlled by the operator in the same manner normally used to control power in the source of intermediate power ranges.

No single failures can cause this event to be more severe. If the operator takes action, the slow power rise will be controlled in the normal manner. If no operator action is taken, scram will terminate the power increase before thermal limits are reached.

No unique safety actions are required to avoid unacceptable safety results for transients as a result of a reactor coolant temperature decrease induced by misoperation of the shutdown cooling heat exchangers.

15.1.4.3 Core and System Performance

The increased subcooling caused by misoperation of the RHR shutdown cooling mode could result in a slow power increase due to the reactivity insertion. This power rise, however, would be terminated by a flux scram before fuel thermal limits are approached. Therefore, only qualitative description of this event is provided.

15.1.4.4 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel clad, pressure vessel, or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.1.4.5 Radiological Consequences

Since this event does not result in any fuel failures, no analysis of radiological consequences is required for this event.

15.1.5 References

1. R. Linford, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802, February 1973.

2. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, (Unit 1: Rev 22, Unit 2: Rev 20).

3. Odar et al., "Safety Evaluation for the General Electric Topical Report: Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," NEDO-24154 and NEDE-24154-P, Volumes I, II and III, 1980.

4. Deleted

5. "NFS:RSA:90-048 EHC System Maximum Combined Flow Limiter (MCFL) Setpoint," W. F. Naughton (Nuclear Fuel Services Manager) tp G. J. Diederich (Station Manager - LaSalle County Station) April 25, 1990. (AIR 373-160-90-00022)

6. Deleted

7. Deleted

8. Deleted

9. Deleted

10. ANF-1358(P)(A), Revision 3 "The Loss of Feedwater Heating Transient in Boiling Water Reactors," Framatome ANP, Dated September 2005.

11. GE Letter LS-2209, H.R. Peffer (GE) to G.R. Crane (ComEd), Subject: LaSalle County Station Unit 1 & 2 Technical Specifications on MSIV Closure SCRAM, dated March 15, 1982.

12. Power Uprate Project Task 900, "Transient Analyses," GE-NE-A1300384-08 Revision 1, September 1999.

13. Deleted

14. Deleted

15. "Migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients," NEDE-32906P Supplement 3-A, Revision 1, April 2010.

16. "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," NEDE-32906P-A, Revision 3, September 2006.

15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

The events listed under the decrease in heat removal by the secondary system include pressure regulator failure closed, generator load rejection with and without operational turbine bypass, turbine trip with and without operational turbine bypass, inadvertent MSIV closure, loss of condenser vacuum, loss of A-C power, loss of feedwater flow, and failure of the RHR shutdown cooling system. The limiting events in terms of minimum critical power ratio were evaluated for the 1999 LaSalle County Station Power Uprate Project and documented in Appendix 15.B and Reference 12.

15.2.1 Pressure Regulator Failure - Closed

The Pressure Regulator Failure - Closed event is analyzed for reload cores for Pressure Regulator Out-of-Service.

15.2.1.1 Identification of Causes and Frequency Classification

Unit 1

The pressure control system controls reactor pressure during plant startup, power generation and shutdown modes of operation. The Mark VI pressure controllers act to ensure that the desired pressure set point is achieved through the positioning of the turbine control valves (CVs) and the steam bypass valves (BPVs) in response to changes in the pressure set point error.

Under steady state operating conditions, the CVs regulate steam pressure; however, whenever the total steam flow delivery exceeds the effective turbine steam flow need or capacity, the BPVs are opened to regulate the pressure and send the excess steam directly to the condenser.

The reactor operator maintains control over the rate of steam production to meet the plant's steam demands – these control functions take place outside of the turbine controllers. The turbine operator uses the operator workstation to set the desired operating pressure set point.

Pressure control is designed to control reactor pressure during the following conditions:

- reactor vessel heat up to rated pressure;
- when the turbine is being brought up to speed and synchronized;
- when reactor steam generation exceeds the turbine steam flow requirements during power operation;
- plant load rejections and turbine trip/generator trips; and
- reactor cool down.

The reactor pressure control algorithm is designed to operate using three pressure transmitter inputs from one of two locations in the steam flow path. In effect, two pressure control strategies are offered, either of which is selectable by the turbine operator. The control strategy offered by the three pressure transmitters tapped into the main steam supply header as it is formed from the reactor vessel dome structure is called reactor vessel (dome) pressure control or VPC. The second control strategy uses three pressure transmitters tapped into the main steam line just upstream of the main stop valves and is called turbine inlet main stream (throttle) pressure control or MSP.

If one of the three processors in the pressure controller failed low, an alarm will be generated and the pressure controller would maintain control of the turbine control valves with no change in pressure. If either one or two of the three pressure transmitters providing input to the pressure controller failed low, the turbine valves would adjust to the pressure sensed by the functioning transmitter and a small change in pressure would occur. The Digital Electro-Hydraulic Controls (DEHC) Pressure and Turbine control system will continue to regulate pressure and will maintain control of the bypass valves and the turbine valves.

The operator should note the alarm indicates the loss of one pressure controller and should verify that the regulator maintains proper control.

If a second processor or the third transmitter failed low, the turbine control valves will close resulting in an increase in reactor pressure.

It is also assumed for purpose of this transient analysis that one pressure controller is out of service, then a single failure occurs in another pressure controller, in addition to the controller that was initially out of service, resulting in a downscale failure of the pressure regulation demand to zero. Should this occur, it could cause full closure of the main turbine control valves as well as inhibit steam bypass flow and thereby increase reactor pressure.

This event is not possible without multiple failures in two separate controllers, however, for this postulated case it is treated as a moderate frequency event.

15.2.1.2 Sequence of Events and System Operation

If one of the three processors in the pressure controller failed low, an alarm will be generated and the pressure controller would maintain control of the turbine control valves with no change in pressure. If either one or two of the three pressure transmitters providing input to the pressure controller failed low, the turbine control valves would adjust to the pressure sensed by the functioning transmitter and a small change in pressure would occur. The Digital Electro-Hydraulic Controls (DEHC) Pressure and Turbine control system will continue to regulate pressure and will maintain control of the bypass valves and the turbine valves.

If a second processor or the third transmitter failed low, the turbine control valves will close resulting in an increase in reactor pressure.

The effect of single failures or operator errors is trivial. The nature of the first assumed failure produces a slight pressure increase in the reactor until the remaining controllers continue control. Since no other action is significant in restoring normal operation if another controller is failed at this time (a second assumed failure), the turbine control valves would continue to close thus raising reactor pressure to the point where a flux or a pressure trip would initiate scram of the reactor.

15.2.1.3 Core and System Performance

No mathematical model was used to evaluate this disturbance. The disturbance is mild, similar to a pressure setpoint change, and no significant reductions in fuel thermal margins occur. This transient is much less severe than the generator and turbine trip transients described in Subsections 15.2.2A and 15.2.3. Only a qualitative result is provided because of the subordinate nature of this event.

If one of the three processors in the pressure controller failed low, an alarm will be generated and the pressure controller would maintain control of the turbine control valves with no change in pressure. If either one or two of the three pressure transmitters providing input to the pressure controller failed low, the turbine control valves would adjust to the pressure sensed by the functioning transmitter and a small change in pressure would occur. The Digital Electro-Hydraulic Controls (DEHC) Pressure and Turbine control system will continue to regulate pressure and will maintain control of the bypass valves and the turbine valves.

If a second processor or the third transmitter failed low, the turbine control valves will close resulting in an increase in reactor pressure.

15.2.1.4 Barrier Performance

As noted above the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel clad, pressure vessel, or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.2.1.5 Radiological Consequences

Because this event does not result in any fuel failures nor any release of primary coolant to either the secondary containment or to the environment there are no radiological consequences associated with this event.

15.2.2A Generator Load Rejection

15.2.2A.1 Identification of Causes and Frequency Classification

Fast closure of the turbine control valves (TCV) is initiated whenever electrical grid disturbances occur which result in significant loss of load on the generator. The turbine control valves are required to close as rapidly as possible to prevent overspeed of the turbine-generator (TG) rotor. Closure of the main turbine control valves will increase system pressure.

a. Generator load rejection with normal turbine bypass operation is categorized as an incident of moderate frequency.

b. Generator load rejection with failure of the turbine bypass operation is categorized as an infrequent incident. Because the new Digital Electro-Hydraulic Controls (DEHC) system is of fault tolerant triple modular redundant (TMR) design (UFSAR section 10.4.4.2), when compared to the previous non redundant analog control system for bypass and turbine control, the overall reliability of the new DEHC system is considered more reliable than the old system. Thus, when compared to the overall failure rate of the bypass system, the frequency of a generator load rejection with failure of the turbine bypass operation remains categorized as an infrequent incident. This event is a transient and as such results in an operating CPR, which does not compromise the MCPR safety limit.

This event is analyzed for every reload since this transient may be the most limiting for setting core thermal limits (see Section 15.A). For GE analyses, methodology outlined in References 1, 3, 20, and 21 may be used for this event. GE utilizes the ODYN plant transient code or the TRACG transient code. Below is a description of the Generator Load Rejection event that outlines the general trends for this event. The results of the reload analysis are given in the reload licensing package which is referenced in the Technical Requirements Manual.

15.2.2A.2 Sequence of Events and System Operation

Complete loss of generator load while the turbine is at 100% NBR power has the largest transient potential for this event.

a. An example sequence of events for loss of generator load with normal operation of the turbine bypass valves (25% bypass capability at LaSalle) (Figure 15.2-1) is as follows (note that the event response may vary slightly based on plant settings):

Time (sec)	Event
-0.050 (approx)	Turbine generator detection of loss of electrical load.
0	Turbine generator power-load-un-balance (PLU) devices trip to initiate turbine control valve fast closure.
0	Turbine generator PLU trip initiates main turbine bypass system operation.
0	Fast turbine control valve closure initiates reactor scram.
0	Fast turbine control valve closure initiates a recirculation pump downshift (EOC RPT)
0.07	Turbine Control valves closed.
0.1	Turbine bypass valves start to open.
0.19	Recirculation pump motor circuit breakers open thus decreasing in-core flow.
1.35, 1.52, 1.67, 1.83 and 2.07	Relief Valves activate sequentially in groups: 1, 2, 3, 4, and 5.

Time (sec)	Event
4.08	High water level trip L8 initiates feedwater trip.
(Est) 5.2, 5.6, 6.0, 6.2, and 7.1	Relief Valves close sequentially in groups: 5, 4, 3, 2, and 1.
24.8 (est)	Turbine bypass valves start to close.
32.5 (est)	Turbine bypass valves closed.
34.56	Turbine bypass valves reopen on pressure signal.
38.31	L2 vessel water level trip initiates MSIV's (see Subsection 15.0.7).
	L2 vessel water level trip initiates RCIC and HPCS (not simulated) and RR pump trip.

b. An example sequence of events for loss of generator load accompanied by failure of the turbine bypass valves (Figure 15.2-2) is as follows (note that the event response may vary slightly based on plant settings):

Time (sec)	Event
-0.050 (approx)	Turbine-generator detection of loss of generator load.
0	Turbine-generator power load unbalance (PLU) devices trip to initiate turbine control valve fast closure.
0	Turbine bypass valves fail to operate.
0	Fast turbine control valve closure initiates reactor scram.

Time (sec)	Event
0	Fast turbine control valve closure initiates a recirculation downshift (EOC RPT).
0.07	Turbine control valves closed.
0.19	Recirculation pump motor circuit breakers open thus decreasing in-core flow.
1.09, 1.20, 1.31, 1.45, and 1.60	Relief valves actuate sequentially in groups 1, 2, 3, 4, and 5.
(Est) 7.4, 7.8, 8.1, 8.5 , and 9.3	Relief valves close sequentially in groups: 5, 4, 3, 2, and 1.
11.38, 12.56	Relief valves actuate sequentially in groups: 1 and 2.
(Est) 17.6, 18.8	Relief valves close sequentially in groups: 2 and 1.
24.3	Group 1 Relief Valves actuated.
32.0(est)	Group 1 Relief Valves closed.

In general with bypass, no corrective action will be taken by the operator for several minutes due to the rapid nature of the transient. Observation is directed at verifying proper bypass valve operation, maintenance of reactor water level at a satisfactory level by the feedwater system, and acceptable reactor pressure as controlled by the pressure regulator.

For either case, with or without bypass valve operation, the operator should verify the relief valve operation. After the brief system discharge to the suppression pool, the unit will continue to expel steam from decay heat via the turbine bypass system.

Reactor pressure will be controlled by the operator, as required, depending on whether a restart or shutdown with cooldown is planned.

Consideration of single failure criterion discloses that a scram trip signal initiates at all power levels greater than or equal to 25% of rated core thermal power. A control valve fast closure scram is generated at powers above 60%. Between 25% and 60% of rated reactor power, a turbine trip followed by a turbine stop valve position scram is initiated from generator protection logic (Reference 17). A recirculation pump trip also is initiated for this same regime. Both of these trip signals satisfy single failure criterion. The pressure relief valves operate independently (individually) on their individual setpoints. Manual operation is independent of this automatic mode. All plant control systems maintain normal operation unless specifically designated to the contrary. The mitigation of pressure increase is accomplished by the reactor protection system functions. The turbine control valve trip and the RPT are designed to satisfy the single failure criterion.

15.2.2A.3 Core and System Performance

Mathematical Model

The computer model used to simulate this event for the initial core is the ODYN model given in Reference 3.

Input Parameters and Initial Conditions

Cycle specific reload analyses are performed at nominal or rated conditions (see section 15.A). The turbine's electrohydraulic control system (EHC) power/load imbalance device detects load rejection before a measurable speed change takes place.

The closure characteristics of the turbine control valves are assumed such that the valves operate in the full arc mode and have a full stroke closure time (full open to fully closed) of 0.15 seconds.

Auxiliary power would normally be independent of any turbine-generator overspeed effect and would be continuously supplied at rated frequency which is the situation when automatic fast transfer to auxiliary power supplies normally occurs. In this simulation, however, for the purposes of worst case analysis, the recirculation pumps are assumed to remain tied to the main generator wherein they will increase in speed with the turbine-generator as it overspeeds momentarily until tripped by the recirculation pump trip (RPT).

The reactor is operating in the manual flow control mode when load rejection occurs. (Results do not differ significantly if the plant had been operating in the automatic flow control mode).

The bypass valve opening characteristics are simulated using the specified delay together with the specified opening characteristic required for bypass system operation.

Although the closure of main steam isolation valves (see Subsection 15.0.7) as activated from the vessel low-water-level trip (L2) is included in the simulation, the water flow increments from the RCIC and HPCS core cooling functions are not included because normal actuation and startup of those ECC systems can take up to 30 seconds before their effects are realized. By the time these events occur, the primary concerns of fuel thermal margin and vessel overpressure effects have already passed. Their use results in less severe effects than those already experienced by the reactor system.

An example of scram position and reactivity characteristics used in the dynamic analysis are designated by Figure 15.2-3.

Results

a. Figure 15.2-1 shows example results of generator load rejection trip from near rated power with normal operation of the turbine bypass equipment (note that the event response may vary slightly based on plant settings).

b. Figure 15.2-2 shows example results of generator load rejection with accompanying failure of the turbine bypass equipment (note that the event response may vary slightly based on plant settings).

The results of the reload analysis are given in Section 15.A and in the reload licensing package referenced in the Technical Requirements Manual.

Consideration of Uncertainties

The full-stroke closure rate of the turbine control valve is conservatively taken at 0.15 seconds. Typically, actual closure rates are more like 0.20 seconds. Clearly, the less time it takes to close, the more severe is the pressurization effect. All systems utilized for protection in this event were assumed to have the poorest allowable response (e.g., relief setpoints at max, scram stroke time, and rod worth characteristics). Plant behavior is therefore, expected to be reduced in severity from the model predictions for this transient.

15.2.2A.4 Barrier Performance

The consequences of these two versions of the load rejection event do not result in any temperatures nor pressures in excess of the criteria for which the fuel clad, pressure vessel, or containment are designed; therefore, these barriers maintain their integrity.

15.2.2A.5 Radiological Consequences

While the consequence of this event does not result in fuel failures, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under defined meteorological and controlled release conditions. If purging of the containment is chosen, the release will have to be in accordance with established technical specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.2.3 Turbine Trip

The Turbine Trip with Bypass event is not analyzed for reload cores. The analysis and results presented are for the initial core.

15.2.3.1 Identification of Causes and Frequency Classification

A variety of turbine or nuclear system malfunctions will initiate a turbine trip. Some examples are: moisture separator / reheater drain tank high levels, large vibrations, operator lockout, loss of control fluid pressure, low condenser vacuum, and reactor high water level.

A turbine trip with bypass is categorized as an incident of moderate frequency. In defining the frequency of this event, turbine trips which occur as a byproduct of other transients such as loss of condenser vacuum or reactor high level trip events are not included. However, spurious low vacuum or high level trip signals which cause an unnecessary turbine trip are included in defining the frequency. This type of division of initiating causes is required to get an accurate event-by-event frequency breakdown.

15.2.3.2 Sequence of Events and Systems Operation

The sequence of events for a turbine trip is similar to that for a generator load rejection. Turbine stop valve closure occurs over a period of approximately 0.1 second. Switches at the stop valves sense the turbine trip and initiate reactor scram. The bypass valves opened normally on a signal from the turbine trip. An RPT is initiated in a similar manner. Safety/relief valves operate to control vessel pressure; following this, the bypass valves reopen on pressure increase to control reactor pressure.

The sequence of events for a turbine trip from high power with an operating bypass (Figure 15.2-4) is as follows:

Time (sec)	Event
0	Turbine trip initiates closure of main stop valves.
	Turbine trip initiates bypass operation.
0.01	Main turbine stop valves reach 90% open position and initiate reactor scram trip.
	Main turbine stop valves reach 90% open position and initiate a recirculation pump downshift (EOC RPT).

Time (sec)	
0.10	Turbine stop valves closed.
	Turbine bypass valves start to open to regulate pressure.
0.19	Recirculation pump motor circuit breakers open causing decrease core flow.
1.55, 1.69, 1.84, 2.02, and 2.30	Relief valves actuated sequentially by groups: 1, 2, 3, 4, and 5.
4.53	Feedwater trip on high water level (L8).
(Est) 5.2, 5.5, 5.9, 6.2 and 7.0	Relief valves close sequentially by groups: 5, 4, 3, 2, and 1.
30.6	Bypass valve initiates to close on pressure signal.
32.2 (est)	Turbine bypass closed.
38.67	Low level trip (L2) initiates a main steam line isolation (see Subsection 15.0.7).
	Low level trip (L2) initiates RCIC and HPCS (not simulated) and RR pump trip.
39.23	Turbine bypass reopen on pressure increase at turbine inlet

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All plant control systems maintain normal operation unless specifically designated to the contrary. Turbine stop valve closure initiates a reactor scram trip via position signals to the protection system. Credit is taken for successful operation of the reactor protection system. Turbine stop valve closure initiates recirculation pump downshift (EOC RPT) thereby reducing the jet pump drive flow. The pressure relief system which operates the relief valves independently when system

pressure exceeds relief valve instrumentation setpoints is assumed to function normally during the time period analyzed.

The turbine trips are analyzed in two parts: the turbine trip with bypass via the REDY model in this section (15.2.3) and the turbine trip without bypass via the ODYN model in Subsection 15.2.3A which follows. The only difference is that failure of the main turbine bypass system is assumed for the entire transient time period analyzed in Subsection 15.2.3A. If the turbine trip occurs at a power level below 25% of rated core thermal power a trip inhibit signal derived from first stage turbine pressure is activated. The normal bypass capacity is adequate to dump the steam load to the condenser without the necessity of shutting down the reactor. All other protection system functions remain functional as before and credit is taken for those protection system trips.

If a turbine trip occurs during normal system operation at greater than or equal to 25% of rated core thermal power, the mitigation of pressure increase is accomplished by the reactor protection system functions. The main turbine stop valve closure trips and downshift (EOC RPT) are redundantly built to satisfy single failure criteria.

Operator actions during this transient normally include the following: verification that auto transfer of generator-supplied buses (if auto transfer does not occur, manual transfer must be made); monitor reactor scram for normal rods in condition; monitor and maintain reactor water level as needed; observe proper turbine operation during coastdown; and depending on conditions, initiate normal post shutdown cooling procedures or maintain pressure for restart of the unit. Checking proper safety/relief valve closure and suppression pool temperature are power generation objectives and are not needed to safely terminate a turbine trip.

15.2.3.3 Core and System Performance

Mathematical Model

The non-linear, dynamic model described in Reference 4 was used to simulate this turbine trip event with the bypass valve operating.

Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with input parameters and initial conditions as tabulated in Table 15.0-2. The reactor is initially operating at 105% NBR power with vessel dome pressure of 1020 psig for these simulations.

Turbine stop valves full stroke closure time is 0.1 second.

A reactor scram is initiated by position switches on the turbine stop valves when the valves are 90% open. This stop valve scram trip signal is automatically bypassed when the reactor is below 25% of rated core thermal power.

Reduction in core recirculation flow is initiated by position switches on the main stop valves, which actuate the RPT circuitry that downshifts the recirculation pumps (EOC RPT).

Conservative EOC scram characteristics are used as is also time of the void reactivity coefficient and the Doppler reactivity coefficient which are the same as used in the generator load rejection analysis of Subsection 15.2.2A.

Results

The turbine trip with bypass operating normally was simulated for 105% NBR steam flow as depicted in Figure 15.2-4.

Neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the flux increase is limited to 165% of rated by the turbine stop valve scram and RPT system. Peak fuel surface heat flux does not exceed 103% of its initial value.

Peak pressure in the bottom of the vessel reaches 1164 psig, which is below the ASME code limit of 1375 psig for the reactor cooling pressure boundary. Vessel dome pressure does not exceed 1138 psig. The severity of turbine trips from lower initial power levels decreases to the point where a scram can be avoided if auxiliary power is available from an external source and the power level is within the bypass capability.

The safety/relief valves are opened and closed sequentially as the stored energy is dissipated and the pressure falls below the setpoints of the valves. Peak nuclear system pressure reaches 1193 psig at the vessel bottom, therefore, the overpressure transient is clearly below the reactor coolant pressure boundary transient pressure limit of 1375 psig. Peak dome pressure does not exceed 1166 psig.

A low power turbine trip with the bypass valves operating results in a steam dump to the condenser without the need to shut down the reactor.

Consideration of uncertainties in this analysis involves the protective system settings, system capacities, and system response characteristics. The most conservative values were used in the analyses; for example, the slowest allowable control rod scram motion was assumed, the EOC all-rod-out condition is assumed for prescram configuration, safety/relief actions occurred at high error limit values, minimum value capacities were utilized for overpressure protection. Such conservatisms are intended to cover uncertainties as defined or anticipated.

15.2.3.4 Barrier Performance

The consequences of this version of turbine trip does not result in any temperatures nor pressures in excess of the criteria for which the fuel clad, pressure vessel, or containment are designed; therefore, these barriers maintain their integrity.

15.2.3.5 Radiological Consequences

While the consequence of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled meteorological and release condition. If purging of the containment is chosen, the release will have to be in accordance with established technical specifications; therefore this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.2.3A Turbine Trip Without Bypass

15.2.3A.1 Identification of Causes and Frequency Classification

A variety of turbine or nuclear system malfunctions will initiate a turbine trip. Some examples are: moisture separator / reheater drain tank high levels, large vibrations, operator lockout, loss of control fluid pressure, low condenser vacuum, and reactor high water level.

A turbine trip without bypass is categorized as an infrequent incident. [HISTORICAL - The frequency is expected to be as follows: 0.0064/plant year with MTBE of 156 years. The basis for this frequency prediction is that the failure rate of the bypass valves is 0.0048, the same as for generator load rejection. Combining this with the turbine trip frequency of 1.22 events per plant year yields the frequency of 0.0064 per plant year. This event is categorized as a transient; its ΔCPR does not compromise the safety limit MCPR.] Because the new Digital Electro-Hydraulic Controls (DEHC) system is of fault tolerate triple modular redundant (TRM) design (UFSAR Section 10.4.4.2), when compared to the previous non redundant analog control system for bypass and turbine control, the overall reliability of the new DEHC system is considered more reliable than the old system. Thus, when compared to the overall failure rate of the bypass system, the frequency of a generator load rejection with failure of the turbine bypass operation remains categorized as an infrequent incident.

This event is analyzed for every reload because this transient may be the most limiting for setting core thermal limits (see Section 15.A). The analysis description to follow is based on GE methodology. Whether TTNBP is bound by LRNBP (Section 15.2.2A) depends on the initial position of the TCV. Cycle specific MCPR operating limits are established which protect TTNBP.

15.2.3A.2 Sequence of Events and System Operation

A typical sequence of events for a turbine trip from high power with accompanying failure of the bypass valves to operate (Figure 15.2-5) is as follows:

Time (sec)	Event
0	Turbine trip initiates closure of main stop valves.
	Turbine bypass valves fail to operate.
0.011	Main turbine stop valves reach 90% open position and initiate reactor scram trip.
	Main turbine stop valves reach 90% open position and initiate a recirculation pump downshift (EOC RPT).
0.1	Turbine stop valves closed.
0.19	Recirculation pump motor circuit breakers open causing decrease in core flow.
1.13, 1.24, 1.36, 1.50 and 1.65	Relief valves actuated sequentially by groups: 1, 2, 3, 4, and 5.
5.98	L8 trip initiate a feedwater pump trip.
(Est) 7.3, 7.7, 8.0, 8.3 & 9.2	Relief valves close sequentially by groups: 5, 4, 3, 2, and 1.
Through 25	Reactuation of relief valves (one or two) to relieve pressure.
26.70	Low level trip (L2) initiates a main steamline isolation (see Subsection 15.0.7).

Time (sec)	Event
	Low level trip (L2) initiates RCIC and HPCS (not simulated) and RR pump trips.

All plant control systems maintain normal operation except the turbine bypass valves fail to open. Turbine stop valve closure initiates a reactor scram trip via position signals to the protection system. Credit is taken for successful operation of the reactor protection system. Turbine stop valve closure initiates recirculation pump downshift (EOC RPT) thereby reducing the jet pump drive flow. The pressure relief system which operates the relief valve independently when system pressure exceeds relief valve instrumentation setpoints is assumed to function normally during the time period analyzed.

This turbine trip is analyzed with the ODYN code to emphasize the early pressurization peak. The failure of the main turbine bypass system is assumed for the entire transient time period analyzed. All other protection system functions remain functional as before and credit is taken for those protection system trips. A low-power turbine trip with failure of the bypass capability will follow the pattern of a high power turbine trip with bypass failure except the RPT and turbine stop valve closure scram trip is normally inoperative. The high flux, high pressure, high water level scrams will still protect the reactor should a single failure occur.

If a turbine trip without bypass occurs during normal system operation at greater than or equal to 25% of rated core thermal power the mitigation of pressure increase is accomplished by the reactor protection system functions. The main turbine stop valve closure trips and RPT trips are redundantly built to satisfy single failure criteria.

15.2.3A.3 Core and System Performance

Mathematical Model

The turbine trip without bypass failure is simulated with the ODYN code as given in Reference 3.

Input Parameters and Initial Conditions

Cycle specific reload analyses are performed at nominal or rated conditions (see section 15.A).

Results

Figure 15.2-5 shows typical results of the turbine trip with accompanying failure of the turbine bypass to operate. Neutron flux increase rapidly because of the void reduction caused by the pressure increase.

This turbine transient would be less severe at low power than it is at rated power because of the following:

a. Below 25% rated core thermal power, the turbine stop valve closure scrams and the turbine control valve closure scram are automatically bypassed.

b. At these lower power levels, the turbine first stage pressure interlock is used to initiate this bypass of scram logic.

c. The protective scram which terminates this transient is initiated by high reactor vessel pressure (safety setpoint 1043 psig).

d. With the turbine bypass system not functioning, the vessel pressure rise is faster than when it is normally functioning; thus the vessel pressure setpoint is reached sooner than for the normal case.

e. The transient is, therefore, turned around at a lower peak pressure and at an earlier time.

Peak surface heat flux and peak fuel centerline temperature remain at relatively low values for the low power event.

Consideration of uncertainties in this analysis involves the protective system settings, system capacities, and system response characteristics. The most conservative values were used in the analyses; for example, the slowest allowable control rod scram motion was assumed, the EOC all-rod-out condition is assumed for prescram configuration, safety/relief actions occurred at high error limit values, minimum value capacities were utilized for overpressure protection. Such conservatisms are intended to cover uncertainties as defined or anticipated.

The results of the reload analysis are given in Section 15.A and in the reload licensing package referenced in the Technical Requirements Manual.

15.2.3A.4 Barrier Performance

The consequences of this version of turbine trip does not result in any temperatures nor pressure in excess of the criteria for which the fuel clad, pressure vessel, or containment are designed; therefore, these barriers maintain their integrity.

15.2.3A.5 Radiological Consequences

Unchanged from initial conclusions in Subsection 15.2.3.5.

15.2.4 Inadvertent MSIV Closure

The inadvertent MSIV closure with direct scram on valve position, as described in this section, is not analyzed for reload cores. The analysis and results presented are for the initial core. Note that the inadvertent MSIV closure with flux scram, described in Section 5.2, is analyzed for reload cores. AREVA documents their disposition of events for MUR PU conditions in Reference 19, which confirms that this event is bounded by the events that are analyzed each reload by AREVA. At each follow-on reload cycle, AREVA confirms the validity of Reference 19. Reference 19 is applicable to both Units 1 and 2.

15.2.4.1 Identification of Causes and Frequency Classification

Various steamline and nuclear system malfunctions, or operator actions, can initiate main steam isolation valve closure. Examples are: low-steamline pressure, high-steamline flow, low water level, or manual action.

Inadvertent MSIV closure is categorized as an incident of moderate frequency irrespective of whether one MSIV or all MSIVs close.

To define the event of all MSIVs closing as the initiating event and not the byproduct of another transient, implies only the following contributions to the frequency considerations: manual action (purposely or inadvertent); spurious signals such as low pressure, low reactor water level, low condenser vacuum, etc.; and finally, equipment malfunctions such as faulty valves or operating mechanisms. A closure of one MSIV may cause an immediate closure of all the other MSIVs

depending on reactor conditions. If this occurs, it is also included in this category. During the main steam isolation valve closure, position switches on the valves provide a reactor scram if the valves in three or more main steamlines are less than 85% open (except for an interlock from the reactor mode switch which permits proper plant startup). Protection system logic, however, permits the test closure of one valve without initiating scram from the position switches.

For the condition of one MSIV closing, one MSIV may be closed at a time for testing purposes. This is done manually. Operator error or equipment malfunction may cause a single MSIV to be closed inadvertently. If reactor power is greater than about 75% when this occurs, an APRM high flux scram or high steamline isolation scram may result. If all MSIVs close as a result of the single closure, the event is included in the frequency group of the preceding paragraph.

15.2.4.2 Sequence of Events and Systems Operation

The sequence of events for the closure of all MSIVs (Figure 15.2-6) is as follows (Initial Core Results):

Time (sec)	Event
0	Initiate closure of all main steamline isolation valves (MSIV).
0.45	MSIVs reach 85% open. (Reference 5)
0.45	MSIV position trip scram initiated. (Reference 5)
1.6	Loss of feedwater begins as turbine loses steam supply.
2.42, 2.51, 2.60, 2.70 and 2.80	Relief valves actuated by groups 1, 2, 3, 4, and 5.
3.0	All main steamline isolation valves closed.
4.47	Recirculation FCV runback on low level alarm (L4) and two feedwater pumps running.
Est. 7.3 7.6, 7.9 and 8.8	Relief valves close by groups 5, 4, 3, 2, and 1.

Time (sec)	Event
10.35	Group 1 relief valves reactuate on high pressure.
10.90	Group 2 relief valves reactuate on high pressure.
Est. 15.9, 17.2	Relief valves reclose by groups 2, and 1.
18.7	Vessel water low level trip (L2) initiates recirculation pump trip. Vessel water low level trip (L2) initiates RCIC & HPCS (not simulated).
20.84	Group 1 relief valves cycle open and closed on pressure.

The MSIV closure event starts with a signal to initiate closure of all main steam line isolation valves (MSIVs).

As all main steam isolation valves close, position switches on these valves initiate a reactor scram when the valves in three or more main steamlines are less than 85% open. This scram signal requires that the reactor pressure is above the Group 1 isolation pressure setpoint and that the reactor mode switch is in the RUN position. Credit is taken for successful operation of the reactor protection system. Normal operation of the pressure relief system logic which initiates the opening of relief valves is also assumed during the time period covered by the analysis. All plant control systems maintain normal operation unless specifically designated to the contrary. Reactor vessel water level would be maintained by RCIC and HPCS when reactor level reaches level 2.

For the closure of a single MSIV, that action will not initiate a reactor scram on valve position because the valve position trip logic is designed to accommodate single MSIV closure during operation at limited power levels. Although the closure of a single MSIV will not initiate a reactor scram on valve position, the resulting transient may initiate a reactor scram on high flux or high reactor pressure. The main steamlines are sized to carry full rated steam flow with one line closed. MSIV testability during normal reactor operation is possible. Credit is taken for the operation of the pressure signals of the NSSS and the flux signals of the reactor protection systems to initiate reactor scram. All plant control systems are assumed to operate normally unless designated to the contrary.

Operator actions should assure that a normal shutdown occurs and that adequate core coverage is maintained for cooling requirements. Other than assurance of adequate reactor pressure relief and requisite cooling following shutdown, there are no safety actions required by the operator.

Consideration of single failures and operator errors shows that mitigation of pressure rise is accomplished by MSIV position switch initiation of reactor scram followed by reactor protection system shutdown of the reactor. Relief valves also operate to limit vessel pressure. Each of these aspects of safety control is designed to single failure criteria and, in this case, additional failures would not alter the results of the analysis. Failure of a single relief valve to open is not expected to have any significant effect because less than 20 psi increase in vessel pressure would occur. The peak pressure is still considerably less than the 1375 psig pressure limit.

15.2.4.3 Core and System Performance

Mathematical Model

An extensive nonlinear dynamic model is employed in the transient analyses for the initial core. The model is described in Reference 4.

Input Parameters and Initial Conditions

The reactor is initially operating at 105% of NB rated power with a vessel dome pressure of 1020 psig. Other plant parameters for the initial core are shown in Table 15.0-2.

The assumptions and conditions are as follows:

a. Automatic circuitry or operator action initiates closure of the main steamline isolation valve(s) which in turn initiates the transient.

b. The main steam isolation valves close in 3 to 5 seconds. The worst case, the 3-second closure time, is assumed for the analysis shown here.

c. Position switches on the valves initiate a reactor scram when the valves are less than 85% open (Reference 5). Closure of these valves inhibits steam flow to the feedwater turbines terminating feedwater flow.

d. Valve closure indirectly causes a trip of the main turbine and generator.

e. Because of the loss of feedwater flow, water level within the vessel decreases sufficiently to initiate trip of the recirculation pump and initiate the HPCS and RCIC systems.

Results

For closure of all MSIVs for the initial core, Figure 15.2-6 shows the changes in important nuclear system variables following simultaneous isolation while the reactor is operating at 105% of NBR steam flow. Peak neutron flux reaches 269% of NBR power flux at approximately 2 seconds. At this time, the nonlinear valve closure characteristic exerts a dominating effect and the assumed conservative scram characteristic has not yet allowed credit for full shutdown of the reactor. No significant increase in fuel surface heat flux nor reduction in MCPR occurs. Water level decreases sufficiently to cause a recirculation pump trip with accompanying initiation of the HPCS and RCIC systems at approximately 18 seconds. However, there is a delay of up to 30 seconds before water supply enters the vessel. There is no change in the thermal margins during the transient.

The nuclear system relief valves begin to open automatically at approximately 2.4 seconds after the isolation is initiated. The valves close sequentially as the stored heat is dissipated and will continue intermittently to discharge steam from decay heat. The peak pressure in the main steamline is 1152 psig. For the initial core analysis, peak pressure at vessel bottom is 1199 psig, clearly below the pressure limits of the reactor coolant pressure boundary. A subsequent evaluation (Reference 5) calculated the peak pressure at vessel bottom to be 1203 psig.

For closure of only one MSIV (such as is permitted for testing purposes), the normal test requirements limit the reactor power to approximately 75% of design conditions to preclude a high flux scram, high pressure scram or complete MSIV isolation of all main steamlines. Only one MSIV is permitted to be closed at a time for testing purposes; this testing mode precludes a reactor scram from the MSIV closure switches on the valve undergoing test. The closure of a single MSIV for testing will not initiate a reactor scram on valve position because the valve position trip logic is designed to accommodate single MSIV closure during operation at limited power levels (<75%).

Inadvertent closure in 3 seconds of an MSIV during 105% NBR steam flow results in flow disturbances sufficient to raise vessel pressure and reactor power enough to cause an APRM high neutron flux scram. No quantitative results are shown for this event because no significant effect is imposed on the reactor coolant pressure boundary. System pressure is regulated via the main turbine bypass system for the other three steamlines.

Inadvertent closure of one or all MSIVs while the reactor is shutdown will produce no significant transient. MSIV closure during plant heatup will be less severe than for the maximum power cases discussed above.

Considerations for uncertainties in the analyses are included in the reactor protection system settings, system capacities, and system response characteristics. In all cases, the most conservative values were used, for example: the slowest allowable control rod motion, the scram worth curve for an all-rods-out condition, minimum valve capacities for overpressure protection, action points on the relief valves were taken at 115% of the nominal setpoint.

The results of the reload analysis are given in Section 15.A and in the reload licensing package which is referenced in the Technical Requirements Manual.

15.2.4.4 Barrier Performance

The consequences of MSIV closure, whether involving all MSIVs or a single MSIV, do not result in any temperatures nor pressures in excess of the criteria for which the fuel clad, pressure vessel, or containment are designed; therefore, these barriers maintain their safety integrity.

The activity released to the suppression pool via the relief valves' discharge is activity in the reactor coolant.

15.2.4.5 Radiological Consequences

While the consequence of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under defined meteorological and controlled release conditions. If purging of the containment is chosen, the release will be done in accordance with established technical specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

As described above, the radiological consequences are based on the normal coolant activity and controlled release administrative procedures. Power uprate to 3489 MWt does not impact either normal coolant activity or the control release procedures, as described in Reference 13.

15.2.5 Loss of Condenser Vacuum

The Loss of Condenser vacuum event is not analyzed for reload cores. The analysis and results presented are for the initial core.

The analysis of this transient for the initial core assumed main and feed water turbine trip at the same condenser vacuum (20-23 in HG). The variations in the actual setpoint for feed water turbine trip would cause a different timing in the sequence of events, but would not significantly change the transient analysis results. For example, if the setpoint were 16 – 18 inches Hg, an approximate 1.5 second delay for feed water turbine trip, after main turbine trip would result (as discussed in 15.2.5.2 and 15.2.5.3). Feed water flow decrease is similarly delayed and vessel level does not experience as severe a decrease as in Figure 15.2-7. Negligible effects on vessel pressure occur as a result of this change, since SRVs and bypass valves control RCS pressure as in the analyzed event. The analysis has not been revised since results of the setpoint change are bounded by the analyzed case. Since this event still has turbine bypass available, it is bounded by the turbine trip no bypass, which is bounded by the load rejection no bypass event, which is analyzed each cycle.

15.2.5.1 Identification of Causes and Frequency Classification

Various system malfunctions which can cause a loss of condenser vacuum due to some single equipment failure are shown below with their estimated vacuum decay rates:

CAUSE	ESTIMATED DECAY RATE
Failure or isolation of steam jet air ejectors	< 1 inch Hg/min.
Loss of sealing steam to shaft gland seals	1 to 2 inches Hg/min.
Opening of vacuum breaker valves	2 to 12 inches Hg/min.
Loss of one or more circulating water pumps	4 to 24 inches Hg/min.

This event is categorized as an incident of moderate frequency.

15.2.5.2 Sequence of Events and Systems Operation

The sequence of events for loss of condenser vacuum (Figure 15.2-7) is as follows (Initial Core Results):

Time (sec)	Event
0	Initiate simulated loss of condenser vacuum at 2 inches Hg/sec.

Time (sec)	Event
5.00	Low condenser vacuum main turbine trip initiated.
	Main turbine trip initiates turbine bypass valve operation.
5.39	Main turbine stop valves reach 90% open position and initiate reactor scram trip and recirculation pump downshift (EOC RPT).
	Turbine stop valves closed and turbine bypass valves start to open to regulate pressure.
5.52	Recirculation pump motor circuit breakers open causing decrease in core flow.
6.50	Low condenser vacuum feedwater turbine trip initiated.
6.65, 6.81, 6.98 and 7.18	Relief valves automatically actuate by Groups 1, 2, 3, and 4.
10.025	Low condenser vacuum initiates turbine bypass valve closure and main streamline isolation valve closure.
Est. 10.325	Turbine bypass valve(s) close.
Est. 11.0, 11.3, 11.9 and 12.7	Relief valves reclose by Groups 4, 3, 2, and 1.
13.42 and 13.85	Pressure relief valves reopen by Groups 1 and 2.
Est. 20.1	Group 2 relief valve closes.

Time (sec)	Event
Est. 23.8	Low vessel level (L2) trip initiates RCIC and HPCS and RR pump trip (not simulated).
Est. 32.2	Group 1 relief valve closes.
40.85	Group 1 relief valves cycle open and closed on pressure.

In establishing the expected sequence of events to simulate plant performance, it was assumed that normal functioning occurred in plant instrumentation and controls, plant protection systems and in the reactor protection systems. The trip signals associated with loss of vacuum at the condenser originate from sensors at the following levels of vacuum.

Vacuum (in Hg)	Action Inititated
27 to 30	Normal vacuum range.
20-23	Main turbine trip via turbine stop valve closures.
16-18	Feedwater Turbine(s) trip.
7-10	MSIV closure and turbine bypass valve(s) closure.

This event does not lead to a general increase in reactor power level. Curtailment of power increase is accomplished by the reactor scram from the main turbine stop valve position switches. This turbine stop valve scram trip signal is automatically bypassed whenever the reactor is below 25% of rated core thermal power; the main turbine steam bypass valves are able to divert steam to the main condenser for this condition.

Operator actions are basically the monitoring of a normal reactor shutdown: verify transfer to incoming power bus; monitor water level in the vessel, observe coastdown, monitor reactor shutdown, initiate essential cooling, monitor vessel pressure.

15.2.5.3 Core and System Performance

Mathematical Model

The GE transient computer model used to simulate the transient event is the same as that described in Subsection 15.1.1.3.

Input Parameters and Initial Conditions

The analysis was performed with plant conditions tabulated in Table 15.0-1 unless otherwise noted. The turbine stop valve full-stroke closure time is 0.1 second. The simulation for a hypothetical case with a conservative 2-inches Hg/sec. vacuum decay rate. Thus, the steam bypass valves are available for several seconds because the bypass signal occurs at a vacuum level of about 10 inches Hg ahead of the trip from the turbine stop valve(s) position switches as they close.

Results

Under this hypothetical 2 inches Hg/sec. vacuum decay condition, the turbine bypass valve and main steamline isolation valve closure would follow main turbine trip by about 5 seconds and feedwater turbine trips by about 3.5 seconds after they initiate the transient. This transient, therefore, is similar to a normal turbine trip with bypass. The effect of main steamline isolation valve closure tends to be minimal since the closure of main turbine stop valves and subsequently the bypass valves have already shutoff the main steamline flow. Figure 15.2-7 shows the transient expected for this event. It is assumed that the plant is initially operating at 104% of NBR steam flow conditions. Peak neutron flux reaches 151% of NBR power while average fuel surface heat flux reaches 104% of rated value. Relief valves open to limit the pressure rise, then sequentially reclose as the stored energy is dissipated. There is no significant decrease in MCPR.

Peak nuclear system pressure is 1159 psig, at the vessel bottom. Clearly, the overpressure transient is below the reactor coolant pressure boundary transient pressure limit of 1375 psig. Vessel dome pressure does not exceed 1134 psig. A comparison of these values to those for turbine trip with bypass failure, at high power shows the similarities between these two transients. The prime differences are the loss of feedwater and main steamline isolation, earlier in the event.

Consideration of uncertainties is dominated by the rate of loss of vacuum in the main turbine condenser because the protective actions are actuated at various levels of condenser vacuum. The severity of the resulting transient is dependent upon the rate at which the vacuum pressure is lost. Typical loss of vacuum due to loss of cooling water pumps or steam jet air ejector problems produces a very slow loss of vacuum rate (minutes, not seconds). Normally the condenser vacuum loss sequentially trips the main turbine and the feedwater turbines, then closes the

MSIVs and the bypass valves. These major events also result in actions which include 1) reactor scram from the turbine stop valve closure and 2) opening of the turbine steam bypass valves when the main turbine trips. A faster rate of vacuum loss reduces the anticipatory action of the scram and the effectiveness of the steam bypass valves because they would close more quickly.

Other uncertainties in the simulation involve protection system settings, system capacities, and system response characteristics. In all cases, the most conservative valves were used in the analysis. For example: the slowest allowable control rod motion for scram, the all-rods out condition for the scram worth curve, minimum valve capacities for overpressure protection, upper tech spec limits on relief valve settings, etc., are utilized in the simulation. Neither the HPCS nor RCIC system was included in this transient.

15.2.5.4 Barrier Performance

The consequences of loss of condenser vacuum, at the postulated maximum rate, do not result in any temperatures nor pressures in excess of the criteria for which the fuel clad, pressure vessel, or containment are designed; therefore, these barriers maintain their safety integrity.

15.2.5.5 Radiological Consequences

While the consequence of this event does not result in fuel failures, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under defined meteorological and controlled release conditions. If purging of the containment is chosen, the release will be in accordance with established technical specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.2.6 Loss of A-C Power

The Loss of A-C Power event is not analyzed for reload cores. The analysis and results presented are for the initial core.

15.2.6.1 Identification of Causes and Frequency Classification

This initiating event is caused by either a loss of both the unit auxiliary and the system auxiliary power transformers or a loss of all grid connections.

a. Causes for interruption or loss of the system auxiliary power transformer can arise from operation or misoperation of the transformer protective circuity due to internal or external electrical faults or from operator error such as tripping the transformer high voltage breakers during normal operation. Causes for interruption or loss of the unit auxiliary power transformer can arise from operation or misoperation of the unit connected main generator/transformers protective circuitry due to internal or external electrical faults, a trip of the unit connected main generator or from operator error such as tripping the main generator output high voltage breakers during normal operation. Loss of both auxiliary power transformers is categorized as an incident of moderate frequency.

b. Loss of all grid connections can result from tornados, ice storms, aircraft accidents, etc., which may, on a massive scale, physically disconnect the plant from the main grid. This incident is classified as infrequent.

15.2.6.2 Sequence of Events and Systems Operation

a. The reactor is subjected to a complex sequence of events when the plant loses all auxiliary power. For the assumed loss of both auxiliary transformers (Figure 15.2-8), the sequence of events is as follows:

Time (sec)	Event
0	Loss of both auxiliary power transformers occurs.
	Recirculation system pump motors are tripped.
	Electric feedwater and condensate and condensate booster pumps are tripped.
	Condenser circulating water pumps are tripped.
10	Low Condenser vacuum initiates a turbine trip.

Time (sec)	Event
	Turbine steam bypass valves operate on turbine trip.
10.1	Reactor scram initiated when turbine stop valves reach 90% open position (switch).
	Turbine stop valves closed and turbine steam bypass valves start to open to regulate pressure.
22.7	Reactor vessel low level trip initiates HPCS and RCIC (not simulated).
22.8	Closure of MSIV's is initiated by low main steamline pressure signal.
25.9	Turbine steam bypass valves closed.
50.1	Group 1 relief valves cycle automatically to relieve pressure.

The operator may maintain reactor water level by use of the RCIC system if the (steam) turbine driven reactor feed pumps should fail. Reactor pressure can be controlled by use of the relief valves. Normal monitoring, such as scram verification, diesel generator start, turbine coastdown vital load carrying by the diesel generators, and normal cooldown, are typical post-event operator actions.

b. For the loss of all grid connections (Figure 15.2-9), a different sequence of events ensues as follows:

Time (sec)	Event
Approx – 0.015	Loss of grid causes turbine-generator to detect a loss of electrical load.
0	Turbine trip initiated by loss of generator load.

Time (sec)	Event
	Turbine-generator PLU trip initiates main turbine control valve fast closure.
	Recirculation system pump motors trip off.
	Circulating water pump trip.
	Condensate and condensate booster pump trip.
	Turbine stop valve closure initiates reactor scram.
	Electric feedwater pump motor is tripped.
0.01	Turbine control valves closed.
0.10	Turbine steam bypass valves open to regulate pressure.
1.61, 1.76, 1.92, 2.12 and 2.56	Relief valves actuated sequentially by Groups 1, 2, 3, 4 and 5.
Est. 5.1, 5.4, 5.8, 6.0 and 6.9	Relief valves reclose sequentially by Groups 5, 4, 3, 2 and 1.
30	Loss of condenser vacuum initiates MSIV closure and turbine steam bypass valve(s) closure.
32.4	Reactor vessel low level 2 trip initiates HPCS and RCIC (not simulated)
50 plus	Group 1 relief valves automatically cycle to regulate pressure.

System operation in both events a and b (above) is normal, i.e., instruments and controls, plant protection circuits, and the reactor protection systems function normally. Operation of the HPCS and RCIC systems are not included in this simulated analysis. Their operation would normally occur at sometime beyond the primary concerns of fuel thermal margin and overpressure effects of this analysis.

In general, the loss of both auxiliary power transformers leads automatic transfer of associated loads to the plant generator (unit aux transformers). Should auto transfer fail, there would be a reduction in power level due to the tripping of the recirculation pumps and due to pressurization effects when the turbine trip follows the reactor scram. Additional failures of systems assumed to function to protect the reactor would not necessarily imply failures of the reactor protection system itself, but it is built to satisfy single failure criteria so no change is expected in the analyzed consequences.

15.2.6.3 Core and System Performance

Mathematical Model

The computer model described in Subsection 15.1.1.3 was used to simulate this event. Operation of the RCIC or HPCS systems was not included in this event, since their startup doesn't provide flow to the reactor in the time period of importance to this simulation.

Input parameters and plant conditions are those given in Table 15.0-1, unless specifically stated at some other value.

Results

a. For the loss of both auxiliary power transformers, Figure 15.2-8 shows graphically the simulated transient. The initial portions of the transient is similar to the loss-of-feedwater transient, however; the LSCS feedwater system is driven by steam for the normal code, hence the simulation is an extreme case for LSCS. Between 10 and 30 seconds a turbine trip, reactor scram, MSIV closure and turbine steam bypass valve closure all occur. There is no significant increase in fuel temperature.

b. For the loss of all grid connections the transient takes the characteristic of the response to a load rejection as discussed in Subsection 15.2.2A. In this transient, the peak neutron flux reaches 150.4% NBR power while the fuel surface heat flux peaks at 101.8% of its initial value. The peak fuel center line temperature rises only 65° F. Peak pressure in the reactor vessel has a maximum valve of 1135 psig.

Consideration of Uncertainties

The most conservative characteristics of the protection features are assumed. Input power level was 105% NBR power for an EOC core condition. Any actual deviations in plant performance are expected to make the results of this event even less severe.

The reactor pressure increase following MSIV closure is expected to automatically actuate the safety/relief valve when their set- points are reached. Cyclic operation of these valves discharges decay heat to the suppression pool, which in turn can be cooled as necessary via the RHR cooling mode; system decay heat is the only driving force for pressure increase.

15.2.6.4 Barrier Performance

The consequences of this event do not result in any significant temperature or pressure transient in excess of the criteria for which the fuel clad, pressure vessel, or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.2.6.5 Radiological Consequences

While the consequence of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under defined meteorological and controlled release conditions. If purging of the containment is chosen, the release will be made in accordance with established technical specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.2.7 Loss of Feedwater Flow

The loss of Feedwater Flow event is not analyzed for reload cores. The analysis and results presented are for the initial core.

GE addressed this event for the Power Uprate Project by use of a generic analysis which is described in greater detail in section 15.B.3.1. In summary, power uprate does not change the ability of RCIC or HPCS to automatically maintain water level above TAF. However, the extra decay heat from power uprate requires more time to restore water level.

An additional steam flow induced process measurement error in the Level 3 scram was accounted for in the Loss of Normal Feedwater event. A lowering in the Level 3 Analytical Limit (AL) setpoint was calculated in Reference 18. It was determined that adequate margin exists to top of active fuel (TAF) uncovery while considering a bounding process measurement error applicable for this event. This error can be absorbed by lowering the L3 AL by an equivalent amount. The consequences of lowering the AL is a bounding reduction in minimum water level in the upper plenum of 12 Inches and a bounding reduction in minimum water level of 12 inches in the vessel downcomer region as discussed in Reference 18. The analysis also shows significant margin to the TAF. The impact of the change is not significant, and no event descriptions or conclusions in the UFSAR need to be modified.

15.2.7.1 Identification of Causes and Frequency Classification

A loss of feedwater flow can occur from pump failures, interruption of driving steam flow, feedwater controller failures, operator gross error, or erroneous input from the high vessel water level (L8) trip signal. This transient disturbance is categorized as an incident of moderate frequency.

15.2.7.2 Sequence of Events and System Operation

The sequence of events for the loss of feedwater flow incident (Figure 15.2-10) is as follows:

Time (sec)	Event
0	Trip initiated on feedwater pumps.
3.49	Recirculation FCV runback initiated by narrow range level L4 and less than two feedwater pumps running.
4.48	Feedwater flow decays to zero.
7.78	Vessel water level (L3) trip initiates reactor scram.
16.99	Vessel water level (L2) trip initiates recirculation pump trip.
	Vessel water level (L2) initiates MSIV isolation (see Subsection 15.0.7).
	Vessel water level (L2) trip initiates HPCS and RCIC system operation (not simulated).
19.99	MSIVs fully closed (see Subsection 15.0.7).
30.38	Group 1 relief valves open automatically.
Est. 35.9	Group 1 safety/relief valves close automatically.
47.46 plus	Group 1 relief valves cycle as needed to relieve pressure at a decreasing rate as decay heat diminishes.

Loss of feedwater flow results in a proportional reduction of reactor vessel inventory thus causing the vessel water level to drop. The first corrective action is a reactor scram via the low level (L3) trip actuation. The reactor protection system responds within 1 second to scram the reactor. The low level (L3) scram trip is capable to meet the single failure criterion. Containment isolation via the MSIVs may also provides a followup scram initiation, however, the reactor is already scrammed and shut down by this time.

Credit is taken for the pressure relief valves to relieve steam pressure created by core decay heat because the turbine steam bypass valves are on the downstream side of the MSIVs which effect the isolation.

Key corrective functions for this event are automatic and designed to satisfy a single failure criterion; therefore, any additional failure in these shutdown methods would not aggravate or change the simulated transient.

The operator performs a monitoring function to verify the following automatic actions: All control rods inserted after the scram, MSIV closure (see Subsection 15.0.7), HPCS and RCIC initiation, self-activation of relief valves on reactor high pressure, recirculation pump trip, and normal turbine coast down. The operator may ensure water level as needed with HPCS and RCIC.

Should a single relief valve fail to close after opening, the reactor will completely depressurize. This situation is discussed in Subsection 15.6.1. In that case the HPCS is capable of maintaining adequate core coverage until such time as the low pressure systems can provide long-term water inventory control.

The timing values following Low water level scram based on the L3 Analytical Limits, are slightly different. However, as described in Reference 18, the impact of the change is not significant.

15.2.7.3 Core and System Performance

The transient computer model described in Subsection 15.1.1.3 was used to simulate this event. Unless otherwise noted, the plant conditions tabulated in Table 15.0-1 were utilized.

The results of this transient simulation are shown in Figure 15.2-10. Feedwater flow terminates at 4.6 seconds, core subcooling decreases thus causing a reduction in power level and pressure. Water level drops until a reactor scram is initiated and continues through the initiation of HPCS (and RCIC); the recirculation system is also tripped. MCPR remains considerably above the safety limit largely because heat flux increases are not experienced.

Peak pressure at the bottom of the vessel reaches 1105 psig; vessel dome pressure does not exceed 1094 psig.

This transient is most severe from high power conditions, because the rate of water level decrease is greatest and the quantity of stored and decay heat to be dissipated are highest for that power conditions.

Operation of the HPCS or the RCIC is not included in the simulation of the first 50 seconds of this transient. Later start up of these pumps has no significance with respect to thermal response of the fuel but relate only to water inventory control after system isolation.

15.2.7.4 Barrier Performance

Peak pressure in the bottom of the vessel reaches 1105 psig, which is below the ASME code limit of 1375 psig for the reactor vessel coolant pressure boundary, vessel dome pressure does not exceed 1094 psig. The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.2.7.5 Radiological Consequences

While the consequence of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under defined meteorological and controlled release condition. If purging of the containment is chosen the release will be in accordance with established technical specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.2.8 Feedwater Line Break

(Refer to Subsection 15.6.6.)

15.2.9 Failure of RHR Shutdown Cooling

The Failure of RHR Shutdown Cooling event is not analyzed for reload cores.

Normally, in evaluating component failure considerations associated with the RHR shutdown cooling mode operation, active pumps or instrumentation (all of which are redundant for safety system portions of the RHR aspects) would be assumed to be the likely errant equipment. For purposes of worst case analysis, the single re-circulation loop suction valve to the redundant RHR loops is assumed to fail. This failure would, of course, still leave two complete RHR loops for LPCI, pool, and containment cooling minus the normal RHR shutdown cooling loop connection. Although the errant valve could be manually manipulated open, it is assumed failed indefinitely. If it is now assumed that the single active failure criterion is applied, the plant operator has one complete RHR loop available with the further selective worst case assumption that the other RHR loop is lost.

Recent analytical evaluations of this event have required additional worst case assumptions. These included:

a. loss of all offsite a-c power,

b. utilization of safety shutdown equipment only, and

c. operator involvement only 10 minutes after coincident assumptions.

These accident-type assumptions certainly would change the initial incident (malfunction of RHR suction valve) from a moderate frequency incident to a classification in the design-basis accident status. However, the event is evaluated as a moderate frequency event with its subsequent limits.

15.2.9.1 Identification of Causes and Frequency Classification

The plant is operating at 3559 MW_t (102% rated power) when a long-term loss of offsite power occurs, causing multiple safety-relief valve actuation (see Subsection 15.2.6) and subsequent heat-up of the suppression pool. Reactor vessel depressurization is initiated to bring the reactor pressure to approximately 100 psig. Concurrent with the loss of offsite power, an additional (divisional) single failure occurs which prevents the operator from establishing the normal shutdown cooling path through the RHR shutdown cooling lines. The operator then establishes a shutdown cooling path for the vessel through the ADS valves.

This event is evaluated as a moderate frequency event. However, for the following reasons it could be considered an infrequent incident:

a. No RHR valves have failed in the shutdown cooling mode in BWR total operating experience.

b. The set of conditions evaluated is for multiple failure as described above and is only postulated (not expected) to occur.

15.2.9.2 Sequence of Events and System Operation

The sequence of events for this incident is as follows:

Approximate Elapsed Time	Event
0	Reactor is operating at 102% rated power when LOSP transient occurs thus initiating plant shutdown.
	Concurrently failure of Division power occurs (i.e., loss of one diesel generator).
10 min	Suppression pool cooling initiated to prevent overheating from SRV actuation.
	Controlled blowdown initiated by operator.
2 to 3 hr	Blowdown to 100 psi completed.
	Attempts to manually open the RHR shutdown cooling suction valve fail.
2 to 3 hrs + 30 min	Operator actuates ADS and completes blowdown to the suppression pool.
	RHR pump discharge from pool is redirected to vessel via LPCI line. Alternate cooling path is now established for shutdown cooling.

Plant instrumentation and control is assumed to be functioning normally except as noted. In this evaluation credit is taken for the plant and reactor protection systems and/or the ESF utilized.

For the early part of the transient, the operator takes identical actions to those described in Subsection 15.2.6; then, the reestablishment of reactor cooling by one or more of the following ways becomes the principal operator activity:

a. Maintain reactor vessel inventory via HPCS, feedwater, or RCIC systems;

b. At about 10 minutes into the transient, initiate suppression pool cooling (note that this analysis of a "worst case" situation assumes that only one RHR heat exchanger is available);

c. Initiate depressurization of the vessel by manual actuation of relief valves;

d. When reactor vessel pressure reaches approximately 100 psi, attempt to open one of the two RHR shutdown valves (assumed unsuccessful, in this analysis);

e. Complete the vessel blowdown via ADS and establish a shutdown cooling path as described in the notes to Figure 15.2-12.

Time required to initiate the necessary steps to maintain reactor pressure and level control is approximately 10 minutes. This is an estimate of how long it would take the operator to initiate the necessary actions; it is not a time by which he must initiate action.

The worst case single failure (loss of Division power) is included in the analysis of this incident. No single failure nor operator error can make the consequences of this event any worse.

15.2.9.3 Core and System Performance

An event that can directly cause reactor vessel water temperature increase is one in which the energy removal rate is less than the decay heat rate. The applicable event is loss of RHR shutdown cooling. This event can occur only during the low pressure portion of a normal reactor shutdown and cooldown when the RHR system is operating in the shutdown cooling mode. The earliest time the shutdown system can be activated is 2 to 3 hours after initiation of shutdown. During this time, MCPR remains high and nucleate boiling heat transfer is not exceeded at any time. Therefore, the core thermal safety margin remains essentially unchanged. Only qualitative results are included below because this transient core behavior has been shown in Subsection 15.2.6.

Table 15.2-4 shows the input parameters and initial conditions used in evaluation of this event.

Results

For most single failures that could result in loss of shutdown cooling, no unique safety actions are required. In these cases, shutdown cooling is simply reestablished using other, normal shutdown cooling equipment. In cases where

both of the RHR shutdown cooling suction valves cannot be opened, alternate paths are available to accomplish the shutdown cooling function (Figure 15.2-11). An evaluation has been performed assuming the worst single failure that could disable the RHR shutdown cooling valves.

This evaluation demonstrates the capability to safely transfer fission product decay heat and other residual heat from the reactor core at the rate such that specified acceptable fuel design limits and the design conditions of the reactor cooling pressure boundary are not exceeded. The evaluation assures that, for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available), the safety function can be accomplished, assuming a worst case single failure.

The alternate cooldown path chosen to accomplish the shutdown cooling function utilizes the RHR and ADS or normal relief valve systems (see Reference 3 and Figure 15.2-12).

The alternate shutdown systems are capable of performing the function of transferring heat from the reactor to the environment using only safety grade systems. The systems are capable of bringing the reactor to a cold shutdown in less than 8 hours after the transient occurs. Even if it is additionally postulated that all of the ADS or relief valves discharge piping also fails, the shutdown cooling function would eventually be accomplished as the cooling water would run directly out of the ADS or safety/relief valves, flooding into the drywell, and then draining through the downcomers into the suppression pool.

The systems have suitable redundancy in components such that, for onsite electrical power operation (assuming offsite power is not available) and for offsite electrical power operation (assuming onsite power is also not available), the system's safety function can be accomplished assuming, even further additional single failures. The systems can be fully operated from the main control room.

The design evaluation is divided into two parts:

a. full power operation to approximately 100 psig vessel pressure, and

b. approximately 100 psig vessel pressure to cold shutdown (14.7 psia) conditions.

<u>Full Power to Approximately 100 psig</u>

Independent of the event that initiated plant shutdown (whether it be a normal plant shutdown or a forced plant shutdown), the reactor is normally brought to approximately 100 psig using either the main condenser or, in the case where the

main condenser is unavailable, the RCIC/HPCS systems, together with the nuclear boiler pressure relief valves.

For evaluation purposes, however, it is assumed that the plant shutdown is initiated by a transient event (loss of offsite power), which results in relief valve actuation and subsequent suppression pool heatup. For this postulated condition, the reactor is shut down and the reactor vessel pressure and temperature are reduced to and maintained at saturated conditions at approximately 100 psig. The reactor vessel is depressurized by manually opening selected relief valves. Reactor vessel makeup water is automatically provided via the RCIC/HPCS systems. While in this condition, the RHR system (suppression pool cooling mode) is used to maintain the suppression pool temperature within shutdown limits.

These systems are designed to routinely perform these functions for both normal and forced plant shutdown. Since the RCIC, HPCS, and RHR systems are divisionally separated, no single failure, together with the loss of offsite power, is capable of preventing pressure reduction to the 100 psig level.

Approximately 100 psig to Cold Shutdown

The following assumptions are used for the evaluation of the procedures for attaining cold shutdown from a pressure of approximately 100 psig:

a. the vessel is at ~100 psig and saturated conditions;

b. a worst case single failure is assumed to have occurred (i.e., loss of a division of emergency power); and

c. there is no offsite power available.

In the event that the RHR's shutdown suction line is not available because of single failure, the first action is to maintain the 100 psig level while personnel gain access and attempt to effect repairs. For example, if a single electrical failure caused the suction valve to fail in the closed position, a handwheel is provided on the valve to allow manual operation. Nevertheless, if for some reason the normal shutdown cooling suction line cannot be repaired, the capabilities described below can satisfy the normal shutdown cooling requirements and thus fully comply with General Design Criterion 34.

The RHR shutdown cooling line valves are in two divisions (Division 1 serves the outboard valve, and Division 2 serves the inboard valve) to satisfy containment isolation criteria. For evaluation purposes, the worst case failure is assumed to be the loss of a division of emergency power because this also prevents actuation of one shutdown cooling line valve. Engineered safety feature equipment available for accomplishing the shutdown cooling function includes (for the selected path):

a. ADS (DC Division 1 and DC Division 2),

b. RHR Loop A (Division 1),

c. HPCS (Division 3),

d. RCIC (DC Division 1), and

e. LPCS (Division 1).

Because availability or failure of Division 3 equipment does not affect the normal shutdown mode, normal shutdown cooling is easily available through equipment powered from only Divisions 1 and 2. It should be noted that, conversely, the HPCS system is always available for cooling injections if either of the other two divisions fails. For failure of Divisions 1 or 2, the remaining systems would be functional:

a. Division 1 Fails, Division 2 and 3 Functional:

Failed Systems	Functional Systems
RHR Loop (A)	HPCS
LPCS	ADS
	RHR Loops B and C
	RCIC

Assuming the single failure is a failure of Division 1 emergency power, the safety function is accomplished by establishing one of the cooling loops described in Activity C1 of Figure 15.2-12.

b. Division 2 Fails, Divisions 1 and 3 Functional:

Failed Systems	Functional Systems
RHR Loops B and C	HPCS
	ADS
	RHR Loop A
	RCIC
	LPCS

Assuming the single failure is the failure of Division 2, the safety function is accomplished by establishing one of the cooling loops described in Activity C2 of Figure 15.2-12. Figures 15.2-13 through 15.2-16 show RHR loops A, B and/or C (simplified).

Impact of RHR Shutdown Cooling Failure on Suppression Pool Temperature

The failure of RHR shutdown cooling was analyzed at 3559 MWt in Reference 11 by evaluating three possible alternate shutdown cooling paths (Activities). As illustrated in Figure 15.2-12, for Division I failure, RHR loops B and C are both available. If loop B alone is utilized, water from the suppression pool is pumped through the RHR heat exchanger and then to the reactor vessel. This path is identified as Activity C1, item b1 (i.e., C1-b1) in Figure 15.2-12. If both loops B and C are utilized, water from the suppression pool is pumped directly to the vessel with loop C, while loop B is used to cool the suppression water in the pool cooling mode. This path is identified as C1-b2 in Figure 15.2-12. For Division II failure, only RHR A is available and the cooling path is similar to that for Activity C1-b1, except that LPCS is also available for this activity. The results of the evaluation at 3559 MWt in Reference 11 show that:

Alternate Shutdown Activity	Peak Suppression Pool Temperature (°F)	Time to Cold Shutdown (hours)
C1-b1	210.4	8.4
C1-b2	205.4	18.4
C2	209.0	13.4

As shown above, for all three Activities, the peak pool temperature is below the design limit of 212° F, and the time to cold shutdown is smaller than 36 hours.

15.2.9.4 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed. Release of coolant to the containment occurs via SRV actuation.

15.2.9.5 Radiological Consequences

While the consequence of this event does not result in fuel failure, it does result in the discharge of normal cooling activity to the suppression pool via SRV operation. Because this activity is contained in the primary containment, there will be no exposures to operating personnel. Because this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under defined meteorological and controlled release conditions. If purging of the containment is chosen, the release is made in accordance with established technical specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.2.10 Loss of Stator Cooling

To protect the generator from overheating from the consequences of a loss of stator cooling, protection logic initiates an automatic runback of the turbine. The Loss of Stator Cooling (LOSC) event is characterized by a very slow turbine control valve (TCV) closure. The slow valve closure does not scram the reactor directly. Subsequently, the steam flow capacity of the turbine is reduced. If the reactor power exceeds the turbine plus bypass capacity, an increase in reactor pressure will occur. Without sufficient turbine plus bypass capacity, the event is terminated by an automatic reactor protection system (PRS) actuation on high reactor pressure or high neutron flux (high APRM scram). The LOSC event was analyzed (Reference 6) and found to be non-limiting. The LOSC event is validated as non-limiting for each reload.

5.2.10.1 Identification of Causes and Frequency Classification

A loss of Stator Cooling event could be initiated as a result of any of the following: low Stator Inlet Flow, low Stator Inlet Pressure, or high Stator Outlet Temperature.

This event is classified as a moderate frequency event.

15.2.10.2 Sequence of Events and System Operation

The loss of stator cooling event is analyzed with the following sequence of events:

A. After receiving an initiating signal for LOSC, Electro-Hydraulic Control (EHC) has a 15 second time delay before starting the Turbine-Generator load runback. The Turbine-Generator load runback starts at 105% steam flow.

B. Once the Turbine-Generator load runback starts the EHC system will start to close the Turbine Control Valves (TCV) at a rate of 30% Rated Steam Flow per minute.

C. Turbine Bypass Valves (TBV) will start opening once the TCV start to close in order to control reactor pressure. Once the TBV are full open, the reactor will begin to pressurize.

D. The Turbine-Generator load runback stops at 20% of Rated Steam Flow to the Turbine.

E. The Event is terminated by a reactor scram on either reactor pressure or neutron flux.

15.2.10.3 Core and System Performance

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

15.2.10.4 Barrier Performance

The consequences of this event do not result in any significant temperature or pressure transient in excess of the criteria for which the fuel clad, pressure vessel, or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.2.10.5 Radiological Consequences

Because this event does not result in any fuel failures nor any release of primary coolant to either secondary containment or to the environment, there are no radiological consequences associated with this event.

15.2.11 References

1. F. Odar et al., "Safety Evaluation for the General Electric Topical Report: Qualification of the One-Dimension Core Transient Model for Boiling Water Reactors," NEDO-24154 and NEDE-24154-P, Volumes I, II and III, 1980

2. Deleted

3. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, (Unit 1: Rev. 22, Unit 2: Rev. 20).

4. R. Linford, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802, April 1973.

5. GE Letter LS-2209, H.R. Peffer (GE) to G.R. Crane (ComEd), Subject: LaSalle County Station Unit 1 & 2 Technical Specifications MSIV Closure Scram, Dated March 15, 1982.

6. GEH Report, “Evaluation of Loss of Stator Water Cooling at LaSalle County Station,” 0000-0154-3327-R2, January 8, 2014.

7. Deleted

8. Deleted

9. Boiling Water Reactor Licensing Methodology Summary, EMF-94-217(P), Rev. 1, Siemens Power Corporation, Nuclear Division.

10. “Backup Pressure Regulator”, SIL 614, Revision 1, March 15, 1999.

11. Power Uprate Project Task 400, "Containment System Response," GE-NE-A1300384-02-01, Revision 3, May 2002.

12. Power Uprate Project Task 900, "Transient Analyses," GE-NE-A1300384-08 Revision 1, September 1999.

13. NDIT LAS-ENDIT-H041, Power Uprate Project S&L Task 21b, "Radiological/Chapter 15 Accidents."

14. Letter, P.D. Doverspike (GE) to D.A. Spencer (ComEd), GE-LPUP-333, "Transmittal of Revised Table 15.2-4a for LaSalle UFSAR Markup," DRF A13-00384-00, dated June 14, 2000.

15. Design Analysis No. L-002874, Rev. 0, “LaSalle County Station Power Uprate Project Task 400: Containment System Response (GE-NG-A1300384-02-01 R3) Revision 3”.

16. Deleted

17. GE Report GE-NE-0000-0036-5084-RO, Rev. 0, "LaSalle 1 and 2 Offrated Analysis Below the PLU Power Lever Power", dated February 2005.

18. “BWR Owners Group Evaluation of Steam Flow Induced Error (SFIE) Impact on the L3 Setpoint Analytic Limit” GEH-NE-0000-0077-4603-R1, October 2008.

19. Deleted

20. "Migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients," NEDE-32906P Supplement 3-A, Revision 1, April 2010.

21. "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," NEDE-32906P-A, Revision 3, September 2006.

TABLE 15.2-1

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TABLE 15.2-2

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TABLE 15.2-3

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TABLE 15.2-4

INPUT PARAMETERS FOR EVALUATION OF FAILURE OF RHR SHUTDOWN COOLING

Rated Power	3489 MWt
Initial power corresponding to	105% rated power
Suppression pool mass (lb_m)	8.52×10^6
RHR (KHX value) (Btu/sec/°F)	438***
Initial vessel condition:	
Pressure (psia)	1055
Temperature (°F)	550.7
Initial primary fluid inventory (lb_m)	7.016×10^5
Initial pool temperature, (°F)	100**
Service water temperature (CSCS), (°F)	107***
Vessel heat capacity (Btu/lb_m/°F)	0.123
HPCS on – off water level (ft) on	40.8
off	47
HPCS flow rate, lb_m/sec	868
LPCI flow rate, lb_m/sec	982

** Supplementary evaluations have been performed, as discussed in Section 6.2.1.8 to verify that an increase in the initial suppression pool temperature (from 100°F to 105°F) would not significantly impact this accident scenario.

*** See Reference 36 noted in Section 6.2

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

The events listed as a decrease in reactor coolant system flow rate include recirculation pump trip, recirculation flow control failure with decreasing flow, recirculation pump seizure, and recirculation pump shaft break. The 1999 power uprate project did not include re-evaluating these events. The bases for not re-evaluating these events is discussed in section 15.B.2.2 and Reference 4.

Events described in this section that result in reduced core flow rates may also result in a core thermal hydraulic instability transient. Refer to Section 15.10 for an overview of this event.

15.3.1 Recirculation Pump Trip

The Recirculation Pump Trip event is not analyzed for reload cores. The analysis and results presented are for the initial core.

The original analysis of dual RR pump trip predicted that level swell would result in a reactor water level 8 turbine trip and a subsequent scram would terminate the event. Actual plant operating experience has shown that level may not reach level 8 and continued operation is possible. Any MCPR concerns do not occur unless thermal hydraulic instability occurs. This is addressed by a manual scram requirement or an operable oscillation power range monitor system.

15.3.1.1 Identification of Causes and Frequency Classification

A re-circulation pump trip (RPT) can be caused by normal operations where hazards are reduced and shutdown enhanced via RPT action. It occurs normally from: reactor vessel level 2 setpoint trip and ATWS high pressure setpoint trip, and re-circulation motor overload protection or overcurrent protection, and from turbine trip or load rejection at power levels greater than or equal to 25% of rated core thermal power.

Random operational failures can also cause re-circulation pump tripping; loss of electrical power to the pumps, operator error and equipment failures.

The transient disturbances from loss of one re-circulation pump or from the loss of both re-circulation pumps are each categorized as events of moderate frequency.

15.3.1.2 Sequence of Events and Systems Operation

a. The sequence of events for a single recirculation pump trip (Figure 15.3-1) is as follows:

Time (sec)	Event
0	Trip of one recirculation pump.
15.0	Jet pump diffuser flow reverses in tripped loop.
40.0	Core flow and power level stabilize at new equilibrium conditions.

b. The sequence of events for trip of both recirculation pumps (Figure 15.3-2) is as follows:

Time (sec)	Event
0	Both recirculation pumps trip.
7.30	Vessel water level (L8) trip initiates main turbine trip.
	Feedwater turbine(s) are tripped off.
	Main turbine trip initiates steam bypass valves.
7.31	Main turbine stop valves reach 90% open position and initiate reactor scram trip.
7.43	Turbine stop valves close completely and turbine steam bypass valves start to open to regulate pressure.
12.14	Group 1 relief valves actuate.
17.4 (est)	Group 1 relief valves close.
34.4 (est)	Turbine steam bypass valves closed.

Time (sec)	Event
41.54	Vessel low level trip (L2) initiates MSIV closure (see Subsection 15.0.7).
	Vessel low level trip (L2) initiates HPCS and RCIC operation (not simulated).
42.35	Turbine steam bypass valves reopen on pressure increase at turbine inlet.

System Operation

a. Tripping a single recirculation pump requires no protection system nor safeguard system operation. Normal functioning of plant instrumentation and controls is assumed in this evaluation. Because no scram occurs, no immediate operator action is required. Normal coastdown occurs and normal power decrease occurs without system effects. The operator verifies that no operating limits are being exceeded, and if necessary, reduces the flow of the remaining operating pump to conform to the single pump flow criteria. The operator must also determine the cause of the trip prior to returning the system to normal via station procedures. No additional effects of single failures need be discussed for this event.

b. Simultaneous, two loop recirculation pump coastdown can occur only from the simultaneous loss of power to both pump drive motors. As in all core flow decrease events, the reduction in core flow results in a greater void fraction which reduces power. Analysis of this event assumes normal functioning of plant instrumentation and controls, and plant protection systems such as RPS. In this transient, credit is taken for vessel level (L8) instrumentation to trip the turbine. Scram trips from the turbine stop valves shutdown the reactor. High steam pressure is limited by the pressure relief valves.

The operator ascertains that the reactor scrams with the turbine trip as a result of reactor level swell. Control of reactor water level is accomplished via automatic feedwater control system setback and HPCS and/or RCIC operation. Monitoring of water level and vessel pressure during normal reactor shutdown are routine operator responsibilities.

The L8 level trip is the first response to initiate corrective action in this transient event. Multiple level sensors are used to detect water level at L8 so that moisture does not carry over to the main turbine. A single failure at this point will neither initiate nor preclude a turbine trip signal, but because the turbine trip circuitry is not built to single failure criterion, any failure at this point would simply delay the pressurization "signature" in time. High moisture levels entering the turbine can cause vibration which also trips the turbine. The scram signals from the turbine to the reactor are designed such that a single failure will neither initiate nor impede the initiation of a reactor scram trip.

15.3.1.3 Core and System Performance

Mathematical Model

The nonlinear, dynamic model described in Reference 1 is used to simulate this event. Minimum specified rotating inertias are assumed. These analyses were performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-1. Prior to pump trip, steady state is assumed at 105% of NB rated power, and 100% recirculation flow, with thermally limited conditions.

Results

The initial conservative conditions chosen for these analyses force the analytical results to be more severe than expected under actual plant conditions. Actual rotating equipment inertia is also expected to be somewhat larger than the minimum design value assumed in these simulations. Any deviations from the assumed conditions are expected to make the results of this event less severe.

a. Figure 15.3-1 shows the results of losing one recirculation pump. The tripped loop diffuser flow reverses in approximately 15 seconds. However, the ratio of diffuser mass-flow to pump mass-flow in the active jet pumps increases considerably to produce approximately 130% of normal diffuser flow (and 50% of normal core rated flow). The ΔCPR calculated for this transient is ~ 0.0.

   The fuel thermal limits are not violated. During this transient, level swell is not sufficient to cause turbine trip from L8, hence there is no reactor scram. System pressures reduce from the initial conditions, and the reactor coolant pressure boundary is not affected.

b. Figure 15.3-2 shows graphically this transient with minimum specified rotating inertia. The ΔCPR calculated for this transient is about ~0.0. No scram is initiated directly by pump trip. The vessel water level swell due to rapid flow coastdown is expected to reach the high level trip thereby shutting down the reactor. The peak pressure of 1075 psig is reached after the turbine trip, but this is well below the design allowable vessel pressure. Subsequent relief valve operation keeps the vessel dome pressure below 1095 psig.

Subsequent events, such as the main steamline isolation and initiation of RCIC and HPCS systems, occurring late in the transient, have no significant effect on the results.

15.3.1.4 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed. Release of coolant to the containment occurs via SRV actuation.

15.3.1.5 Radiological Consequences

While the consequence of this event does not result in fuel failure, it does result in the discharge of normal cooling activity to the suppression pool via SRV operation. Because this activity is contained in the primary containment, there will be no exposures to operating personnel. Because this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge to the environment under defined meteorological and controlled release conditions. If purging of the containment is chosen, the release is made in accordance with established technical specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.3.2 Recirculation Flow Control Failure-Decreasing Flow

The Recirculation Flow Control Failure - Decreasing Flow event is not analyzed for reload cores. The analysis and results presented are for the initial core.

15.3.2.1 Identification of Causes and Frequency Classification

Master flow controller malfunctions can cause a decrease in core coolant flow. A downscale failure of either the master power controller or the flux controller will generate a zero flow demand signal to both recirculation flow controllers. Each individual valve actuator has a velocity limiter which limits the maximum valve stroking rate to 11% per second. A postulated failure of the input demand signal, which is utilized in both loops, can decrease flow at the maximum valve stroking rate established by the loop limiter.

Failure within either loop controller or valve positioner can result in a maximum valve stroking rate as limited by the capability of the valve hydraulics.

A recirculation flow control failure is categorized as an incident of moderate frequency.

15.3.2.2 Sequence of Events and Systems Operations

a. The sequence of events for the fast closure of one main recirculation flow control valve (Figure 15.3-3) is as follows:

Time (sec)	Event
0	Initiate fast closure of one main recirculation flow control valve.
2.65	Jet pump diffuser flow reverses in the affected loop.
3.38	Recirculation flow control valve at minimum flow position.
7.21	High vessel water level trip (L8) initiates main turbine and feedwater turbine trip(s). Main turbine trip initiates steam bypass valve operation.
7.22	Main turbine stop valves reach 90% open position and initiate reactor scram trip and recirculation pump downshift (EOC RPT).
7.32	Main turbine stop valves closed and steam bypass valves start to open to

Time (sec)	Event
	regulate pressure.
7.35	Recirculation pump motor circuit breaker opens to cause decrease in core flow.
11.99	Group 1 pressure relief valves actuate.
17.4 (est)	Group 1 pressure relief valves close.
33.5 (est)	Turbine steam bypass valves start to close.
34.4 (est)	Turbine steam bypass valves closed.
41.89	Vessel low level trip (L2) initiates MSIV closure (see Subsection 15.0.7).
	Vessel low level trip (L2) initiates HPCS and RCIC systems and RR pump trip (not simulated).
43.08	Turbine steam bypass valves reopen on pressure increase at turbine inlet.
44.7 (est)	Turbine steam bypass valves reclose.

b. The sequence of events for the fast closure of both main recirculation flow control valves (Figure 15.3-4) is as follows:

Time (sec)	Event
0	Initiate fast closure of both main recirculation flow control valves.
6.43	High vessel water level (L8) trip initiates main turbine trip.
	High vessel water level (L8) trip initiates feedwater turbine trip(s).

Time (sec)	Event
6.44	Main turbine stop valves reach 90% open position and initiate reactor scram trip and recirculation pump downshift (EOC RPT).
6.53	Turbine stop valves closed and turbine steam bypass valves start to open to regulate pressure.
6.57	Recirculation pump motor circuit breakers open to cause decrease in core flow to natural circulation state.
7.0 (est)	Both recirculation flow control valves are at minimum flow position.
9.41	Relief valves actuate by Groups 1 and 2.
(est)13.8,14.7	Relief valves close by Groups 2 and 1.
31.0 (est)	Turbine steam bypass valves start to close.
32.3 (est)	Turbine steam bypass valves closed.
40.08	Vessel low level trip (L2) initiates MSIV closure (see Subsection 15.0.7).
	Vessel low level trip (L2) initiates HPCS and RCIC systems and RR pump trip (not simulated).
40.74	Turbine steam bypass valves reopen on pressure increase at turbine outlet.

The operator should ascertain that the reactor scrammed from a turbine trip resulting from reactor water swell. The operator should verify that no operating limits have been exceeded, then regain control of vessel water level through feedwater system, HPCS, or RCIC operation. Monitoring of level and pressure control should continue after reactor shutdown. Also, whether a single FCV closure

or a double FCV closure, the cause of the transient should be determined prior to returning the system to normal.

The scram for either case originates from the high water level trip (L8) with credit taken from normal operation of plant instruments and controls. Any single failures or operator errors could lead no further than to the dual recirculation loop close down.

15.3.2.3 Core and System Performance

The nonlinear dynamic model described briefly in Subsection 15.1.1.3 was used to simulate these transient events. The less negative void coefficient of Table 15.0-1 was used for these analyses.

a. Failure of either loop's controller or valve positioner can result in a maximum stroking rate of 30% per second as limited by the FCV hydraulics.

b. A down scale failure of either the master power controller or the flux controller will generate a zero demand signal to both recirculation flow controllers. Each individual FCV actuator circuit has a velocity limiter which restricts maximum stroking rate to 11% per second. The recirculation loop flow is allowed to decrease to approximately 20% of rated flow. This is the expected flow when the FCVs are maintained at the minimum open position at high pump speed.

The transients described here start at 105% NBR power. All other inputs and response functions are assumed normal unless otherwise noted.

Results

The results of these transients are similar to the trip of a single and both recirculation pumps; however, they are less severe than the transient resulting from the simultaneous trip of both recirculation pumps. Thermal consequences are insignificant and because vessel pressure remains below the design value, there is no threat to the reactor coolant pressure boundary.

a. Figure 15.3-3 shows the maximum FCV stroking rate as limited only by the valve positioner hydraulics. It is similar to the trip-of-one-recirculation-pump transient except that the core flow decreases more rapidly, thus causing water level to swell to the L8 high level trip setpoint at an earlier time. The main turbine trip initiates scram and recirculation pump downshift (EOC RPT). MCPR does not decrease significantly and fuel

margins are not threatened. The pressure increases to a value above the setpoint of the first group of relief valves. Pressure in the vessel dome is limited to 1095 psig.

b. Figure 15.3-4 depicts the expected transient which is similar to that resulting from a two pump trip of Subsection 15.3.1. The rate of flow decrease in the recirculation loops is somewhat slower; however, because of the limiter operation, thus the subsequent events occur later in time. The initial flow decrease is also slower than that experienced with the closure of valve at its maximum rate, as described in a above. Two groups of pressure relief valves open as vessel pressure exceeds their setpoints following the high water level trip. Steam is discharged briefly to the suppression pool. Peak pressure at the vessel bottom reaches 1108 psig; at the vessel dome, 1095 psig. Both are well below the ASME Code limit for the reactor vessel coolant pressure boundary.

Initial conditions chosen for these analyses are conservative and tend to force analytical results to be more severe than otherwise expected. These analyses are unaffected by variations in pump/pump motor and drive line inertias because the FCV is the cause of rapid recirculation flow decreases.

15.3.2.4 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed. Release of coolant to the containment occurs via SRV actuation.

15.3.2.5 Radiological Consequences

While the consequence of this event does not result in fuel failure, it does result in the discharge of normal cooling activity to the suppression pool via SRV operation. Because this activity is contained in the primary containment, there will be no exposures to operating personnel. Because this event does not result in an uncontrolled release to the environment, the plant containment or discharge it to the environment under defined meteorological and controlled release conditions. If purging of the containment is chosen, the release is made in accordance with established technical specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.3.3 Recirculation Pump Seizure

The Recirculation Pump Seizure event is not analyzed for reload cores. The

analysis and results presented are for the intial core

15.3.3.1 Identification of Causes and Frequency Classification

The postulated recirculation pump seizure represents an extremely unlikely event of instantaneous stoppage of the pump-motor shaft of one recirculation pump. This produces a very rapid decrease of core flows as a result of the large hydraulic resistance introduced by the stopped rotor.

A recirculation pump seizure is categorized as an event of limiting fault frequency, an accident in other words. Actual data are not available for this postulated event.

15.3.3.2 Sequence of Events and Systems Operation

The sequence of events for this hypothetical incident (Figure 15.3-5) is as follows:

Time (sec)	Event
0	Single pump seizure was initiated.
0.58	Jet pump diffuser flow reverses in seized loop.
3.58	High vessel water level (L8) trip initiates main turbine trip.
	High vessel water level (L8) trip initiates feedwater turbine trip.
	Main turbine trip initiates bypass operation.
3.59	Main turbine stop valves reach 90% open position and initiate reactor scram trip and recirculation pump downshift (EOC RPT).
3.68	Turbine stop valves closed and turbine bypass valves start to open to regulate pressure.

Time (sec)	Event
3.72	Recirculation pump motor circuit breakers open causing decrease in core flow.
6.31 and 6.72	Relief Valves actuated by Groups 1 and 2.
Est 10.8 and 11.9	Relief Valves close by Groups 2 and 1.
Est 28.8	Turbine bypass valves start to close.
30.18	Turbine bypass valves closed.
37.79	Turbine bypass valves reopen on pressure increase.
38.17	Low level trip (L2) initiates MSIV closure (see Subsection 15.0.7). Low level trip (L2) initiates RCIC and HPCS and RR pump trip (not simulated).
40.51	Turbine bypass valves closed.
41.17	MSIVs closed (see Subsection 15.0.7).

The simulation of the expected sequence of events relied upon normal functioning of plant instruments and controls of the plant protection equipment, and of the reactor protection systems. Operation of RCIC and HPCS systems, though not included in the simulation are expected to be utilized normally. Their use maintains adequate water level which is not a problem for this event. In fact, the reactor scrams from a turbine trip which is caused by water level swell (L8).

No operator actions are required, however, the operator can regain control of reactor water level through RCIC operation or via restart of the motor driven feedwater pump. Normally the operator would ascertain that the reactor scrams from the L8 trip and that coastdown is normal with pressure controlled via ADS if needed.

Single failures in the scram logic originated via the high water level trip (L8) are the same as explained in Subsection 15.3.1.2, trip of two recirculation pumps.

15.3.3.3 Core and System Performance

This transient event is assumed to occur as a consequence of an unspecified, instantaneous stoppage of one recirculation pump shaft while the reactor is operating at 104% NBR power. The reactor is assumed to be operating at thermally limiting conditions. The void coefficient is adjusted to the most conservative value, i.e., the least negative value in Table 15.0-1.

The nonlinear dynamic model described briefly in Section 15.1.1.3 was used to simulate this event for the initial core.

Unless otherwise noted, the analyses were performed with plant conditions shown in Table 15.0-1.

Results

Core coolant flow drops rapidly upon abrupt stoppage of a recirculation pump; it reaches a minimum value of 51.6% flow at about 1.1 second. Natural circulation is then established. The water level swell produces a trip of both the main turbine and the feedwater turbines; this in turn initiates a turbine stop valve (closure) scram and a recirculation pump downshift (EOC RPT). After the recirculation pump downshift, the initial core MCPR decreased by an insignificant amount before the fuel surface heat flux begins to drop, thus restoring a greater thermal margin. The turbine trip, which occurs after the time interval in which MCPR decreased, does not significantly retard the decrease in heat flux. Therefore, there is no threat to fuel thermal limits and the design basis is satisfied.

Considerations of uncertainties are included in the GETAB analysis.

As documented in Reference 2, no reload analysis of this event is required.

15.3.3.4 Barrier Performance

Steam flow through the turbine bypass valves and momentary opening of some of the safety/relief valves (automatically) limit the vessel pressure to acceptable values. The reactor coolant pressure boundary is not threatened by overpressure because the fuel temperature and MCPR limits are not compromised, the fuel barrier (clad) is also not threatened.

15.3.3.5 Radiological Consequences

While the consequence of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity

bottled up in the containment or discharge it to the environment under defined meteorological and controlled release conditions. If purging of the containment is chosen the release will be in accordance with established technical specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.3.4 Recirculation Pump Shaft Break

The Recirculation Pump Shaft Break event is not analyzed for reload cores. The analysis and results presented are for the initial core.

15.3.4.1 Identification of Causes and Frequency Classification

The breaking of the recirculation pump shaft is a postulated event which is categorized in the functional sense as a design-basis accident. The frequency classification is, therefore, that of a limiting fault. The recirculation pump was designed to very rigid standards and codes. It is instrumented, monitored, and controlled to assure safe and orderly operation. It was designed to modern seismic and environmental conditions. Refer to Chapter 5.0 for specific mechanical considerations and to Chapter 7.0 for electrical aspects of the design.

The analysis of this postulated failure was conducted for single and two loop operation. This analysis indicates that the postulated accident is very mild in relation to other design-basis accidents such as the LOCA. It is bounded by the more limiting case of recirculation pump seizure recorded in Subsection 15.3.3.

15.3.4.2 Sequence of Events and Systems Operation

In the extremely unlikely event of a recirculation pump shaft break, a very rapid decrease of core flow results from severance of the driving motor from the large hydraulic resistance of the recirculation flow loop. The core flow decrease results in water level swell in the reactor vessel. When the vessel water level reaches the high water level setpoint (L8), the main turbine trips and the feedwater turbine also trips: subsequently, reactor scram and RPT are initiated from main turbine control valve position switches. Eventually, the vessel water level is controlled by the HPCS and/or RCIC flow or start of a motor driven feedwater pump. Pressure relief is automatically handled by the safety relief valves (SRVs).

The system operation in this accident is similar to that which is recorded in Subsection 15.3.3. Operator actions include the normal verification of scram, regaining control of water level for the reactor, and monitoring of pressure after shutdown.

Normal operation of plant instrumentation and controls was assumed for this event. Operation of the HPCS and/or RCIC is a backup to the normal recovery of feedwater supply via a motor-driven feedwater pump. Effects of single failures in the vessel high level trip (L8) or in the reactor protection system trip are identical to those previously presented in Subsection 15.3.1.2. Examination of other equipment failures in the recirculation system has led to the conclusion that no other credible failure exists to aggravate conditions worse than those represented in the bounding case of Subsection 15.3.3.

15.3.4.3 Core and System Performance

This event has less limiting consequences than the pump seizure event of Subsection 15.3.3, hence only qualitative comparison is provided here.

For the pump seizure event, the loop flow decreases faster than the normal flow coastdown because of the large hydraulic resistance introduced by the stopped rotor. For the pump shaft break event, the hydraulic resistance associated with the movable rotor is less than that encountered with the seized rotor. Therefore, the core flow decreased more slowly following a pump shaft break. The potential effects on the core are, therefore, bounded by those reported for the pump seizure event.

Results

For the pump shaft break, the core heat flux decrease more rapidly than the rate at which heat is removed by the coolant, therefore, there is no threat to fuel thermal limits.

Operation of the turbine steam bypass valves and automatic momentary opening of some of the safety relief valves (SRVs) limit the vessel pressure well within the range allowed by the ASME vessel code. Therefore, the reactor coolant pressure boundary is not threatened by overpressure.

15.3.4.4 Barrier Performance

As noted in the above comparison and by reference to the simulation results for the pump seizure transient (15.3.3), the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel clad, pressure vessel, or containment are designed. There is, therefore, no effect on the primary pressure containment barriers.

15.3.4.5 Radiological Consequences

While the consequence of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via relief valve operation. Because this activity is contained in the primary containment, there will be no exposure to operating personnel. Because this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under defined meteorological and controlled release condition. If purging of the containment is chosen the release is made in accordance with established technical specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.3.5 References

1. R. Linford, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802, February 1973.

2. General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, (Unit 1: Rev. 22, Unit 2: Rev. 20).

3. Intentionally Deleted.

4. Power Uprate Project Task 900, "Transient Analyses," GE-NE-A1300384-08 Revision 1, September 1999.

5. Design Analysis L-003505, Revision 0, "Disposition of Events Summary for the LaSalle MUR Power Uprate," July 2010.

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

15.4.1 Rod Withdrawal Error - Low Power

The Rod Withdrawal Error- Low Power events are not analyzed for reload cores. The analysis and results presented are for the initial core.

This event was evaluated for the effect of 105% power uprate (Reference 25) and it was determined that this was not a bounding event.

15.4.1.1 Control Rod Removal Error During Refueling

15.4.1.1.1 Identification of Causes and Frequency Classification

This event considers an arbitrary full withdrawal of the most reactive control rod during refueling. Such an event is categorized by the frequency of a limiting fault. The probability of the initiating causes alone is considered low enough to warrant its being categorized as an infrequent incident because there is no practical set of circumstances which can result in an inadvertent RWE while in the REFUEL mode.

15.4.1.1.2 Sequence of Events and Systems Operation

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

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

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

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

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

15.4.1.1.3 Core and System Performance

Subsection 4.3.2 contains the shutdown margin analysis.

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

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

15.4.1.1.4 Barrier Performance

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

15.4.1.1.5 Radiological Consequences

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

15.4.1.2 Continuous Rod Withdrawal During Reactor Startup

15.4.1.2.1 Identification of Causes and Frequency Classification

Control rod withdrawal errors are not considered credible in the startup power range. The RWM prevents the operator from selecting and withdrawing an out-of-sequence control rod.

The probability of initiating causes (or multiple errors) for this event alone is considered low enough to warrant its being categorized as an infrequent incident. The probability of further development of this event is extremely low because it is contingent upon the failure of the RWM system concurrent with high worth rod, out-of-sequence rod selection contrary to procedures, plus operator nonacknowledgment of continuous alarm annunciations prior to safety system actuation.

Not withstanding the design provisions to preclude rod withdrawal, a special analysis is included (Subsection 15.4.1.2.3.1) to show that continuous withdrawal of an out of sequence rod in the startup range results in acceptable peak fuel enthalpies less than the licensing basis criteria.

15.4.1.2.2 Sequence of Events and Systems Operation

Control rod withdrawal errors are not considered credible in the startup and low power ranges. The RWM prevents the operator from selecting and withdrawing an out-of-sequence control rod.

Continuous control rod withdrawal errors during reactor startup are precluded by the RWM. The RWM prevents the withdrawal of an out-of-sequence control rod in the 100% to 75% control rod density range and limits rod movement to the banked position mode of rod withdrawal from the 75% rod density to the preset power level. Since only in-sequence control rods can be withdrawn in the 100% to 75% control rod density and control rods are withdrawn in the banked position mode from the 75% control rod density point to the preset power level, there is no basis for the continuous control rod withdrawal error in the startup and low power range. The low power range is defined as zero power to the RWM low power setpoint, i.e., 10% of rated core power. For RWE above low power setpoint see Section 15.4.2.

No operator actions are required to preclude this event since the plant design as discussed above prevents its occurrence.

For any operation involved in a possible initiating failure (or error), the necessary safety actions are taken (rod blocks) prior to any possible subsequent single failure.

15.4.1.2.3 Core and System Performance

The functions of the RWM prevent erroneous selection and withdrawal of an out-of-sequence control rod. Thus, the core and system performance is not affected by such an operator error.

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

15.4.1.2.3.1 Special Analysis

The continuous control rod withdrawal transient analysis in the startup range was performed to demonstrate that the licensing basis criteria for fuel failure will not be exceeded when an out- of-sequence control rod is withdrawn at the maximum

allowable normal drive speed. The sequence and timing assumed in this special analysis is shown in Table 15.4-1.

The rod worth minimizer (RWM) constraints on rod sequences will prevent the continuous withdrawal of an out-of-sequence rod. This analysis was performed to demonstrate that, even for the unlikely event where the RWM fails to block the continuous withdrawal of an out-of-sequence rod, the licensing basis criteria for fuel failure is still satisfied.

The methods and design basis used for performing the detailed analysis for this event, are similar to those previously approved for the control rod drop accident (CRDA) (References 1, 2, and 3). Additional simplified point model kinetics calculations were performed to evaluate the dependence of peak fuel enthalpy on the control blade worth. For the detailed calculation, the 50% control rod density pattern was selected as the initial starting condition which is consistent with the approved design basis for the CRDA (References 1, 2, and 3).

The licensing basis criterion for CRDA is the contained energy of a fuel pellet located in the peak power region of the core. The peak fuel pellet enthalpy for CRDA shall not exceed 280 cal/g-UO_2. The peak fuel pellet enthalpy associated with the threshold for fuel failures is 170 cal/g-UO_2.

15.4.1.2.3.2 Methods of Analysis

Since the rod worth calculations using the approved design basis methods (References 1, 2, and 3) use three-dimensional geometry, it is not practical to do a detailed analysis of this event parameterizing control rod worths. Therefore, the methods of analysis employed were to perform a detailed evaluation of this event for a typical BWR and control rod worth (1.6% Δk) and to use a point model calculation to evaluate the results over the expected ranges of out-of-sequence control rod worths. The detailed calculations are performed to demonstrate (1) the consequences of this event over the expected power operating range and (2) the validity of the approximate point model calculation. The point model calculation will demonstrate that the licensing criteria for fuel failure is easily satisfied over the range of expected out-of-sequence control rod worths. These methods are described in more detail below.

The following calculations were performed assuming an APRM scram setpoint of 15%, however, the actual analytical setpoint is 25%. A subsequent evaluation (Reference 10) determined that if the scram is assumed to occur at the 25% setpoint (without credit for the IRM scram), the resulting fuel enthalpy would be less than 100 cal/g, meeting the fuel failure threshold of 170 cal/g. Additionally, the same evaluation also showed that if only the 120% APRM scram is credited, the resulting fuel enthalpy would be less than 170 cal/g. The following analyses are retained for

historical information, and because they were the basis behind the Reference 10 evaluation.

The methods used to perform the detailed calculation are identical to those used to perform the design basis control rod drop accident with the following exceptions:

a. The rod withdrawal rate is 3.6 ips rather than the blade drop velocity of 3.11 fps.

b. Scram is initiated either by the IRM or 15% APRM scram in the startup range. The IRM system is assumed to be in the worst bypass condition allowed by technical specifications.

c. The blade being withdrawn inserts along with remaining drives at technical specification insertion rates upon initiation of scram signal.

Examination of a number of rod withdrawal transients in the low power startup range, using an R-Z model, has shown clearly that higher fuel enthalpy addition would result from transients starting at the 1% power level rather than from lower power levels. The analysis further shows that for continuous rod withdrawal from these initial power levels (1% range) the APRM 15% power level scram is likely to be reached as soon as the degraded (worst bypass condition) IRM scram. Consequently, credit is taken for either the IRM or APRM 15% scram in meeting the consequences of this event. The transients for this response were initiated at 1% of power and were performed using the 15% APRM scram.

An initial point kinetics calculation was run to determine the line to scram based on an APRM scram setpoint of 15% power and an initial power level of 1%. From this time and the maximum allowable rod withdrawal speed, it is possible to show the degree of rod withdrawal before reinsertion due to the scram. From this information Figure 15.4-1, showing the modified effective reactivity shape, was constructed.

The point model kinetics calculations use the same equations employed in the Adiabatic Approximation described on page 4-1 of Reference 1. The rod reactivity characteristics and scram reactivity functions are input identical to the adiabatic calculations, and the Doppler reactivity is input as a function of core average fuel enthalpy. The Doppler reactivity feedback function input to the point model calculations was derived from the detailed analysis of the 1.6% rod worth case described above. This is a conservative assumption for higher rod worths since the power peaking and hence spatial Doppler feedback will be larger for higher rod worths. As will be seen in the results section, maximum enthalpies resulted from cases initiated at 1% of rated power. In this power range the APRM will initiate scram at 15% of power; hence, the APRM 15% power scram was used for these

calculations thereby eliminating the need to perform the spatial analysis required for the IRM scram. All other inputs are consistent with the detailed transient calculation.

The point model kinetics calculations results in core average enthalpies. The peak enthalpies were calculated using the following equation:

where

$\hat{h}$ = $ho + (P/A)_T (\bar{h}_f - ho)$;

h = Final peak fuel enthalpy;

$\hat{h}_o$ = Initial fuel enthalpy;

$\bar{h}_f$ = Final core average fuel enthalpy; and

$(P/A)_T$ = Total peaking factor (radial peaking) * (axial peaking) *(local fuel pin peaking).

For these calculations, the (radial x axial) peaking factors as a function of rod worth were obtained from the calculations performed in Section 3.6 of Reference 2 and are shown in Figure 15.4-2. It was conservatively assumed that no power flattening due to Doppler feedback occurred during the course of the transient.

15.4.1.2.3.3 Results

The result of the Reference 10 evaluation was that, with no credit being taken for the IRM scram, and with an APRM scram at the analytical setpoint of 25%, the peak fuel enthalpy would be less than 100 cal/g. If only the 120% APRM scram is credited, the peak fuel enthalpy would be less than 170 cal/g. The following discussion pertains to the original analysis, with credit given for a 15% APRM scram and an IRM scram. These original analyses are the basis behind the more recent evaluation performed in Reference 10.

The reactivity insertion resulting from moving the control rod is shown in Figure 15.4-1 for the point kinetics calculations. The core average power versus time and the global peaking factors from Section 3.6 of Reference 2 are shown in Figures 15.4-3 and 15.4-4 respectively. The results of the point kinetics calculation are summarized in Table 15.4-2 along with the results of the detailed analysis.

From Figure 15.4-3 and Table 15.4-2, it is shown that the core average energy deposition is insensitive to control rod worth; therefore, the only change in peak enthalpy as a function of rod worth will result from differences in the global peaking

which increases with rod worth. Comparison of the global peaking factors shown in Figure 15.4-2 with the values used in the detailed calculations demonstrates that the Reference 2 values are reasonable for their application in this study. For all cases, the peak fuel enthalpy is well below the licensing design criteria of 170 cal/gm.

Cases 4 and 5 of Table 15.4-2 show that the point kinetics calculations give conservative results relative to the detailed evaluations. The primary difference is that the global peaking will flatten during the transient due to Doppler feedback. This is accounted for in the detailed calculation but the point kinetics calculations conservatively assumed that the peaking remains constant at its initial value.

The differences in core average and peak enthalpy between cases 1 and 5 are due to the fact that for case 1 the scram was initiated by the 15% APRM scram setpoint, whereas, in case 5 the scram was initiated by the IRM's. As seen by Figure 15.4-4, this occurred at a core average power of 21%. Since the APRM trip point will be reached first, it is reasonable to take credit for the APRM scram.

15.4.1.2.3.4 Conclusions

From this study the following conclusions can be stated:

a. The resultant peak fuel enthalpies due to the continuous withdrawal of an out-of-sequence rod in the startup range results in peak fuel enthalpies which are significantly less than the licensing basis criteria of 170 cal/gm.

b. The point model calculations used to assess the sensitivity of peak enthalpy as a function of control rod worth are in good agreement with, and slightly conservative relative to the more detailed design basis model which is employed to evaluate the continuous rod withdrawal transient in the startup range.

15.4.1.2.4 Barrier Performance

An evaluation of the barrier performance was not made for this event because resultant peak fuel enthalpies, due to continuous rod withdrawal, are significantly less than the licensing basis criteria.

15.4.1.2.5 Radiological Consequences

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

15.4.2 Rod Withdrawal Error - at Power

15.4.2.1 Identification of Causes and Frequency Classifications

While operating in the power range in a normal mode of operation the reactor operator makes a procedural error and continuously withdraws the maximum worth control rod until the RBM inhibits further withdrawal. The cycle specific analyses of this event typically are performed unblocked.

The rod withdrawal error (RWE) event is considered a moderate frequency event.

15.4.2.2 Sequence of Events and Systems Operation

Typical sequence of events for this rod withdrawal error at power is as follows:

Elapsed Time (sec)	Event
0	Core is assumed to be operating at rated conditions when the operator selects and continuously withdraws the maximum worth control rod.
~1	The local power in the vicinity of the control rod increases; total core power increases.
~5	LPRM's indicate excessive localized peaking, but the operator ignores this warning and continues withdrawal.
~15	The RBM subsystem indicates excessive localized peaking, but the operator ignores this warning and continues withdrawal.
~20	The RBM system initiates a rod block to preclude further rod withdrawal.
~40	Reactor core stabilizes at a higher power level.
~60	Operator inserts control rod to reduce core power.

Elapsed Time (sec)	Event
~80	Reactor core stabilizes at rated conditions.

The focal point of this event is localized to a small portion of the core; therefore, although reactor control and instrumentation is assumed to function normally, credit is taken only for the RBM subsystem.

Under most normal operating conditions no operator action is required since the transient which occurs would be very mild. Should the peak linear power design limits be exceeded, the nearest Local Power Range Monitor (LPRM) would detect this phenomenon and sound an alarm. The operator must acknowledge this alarm and take appropriate action to rectify the situation.

If the rod withdrawal continues, the rod block monitor (RBM) system would sound alarms, at which time the operator would acknowledge the alarm and take corrective action. Even for the most severe conditions (i.e., for highly abnormal control rod patterns, operating conditions, and assuming that the operator ignores all alarms and warnings and continues to withdraw the control rod), the RBM system will inhibit further withdrawal of the control rod before the fuel reaches the point of boiling transition or the 1% circumferential strain limit imposed on the clad.

Due to this positive reactivity insertion, the core average power increases. The local power in the vicinity of the withdrawn control rod also increases. Although operator errors initiated and continued the rod withdrawal, the RBM subsystem terminates this transient. The RBM subsystem is designed to be single failure proof; therefore, termination of the event is assured.

15.4.2.3 Core and System Performance

Mathematical Model

The cycle-specific analysis considers the continuous withdrawal of the maximum worth control rod, at its maximum drive speed, from the reactor which is assumed to be operating at rated power with a control rod pattern which results in the core being placed on thermal design limits.

For this transient, the reactivity insertion rate is very slow; therefore, it is adequate to assume that the core has sufficient time to equilibrate (i.e., that both the neutron flux and heat flux are in phase). Making use of this assumption, the transient is

calculated using a steady-state, three-dimensional, coupled nuclear-thermal-hydraulics computer program. All spatial effects are included in the calculation. For GE fuel reloads using GE methodology, the program described in Reference 5 is used.

The primary output from this code, in addition to the basic nuclear parameters, is: the variation of the Linear Heat Generator Rate (LHGR); the variation of the Minimum Critical Power Ratio (MCPR); the total reactor power; and the variation of the in-core instruments during the transient. A detector response code uses the calculated instrument responses to determine the RBM setpoints as a function of the position of the withdrawn error rod. Finally, the delta-MCPR associated with each rod block set point is tabulated. Generally, the delta-MCPR increases as the error rod is withdrawn to its fully withdrawn position.

Analysis

The analytical methods and assumptions which are used in evaluating the consequences of this accident are considered to provide a realistic, yet conservative assessment of the consequences.

The number of possible RWE transients is extremely large due to the number of control rods and the wide range of exposures and power levels. In order to encompass all of the possible RWEs which could conceivably occur, a limiting analysis is defined such that conservative assessment of the consequences is achieved.

The conservative assumptions are:

a. The assumed error is a continuous withdrawal of the maximum worth rod at its maximum drive speed.

b. The core is assumed to be operating at rated conditions.

c. The reactor is presumed to be in its most reactive state and devoid of all xenon. This insures that the maximum amount of excess reactivity which must be controlled by the movable control rods is present.

d. Furthermore, it is assumed that the operator has fully inserted the maximum worth rod prior to its removal and selected the remaining control rod pattern in such a way as to approach thermal limits in the fuel bundles in the vicinity of the rod to be withdrawn. (See Figure 15.4-5 for the initial core pattern.) It should be emphasized that this control rod configuration would be highly abnormal and could only be achieved by deliberate operator action or by numerous operator errors.

e. The operator is assumed to ignore all warnings during the transient.

f. Of the four LPRM strings, nearest to the control rod being withdrawn, the two highest reading LPRM during the transient are assumed to have failed. (One of the two instrument channels is assumed to be bypassed and out-of-service. The A and C LPRM chambers input to one channel while the B and D chambers input to the other. The channel with the greatest response is assumed to be bypassed).

These conservative assumptions provide a high degree of assurance that the analyzed transient bounds all possible RWEs. For cycle specific results, refer to the reload licensing package which is referenced in the LaSalle Administrative Technical Requirements.

RBM Subsystem Operation

The RBM subsystem minimizes the consequences of a RWE by inhibiting the motion of the control rod before the safety limits are exceeded. The RBM has three trip levels (rod withdrawal permissive removed). The trip levels may be adjusted. The highest trip level is set so that the safety limit is not exceeded. The lower two trip levels are intended to provide a warning to the operator. The trip levels are automatically varied with reactor coolant flow to protect against fuel damage at lower flows. The variation is set to assure that no fuel damage will occur at any indicated coolant flow. The operator may encounter any number (up to three) of trip points depending on the starting power of a given control rod withdrawal, but normally only the highest trip level will be selected. The lower two points may be passed up (reset) by manual operation of a push button. The reset permissive is actuated (and indicated by a light) when the RBM approaches the trip point. The operator should then assess his local power and either reset or select a new rod. The highest (power) trip point cannot be reset.

Results

The results for the RWE are reported in the Reload Licensing Onsite Review

Package. The cycle specific reload licensing package typically contains the results of the RWE analysis which consists of the delta-MCPR for the unblocked RWE event.

Using the results, the RBM setpoint may be chosen to allow an acceptable RWE delta-MCPR. The chosen setpoint ensures that neither localized boiling transition nor gross boiling transition occurs, nor is the 1% circumferential strain limit on the cladding violated.

If the reactor response is acceptable with the error rod fully withdrawn, then no credit need be taken for the RBM subsystem in analyzing the RWE event. In this case, the cycle-specific licensing documents may report only the unblocked (fully withdrawn) RWE delta-MCPR, and the unblocked RBM setpoint is used.

For the cycle specific results see the reload licensing package which is referenced in the Technical Requirements Manual. Figures 15.4-6 through 15.4-9 illustrate typical RWE response data.

The variation in the MCPR and MLHGR, as a function of withdrawal of the highest worth rod for the initial core, is presented in Figures 15.4-6 and 7, respectively. The bundles presented in Figures 15.4-6 and 7 represent the envelope of the MCPR and the MLHGR for each two-foot interval during the transient. Variation in the total reactor power is also shown in these figures. Although these figures show the change in thermal limits from the fully inserted to the fully withdrawn position, the control rod is automatically blocked at 5.0 feet, even under the worst set of assumptions. The variation in the signal response for the initial core of the two independent channels is shown in Figures 15.4-8 and 9.

15.4.2.4 Barrier Performance

The cycle-specific minimum critical power ratio (MCPR) limiting condition for operation (LCO) is determined for each reload core based on bounding events for the cycle. The MCPR LCO is calculated to preclude violation of the fuel cladding integrity safety limit.

An evaluation of the barrier performance was not made for this event since this is a localized event with very little change in the average core characteristics. Typically, an increase in total core power is less than 5% and the changes in pressure are negligible.

15.4.2.5 Radiological Consequences

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

15.4.3 Control Rod Misoperation

Limiting control rod operation anomalies have been analyzed in Subsections 15.4.1 and 15.4.2.

15.4.4 Abnormal Startup of Idle Recirculation Pump

The current analysis for idle loop startup supports a delta T of 50 °F between the idle loop and the reactor pressure vessel. This is consistent with the Technical Specifications that requires verification of the difference between the reactor coolant temperature in the recirculation loop to be started and the RVP coolant temperature to be ≤ 50 °F.

The Abnormal Startup of an Idle Recirculation Pump event is analyzed for a full GE9 core (Reference 11, GE Report B33-00296-03P). This event is not analyzed for reload cores.

15.4.4.1 Identification of Causes and Frequency Classification

A typical idle loop startup transient progresses as follows from the time the pump is started at time zero.

Time (seconds)	Event
0.0	Plant operating with one recirculation loop only [35% power, 47% Flow]. Start idle recirculation loop pump motor.
4.6	Peak neutron flux
23.1	Peak Average Surface Heat Flux
60+	Reactor State settles at new steady state conditions

This hypothetical event is categorized as a moderate frequency event (reference UFSAR Table 15.0-2).

15.4.4.2 Sequence of Events and Systems Operation

For comparison purposes only, the normal sequence of operator actions expected in starting an idle loop is as follows:

a. Adjust rod pattern as necessary for new power level following idle loop start and to comply with the Core Operating Limits Report (COLR) thermal limits for single loop operation.

b. Verify that idle pump suction and discharge valves are open and the flow control valve is at or below the maximum position for pump upshift. Allow temperature of idle loop to equalize in accordance with technical specifications.

c. With the flow control valve at or below the maximum position and the feedwater flow cavitation interlock satisfied, the idle pump is started to fast speed.

d. Re-adjust flow of the running loop downward to less than half of the rated flow in accordance with technical specifications.

e. Start the idle loop pump and adjust flow to match the adjacent loop flow and monitor reactor power.

f. Re-adjust power as necessary to satisfy plant requirements per standard procedure.

(Time to do the above work is approximately 0.5 hours)

For the postulated transient, system operation is assumed to adjust to the abnormal startup. No operator actions are needed for this transient, in fact, the transient is over before the operator can insert rods to limit the flux peak. No scram occurs. Only post event monitoring is needed to verify that no core limits were exceeded. After temperature equilibrium is re-established, the temperature transient effects on the reactor and recirculation system should be reviewed.

15.4.4.3 Core and System Performance

For the mathematical model, input parameters, and initial conditions, see Reference 11.

Results

A typical transient response to the incorrect startup of a cold, idle recirculation loop is shown in Figures 15.4-10a through 15.4-10d. Shortly after the idle recirculation pump begins to move, a surge of flow through the started jet pump diffusers causes the core inlet flow and subcooling to rise sharply. The influx of cool water causes the voids in the vessel to collapse and core neutron flux to rise. Also, as the jet pump diffuser flow in the just started loop increases, the diffuser flow in the other loop begins to drop off. As the event continues, the core neutron flux may increase to the high flux trip setpoint where a reactor scram is initiated or the event may settle out at new steady-state condition. In this analysis, reactor scram is not initiated and the core settles to a new power level. Figure 15.4-10a shows that, at about 23 seconds, the power begins to decrease and Figure 15.4-10c shows that the core inlet subcooling begins to decrease. These results show that, at about 23 seconds, the slug of cold water has traversed the system and the subcooling begins to decrease.

Due to the nature of this transient, it has been analyzed at initial conditions on the Power-to-Flow Map that intentionally avoid a scram such that the resulting challenge to core thermal limits is bounding. To protect the MCPR Safety Limit, an additional barrier to restarting an idle recirculation pump has been added to the operating procedure for fast pump start. This barrier involves the operator verifying (and applying) the single loop operation thermal limit corrections prior to restarting an idle pump in fast speed.

The core wide transient results are available in Reference 11.

15.4.4.4 Barrier Performance

The consequences of this postulated event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed.

Therefore, there is no effect on the primary pressure containment barriers.

15.4.4.5 Radiological Consequences

None

15.4.5 Recirculation Flow Control Failure with Increasing Flow

The bases for not re-evaluating this transient event for power uprate are discussed in section 15.B.2.2 and Reference 23.

The Recirculation Flow Control Failure with Increasing Flow event is not analyzed for reload cores. The analysis and results presented are for the initial core.

Flow dependent MCPR and LHGR limits are established to support operation at off-rated core flow conditions based on the CPR and heat flux changes experienced by the fuel during slow flow excursions. The slow flow excursion event assumes a failure of the recirculation flow control system such that the core flow increases slowly to the maximum flow attainable. The increase in flow creates the potential for a significant increase in core power and heat flux.

15.4.5.1 Identification of Causes and Frequency Classification

Failure of the master controller or neutron flux controller is postulated to cause a maximum rate increase in the recirculation flow hence in the core coolant flow rate. Failure within a recirculation loop's flow controller can also cause an increase in core flow rate. To maximize the effect of this postulated transient, the reactor is considered to be operating at the low end of the rated-flow-control range, i.e., at 56% NBR power and 35% core flow on the 105% NBR flow control characteristic curve for the reactor.

This transient disturbance is categorized as an event of moderate frequency.

15.4.5.2 Sequence of Events and Systems Operation

Simulations were made for the fast opening of one recirculation flow control valve (FCV) (Figure 15.4-11) and for the fast opening of both recirculation flow control valves (Figure 15.4-12).

a. The sequence of events for the fast opening of one recirculation FCV as a result of failure of single loop control is as follows:

Time (sec)	Event
0	Simulate failure of single loop control.
1.15	Reactor APRM high flux scram trip initiated.
6.0 (est)	Turbine control valves start to close upon falling turbine pressure.
12.1 (est)	Turbine control valves closed.
30.4 (est)	Feedwater decreases upon rising water level.
58.78	Vessel water level (L8) trip initiates main turbine trip.
	Main turbine trip initiates steam bypass valve operation.
58.79	Main turbine stop valves reach 90% open position and initiate recirculation pump downshift (EOC RPT).
58.88	Main turbine stop valves closed. Steam bypass valves do not reopen as turbine inlet pressure remains below pressure regulator setpoint.

Time (sec)	Event
58.92	Recirculation pump motor circuit breakers open to cause decrease in core flow to natural circulation condition.
100 +	Turbine steam bypass valves open to regulate pressure. New thermal equilibrium established.

b. The sequence of events for the fast opening of both recirculation FCV's as a result of failure of the master controller is as follows:

Time (sec)	Event
0	Initiate failure of master controller.
1.2	Reactor APRM high flux scram trip initiated.
1.2	Turbine control valves start to close upon falling turbine pressure.
9.7	Turbine control valves closed. Turbine pressure below pressure regulator set-points.
15.7	Vessel water level (L8) trip initiates main turbine and feedwater turbine trips.

Each of these transient events was analyzed with credit for normal functioning of plant instrumentation and controls and the reactor protection system. Operation of engineered safeguards systems is not expected for either of these events. Both of these transients lead to a rapid rise in reactor flux. Corrective action first occurs via the high flux trip (not flow referenced). As a part of the reactor protection system, which is designed to single failure criteria, this trip is assured and reactor operator actions are not of concern in scramming the reactor.

Normal operator actions are expected following the high flux scram; these include: observation that all rods are inserted; maintenance of reactor water level and pressure to prevent MSIV's from isolating the reactor; reset of the mode switch to start up; and maintenance of condenser vacuum and turbine seals to enable reactor restart. Coastdown monitoring and reindexing the recirculation flow controller (manual position) are additional operators actions needed to reestablish a reactor restart configuration.

15.4.5.3 Core and System Performance

The nonlinear dynamic model described briefly in Subsection 15.1.1.3 was used to simulate these events. The plant conditions tabulated in Table 15.0-1 were used in the analysis, unless otherwise noted. The rod worth and the void reactivity coefficient are assumed equal to the technical specification requirements and represent end of cycle, all rods out conditions.

In each event, the most severe transient results when the initial conditions are established for operation at the low end of the 105% NBR flow line. Maximum stroking rates for the FCV's were uses for each separate event: 11% per second for the failure of the master controller driving two loops where the limit is then set by each individual controller, and 30% per second for single loop controller failure as determined by the FCV hydraulics.

Results

Figure 15.4-11 depicts the results for the fast opening of one recirculation FCV. The neutron flux reaches 282% of NBR flux; the surface heat flux reaches 79% of rated heat flux and approximately 2.15 seconds. Fuel centerline temperature increases by only 350° F and the design-basis is satisfied. MCPR remains considerably above 1.06 (original analysis MCPR safety limit).

Figure 15.4-12 presents the results for the fast opening of both recirculation FCV's. The rapid increase in core coolant flow causes an increase in neutron flux, which initiates a reactor high flux scram (not flow referenced). The peak neutron flux reaches 282% of rated level in about 2 seconds and the accompanying transient fuel surface heat flux reaches 76.2% of rated heat flux. This does not exceed the steady state power flow control line. MCPR remains considerably above 1.06 (original analysis MCPR safety limit).

The changes in nuclear system pressures are not significant with regard to overpressurization of the vessel. Pressure decreases over most of the transient.

Variations of valve speed, void reactivity characteristics, or scram time and rod worth are all expected to be in the direction to reduce the severity of the transient analyzed here.

15.4.5.4 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed. Therefore, there is no effect on the primary pressure containment barriers.

15.4.5.5 Radiological Consequences

None.

15.4.6 Chemical and Volume Control System Malfunctions

Not applicable.

15.4.7 Misplaced Fuel Assembly and Subsequent Operation

The Misplaced Fuel Assembly with Subsequent Operation event (also known as a Fuel Loading Error) is evaluated for each reload core.

15.4.7.1 Identification of Causes and Frequency Classification

The event hypothesized for this subsection is the improper loading of a fuel assembly and subsequent operation of the core. There are two ways in which an assembly can be misplaced. The first, known as a mislocation, involves the assembly being loaded into an incorrect core location. The second, known as a misorientation, involves the assembly being loaded into the correct location but in an incorrect orientation.

The GE Method classifies these events as infrequent incidents because core verification is credited. (Reference 5).

15.4.7.2 Sequence of Events and Systems Operation

The mislocation incident typically follows the sequence of events given below:

1. During core loading operation, a fuel assembly is placed in the wrong location in the core.

2. Subsequently, the fuel assembly intended for that location is placed in the location intended for the previous bundle.

3. The placement error is not observed during core verification.

4. The plant is brought to full power operation with these fuel assembly location anomalies. In a reload core the possibility exists for the mislocated assembly to be a bundle scheduled for discharge (intended to remain in the fuel pool), in which case the second step above would be skipped.

The misorientation incident follows the sequence of events given below:

1. During core loading operations a fuel assembly is loaded into its correct location but is misoriented (rotated) by 90°, 180° or 270° from its proper facing in the fuel cell.

2. The placement error is not observed during core verification.

3. The plant is brought to full power operation with the fuel assembly orientation anomaly.

No corrective action occurs because of the lack of detection of the mispositioning during core verification. The analysis assumes the worst case situation so no further errors or failures are considered.

15.4.7.3 Core and System Performance

The concern with a mislocated assembly is that it could lead to an assembly in the core running at a higher power than indicated by the core monitoring system. This is keyed to LPRM readings. A low reactivity fuel assembly mislocated next to an LPRM will lead to a lower reading than is representative of the three symmetric properly loaded bundles. A high reactivity fuel assembly mislocated away from an LPRM will have its power tracked by a symmetric properly loaded low reactivity assembly. The change then exists that, when the core monitoring system indicates the core is on limits, some fuel assemblies are actually exceeding their thermal limits.

The concern with a misoriented assembly lies primarily with the individual fuel pin power response. The core monitoring system assumes that the fuel pins are loaded as expected. A rotation of the bundle will also lead to an axial lean (due to spacer buttons). This leads to an axial varying water gap which, even for lattices designed symmetrically, will distort individual pin powers. The chance then exists that, when the core monitoring system indicates the core is on limits, some fuel pins are actually exceeding their thermal limits.

The effects of a misplaced (misoriented/mislocated) fuel assembly are evaluated for each reload.

The approach used for these events is summarized below.

A bounding delta critical power ratio (ΔCPR) for a mislocation event exists. Should this be judged to be limiting, a specific calculation can be performed to gain additional margin. Standard three-dimensional BWR simulation is used to calculate core performance when an explicit analysis is required.

The misorientation ΔCPR response is typically calculated for the fresh fuel each cycle. However, based on previous results and fuel pin orientation, the bounding mislocation ΔCPR can be applied to the misorientation event as well. The approved lattice calculation code (e.g. TGBLA) is used to evaluate pin power response as a function of misorientation.

These ΔCPRs are used along with other AOO events in setting the plant Operating Limit Minimum Critical Power Ratio (OLMCPR).

15.4.7.4 Barrier Performance

For this event the SLMCPR and the transient LHGR limit are protected and therefore the fuel cladding is protected. The fuel misloading and misorientation events are highly localized events. No perceptible change in the core pressure would be observed.

15.4.7.5 Radiological Consequences

Radiological consequences are limited by the protection of the SLMCPR and the transient LHGR limit.

15.4.8 Spectrum of Rod Ejection Accidents

Not applicable; this is a PWR event.

The BWR has precluded this event by incorporating into its design mechanical equipment which restricts any movement of the control rod drive system assemblies. The control rod drive housing support assemblies are described in Chapter 4.0.

15.4.9 Control Rod Drop Accident (CRDA)

15.4.9.1 Identification of Causes and Frequency Classification

Causes

The control rod drop accident is the result of a postulated event in which a high-worth control rod within the constraints of a rod drop accident sequence (such as the banked-position withdrawal sequence or the analyzed rod position sequence), drops from the fully inserted or intermediate position in the core. The high-worth rod becomes decoupled from its drive mechanism. The drive mechanism is withdrawn but the decoupled control rod is assumed to be stuck in place. At a later most adverse moment, the control rod suddenly falls free and drops to the control rod drive position. This results in the removal of large negative reactivity from the core and results in a localized power excursion. This accident encompasses the consequences of all such reactivity control system excursion via postulating the worst possible combination of rod-worth and core conditions.

A more detailed discussion is given in Reference 6.

Frequency

The rod worth minimizer (RWM) limits the worth of any control rod which could be dropped by regulating the withdrawal sequence. This system prevents the movement of an out-of-sequence rod in the 100% to 75% rod density range, and from the 75% rod density point to the preset power level the RWM will only allow banked-position mode rod withdrawals or insertions. A control rod drop accident sequence (such as the banked position withdrawal or the analyzed rod position) is described in Reference 1 for a typical BWR.

The CRDA is categorized as a limiting fault because it is not expected to occur during the lifetime of the plant; but, as postulated herein, it has consequences that include the potential for the release of radioactive material from the fuel. Such an event would be an accident.

15.4.9.2 Sequence of Events and Systems Operation

Before the control rod drop accident (CRDA) is possible, the sequence of events presented below must occur. No operator actions are required to terminate this transient.

Approximate Elapsed Time	Event
-	Reactor is operating near critical rod pattern.
-	Maximum worth control rod blade becomes decoupled from the CRD.
-	Operator selects and withdraws the CRD of the decoupled rod.
-	Decoupled control rod sticks in the fully inserted position or an intermediate bank position.
-	Control rod becomes unstuck and drops to the drive position at the maximum speed assumed in the analysis.
<1 second	Reactor goes on a positive period and the initial power increase is terminated by the Doppler coefficient.
< 1 second	APRM 120% power signal scram reactor.
< 1 second	Main steamline radiation monitors detect high radiation and initiate closure of the main steamline isolation valves and annunciate a control room alarm.
<5 seconds	Scram terminates accident.
< 6 seconds	Main steamline isolation valves are fully closed (not simulated).
≤ 15.0 minutes	The main condenser mechanical vacuum pump is manually shutdown.

The unlikely set of circumstances, referenced above, makes possible the rapid removal of a control rod. The dropping of the rod results in high reactivity in a small region of the core. For large, loosely coupled cores, this would result in a highly peaked power distribution and subsequent operation of shutdown mechanisms. Significant shifts in the spatial power generation would occur during the course of the excursion.

The termination of this excursion is accomplished by automatic APRM signal scram and the inherent Doppler shutdown mechanism. Therefore, no operator action is required during the excursion. Although other normal plant instrumentation and controls are assumed to function, no credit for their operation is taken in the analysis of this event.

The system which mitigates the consequences of this event is the APRM scram. The APRM scram system is designed to single failure criteria. Therefore, termination of this transient is assured under single failure assumptions. No credit is taken for the closure of the Main Steamline Isolation Valves in the analysis.

The operator actions which are required to mitigate the consequences of this event are the manual shutdown of the main condenser mechanical vacuum pump and closure of the upstream isolation valves following the receipt of the Main Steamline High Radiation Trip Alarm which is a symptom of a CRDA. Per an analysis performed by GE, the mechanical vacuum pump must be manually tripped within 15 minutes of the accident to ensure that the doses remain less than the limits specified in USNRC Standard Review Plan 15.4.9 (reference 7). The upstream isolation valves that provide condenser suction for both the Off Gas system and mechanical vacuum pump are subsequently closed. Both of these operator actions which are required by the LaSalle abnormal procedures are performed in the control room.

No operator error (in addition to the one that initiates this event) can result in a more limiting case because the reactor protection system automatically terminates the transient independent of the operator.

15.4.9.3 Core and System Performance

Historically, ComEd utilized the generic General Electric Banked Position Withdraw Sequence methodology (References 1 and 5) to protect the 280 cal/gm fuel damage limit. This analysis was a bounding and conservative generic calculation. As with most generic analyses, it can also be unnecessarily restrictive. In the early 90s, ComEd received NRC approval (Reference 16) to perform in-house design calculations. Using this in-house ability, ComEd began to perform cycle specific CRDA analyses. Using cycle specific calculations, ComEd/Exelon is able to modify the

original BPWS sequence to remove some of the unnecessary conservatism (typically, elimination of some of the banked positions, Reference 18.) These sequences are referred to as the analyzed rod position sequence.

Control rod patterns analyzed in the cycle specific CRDA analyses follow predetermined sequencing rules. This sequence applies to all control rod movement from the all rods in condition to the Low Power Setpoint (LPSP). These rules include the designation of control rod groups. The banked positions are established to limit the maximum incremental control rod worth such that the 280 cal/gm design limit is not exceeded. Cycle specific analyses ensure that the 280 cal/gm fuel damage limit is not exceeded during worst case scenarios. These worst case scenarios account for a limited number of inoperable control rods with a specified separation criteria. Specific evaluations or analyses can be performed for a typical operating conditions, e.g. fuel leaker suppression.

Simplified shutdown sequences that eliminate the group banking requirements have been generically bounded in Reference 29.

15.4.9.3.1 Evaluations

Mathematical Model

The analytical methods, assumptions and conditions for evaluating the excursion aspects of the control rod drop accident are described in detail in References 2, 3, and 6. They are considered to provide a realistic yet conservative assessment of the associated consequences. The data presented in Reference 1 shows that the banked-position mode reduces the control rod worths to the degree that the detailed analyses presented in References 2, 3, and 6 are not necessary.

For the initial core, compliance checks were made to ascertain that the maximum rod worth for analysis of the BWR-5 core, did not exceed 1%Δ k. When this criteria was not met, the bounding analyses were performed. The rod worths were determined using the BWR Simulator Model described in Reference 1. When necessary, detailed evaluations were made using the methods described in References 2, 3, and 6. See Table 15.4-5 for the worst cases identified from these compliance checks.

Note that the maximum incremental rod worth is constrained to very low values, which means that the postulated CRDA cannot result in peak enthalpies in excess of 280 calories per gram for any plant condition. These data indicate the maximum control rod worth is well below the rod worth required to cause a CRDA which could result in a peak fuel enthalpy of 280 cal/gm.

For reload cores that followed standard BPWS (prior to early 1990's), this event has been generically bounded as described in Reference 5.

Input Parameters and Initial Conditions

At the time of the control rod drop accident the core is assumed to be at a cycle point which results in the highest control rod worth. The core is also assumed to contain no xenon, to be in a hot-startup condition and, to have the control rods in sequence A and be near critical. The assumption to remove xenon, which competes well for neutron absorptions, increases the fractional absorptions, or worth, of the control rods

Results

The conclusion is that the 280 cal/gm design limit is not exceeded and failure of fuel cannot result naturally from a CRDA. This result is considerably less than an assumed failure of 770 fuel rods (for GE 8x8 fuel). Nevertheless, for analytical purposes, radiological evaluations are made for the assumed failure of 770 fuel rods (for GE 8x8 fuel) for the plant operating conditions indicated in Table 15.4-6 and the design-basis analysis of Reference 7.

The 1999 power uprate core design has sufficient enthalpy margin for the cladding to assure that the assumed number of fuel rods damaged and the fraction of damaged fuel assumed to melt (due to a control rod drop accident) are not affected (Reference 24).

15.4.9.4 Barrier Performance

An evaluation of barrier performance was not made for this accident since it is a highly localized event with no significant change in the gross core temperature or pressure.

15.4.9.5 Radiological Consequences

The analysis is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining the adequacy of the plant design to meet 10 CFR 100 guidelines. This analysis is referred to as the "design basis analysis".

The design basis analysis is based on USNRC Standard Review Plan 15.4.9 (Reference 7). The models, assumptions and computer program used for this analysis are described in Reference 8. Specific values of parameters used for evaluation of this accident under design basis assumptions are presented in Table 15.4-6.

Fission Product Release from Fuel

The failure of 770 fuel rods is used for this analysis. The mass fraction of the fuel in the damaged rods which reaches or exceeds the initiation temperature of fuel melting (taken as 2804 °C) is estimated to be 0.0077.

Fuel reaching melt conditions is assumed to release 100 percent of the noble gas inventory and 50 percent of the iodine inventory to the coolant. The remaining fuel in the damaged rods is assumed to release 10 percent of both the noble gas and iodine inventories.

A maximum equilibrium inventory of fission products in the core based on 1000 days of continuous operation at 3458 Mwt is assumed. The rods which fail are assumed to have operated at 1.5 times the average fuel rod power. No delay time is considered between departure from the above power condition and the initiation of the accident.

Fission Product Transport to the Environment

The transport pathway consists of carryover with steam to the turbine-condenser prior to MSIV closure and subsequent release from the condenser to the environment. The release to the environment is via condenser leakage if the condenser is isolated or via the Mechanical Vacuum Pump (MVP) flow path if the pump is operating.

Based on the conservative analysis assumptions of Standard Review Plan 15.4.9, it is assumed that 100 percent of the noble gases and 10 percent of the iodines released to the coolant are carried to the condenser before MSIV closure is complete.

Of the activity reaching the condenser, 100 percent of the noble gases and 10 percent of the iodines (the iodine fraction is diminished by partitioning and plateout) are assumed to remain airborne in the condenser and available for release.

The activity release path from the condenser and potential radiological consequences vary depending on whether or not the MVP is operating. Therefore, separate radiological evaluations have been performed for accident scenarios with and without assumed operation of the MVP. Analyses have been performed for each of the three cases described below.

Case 1 assumes that the MVP flow path is isolated for the duration of the accident. This assessment follows the standard scenario described in Reference 7. In accordance with Reference 7, the activity airborne in the condenser is assumed to leak directly to the environment at a rate of 1 percent per day. Radioactive decay is accounted for during residence in the condenser and is neglected after release to the environment. No credit is taken for holdup and decay in the turbine building. The release from the turbine building is assumed to occur at ground level. The resulting time-dependent activity airborne in the condenser is presented in Table 15.4-7. The cumulative release to the environment is presented in Table 15.4-8.

Case 2 is based on manual isolation of the MVP. Operation of the MVP is not terminated by an automatic isolation signal in event of a CRDA. Consequently, plant procedures require that operator action be taken to manually trip the pump

no more than 15 minutes after the event has occurred. The design basis analysis for a CRDA which occurs at a time when condenser vacuum is being maintained by the MVP therefore assumes that activity will be removed from the condenser by the MVP for a period of 15 minutes. Activity removed by the MVP will be discharged to the environment as an elevated release from the plant main stack. The maximum removal rate by the MVP is estimated to be 3157 percent per day. Subsequent to isolation of the MVP, the condenser is assumed to leak at a rate of 1 percent per day at ground level. The activity airborne in the condenser for this case is presented in Table 15.4-9. The cumulative release to the environment is presented in Table 15.4-10.

Case 3 has been included to illustrate the upper bound consequences calculated to result in the unlikely event that manual isolation of the MVP did not occur. In this case, the MVP is assumed to run continuously and discharge activity via the plant stack at the maximum removal rate of 3157 percent per day for the duration of the accident. The activity airborne in the condenser for this case is presented in Table 15.4-11. The cumulative release to the environment is presented in Table 15.4-12.

Results

The calculated radiological consequences for each of the three cases evaluated are presented in Table 15.4-13. The results of the design basis analysis for Case 1 (no MVP operation) and for Case 2 (manual MVP isolation within 15 minutes) are, in each case, well within the guidelines of 10 CFR 100. The results for hypothetical Case 3, based on failure of the required manual isolation to occur, are also small in comparison to 10 CFR 100 limits.

A radiological reassessment (Reference 27 and 30) of the control rod drop accident for Siemens fuel was made based on (a) the core fission product inventory for higher burnup fuel, (b) reactor core power of 3458 MWt (105%), and (c) thyroid dose conversion factors derived from ICRP-30. The reassessment shows a maximum relative increase in radiological exposure of 1.08. But, for an assumed failure of 850 Siemens 9x9 fuel rods with 72 rods per fuel bundle and the conservative assumption of 0.0077 fraction fuel melt, the relative change in radiological exposures is (1.08)(850/770)(60/72) = 0.99. That is, the radiological exposures from an assumed failure of 850 Siemens 9x9 fuel rods is 0.99 times the radiological exposures shown in Table 15.4-13.

A radiological evaluation (References 19 and 24) of the control rod drop accident for 105% licensed core thermal uprate was done. This evaluation involved scaling of the results in Table 15.4-13 based on an increase of power from 3458 MWt to 3559 MWt (102% of 3489 MWt). It was determined that a scaling factor of 1.029 (3559/3458) should be applied to all isotopes except I-131. The scaling factor was applied to the whole body dose. The higher burnup results in an additional increase

to I-131 of 1.097. This results in an I-131 unique scaling factor of 1.129 (1.097 x 1.029) which is applied to the thyroid dose.

As outlined in Reference 20, the reload analyses performed for L1C09 and L2C08 bound the extended burnup to support 24-month fuel cycles. These analyses included, in part, the fuel handling accident, cask drop accident, and the control rod drop accident. The dose consequences for the fuel handling and control rod drop accidents remain bounded by the corresponding Updated Final Safety Analysis Report (UFSAR) Chapter 15 analyses previously performed for General Electric Company (GE) supplied fuel. The dose consequences for the cask drop accident increased by an insignificant amount.

Below is the listing of the parameters (References 20, 26 and 27) that are the basis for the high exposure ATRIUM-9B Source Term used in the L1C09 and L2C08 analyses for ATRIUM-9B fuel. The units given below are Megawatt-days per Metric Ton (tonne) of Uranium (MWd /MTU) and Megawatts thermal (MWt).

CRDA: 60,000 MWd/MTU; 3458 MWth core power; 2034 core residence days.

Below is the listing of the parameters (Reference 30) that are the basis for the high exposure ATRIUM-10 Source Term used in the analyses for ATRIUM-10 fuel at 3489 MWt core power. Reference 34 confirms that these values remain applicable for the ATRIUM-10 reload fuel and the ATRIUM 10XM LTA at MUR conditions.

CRDA: 60,000 MWd/MTU; 3910 MWT; 2078 core residence days.

15.4.10 References

1. C. J. Paone, "Banked Position Withdrawal Sequence," NEDO-21231, January 1997.

2. R. C. Stirn, et al., "Rod Drop Accident Analysis for Large BWRs," NEDO-10527, Supplement 1, July 1972.

3. R. C. Stirn, "Rod Drop Accident Analysis for Large Boiling Water Reactors Addendum No. 2 Exposed Cores," NEDO-10527, Supplement 2, January 1973.

4. Intentionally deleted.

5. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, (Unit 1: Rev. 22, Unit 2 Rev: 20).

6. R. C. Stirn, et al., "Rod Drop Accident Analysis for Large Boiling Water Reactors," NEDO-10527, March 1972.

7. USNRC Standard Review Plan, NUREG-0800, July 1981.

8. H. A. Careway, V. D. Nguyen, P. P. Stancavage, "Radiological Accident Evaluation - The CONAC03 Code," December 1981.

9. Deleted

10. Letter, N. Sadeghi (GE) "25% APRM Analytical Limit Scram vs. 15% APRM Scram Setpoint," DRF A12-00038-4, November 13, 1996.

11. GE Report B33-00296-03P, "Updated Transient Analysis: Abnormal Start-up Of An Idle Recirculation Loop For LaSalle County Nuclear Station, Units 1 and 2," March 1998.

12. NEDO-10802, "Analytical Methods of Plant Transient Evaluations For The General Electric Boiling Water Reactor," April 1973.

13. NFM:BSS 98-070, "Generic Application of Steady-State Rod Withdrawal Error Event Methods to Control Rod Drive Withdrawal Speed up to 5.14 inches per second," July 10, 1998.

14. Intentionally Deleted.

15. Deleted

16. Commonwealth Edison Topical Report NFSR-0091, “Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods,” Revision 0 and Supplements 1 and 2, December 1991, March 1992, and May 1992, respectively.

17. ComEd Document, BSA-M-96-08, Rev. 2, “New BWR Fuel Radiological Source Term Assessment for LaSalle”, November 21,1996.

18. ComEd Letter, “Dresden Nuclear Power Station Units 2 and 3, Quad Cities Nuclear Power Station Units 1 and 2, LaSalle County Nuclear Power Station Units 1 and 2, Revised Control Rod Sequencing Methods, NRC Docket Nos. 50-237/249, 50-254/265, and 50-373/374,” P. L. Piet to T. E. Murley, January 27, 1993.

19. Letter from R. M. Krich, Commonwealth Edison (ComEd) Company, to U.S. NRC, “Request for License Amendment for Power Uprate Operation,” dated July 14, 1999, with Attachment E: General Electric Nuclear Energy, Licensing Topical Report NEDC-32701P, Revision 2, "Power Uprate Safety Analysis Report for LaSalle County Station, Units 1 and 2," dated July 1999 (Proprietary).

20. Letter from C. G. Pardee, Commonwealth Edison (ComEd) Company, to U.S. NRC, "Supplement to the License Amendment Request for Power Uprate Operation," dated 04/14/2000.

21. Letter from D. M. Skay, U.S. NRC, to O. D. Kingsley, Commonwealth Edison (ComEd) Company, "LaSalle - Issuance Of Amendments Regarding Power Uprate (TAC NOS. MA6070 AND MA6071)," dated May 9, 2000 (OL Amendments 140/125).

22. "Safety Evaluation By The Office Of Nuclear Reactor Regulation Related To Amendment No. 140 To Facility Operating License No. NPF-11 And Amendment No. 125 To Facility Operating License No. NPF-18; Commonwealth Edison Company Lasalle County Station, Units 1 And 2; Docket Nos. 50-373 And 50-374" dated May 9, 2000.

23. Power Uprate Project Task 900, "Transient Analyses," GE-NE-A1300384-08, Revision 1, September 1999.

24. ENDIT H041, S&L Task 21b, "Radiological - Chapter 15 Accidents."

25. Power Uprate Project Task 904, "Accident Analyses (non-LOCA)," GE-NE-A1300384-36-01, Revision 0, August 1999.

26. Memorandum NFM:BSA 00-028 from R. W. Tsai, Nuclear Fuel Management, to F. A. Spangenberg III, Regulatory Assurance, "Summary of Key Input Parameters for the High Exposure Siemens ATRIUM-9B Source Term," dated April 13, 2000 DG00-000413.

27. Letter JH:96:188 from J. H. Riddle (Siemens) to Dr. R. J. Chin (ComEd), "Radioactive Release Analysis Source Term Values," dated May 20, 1996 DG96-001013.

28. Deleted

29. "Improved BPWS Control Rod Insertion Process," NEDO-33091-A Revision 2, July 2004.

30. Letter, D.E. Garber (FANP) to F.W. Trikur (Exelon), "Radioactive Source Term Values for ATRIUM-10 Fuel at LaSalle," DEG:01:170 dated October 22, 2001.

31. Intentionally Deleted.

32. Deleted

33. Deleted

34. Design Analysis L-003509, Revision 0, "Evaluation of Appendix R, Station Blackout, Containment and Source Terms for LaSalle MUR Power Uprate," July 2010.

TABLE 15.4-1

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TABLE 15.4-2

SUMMARY OF RESULTS FOR DETAILED AND POINT KINETICS EVALUATIONS OF CONTINUOUS ROD WITHDRAWAL IN THE STARTUP RANGE

CASE	CONTROL ROD WORTH (%Δk)	$\bar{h}f$(cal/gm)	P/A [c]	$\hat{h}$(cal/gm)
1	1.6	17.3	24.2	42.7
2	2.0	17.3	30.9	50.0
3	2.5	17.2	46.0	58.5
4	1.6 [a]	18.3	19.7[b]	56.2
5	1.6 [d]	18.3	19.7	59.6

(a) Detailed transient calculation. All other data reported are for point kinetics calculations.

(b) The P/A = 19.7 is the initial value. For the detailed analysis this value will decrease during the course of the transient since the power shape will flatten due to Doppler feedback.

(c) P/A = global peaking factor (Radial x Axial).

(d) Point kinetics calculation with IRM initiated scram and 3-D simulator global peaking.

TABLE 15.4-3

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TABLE 15.4-4

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TABLE 15.4-5

INCREMENTAL ROD WORTHS USING BANK-POSITION
WITHDRAWAL SEQUENCE WORST CASE FOR EACH OF THE GIVEN ROD GROUPS
(INITIAL CYCLE ANALYSES)

CORE CONDITION	CONTROL ROD GROUP*	BANKED AT NOTCH	CONTROL ROD (X,Y)	DROPS FROM-TO	Δk
BOC-1 Sequence A G1 through G4 W/D all others at 0	7	12	26-35	0 - 48	.004658
BOC-1 Sequence A G1 through G4 W/D all others at 0	8	12	26-43	0 - 48	.002518
BOC-1 Sequence A G1 through G4 W/D G5 through G8 at 12 G10 at 0	9	4	30-31	0 - 8	.002154
BOC-1 Sequence A G1 through G4 W/D G5 through G8 at 12 G9 at 0	10	4	22-31	0 - 8	.002141

NOTE: The following assumptions were made to ensure that the rod worths were conservatively high for the banked-position withdrawal sequence:

a. BOC,
b. hot startup,
c. no xenon.

* For definition of rod groups, see Figures 4.3-27 and 4.3-28.

TABLE 15.4-6

(Sheet 1 of 2)

INPUT PARAMETERS AND INITIAL CONDITIONS: ROD DROP ACCIDENT

	DESIGN-BASIS ASSUMPTIONS
I. Data and assumptions used to estimate radioactive source from postulated accidents	
A. Power level corresponding to 102% of rated core thermal power	3559 Mwt (Reference 24)
B. Radial peaking factor	1.5
C. Fuel Damaged	770 rods (GE), 850 rods (FANP)
D. Fraction of damaged fuel assumed to melt	0.0077
E. Fractions of fission products in damaged fuel assumed release to coolant	Subsection 15.4.9.5
F. Iodine fractions (1) Organic (2) Elemental (3) Particulate	 0 1 0
G. Release of activity to environment by nuclide (1) Case 1 (2) Case 2 (3) Case 3	 Table 15.4-8 Table 15.4-10 Table 15.4-12
II. Data and assumptions used to estimate activity released	
A. Condenser leak rate (%/day)	1.0
B. Mechanical vacuum pump operating period (1) Case 1 (2) Case 2 (3) Case 3	 NA 15 min continuous
C. Condenser discharge rate via MVP (%/day) (1) Case 1 (2) Case 2 (3) Case 3	 0 3157 3157

TABLE 15.4-6

(Sheet 2 of 2)

INPUT PARAMETERS AND INITIAL CONDITIONS: ROD DROP ACCIDENT

III.	Dispersion data	
A.	Exclusion Area Boundary and LPZ distances (m)	Section 15.0.5, Item b
B.	Ground Level Release χ/Q's for time intervals of (1) 0 - 2 hr - EAB/LPZ (2) 2 - 8 hr - LPZ (3) 8 - 24 hr - LPZ (4) 1 - 4 day - LPZ (5) 4 - 30 day - LPZ	 5.1E-4/1.0E-5 1.0E-5 6.7E-6 2.6E-6 6.5E-7
C.	Elevated Release χ/Q's for time intervals of (1) 0 - 1/2 hr - EAB/LPZ (2) 1/2 - 2 hr - EAB/LPZ (3) 2 - 8 hr - LPZ (4) 8 - 24 hr - LPZ (5) 1 - 4 day - LPZ (6) 4 - 30 day LPZ	 8.4E-5/8.9E-6 2.6E-6/1.6E-6 9.2E-7 5.5E-7 2.5E-7 8.2E-8
IV.	Dose Data	
A.	Method of dose calculation	Reference 8
B.	Dose conversion assumptions	Reference 8
C.	Peak activity concentrations in condenser (1) Case 1 (2) Case 2 (3) Case 3	 Table 15.4-7 Table 15.4-9 Table 15.4-11
D.	Doses	Table 15.4-13

TABLE 15.4-7

CONTROL ROD DROP ACCIDENT - DESIGN BASIS (CASE 1 - NO MVP OPERATION)
AIRBORNE ACTIVITY IN CONDENSER (CURIES)

ISOTOPE	1-MIN	15-MIN	30-MIN	2-HOUR	4-HOUR	8-HOUR	12-HOUR	1-DAY	4-DAY	30-DAY
I-131	2.29E 03	2.28E 03	2.28E 03	2.27E 03	2.25E 03	2.21E 03	2.18E 03	2.08E 03	1.56E 03	1.28E 02
I-132	3.33E 03	3.10E 03	2.87E 03	1.83E 03	1.00E 03	2.99E 02	8.94E 01	2.39E 00	8.76E-10	0.0
I-133	4.78E 03	4.74E 03	4.70E 03	4.47E 03	4.18E 03	3.65E 03	3.19E 03	2.13E 03	1.87E 02	1.35E-07
I-134	5.20E 03	4.32E 03	3.54E 03	1.08E 03	2.22E 02	9.40E 00	3.97E-01	3.00E-05	0.0	0.0
I-135	4.51E 03	4.40E 03	4.28E 03	3.66E 03	2.96E 03	1.95E 03	1.28E 03	3.61E 02	1.84E-01	0.0
TOTAL I	2.01E 04	1.88E 04	1.77E 04	1.33E 04	1.06E 04	8.12E 03	6.74E 03	4.57E 03	1.74E 03	1.28E 02
KR-83M	2.81E 04	2.57E 04	2.34E 04	1.32E 04	6.21E 03	1.36E 03	2.99E 02	3.16E 00	4.41E-12	0.0
KR-85M	6.05E 04	5.84E 04	5.62E 04	4.45E 04	3.26E 04	1.75E 04	9.43E 03	1.47E 03	2.06E-02	0.0
KR-85	2.72E 03	2.72E 03	2.72E 03	2.72E 03	2.71E 03	2.71E 03	2.70E 03	2.69E 03	2.61E 03	2.00E 03
KR-87	1.15E 05	1.02E 05	8.87E 04	3.91E 04	1.31E 04	1.48E 03	1.67E 02	2.40E-01	0.0	0.0
KR-88	1.64E 05	1.55E 05	1.46E 05	1.01E 05	6.20E 04	2.33E 04	8.77E 03	4.67E 02	1.06E-05	0.0
KR-89	1.65E 05	7.72E 03	2.91E 02	8.26E-07	0.0	0.0	0.0	0.0	0.0	0.0
XE-131M	1.43E 03	1.42E 03	1.42E 03	1.42E 03	1.41E 03	1.39E 03	1.38E 03	1.33E 03	1.09E 03	1.84E 02
XE-133M	2.08E 04	2.07E 04	2.06E 04	2.02E 04	1.97E 04	1.86E 04	1.76E 04	1.50E 04	5.62E 03	1.15E 00
XE-133	4.98E 05	4.98E 05	4.97E 05	4.92E 05	4.87E 05	4.75E 05	4.64E 05	4.32E 05	2.82E 05	7.00E 03
XE-135M	8.99E 04	4.83E 04	2.49E 04	4.61E 02	2.27E 00	5.47E-05	1.32E-09	0.0	0.0	0.0
XE-135	6.43E 04	6.32E 04	6.20E 04	5.53E 04	4.74E 04	3.49E 04	2.57E 04	1.02E 04	4.07E 01	0.0
XE-137	3.65E 05	2.90E 04	1.92E 03	1.62E-04	6.00E-14	0.0	0.0	0.0	0.0	0.0
XE-138	3.96E 05	1.99E 05	9.58E 04	1.17E 03	3.30E 00	2.63E-05	2.09E-10	0.0	0.0	0.0
TOTAL NG	1.97E 06	1.21E 06	1.02E 06	7.72E 05	6.72E 05	5.77E 05	5.30E 05	4.63E 05	2.91E 05	9.18E 03

NOTE: For power uprate to 3559 MWt (102% of 3489 MWt), these values should be increased by a factor 1.029 for all isotopes except I-131. I-131 should be increased by a factor of 1.129 (Reference 24).

TABLE 15.4-8

CONTROL ROD DROP ACCIDENT - DESIGN BASIS (CASE 1 - NO MVP OPERATION)
ACTIVITY RELEASED TO ENVIRONMENT (CURIES)

ISOTOPE	1-MIN	15-MIN	30-MIN	2-HOUR	4-HOUR	8-HOUR	12-HOUR	1-DAY	4-DAY	30-DAY
I-131	1.59E-02	2.38E-01	4.76E-01	1.90E 00	3.78E 00	7.50E 00	1.12E 01	2.18E 01	7.59E 01	2.24E 02
I-132	2.32E-02	3.35E-01	6.46E-01	2.09E 00	3.23E 00	4.20E 00	4.49E 00	4.61E 00	4.62E 00	4.62E 00
I-133	3.32E-02	4.96E-01	9.88E-01	3.85E 00	7.46E 00	1.40E 01	1.97E 01	3.28E 01	5.67E 01	5.91E 01
I-134	3.63E-02	4.98E-01	9.06E-01	2.20E 00	2.66E 00	2.77E 00	2.77E 00	2.77E 00	2.77E 00	2.77E 00
I-135	3.13E-02	4.64E-01	9.17E-01	3.39E 00	6.14E 00	1.02E 01	1.28E 01	1.64E 01	1.79E 01	1.79E 01
TOTAL I	1.40E-01	2.03E 00	3.93E 00	1.34E 01	2.33E 01	3.86E 01	5.09E 01	7.84E 01	1.58E 02	3.09E 02
KR-83M	1.96E-01	2.81E 00	5.37E 00	1.65E 01	2.43E 01	2.96E 01	3.07E 01	3.11E 01	3.11E 01	3.11E 01
KR-85M	4.21E-01	6.20E 00	1.22E 01	4.35E 01	7.54E 01	1.16E 02	1.38E 02	1.59E 02	1.63E 02	1.63E 02
KR-85	1.89E-02	2.83E-01	5.66E-01	2.26E 00	4.53E 00	9.04E 00	1.36E 01	2.70E 01	1.07E 02	7.03E 02
KR-87	8.05E-01	1.13E 01	2.12E 01	5.91E 01	7.89E 01	8.78E 01	8.88E 01	8.90E 01	8.90E 01	8.90E 01
KR-88	1.14E 00	1.67E 01	3.23E 01	1.09E 02	1.75E 02	2.41E 02	2.66E 02	2.80E 02	2.81E 02	2.81E 02
KR-89	1.28E 00	6.27E 00	6.51E 00	6.52E 00	6.52E 00	6.52E 00	6.52E 00	6.52E 00	6.52E 00	6.52E 00
XE-131M	9.90E-03	1.48E-01	2.97E-01	1.18E 00	2.36E 00	4.70E 00	7.01E 00	1.38E 01	4.99E 01	1.82E 02
XE-133M	1.44E-01	2.16E 00	4.31E 00	1.71E 01	3.37E 01	6.56E 01	9.58E 01	1.77E 02	4.64E 02	6.36E 02
XE-133	3.46E 00	5.19E 01	1.04E 02	4.13E 02	8.21E 02	1.62E 03	2.40E 03	4.64E 03	1.52E 04	3.46E 04
XE-135M	6.38E-01	7.15E 00	1.08E 01	1.47E 01	1.47E 01	1.47E 01	1.47E 01	1.47E 01	1.47E 01	1.47E 01
XE-135	4.47E-01	6.65E 00	1.32E 01	4.98E 01	9.25E 01	1.60E 02	2.11E 02	2.94E 02	3.50E 02	3.50E 02
XE-137	2.78E 00	1.57E 01	1.67E 01	1.68E 01	1.68E 01	1.68E 01	1.68E 01	1.68E 01	1.68E 01	1.68E 01
XE-138	2.82E 00	3.07E 01	4.54E 01	5.88E 01	5.90E 01	5.90E 01	5.90E 01	5.90E 01	5.90E 01	5.90E 01
TOTAL NG	1.42E 01	1.58E 02	2.73E 02	8.08E 02	1.40E 03	2.43E 03	3.35E 03	5.81E 03	1.68E 04	3.71E 04

NOTE: For power uprate to 3559 MWt (102% of 3489 MWt), these values should be increased by a factor 1.029 for all isotopes except I-131. I-131 should be increased by a factor of 1.129 (Reference 24).

TABLE 15.4-9

CONTROL ROD DROP ACCIDENT - DESIGN BASIS (CASE 2 - MVP TRIPPED AT 15 MINUTES)
AIRBORNE ACTIVITY IN CONDENSER (CURIES)

ISOTOPE	1-MIN	15-MIN	30-MIN	2-HOUR	4-HOUR	8-HOUR	12-HOUR	1-DAY	4-DAY	30-DAY
I-131	2.24E 03	1.64E 03	1.64E 03	1.63E 03	1.62E 03	1.59E 03	1.57E 03	1.50E 03	1.12E 03	9.18E 01
I-132	3.25E 03	2.23E 03	2.07E 03	1.32E 03	7.20E 02	2.15E 02	6.44E 01	1.72E 00	6.31E-10	0.0
I-133	4.68E 03	3.41E 03	3.39E 03	3.22E 03	3.01E 03	2.63E 03	2.30E 03	1.53E 03	1.35E 02	9.69E-08
I-134	5.08E 03	3.11E 03	2.55E 03	7.79E 02	1.60E 02	6.77E 00	2.86E-01	2.16E-05	0.0	0.0
I-135	4.41E 03	3.17E 03	3.08E 03	2.63E 03	2.13E 03	1.40E 03	9.19E 02	2.60E 02	1.33E-01	0.0
TOTAL I	1.97E 04	1.36E 04	1.27E 04	9.58E 03	7.64E 03	5.85E 03	4.85E 03	3.29E 03	1.26E 03	9.18E 01
KR-83M	2.75E 04	1.85E 04	1.68E 04	9.54E 03	4.47E 03	9.80E 02	2.15E 02	2.27E 00	3.17E-12	0.0
KR-85M	5.92E 04	4.20E 04	4.04E 04	3.20E 04	2.35E 04	1.26E 04	6.79E 03	1.06E 03	1.49E-02	0.0
KR-85	2.66E 03	1.96E 03	1.96E 03	1.95E 03	1.95E 03	1.95E 03	1.95E 03	1.94E 03	1.88E 03	1.44E 03
KR-87	1.13E 05	7.31E 04	6.38E 04	2.82E 04	9.46E 03	1.07E 03	1.20E 02	1.73E-01	0.0	0.0
KR-88	1.61E 05	1.12E 05	1.05E 05	7.28E 04	4.47E 04	1.68E 04	6.32E 03	3.36E 02	7.60E-06	0.0
KR-89	1.61E 05	5.56E 03	2.09E 02	5.95E-07	0.0	0.0	0.0	0.0	0.0	0.0
XE-131M	1.39E 03	1.03E 03	1.02E 03	1.02E 03	1.01E 03	1.00E 03	9.92E 02	9.59E 02	7.81E 02	1.32E 02
XE-133M	2.03E 04	1.49E 04	1.49E 04	1.46E 04	1.42E 04	1.34E 04	1.27E 04	1.08E 04	4.05E 03	8.25E-01
XE-133	4.87E 05	3.58E 05	3.58E 05	3.54E 05	3.50E 05	3.42E 05	3.34E 05	3.11E 05	2.03E 05	5.04E 03
XE-135M	8.79E 04	3.48E 04	1.79E 04	3.32E 02	1.63E 00	3.94E-05	9.51E-10	0.0	0.0	0.0
XE-135	6.30E 04	4.55E 04	4.46E 04	3.98E 04	3.41E 04	2.51E 04	1.85E 04	7.35E 03	2.93E 01	0.0
XE-137	3.57E 05	2.09E 04	1.38E 03	1.17E-04	4.32E-14	0.0	0.0	0.0	0.0	0.0
XE-138	3.87E 05	1.44E 05	6.89E 04	8.44E 02	2.38E 00	1.89E-05	1.50E-10	0.0	0.0	0.0
TOTAL NG	1.93E 06	8.72E 05	7.35E 05	5.55E 05	4.84E 05	4.15E 05	3.82E 05	3.34E 05	2.10E 05	6.61E 03

NOTE: For power uprate to 3559 MWt (102% of 3489 MWt), these values should be increased by a factor 1.029 for all isotopes except I-131. I-131 should be increased by a factor of 1.129 (Reference 24).

TABLE 15.4-10

CONTROL ROD DROP ACCIDENT - DESIGN BASIS (CASE 2 - MVP TRIPPED AT 15 MINUTES)
ACTIVITY RELEASED TO ENVIRONMENT (CURIES)

ISOTOPE	1-MIN	15-MIN	30-MIN	2-HOUR	4-HOUR	8-HOUR	12-HOUR	1-DAY	4-DAY	30-DAY
I-131	4.96E 01	6.41E 02	6.41E 02	6.42E 02	6.43E 02	6.46E 02	6.49E 02	6.56E 02	6.95E 02	8.02E 02
I-132	7.23E 01	9.04E 02	9.04E 02	9.05E 02	9.06E 02	9.07E 02	9.07E 02	9.07E 02	9.07E 02	9.07E 02
I-133	1.04E 02	1.34E 03	1.34E 03	1.34E 03	1.34E 03	1.34E 03	1.35E 03	1.36E 03	1.38E 03	1.38E 03
I-134	1.13E 02	1.35E 03	1.35E 03	1.35E 03	1.35E 03	1.35E 03	1.35E 03	1.35E 03	1.35E 03	1.35E 03
I-135	9.78E 01	1.25E 03	1.25E 03	1.25E 03	1.25E 03	1.26E 03	1.26E 03	1.26E 03	1.26E 03	1.26E 03
TOTAL I	4.37E 02	5.48E 03	5.48E 03	5.48E 03	5.49E 03	5.50E 03	5.51E 03	5.53E 03	5.59E 03	5.70E 03
KR-83M	6.11E 02	7.58E 03	7.58E 03	7.59E 03	7.60E 03	7.60E 03	7.60E 03	7.60E 03	7.60E 03	7.60E 03
KR-85M	1.31E 03	1.67E 04	1.67E 04	1.67E 04	1.68E 04	1.68E 04	1.68E 04	1.68E 04	1.68E 04	1.68E 04
KR-85	5.89E 01	7.62E 02	7.62E 02	7.63E 02	7.65E 02	7.68E 02	7.71E 02	7.81E 02	8.38E 02	1.27E 03
KR-87	2.51E 03	3.06E 04	3.06E 04	3.07E 04	3.07E 04	3.07E 04	3.07E 04	3.07E 04	3.07E 04	3.07E 04
KR-88	3.57E 03	4.49E 04	4.49E 04	4.50E 04	4.50E 04	4.51E 04	4.51E 04	4.51E 04	4.51E 04	4.51E 04
KR-89	4.00E 03	1.82E 04	1.82E 04	1.82E 04	1.82E 04	1.82E 04	1.82E 04	1.82E 04	1.82E 04	1.82E 04
XE-131M	3.09E 01	4.00E 02	4.00E 02	4.00E 02	4.01E 02	4.03E 02	4.04E 02	4.09E 02	4.35E 02	5.30E 02
XE-133M	4.50E 02	5.81E 03	5.81E 03	5.82E 03	5.84E 03	5.86E 03	5.88E 03	5.94E 03	6.15E 03	6.27E 03
XE-133	1.08E 04	1.40E 05	1.40E 05	1.40E 05	1.40E 05	1.41E 05	1.41E 05	1.43E 05	1.50E 05	1.64E 05
XE-135M	1.99E 03	1.96E 04	1.96E 04	1.96E 04	1.96E 04	1.96E 04	1.96E 04	1.96E 04	1.96E 04	1.96E 04
XE-135	1.40E 03	1.79E 04	1.79E 04	1.79E 04	1.80E 04	1.80E 04	1.80E 04	1.81E 04	1.81E 04	1.81E 04
XE-137	8.68E 03	4.50E 04	4.50E 04	4.50E 04	4.50E 04	4.50E 04	4.50E 04	4.50E 04	4.50E 04	4.50E 04
XE-138	8.79E 03	8.42E 04	8.42E 04	8.42E 04	8.42E 04	8.42E 04	8.42E 04	8.42E 04	8.42E 04	8.42E 04
TOTAL NG	4.42E 04	4.31E 05	4.31E 05	4.32E 05	4.32E 05	4.33E 05	4.33E 05	4.35E 05	4.43E 05	4.58E 05

NOTE: For power uprate to 3559 MWt (102% of 3489 MWt), these values should be increased by a factor 1.029 for all isotopes except I-131.
I-131 should be increased by a factor of 1.129 (Reference 24).

TABLE 15.4-11

CONTROL ROD DROP ACCIDENT - DESIGN BASIS (CASE 3 - MVP OPERATING CONTINUOUSLY)

AIRBORNE ACTIVITY IN CONDENSER (CURIES)

ISOTOPE	1-MIN	15-MIN	30-MIN	2-HOUR	4-HOUR	8-HOUR	12-HOUR	1-DAY	4-DAY	30-DAY
I-131	2.24E 03	1.64E 03	1.18E 03	1.64E 02	1.17E 01	5.98E-02	3.06E-04	4.09E-11	0.0	0.0
I-132	3.25E 03	2.23E 03	1.49E 03	1.32E 02	5.19E 00	8.07E-03	1.25E-05	4.71E-14	0.0	0.0
I-133	4.68E 03	3.41E 03	2.44E 03	3.22E 02	2.17E 01	9.86E-02	4.48E-04	4.19E-11	0.0	0.0
I-134	5.08E 03	3.11E 03	1.84E 03	7.80E 01	1.16E 00	2.54E-04	5.57E-08	0.0	0.0	0.0
I-135	4.41E 03	3.17E 03	2.22E 03	2.64E 02	1.54E 02	5.25E-02	1.79E-04	7.10E-12	0.0	0.0
TOTAL I	1.97E 04	1.36E 04	9.17E 03	9.59E 02	5.52E 01	2.19E-01	9.45E-04	8.99E-11	0.0	0.0
KR-83M	2.75E 04	1.85E 04	1.21E 04	9.55E 02	3.22E 01	3.68E-02	4.19E-05	6.22E-14	0.0	0.0
KR-85M	5.92E 04	4.20E 04	2.91E 04	3.21E 03	1.70E 02	4.74E-01	1.32E-03	2.89E-11	0.0	0.0
KR-85	2.66E 03	1.96E 03	1.41E 03	1.96E 02	1.41E 01	7.31E-02	3.79E-04	5.30E-11	0.0	0.0
KR-87	1.13E 05	7.31E 04	4.59E 04	2.82E 03	6.83E 01	4.00E-02	2.35E-05	4.73E-15	0.0	0.0
KR-88	1.61E 05	1.12E 05	7.56E 04	7.29E 03	3.22E 02	6.30E-01	1.23E-03	9.18E-12	0.0	0.0
KR-89	1.61E 05	5.56E 03	1.51E 02	5.96E-08	0.0	0.0	0.0	0.0	0.0	0.0
XE-131M	1.39E 03	1.03E 03	7.38E 02	1.02E 02	7.33E 00	3.76E-02	1.93E-04	2.62E-11	0.0	0.0
XE-133M	2.03E 04	1.49E 04	1.07E 04	1.46E 03	1.02E 02	5.03E-01	2.48E-03	2.95E-10	0.0	0.0
XE-133	4.87E 05	3.58E 05	2.57E 05	3.55E 04	2.53E 03	1.28E 01	6.51E-02	8.51E-09	0.0	0.0
XE-135M	8.79E 04	3.48E 04	1.29E 04	3.33E 01	1.18E-02	1.48E-09	0.0	0.0	0.0	0.0
XE-135	6.30E 04	4.55E 04	3.21E 04	3.98E 03	2.46E 02	9.42E-01	3.60E-03	2.01E-10	0.0	0.0
XE-137	3.57E 05	2.09E 04	9.94E 02	1.17E-05	0.0	0.0	0.0	0.0	0.0	0.0
XE-138	3.87E 05	1.44E 05	4.96E 04	8.45E 01	1.72E-02	7.10E-10	0.0	0.0	0.0	0.0
TOTAL NG	1.93E 06	8.72E 05	5.29E 05	5.56E 04	3.49E 03	1.56E 01	7.44E-02	9.12E-09	0.0	0.0

NOTE: For power uprate to 3559 MWt (102% of 3489 MWt), these values should be increased by a factor 1.029 for all isotopes except I-131.

I-131 should be increased by a factor of 1.129 (Reference 24).

TABLE 15.4-12

CONTROL ROD DROP ACCIDENT - DESIGN BASIS (CASE 3 - MVP OPERATING CONTINUOUSLY)

ACTIVITY RELEASED TO ENVIRONMENT (CURIES)

ISOTOPE	1-MIN	15-MIN	30-MIN	2-HOUR	4-HOUR	8-HOUR	12-HOUR	1-DAY	4-DAY	30-DAY
I-131	4.96E 01	6.41E 02	1.10E 03	2.12E 03	2.27E 03	2.28E 03	2.28E 03	2.28E 03	2.28E 03	2.28E 03
I-132	7.23E 01	9.04E 02	1.51E 03	2.61E 03	2.72E 03	2.72E 03	2.72E 03	2.72E 03	2.72E 03	2.72E 03
I-133	1.04E 02	1.34E 03	2.29E 03	4.35E 03	4.64E 03	4.66E 03	4.66E 03	4.66E 03	4.66E 03	4.66E 03
I-134	1.13E 02	1.35E 03	2.14E 03	3.24E 03	3.29E 03	3.29E 03	3.29E 03	3.29E 03	3.29E 03	3.29E 03
I-135	9.78E 01	1.25E 03	2.13E 03	3.94E 03	4.17E 03	4.18E 03	4.18E 03	4.18E 03	4.18E 03	4.18E 03
TOTAL I	4.37E 02	5.48E 03	9.16E 03	1.63E 04	1.71E 04	1.71E 04	1.71E 04	1.71E 04	1.71E 04	1.71E 04
KR-83M	6.11E 02	7.58E 03	1.25E 04	2.12E 04	2.19E 04	2.20E 04	2.20E 04	2.20E 04	2.20E 04	2.20E 04
KR-85M	1.31E 03	1.67E 04	2.83E 04	5.14E 04	5.42E 04	5.43E 04	5.43E 04	5.43E 04	5.43E 04	5.43E 04
KR-85	5.89E 01	7.62E 02	1.31E 03	2.52E 03	2.70E 03	2.72E 03	2.72E 03	2.72E 03	2.72E 03	2.72E 03
KR-87	2.51E 03	3.06E 04	4.99E 04	8.03E 04	8.23E 04	8.23E 04	8.23E 04	8.23E 04	8.23E 04	8.23E 04
KR-88	3.57E 03	4.49E 04	7.53E 04	1.33E 05	1.39E 05	1.39E 05	1.39E 05	1.39E 05	1.39E 05	1.39E 05
KR-89	4.00E 03	1.82E 04	1.87E 04	1.87E 04	1.87E 04	1.87E 04	1.87E 04	1.87E 04	1.87E 04	1.87E 04
XE-131M	3.09E 01	4.00E 02	6.87E 02	1.32E 03	1.42E 03	1.42E 03	1.42E 03	1.42E 03	1.42E 03	1.42E 03
XE-133M	4.50E 02	5.81E 03	9.98E 03	1.91E 04	2.05E 04	2.06E 04	2.06E 04	2.06E 04	2.06E 04	2.06E 04
XE-133	1.08E 04	1.40E 05	2.40E 05	4.61E 05	4.94E 05	4.96E 05	4.96E 05	4.96E 05	4.96E 05	4.96E 05
XE-135M	1.99E 03	1.96E 04	2.68E 04	3.11E 04	3.11E 04	3.11E 04	3.11E 04	3.11E 04	3.11E 04	3.11E 04
XE-135	1.40E 03	1.79E 04	3.05E 04	5.71E 04	6.07E 04	6.09E 04	6.09E 04	6.09E 04	6.09E 04	6.09E 04
XE-137	8.68E 03	4.50E 04	4.72E 04	4.73E 04	4.73E 04	4.73E 04	4.73E 04	4.73E 04	4.73E 04	4.73E 04
XE-138	8.79E 03	8.42E 04	1.13E 05	1.29E 05	1.29E 05	1.29E 05	1.29E 05	1.29E 05	1.29E 05	1.29E 05
TOTAL NG	4.42E 04	4.31E 05	6.54E 05	1.05E 06	1.10E 06	1.11E 06	1.11E 06	1.11E 06	1.11E 06	1.11E 06

NOTE: For power uprate to 3559 MWt (102% of 3489 MWt), these values should be increased by a factor 1.029 for all isotopes except I-131.

I-131 should be increased by a factor of 1.129 (Reference 24).

TABLE 15.4-13

CONTROL ROD DROP ACCIDENT RADIOLOGICAL EFFECTS

		DOSE (REM) (Note 1)			
		EAB (2HR)		LPZ (30 DAY)	
Case	Mechanical Vacuum Pump Operation	Thyroid	Whole Body	Thyroid	Whole Body
1	None	0.72	0.052	0.17	0.0030
2	15 Minutes	41.3	5.9	4.5	0.63
3	Continuous	72.1	8.7	8.8	0.99
10 CFR 100 Limits		300	25	300	25

Note 1: For 105% licensed core thermal power uprate to 3489 MWt, the most limiting dose consequence (Case 2) was evaluated at 3559 MWt (i.e., 102% of 3489 MWt). It was determined that the whole body dose increases by 1.029 and the thyroid dose by 1.129 (See Section 15.4.9.5). This results in a 2 hour EAB thyroid dose of 46.7 rem and a whole body dose of 6.07 rem for Case 2. (Reference 24)

15.5 INCREASE IN REACTOR COOLANT INVENTORY

15.5.1 Inadvertent HPCS Pump Start

The Inadvertent HPCS Pump Start event is not analyzed for reload cores. The analysis and results presented are for the initial core.

This transient event typically is not part of the standard reload core licensing analysis. As discussed in reference 2, this event is bounded by the Feedwater Controller Failure Maximum Demand which was re-evaluated for Power Uprate and is discussed in section 15.B.3.1.

15.5.1.1 Identification of Causes and Frequency Classification

Inadvertent startup of the HPCS system is caused by manual operator action and is analyzed because it is the largest auxiliary source of cold water. The event introduces cold water into the upper core plenum and quenches some of the steam, thus causing some depressurization. In automatic mode, the recirculation flow control modulates core flow to compensate for power level changes; in manual mode no flow adjustments are made and the excursions of related system variables are greater. The inadvertent start of an HPCS pump is estimated to occur with moderate frequency.

15.5.1.2 Sequence of Events and Systems Operation

The sequence of events for the inadvertent start of the HPCS (Figure 15.5-1) is as follows:

Time (sec)	Event
0	Begin HPCS cold water injection.
6	Full flow established for HPCS.
10+	Depressurization effect stabilized.

In order to simulate this expected sequence of events, the analysis assumes normal functioning of the plant instrumentation and controls which directly respond to this event, namely the pressure regulator and the vessel level control.

Inadvertent operation of the HPCS results in a mild depressurization. Corrective action by the pressure regulator and/or level control is expected to establish a new stable operating state. The effect of a single failure in the pressure regulator will aggravate the transient depending upon the nature of the failure. These pressure regulator failures have been analyzed in Subsections 15.1.3 and 15.2.1. The effect of a single failure in the level control system has rather straightforward consequences, including level rise or fall by improper control of the feedwater

system. Increasing level will trip the turbine and automatically trip the HPCS system off. This trip signature is already described in the failure of the feedwater controller with increasing flow. Decreasing level will automatically initiate scram at the L3 level trip and will have a signature similar to loss of feedwater control with decreasing flow.

Operation of other engineered safeguards is not required for this transient and therefore has not been assumed in the analysis.

With the recirculation system in either the automatic or manual mode, relatively small changes would be experienced in plant conditions. The operator should, after hearing the alarm that the HPCS has commenced operation, check reactor water level and drywell pressure. If conditions are normal, the operator should shut down the HPCS system.

15.5.1.3 Core and System Performance

Mathematical Model

The detailed nonlinear dynamic model used to simulate this transient has been briefly described in Subsection 15.1.1.3. Analysis has been made for the manual mode of recirculation flow control. Important analytical factors, including reactivity coefficient and feedwater temperature change, have been assumed to be at the worst conditions, so that any deviations in the actual plant parameters will produce a less severe transient.

Input Parameters and Initial Conditions

The reactor is operating at 105% of NB rated power with thermally limited conditions. The water temperature of the HPCS system is assumed to be 40°F with an enthalpy of 11 Btu/lb. Other plant parameters are as shown in Table 15.0-1.

Results

Figure 15.5-1 shows the simulated transient event for the manual flow control mode, which begins with the introduction of cold water into the upper core plenum. Within 6 seconds, the full HPCS flow is established at approximately 5.4% of the rated feedwater flow rate. No delays were considered because they are not relevant to the analysis of the inadvertent HPCS pump start.
Addition of cooler water to the upper plenum causes a reduction in steam flow, which results in some depressurization as the pressure regulator responds to the event. In the automatic flow control mode, following a momentary decrease, neutron power settles out at a level slightly above operating level. In manual mode, the flux level settles out slightly below operating level. In either case, pressure and thermal variations are experienced. The ΔCPR for this event is essentially zero, thus the safety limit MCPR is not compromised.

Although no aspects of reactor design are threatened by these events, the analyses assume conservative (most severe) characteristics for the input analytical factors, including reactivity coefficient and feedwater temperature changes. Actual plant parameter deviations are expected to make the results of this event less severe.

15.5.1.4 Barrier Performance

The slight pressure reduction from the initial conditions (Figure 15.5-1) does not exceed the criteria for which the barriers are designed; therefore, these barriers maintain their integrity and function as designed.

15.5.1.5 Radiological Consequences

This event does not result in any release of radioactivity, therefore analysis of radiological consequences is not required.

15.5.2 Chemical and Volume Control System Malfunction

Not applicable.

15.5.3 BWR Transients

These transients have been appropriately analyzed in Sections 15.1 and 15.2 of this chapter.

15.5.4 References

1. Deleted

2. Power Uprate Project Task 900, "Transient Analysis," GE-NE-A1300384-08 Revision 1, September 1999.

3. Deleted

15.6 DECREASE IN REACTOR COOLANT INVENTORY

15.6.1 Inadvertent Safety/Relief Valve Opening

The inadvertent Safety/Relief Valve Opening event is not analyzed for reload cores.

15.6.1.1 Identification of Causes and Frequency Classification

The cause of an inadvertent opening of a safety/relief valve is attributed to malfunction of the valve or to the operator which controls valve opening. For a single safety/relief valve, the manual opening and closing circuitry is redundant to provide for single failure. The safety function and the relief function are separately controlled; further, by group action, the SRV's are redundant functionally and have single failure capability to perform the depressurization/safety functions.

Simply, a postulated fail open situation is analyzed here for completeness.

This transient disturbance is categorized as an infrequent event for the direct-acting Crosby SRV's used at LaSalle; however, due to a lack of comprehensive data on this newly designed valve, it is being analyzed here as an incident of moderate frequency.

15.6.1.2 Sequence of Events and System Operation

The sequence of events for this postulated incident is outlined as follows:

Time (sec)	Event
0	Initiate opening of 1 safety/relief valve.
0.5 (est)	Relief valve reaches full steam flow.
15.0 (est)	Reactor/power equipment establishes a new steady state operation condition, suppression pool accepts blowdown.

The normal functioning of plant instrumentation and controls is assumed for this incident; specifically the operation of the pressure regulator and vessel level control systems is assumed normal. Failures of additional equipment such as in feedwater system, the pressure regulator, or in the recirculation system are addressed elsewhere in Chapter 15.0 accident.

For the simple failure postulated here, the plant operator must reclose the valve as quickly as possible and check that reactor and turbine-generator return to normal. If the valve cannot be closed, plant power reduction by reducing reactor recirculation flow and subsequent feedwater flow reduction should be initiated as required by procedures and/or Technical Specifications. The suppression pool bulk temperature should be monitored to assure that plant shutdown occurs when the suppression pool temperature setpoint is reached. Initiation of suppression pool cooling is a power generation objective as long as the suppression pool setpoint is not attained. Mandatory shutdown of the plant is necessary at and beyond that setpoint.

15.6.1.3 Core and System Performance

Mathematical Model

This event is not limiting from a core performance standpoint. A heat-balance with time-temperature output is adequate to describe a bounding analysis for this event.

Input Parameters and Initial Conditions

It is assumed that the reactor is operating at an initial power level corresponding to 105% of rated steam flow conditions when a safety/relief valve opens inadvertently. Manual recirculation flow control is assumed. Discharge flow through the open valve at normal plant operating conditions is approximately 895,000 lb/hr of steam.

The 13 SRV's (Unit 2 has a total of 13 valves) at LSCS have rated steam flows which range from 862-906,000 lb/hr. Evaluation at a flow of 895,000 lb/hr provides a time-temperature response within 1 1/2% of that from the maximum possible flow condition for any of these Crosby valves at LSCS.

Results

The opening of a safety/relief valve allows steam to be discharged to the suppression pool. The sudden increase in the rate of steam flow leaving the reactor vessel causes a mild depressurization transient of brief duration. The pressure regulator senses this nuclear system pressure decrease and within a few seconds closes the turbine control valve to stabilize reactor vessel pressure at a slightly lower value. Reactor power settles very nearly at the initial power level. Thermal margin on the fuel decreases only slightly through this transient and no fuel damage results. MCPR is essentially unchanged. The safety limit margin is unaffected.

Figures 15.6-1 and 15.6-2 show the rise in suppression pool temperature for a stuck-open relief valve from high power operation and from hot standby respectively. Supplementary evaluations have been performed, as discussed in Section 6.2.1.8 and 15.0.8, to verify that an increase in the initial suppression pool temperature (from 100°F to 105°F) would not significantly impact this accident scenario.

15.6.1.4 Barrier Performance

The transient resulting from an inadvertently opened safety/relief valve is totally within the range of normal load following by this recirculation system. The feedwater system is adequate to maintain level. Therefore, this is no significant effect on the Reactor Coolant Pressure boundary nor on the containment integrity, either by pressure or temperature.

15.6.1.5 Radiological Consequences

While the consequence of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposures to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under defined meteorological and controlled release conditions. If purging of the containment is chosen the release will be in accordance with the established technical specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.6.2 Instrument Line Break

The Instrument Line Break event is not analyzed for reload cores.

15.6.2.1 Identification of Causes and Frequency Classification

There is no specific event or circumstance identified which results in the failure of an instrument line. These lines are designed to high quality engineering codes and standards and seismic and environmental requirements. However, for the purpose of evaluating the consequences, a small steam or liquid line (connected to the primary coolant system) is postulated to fail inside or outside the primary containment but inside the secondary containment. In order to bound the event, it is assumed that the instrument line instantaneously and circumferentially breaks outside the drywell but within the secondary containment at a location where immediate detection is not automatic or apparent, and where isolation of the break might not be possible. This kind of a break can also occur within the drywell, however, the associated effects are not as significant as those resulting from the analyzed failure which represents the envelope evaluation for small line break relative to sensitivity of detection.

The instrument line failure results in release of primary system coolant to the secondary containment (until the reactor is depressurized), but the event is far less limiting than the postulated events described in Subsections 15.6.4, 15.6.5, and 15.6.6. This event is estimated to occur with a limiting fault frequency.

15.6.2.2 Sequence of Events and Systems Operation

The sequence of events for this accident is as follows:

Approximate Elapsed Time	Event
0	Instrument line fails.
≤ 10 min	Identification of break.
≤ 10 min	Activation of RHR and initiation of reactor shutdown, including SGTS operation.
5 hours	Reactor vessel depressurized and break flow terminated.

Normal plant instrumentation and controls are assumed to be fully operational during the entire plant transient to ensure positive identification of the break and safe shutdown of the plant. Minimum reactor and plant protection system operations are assumed for the analysis, e.g., minimum ECCS flow and pool cooling capability.

The operator, after identification of the break isolates the affected instrument line if possible. Depending on which line is broken, the operator determines whether to continue plant operation until a scheduled shutdown can be made or to proceed with an immediate, orderly plant shutdown, including initiation of SGTS.

As a result of increased radiation, temperature, humidity, fluid, and noise levels within the containment, operator action can be initiated by any one or any combination of the following signals:

a. Operator comparing readings with several instruments monitoring the same process variable such as reactor level, jet pump flow, steam flow, and steam pressure.

b. By annunciation of the control function, either high or low in the control room.

c. By a half-channel scram if rupture occurred on a reactor protection system instrument line.

d. By a general increase in the area radiation monitor readings throughout the reactor building.

e. By increases in area temperature monitor readings in the reactor building.

f. By actuations of the leak detection system.

15.6.2.3 Core and System Performance

Mathematical Model

The analytical methods and associated assumptions which are used to evaluate the consequences of this accident are considered to provide a realistic, yet conservative assessment of the consequences.

The analytical techniques which are used to evaluate the consequences of this event are consistent with well-established heat transfer and mass blowdown calculational models. The instrument line is assumed to fail external to the primary containment and inside the secondary containment, resulting in the release of primary coolant to the reactor building. In addition to the normal levels of iodine in the reactor coolant, consideration is given to additional iodine release to the primary coolant from the fuel as a consequence of reduction in power and reactor vessel pressure.

This event was conservatively analyzed. As a result of this conservative approach and the fact that a more limiting line break event is analyzed in Section 6.3, specific uncertainties were not evaluated for this event.

As a consequence of this accident, the reactor is scrammed and the reactor vessel cooled and depressurized over a 5-hour period.

Input Parameters and Initial Conditions

The reactor is operating at design power conditions when a failure occurs in one of the instrument lines which is connected to the primary coolant system and penetrates the primary containment.

The following assumptions and conditions are the basis for the mass loss during the 5-hour period:

a. Shutdown and depressurization initiated at 10 minutes after break occurs.

b. Normal depressurization and cooldown of reactor pressure vessel.

c. Line contains a 1/4-inch diameter flow-restricting orifice inside the primary containment.

d. Homogeneous critical blowdown flow model (Reference 1) is applicable and flow is critical at the orifice.

Results

Instrument line breaks, because of their small size, are substantially less limiting from a core and systems performance standpoint than the events examined in Subsections 15.6.4, 15.6.5, and 15.6.6. Specifically these breaks are bounded by the steamline break (Subsection 15.6.4). In addition, instrument line breaks are included within the spectrum considered in ECCS performance calculations discussed in detail in Section 6.3.

Because instrument line breaks result in a slower rate of coolant loss and are bounded by the calculations referenced above, the results presented here are qualitative rather than quantitative. As the rate of coolant loss is slow, an orderly reactor system depressurization follows reactor scram, and the primary system is cooled down and maintained without ECCS actuation. No fuel damage or core uncovery occurs as a result of this accident.

As a consequence of depressurization and possible reactor scram, it is expected that additional iodine and noble gas activity will be released from those fuel rods which may have experienced prior cladding defects during normal operation (Reference 3). The noble gases, being only slightly soluble in the reactor coolant, will, for the most part, be released to the reactor vessel vapor dome. However, the released iodine is assumed to remain in the coolant and is discharged from the vessel in proportion to the mass of the coolant released.

The total integrated mass of fluid released into the secondary containment via the break during the blowdown is 25,000 pounds. Of this total, 6000 pounds flash to steam. Release of this mass of coolant results in a secondary containment pressure which is well below the design pressure.

15.6.2.4 Barrier Performance

No fuel cladding perforation or core uncovery occurs as a result of this event.

The release of primary coolant through the orificed instrument line could result in an increase in the normal secondary containment pressure and isolation of the normal ventilation system. The peak pressure reached will be governed primarily by the mass blowdown rate, coolant temperature, condensation factor, and to a lesser degree by the containment leak rate. In any event, the peak pressure will be well below the containment design pressure.

15.6.2.5 Radiological Consequences

To the present time the NRC has not issued any specific guidelines for evaluating this event. Therefore, no comparison can be made between a realistic and an NRC guided analysis. However, Table 15.6-1 is included and itemizes those parametric values applicable to a realistic analysis. The specific models and assumptions and the program used for computer evaluation are described in Reference 2. The analyzed leakage path is shown in Figure 15.6-3.

The radiological exposures are based on the assumption that the activity released to the containment is proportional to the mass loss. In addition to the activity contained in the coolant prior to blowdown, additional activity may be released as a consequence of vessel depressurization and possible reactor scram. This additional release is taken into consideration in evaluating the radiological exposures and is based on experimental data collected from BWR reactor shutdowns on similar plants (Reference 3).

The activity airborne within the secondary containment is a function of the primary coolant activity, blowdown rate, condensation rate, fraction of liquid which flashes to steam, and leakage rate from the containment. It is assumed that normal ventilation occurs for the first 10 minutes, followed by building isolation and initiation of the SGTS for the remainder of the event. Correlating the effects of these parameters and considering a combined washout-plateout factor of 2, the activity airborne in the secondary containment as a function of time is presented in Table 15.6-2.

The fission product activity released to the environment is based on a ventilation rate from the secondary containment 110,000 cfm for the first 10 minutes and a

standby gas treatment system iodine removal efficiency of 95% with a rate of 4,000 cfm for the duration of the accident. The iodine released as a function of time is presented in Table 15.6-3.

The radiological exposures have been evaluated for the meteorological conditions defined in Subsection 15.0.2. The doses are presented in Table 15.6-4.

Based on Reference 15, the normal reactor coolant fission product inventory source term and reactor dome pressures do not change for power uprate to 3489 MWt. Therefore, the radiological exposures for an instrument line break outside containment are not impacted by 105% power rate.

15.6.3 Steam Generator Tube Failure

Not applicable.

15.6.4 Steam System Pipe Break Outside Containment

The Steam System Pipe Break Outside Containment event is not analyzed for reload cores.

This event is not fuel type dependent.

15.6.4.1 Identification of Causes and Frequency Classification

This event involves the postulation of a large steamline pipe break outside containment. It is assumed that the largest steamline, the main steamline, instantaneously and circumferentially breaks at a location downstream of the outermost isolation valve. This evaluation therefore represents the envelope evaluation of steamline failures outside containment.

The plant is designed to detect such an occurrence immediately, initiate isolation of the broken line, and actuate the necessary protective features. The main steamlines are designed to high quality engineering codes and standards and restrictive seismic and environmental requirements. However, for the purpose of evaluating the consequences of a postulated large steamline rupture, the failure of a main steamline is assumed to occur. This event is categorized as a limiting fault with respect to frequency classification.

15.6.4.2 Sequence of Events and Systems Operation

Release of radioactive materials outside the secondary containment results from postulated breaches in the reactor coolant pressure boundary or the steam power conversion system boundary. A break spectrum analysis for the complete range of reactor conditions (Section 6.3) indicates that the limiting fault event for breaks outside the containment is a complete severance of one of the main steamlines. The sequence of events and approximate time required to reach the event is as follows:

Time (sec)	Event
0	Guillotine break of one main steamline.
~ 0.5	High steamline flow signal initiates MSIV closure.
<1.0	Reactor begins scram.
≤5.5	Main steamline isolation valves fully closed.
10	Safety/relief valves open automatically on high vessel pressure and maintain vessel pressure at approximately 1100 psi.
30	Normally RCIC and HPCS would initiate on low water level (no credit taken for RCIC or HPCS in this analysis).
330	Reactor water level begins to drop slowly due to loss of steam through the safety/relief valves; reactor pressure still at approximately 1100 psi.
600	Operator initiates ADS or manually controls relief valves. Vessel depressurizes rapidly.
830	Low-pressure ECCS systems initiated; core effectively reflooded and cladding temperature heatup terminated. No fuel rod failure (Subsection 6.3.3).

A postulated guillotine break of one of the four main steamlines outside the containment results in mass loss from both ends of the break. The flow from the upstream side is initially limited by the flow restrictor upstream of the inboard isolation valve. Flow from the downstream side is initially limited by the total area of the flow restrictors in the three broken lines. Subsequent closure of the MSIV's

further limits the flow when the valve area becomes less than the limiter area and finally terminates the mass loss when the full closure is reached.

The effect of single failures has been considered in analyzing this event. The ECCS aspects are covered in Section 6.3. The break detection and isolation considerations are defined in Sections 7.3 and 7.6. All of the protective sequences for this event can accommodate single component failure and single operator error and still complete the necessary safety action. A discussion of plant, reactor protection system, and ESF actions is given in Sections 6.3, 7.3, and 7.6.

Normally the operator monitors vessel pressure and water inventory to maintain core cooling. Without operator action, the RCIC would initiate automatically on low water level following isolation of the main steam supply system (i.e., MSIV closure). The core would be covered throughout the accident and there would be no fuel damage. Without taking credit for the RCIC water makeup capability and assuming HPCS failure. ADS will auto initiate to ensure termination of the accident without fuel damage.

15.6.4.3 Core and System Performance

Quantitative results (including mathematical models, input parameters, and consideration of uncertainties) for this event are given in Section 6.3. The temperature and pressure transients resulting from this accident are insufficient to cause fuel damage; there are no fuel cladding perforations as a consequence of this event.

15.6.4.4 Barrier Performance

Since this break occurs outside the containment, barrier performance within the containment envelope is not applicable. Details of the results of this event can be found in Subsection 6.2.3.

The following assumptions and conditions are used in determining the mass loss from the primary system from the inception of the break to full closure of the MSIVs (no operator action is needed in this interval):

a. The reactor is operating at the power level associated with maximum mass release.

b. Nuclear system pressure is 1055 psia and remains constant during closure.

c. There is an instantaneous circumferential break of the main steamline.

d. Isolation valves start to close at 0.5 second on high flow signal and are fully closed at 5.5 seconds.

e. The Moody critical flow model (Reference 1) is applicable.

f. Level rise time is conservatively assumed to be 1 second. Mixture quality is conservatively taken to be a constant 7% (steam weight percentage) during mixture flow.

Initially only steam will issue from the broken end of the steamline. The flow in each line is limited by critical flow at the limiter to a maximum of 200% of rated flow for each line.

For the NRC analysis of this event, an assumption is made that rapid depressurization of the RPV allows the water level to rise quickly enough to allow a flow of steam water mixture from the main steamline break until the MSIV is closed on that line.

For the realistic analysis of this event using the most probable operating condition prior to the postulated break, the calculated two-phase mixture level in the RPV does not reach the elevation of the main steam nozzles before the MSIV closes. Therefore, only steam is released from the break during the event.

Aside from this acknowledged difference in source terms, there is a major difference in methodology for calculating the radiological consequences for the design-basis analysis and for the realistic analysis. Each is treated separately in the next topic.

A schematic of the release path is shown in Figure 15.6-4.

15.6.4.5 Radiological Consequences

The discussion and specific numerical values provided, describes that used in the original design basis analysis evaluating radiological consequences. The information provided represents the then state of the art. As mentioned above, this accident analysis is not revised for reload cores.

Two separate radiological analyses are provided for this event. The first is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet 10 CFR 100 guidelines. This analysis is referred to as the "design-basis analysis". The second is based on assumptions considered to provide a realistic conservative estimate of the radiological consequences. This analysis is referred to as the "realistic analysis".

A schematic of the release path is shown in Figure 15.6-4.

Design-Basis Analysis

The design-basis analysis utilizes NRC Standard Review Plan 15.6.4 and NRC Regulatory Guide 1.5. The specific models and assumptions and the program used

for computer evaluation are described in Reference 4. Specific values of parameters used in the evaluation are presented in Table 15.6-5.

There is no fuel damage as a result of this event. The only activity available for release from the break is that which is present in the reactor coolant and steamlines prior to the break. The iodine concentration in the reactor coolant is then given by (μCi/gm):

I-131	0.039
I-132	0.360
I-133	0.270
I-134	0.720
I-135	0.390

Because of its short half-life, N-16 is not considered in the analysis.

The transport pathway is a direct, unfiltered release to the steam tunnel, which is a part of the secondary containment. The MSIV detection and closure time of 5.5 seconds results in a discharge of 14,000 pounds of steam and 86,000 pounds of liquid from the break. Assuming all the activity in this discharge becomes airborne, the release of activity to the environment is presented in Table 15.6-6.

This level of activity is consistent with an off-gas release rate of 300,000 μCi/sec after a 30-minute delay.

The calculated exposures for the design-basis analysis are presented in Table 15.6-8 and are a small fraction of the guidelines of 10 CFR 100.

Realistic Analysis

The realistic analysis is based on a plausible but still conservative assessment of this event. The specific models and assumptions and the program used for computer evaluation are described in Reference 2. Specific values of parameters used in the evaluation are presented in Table 15.6-5.

Since there is no fuel rod damage as a consequence of this event, the only activity released to the environment is that associated with the steam and liquid discharged from the break.

The activity released from the event is a function of the coolant activity, valve closure time, and mass of coolant released. A portion of the released coolant exists as steam prior to the blowdown, and as such does not contain the same concentration per unit of mass as does the steam generated as a consequence of the blowdown. Therefore, it is necessary to subtract the initial steam mass from the

total mass released and assign to it only 2% of the iodine activity contained by an equivalent mass of primary coolant.

The following assumptions are used in the calculation of the quantity and types of radioactive material released from the reactor coolant pressure boundary:

a. The amount of coolant discharged is that calculated in the analysis of the nuclear system transient. The mass loss is 36,000 pounds of steam.

b. The concentrations of biologically significant radionuclides contained in the primary coolant are as follows:

I-131	0.013 μCi/gm
I-132	0.120 μCi/gm
I-133	0.089 μCi/gm
I-134	0.240 μCi/gm
I-135	0.130 μCi/gm

Measurements made on BWR's of the current generation show the activity ratio between the main turbine condensate and reactor coolant to be on the order of 0.5% to 2%. For the purpose of this evaluation, the conservative assumption is made that the activity per pound of steam is equal to 2% of the activity per pound of reactor water.

c. The noble gas discharge rate after a 30-minute holdup is assumed to be 0.1 Ci/sec, an unusually high normal discharge rate. This assumption permits direct computation of the amount of noble gas activity leaving the reactor vessel at the time of the accident. The result is that 0.45 Ci of noble gas activity leaves the reactor vessel during each second that the isolation valve is open.

d. Because of the short half-life of N-16, the radiological effects from this isotope are of no major concern and are not considered in the analysis.

Based on the above considerations, the amount of activity which is available for atmospheric dispersion is presented in Table 15.6-7.

The calculated exposures for this event are presented in Table 15.6-8. As noted in comparative Table 15.6-8, these values are a small fraction of the 10 CFR 100 guidelines.

Power Uprate Evaluation

Based on Reference 15, the pre-power uprate calculations for mass release from a main steam line break were determined to be bounding for 105% power uprate (3489 MWt). The MSIV closure time and design basis coolant activities are not impacted by power uprate. Therefore, the radiological dose consequences presented in Table 15.6-8 remain bounding.

15.6.5 Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary

This event involves the postulation of a spectrum of piping breaks inside containment varying in size, type, and location. The break type includes steam and/or liquid process system lines. This event is also coupled with severe natural environmental conditions and includes earthquake coincidence.

The event has been analyzed quantitatively in Sections 6.2, 6.3, 7.1, 7.3, and 8.3. Therefore, the following discussion provides only new information not presented in the subject sections. All other information is covered by cross-referencing.

The release of radioactive fission products directly into the containment results from these postulated pipe breaks in the primary coolant pressure boundary. Possibilities for all pipe-break sizes and locations are examined in Sections 6.2 and 6.3, including the severance of small process system lines, the main steamlines upstream of the flow restrictors, and the recirculation loop pipelines. The most severe nuclear system effects and the greatest release of radioactive material to the containment result from a complete circumferential break of one of the two recirculation loop pipelines. The minimum required functions for the reactor and plant protection systems are discussed in Sections 6.3, 7.3, 7.6, and 8.3.
The postulated event represents an envelope evaluation for all liquid or steamline failures inside containment.

The LOCA analysis for LaSalle is plant specific and not cycle specific for GE methodology. Three different analysis methodologies may be used to determine the effects of the LOCA in accordance with the requirements of 10 CFR 50.46 and Appendix K. The three methodologies are SAFE/REFLOOD, SAFER/GESTR, and SAFER/PRIME. The method used is indicated in the Technical Requirements Manual and in Section 15.A of the UFSAR. The information contained in this subsection pertains to the original LOCA analysis which used the SAFE/REFLOOD methodology and was applicable for the first two cycles on both units. The SAFER/GESTR methodology is described in References 5 and 6 for GE fuel. The SAFER/PRIME methodology is described in Reference 21 for GE fuel.

15.6.5.1 Identification of Causes and Frequency Classification

There are no realistic, identifiable events which would result in a pipe break inside the primary containment of the magnitude required to cause a loss-of-coolant accident coincident with a safe shutdown earthquake and a single active component failure. The subject piping is designed, built, and analyzed to strict emergency code and standards criteria, and to severe seismic and environmental conditions. However, since such an accident provides an upper limit estimate to the resultant effects for this category of pipe breaks, it is evaluated without the cause being identified.

The loss-of-coolant accident resulting from a pipe break inside the primary containment is assigned the frequency of a limiting fault.

15.6.5.2 Sequence of Events and Systems Operation

The sequence of events associated with this accident is shown in Table 6.3-3.

Following the pipe break and scram, the MSIV will begin closing on the vessel low-low-low water level signal. The low-low water level or high drywell pressure signal will initiate HPCS, LPCS, and LPCI systems.

Because automatic actuation and operation of the ECCS is a system design basis, no operator action is required for the accident. However, by procedural requirement, the operator will perform the following monitoring actions.

After checking that all rods are inserted at time 0 plus approximately 10 seconds, the operator determines plant condition by observing the annunciators. After observing that the ECCS flows are initiated on low water level, the operator checks that the diesel generators have started and are on standby condition. The operator initiates the operation of the RHR system heat exchangers in the suppression pool cooling mode and gives instruction to put the service water systems into service. After the RHR system and other auxiliary systems are in proper operation, the operator monitors the hydrogen concentration in the drywell for proper activation of the recombiner, if necessary.

Single failures and operator errors have been adequately considered in the analysis of the entire spectrum of primary system breaks. The consequences of a LOCA with considerations for SCF and SOE occurrence are shown to be fully accommodated without the loss of any required safety function.

15.6.5.3 Core and System Performance

The analytical methods and associated assumptions which are used in evaluating the consequences of this accident provide an ultraconservative assessment of the expected consequences of this very improbable event.

The details of these calculations, their justification, and bases for the models are developed in Sections 6.3, 7.3, 7.6, and 8.3.

Results of this event are given in detail in Section 6.3 for compliance with 10CFR50.46 requirements. The temperature and pressure transients resulting from this accident are insufficient to cause perforation of the fuel cladding. Therefore, no fuel damage results from this accident.

Postaccident tracking instrumentation and control is assured. Continued long-term core cooling is demonstrated. Radiological input is minimized, and calculated radiological exposures are within limits. Continued operator control and surveillance are adequately enabled.

15.6.5.4 Barrier Performance

The design basis for the containment is to maintain its integrity and experience acceptable stresses after the instantaneous rupture of the largest single primary system piping within the structure while also accommodating the dynamic effects of the pipe break at the same time an SSE is also occurring. Therefore, any postulated loss-of-coolant accident does not result in exceeding the containment design limit. For details and results of the analyses, see Sections 3.8, 3.9, and 6.2.

15.6.5.5 Radiological Consequences

The LSCS LOCA was analyzed (Reference 11) using a plant power level of 3559 MWt, a conservative set of assumptions, and as-built design input parameters compatible for AST and the TEDE dose criteria. The numeric values of the critical design inputs were conservatively selected to assure an appropriate prudent safety margin against unpredicted events in the course of an accident and compensate for large uncertainties in facility parameters, accident progression, radioactive material transport, and atmospheric dispersion.

The design inputs used for the design analyses were extracted from LSCS licensing basis documents, Updated Final Safety Analysis Report (UFSAR) sections, existing calculations, design basis documents, and regulatory guidance documents. Key parameters used in the LOCA analysis are summarized in Table 15.6-27.

EAB, LPZ, CR, and TSC doses for LSCS were calculated using the guidance in Regulatory Guide 1.183, and the TEDE dose criteria. In addition to direct shine to control room operators, the DBA LOCA calculation was performed for the following post-LOCA release paths:

- Primary containment leakage,
- ECCS leakage, and
- MSIV leakage..

In general, credit is taken only for those active accident mitigation features that are classified as safety related, are required to be operable by TS, are powered by emergency power sources, and are automatically actuated. Exceptions are the following:

- The CR emergency ventilation system is designed to automatically initiate; however, the LOCA analysis assumes manual action timing to address single failures.
- The alignment of an MSIV drain line to direct MSIV leakage to the condenser is manually initiated.
- The seismically rugged portions of steam piping and the condenser are not classified as safety related.
- The SLC system is credited for suppression pool pH control. The SLC system is manually initiated.

The numeric values that are chosen as inputs to analyses required by 10 CFR 50.67 are compatible to AST and TEDE dose criteria and selected with the objective of maximizing the postulated dose. The use of a 10% lower makeup flow rate for the CR and a minimum CR recirculation flow rate, and use of worst-case ground release X/Q values, demonstrate the inherent conservatisms in the plant design and post-accident response analysis. These results can be seen in Table 15.6-29.

15.6.5.5.1 Assumptions on Transport in the Primary Containment

For LSCS, the radioactivity release from the reactor is assumed to mix instantaneously and homogeneously throughout the drywell. No mixing between the drywell and the wetwell airspace is assumed for the first two hours. This is based on an assumption that the initial blowdown occurs before fuel damage commences, and that AST source terms are based on a non-mechanistic loss of ECCS flow to the reactor for two hours. After ECCS flow restoration, the rapid steaming of ECCS liquids are assumed to quickly displace significant fractions of the airborne activity in the drywell through downcomers into the suppression chamber, providing the mixing mechanism. Conservatively, no credit is taken for suppression pool scrubbing during this flow. Therefore, after two hours, complete mixing of activity in the drywell volume to the suppression chamber airspace is assumed. The RADTRAD containment compartment volume parameter and MSIV leakage flow rates implement this treatment.

With the exception of noble gases, all fission products released from the fuel to the containment are also assumed to instantaneously and homogeneously mix in the suppression pool at the time of release. RADTRAD models for ECCS leakage treat the suppression pool water as the compartment to which core activity is released.

Radioactivity in containment is reduced only by natural deposition, decay, and leakage. For LSCS, the RADTRAD computer program, including the Powers Natural Deposition algorithm based on NUREG/CR-6189, is used for modeling

aerosol deposition in primary containment. No natural deposition is assumed for elemental or organic iodine. The lower bound (i.e., 10%) level of deposition credit is used. Suppression pool scrubbing is not credited. Neither drywell nor wetwell spray is credited as a removal mechanism. Analyses demonstrate that suppression pool pH is maintained greater than seven, so iodine re-evolution is not assumed.

Decay of radioactivity is credited in the drywell prior to release. RADTRAD's decay plus daughter option is used.

Leakage from the primary containment is postulated to be released directly to the environment without mixing in the Reactor Building free air volume.

15.6.5.5.2 Post-LOCA Containment Leakage

Primary containment leakage is assumed to be controlled to an La rate of 1.0% per day, with no reduction after the first 24 hours.

The entire leakage is treated as being to the secondary containment. The exhaust from secondary containment is filtered through the SGT system filter train, following a 15 minute drawdown period with the filtration not credited. After drawdown, SGT system high efficiency particulate air (HEPA) and charcoal filters are available to reduce the released activity.

Other than leakage through the MSIVs, there are no other leakage pathways that bypass secondary containment at LSCS. Because of the use of the MSIV – Isolated Condenser Leakage Treatment Method (MSIV-ICLTM), MSIV leakage bypasses secondary containment and is released through the seismically rugged Turbine Condenser system, as discussed below.

15.6.5.5.3. Containment Leakage Source Term

The BWR core inventory fractions listed in Table 1 of Regulatory Guide 1.183 are released into the containment at the release timing shown in Table 4 of Regulatory Guide 1.183. Since the post-LOCA minimum suppression pool water pH is greater than 7.0 for the duration of the accident, the chemical form of radioiodine released into the containment is assumed to be 95% CsI, 4.85% elemental iodine, and 0.15% organic iodide. With the exception of elemental and organic iodine and noble gases, the remaining fission products are assumed to be in particulate form. The plant-specific isotopic fission product core activities, in units of Ci/MWt (Table 15.6-28), were calculated using the core thermal power level.

15.6.5.5.4. Containment Purging

Purging of containment is not a routine activity at LSCS. TS SR 3.6.1.3.1 identifies purposes for containment purging at LSCS as inerting, de-inerting, pressure control, ALARA or air quality considerations for personnel entry, and surveillances

that require valves to be open. TS 3.6.3.2 provides limitations on use for inerting and deinerting at power.

15.6.5.5.5. Post-LOCA ECCS Leakage

The ECCS fluid systems that are located in the Reactor Building, recirculate suppression pool water outside of the primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. The radiological consequences from the postulated leakage are analyzed and combined with the radiological consequences from other fission product release paths to determine the total calculated radiological consequences from the LOCA.

15.6.5.5.6. ECCS Leakage Source Term

With the exception of noble gases, fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the suppression pool water at the time of release from the core. The total ECCS leakage from all components in the ECCS systems is assumed to be 5 gpm, which is assumed to start immediately after the onset of a LOCA. With the exception of iodine, remaining fission products in the recirculating liquid are assumed to be retained in the pool water. Since the post-LOCA temperature of suppression pool water recirculated through the ECCS system is less than 212°F, 10% of the iodine activity in the leaked liquid is assumed to become airborne. The reduction in ECCS leakage activity by dilution in the Reactor Building volume is not credited. The radioiodine that is postulated to be available for release to the environment due to ECCS leakage is assumed to be 97% elemental and 3% organic.

15.6.5.5.7. MSIV Leakage Release Pathway

The previous MSIV leakage rate limits of 100 scfh per steam line and a total of 400 scfh for all four lines was changed to 200 scfh for any one line and a total of 400 scfh for all four lines. These limits continue to apply at test pressures of > 25 psig.

Outboard MSIV failure is assumed as the single active failure since this maximizes the volume of piping in which the fluid is depressurized, minimizing deposition.

Inboard piping on one main steam line is not credited to conservatively simulate the impact of a LOCA involving a steam line break inside containment.

15.6.5.5.8. MSIV Leakage Source Term

The activity available for release via MSIV leakage is assumed to be that activity released into the drywell for evaluating containment leakage per Regulatory Guide 1.183.

15.6.5.5.9. Modeling of Deposition Credit in Pipes and Condenser

LSCS has previously been analyzed and licensed to no longer credit an MSIV Leakage Control system other than the MSIV-ICLTM, and to credit seismically analyzed portions of the Turbine Condenser system. This system has previously been shown to be seismically rugged as discussed in UFSAR Section 6.8. This historical evaluation is based on methodology described in NEDC-31858P A (Reference 7). That analysis was based on a design basis recirculation line break. In the AST LOCA calculation, the analysis of MSIV leakage is updated to reflect AST parameters related to release timing and chemical makeup and NRC-approved approaches regarding fission product settling and deposition, as discussed below.

15.6.5.5.10. Aerosol Settling

Modeling of aerosol settling is based on methodology used by the NRC in Accident Evaluation Branch (AEB) 98 03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG 1465) Source Term," with some additional conservatism based on LSCS specific parameters. For aerosol settling, only horizontal piping runs are credited, and only the horizontal projected area of horizontal piping is considered as the settling area.

This analysis implements a 20-group settling velocity distribution rather than the AEB 98 03 single, median value, model. The same settling velocity probability distribution function was applied. This is conservative because it does not consider such phenomena as thermophoresis, diffusiophoresis, flow irregularities, and hygroscopicity, which would serve to increase the rate of aerosol deposition and settling. The settling velocity distribution is a function of a randomly sampled range of the three particle parameters (i.e., density (logarithmically distributed), diameter (uniformly distributed), and shape (uniformly distributed)) and three constants (i.e., gravitational acceleration, Cunningham slip factor, and viscosity). The range of each particle parameter is referenced in AEB-98-03.

By implementing a conservative, semi-continuous, probability-weighted, 20-group step function to simulate the varied population of particulate in a given Main Steam (MS) system volume, as opposed to a single median value, this model accounts for the uneven settling of "easier to remove particles" versus "difficult to remove particles."

The analysis takes no credit for any aerosol deposition after 24 hours. This conservative aerosol deposition treatment, and the conservatisms in the AEB 98 03 conclusions, account for uncertainty in the AEB-98-03 model.

15.6.5.5.11. Elemental and Organic Iodine Removal

Because elemental iodine deposition is not gravity dependent, deposition is credited in both horizontal and vertical piping on all surface areas. For conservatism, no

credit is taken for deposition in the drain lines that provide the previously licensed alternate drain path to the condenser. All MS drain lines are routed to the condenser at a point below the condenser tubing.

Credit is taken for deposition in the condenser, but only the deposition area of the horizontal surface of the wetwell of the high pressure condenser. The condenser tubing provides a surface area that is orders of magnitude larger than that of the credited bottom surface area. No credit is taken for any organic iodine removal in piping or the condenser.

Re-suspension of deposited elemental iodine is conservatively treated as organic iodine and immediately released. Re-suspension of iodine from steel surfaces was simulated by applying the model developed by J. E. Cline & Associates, Inc. and Science Applications International Corporation (SAIC). The immediate release of re-suspended iodine, directly to the environment, in organic form is a conservative assumption due to inherent holdups in this release and tortured paths through which this activity will be transported. Therefore, this simulation conservatively models the re-suspension effects of elemental iodine.

15.6.5.5.12. Condenser Credit Treatment and Conservatisms

The condenser is treated as a well-mixed volume. The credited deposition area for elemental iodines includes walls and the base, which includes the wetwell. For aerosols, only the base/wetwell surfaces are credited since the removal is by gravitational settling. No organic iodine removal credit is taken.

In general, the crediting of steam line piping and the condenser results in dose contributions being dominated by noble gases and organic iodine. Even so, the treatment of aerosols and elemental iodine is conservative, for the following reasons.

1. The drain lines, which are not credited for settling or deposition, enter the condenser below the condenser tubing. Expected exhaust paths are through:
   (1) the turbine shaft seals to the gland seal condenser exhaust (unpowered),
   (2) through the condenser vacuum breaker if in use prior to a Mode 3 LOCA, or
   (3) through shell leakage.

2. The first two paths are well above the condenser tubing. For aerosols, the direction for the first two paths requires that they "settle up" through the condenser tubes. For elemental iodine, the neglected surface area of the condenser tubes far exceeds the credited wall and wetwell surface.

3. The condenser shell path is one of the assumed paths for non-condensable gas entry when the condenser is at vacuum. However, at loss of vacuum conditions, its out-leakage equivalent path resistance would be expected to be significantly less that the gland seal path. Therefore, general shell leakage, which could be above or below the condenser tubes, is expected to be small.

15.6.5.5.13. Determination of MSIV Leakage Rates in Various Main Steam Line (MSL) volumes

The radioactivity associated with MSIV leakage is assumed to be released directly from the primary containment and into the MSLs. MSIV leakage has separate limits and a separately analyzed dose; therefore, it is not included in the La fraction limit and is instead separately controlled.

MSIV leakage assumed in the LOCA analysis is 400 scfh total for all MSLs and 200 scfh for any one MSL, when tested at or greater than 25 psig. The leakage rate and inboard piping flow rate associated with a 200 scfh leakage rate is adjusted for pressure and temperature differences.

Flow rates out of the condenser are similarly calculated with the assumption of a condenser air space temperature of 120°F for the accident duration. This rate applies to any condenser opening such as turbine seals, condenser shell leakage, or open vacuum breakers that may be in use under Mode 3 conditions.

Determination of inboard steam line, outboard steam line, and condenser effective filter efficiencies is determined using AEB 98-03 formulations and settling and deposition velocities.

15.6.5.5.14. Recirculation Line Rupture Versus MSL Rupture

10 CFR 50, Appendix A, defines LOCAs as those postulated accidents that result from a loss of coolant inventory at rates that exceed the capability of the reactor coolant makeup system. Leaks up to a double-ended rupture of the largest pipe of the RCS are included. The LOCA is a conservative surrogate accident that is intended to challenge selective aspects of the facility design. The DBA for the safety related system design is a LOCA. This LOCA leads to a specific combination of dynamic, quasi-static, and static loads in time. The thermal transients due to other postulated events, including the MSLB inside the drywell, do not impose maximum challenge to the drywell pressure boundary and fuel integrity. The LOCA results in the maximum core damage and fission product release as shown in Regulatory Guide 1.183, Table 1. Therefore, a recirculation line rupture is considered to be the limiting event with respect to radiological consequences.

Regulatory Guide 1.183, Appendix A, Section 6.5 allows reduction in MSIV releases that is due to holdup and deposition in MS piping downstream of the MSIVs and in

the main condenser, including the treatment of air ejector effluent by offgas systems, if the components and piping systems used in the release path are capable of performing their safety function during and following a safe shutdown earthquake. Although postulating a MSLB in one steam line inside the drywell would maximize the dose contribution from the MSIV leakage, the MSLB is not a credible event during a LOCA since the MS piping is designed to withstand the safe shutdown earthquake.

15.6.5.5.15. CR/AEER Model

The LSCS CR envelope has historically been treated as consisting of the CR and AEER, with a shared filtered emergency makeup system and separate filtered recirculation systems. In the AST LOCA analysis, standard continuous occupancy assumptions are applied to the CR. However, AEER occupancy is only required for the safety related action of starting the fan that provides containment air mixing as required per 10 CFR 50.44(c)(1) for combustible gas control. This mission is assumed to be performed by an operator not assigned full time to the CR, but dispatched from the CR. The total expected time for this mission outside of the CR is nine minutes. The dose analysis is conservatively based on 30 minutes. Worst-case timing for this operation is assumed starting at time zero because of exposure to releases during reactor enclosure drawdown. No credit is taken for any filtration provided by the makeup filter or AEER recirculation filter system. On this basis, the features that control radioactivity in the AEER, such as filtered intake, filtered recirculation, and positive pressurization are not required for this mission.

The CR and AEER share a makeup filter system, but have separate recirculation filter systems. Nominally, 37.5% of the makeup flow is directed to the CR and 62.5% is directed to the AEER. In the AST LOCA analysis, splits of 25%, 37.5%, and 50% to the CR are analyzed in this distribution with the balances directed to the AEER. The bounding values for dose analysis purposes were used to demonstrate 10 CFR 50.67 compliance.

The CR/AEER makeup filter charcoal adsorber credit is based on 90% efficiency for elemental and organic iodines, rather than the historically credited 95%.

Because of the presence of HEPA filtration in the makeup filter train, aerosol removal efficiency is credited at 99%, rather than the historically credited 99.95%. No aerosol removal is credited in the CR or AEER recirculation filter trains.

Recirculation filter bypass for the CR is assumed to be at 5% of the minimum CR supply flow. That is 900 cfm for the CR recirculation filter. Inleakage upstream of the CR recirculation filters and upstream of the supply fans are addressed separately as discussed below.

A CR recirculation unfiltered inleakage rate of 2400 cfm is assumed. This is 200% of the historically assumed value, and conservatively well above tracer gas testing

results, to provide operational margin to be managed under the Control Room Envelope Habitability Program.

In addition to the unfiltered inleakage and the 5% filter bypass, another 50 cfm of unfiltered inleakage is assumed into the ductwork downstream of the CR recirculation filters and upstream of the supply fans for the CR. The allowance is based on historical estimates of maximum credible leakage, now multiplied by approximately a factor of seven.

15.6.5.5.16. Shine Doses to CR from External Sources

The pre-AST UFSAR shine doses, and supporting analyses, have been reviewed and the largest contributors re-evaluated on a conservative AST basis. These were shine from plate-out of activity on the refuel floor and control building filters. External cloud doses were also reanalyzed for possible AST effects. Resulting external dose contributions are small.

15.6.5.5.17. Vital Area Accessibility

The LOCA analysis establishes that vital areas remain accessible. Vital areas outside of the CR are:

1. The AEER, with the associated mission dose to start fans that provide containment air mixing for post-LOCA combustible gas control purposes, and

2. The TSC, which is assumed to require occupancy equivalent to the CR.

Assessment of these analyses shows these areas to be accessible, with doses within 10 CFR 50.67 CR dose limits.

Based on evaluations in the AST LOCA calculation, the dose for occupancy of the TSC, and for the safety related mission to the AEER are within 10 CFR 50.67 CR dose limits. Existing analyses for other locations and pathways as described in UFSAR Section 12.3 were reviewed.

15.6.5.5.18. Suppression Pool pH Control

Suppression pool pH was evaluated over the 30-day duration of the DBA LOCA and demonstrated that the pH will remain above 7.0. Therefore, no iodine conversion to elemental with re-evolution is considered in the LOCA calculation. The control of pH also significantly limits the potential for airborne release from subcooled ECCS leakage inside and outside of secondary containment. SLC system injection is required for pH control within approximately 3.5 hours of the start of the LOCA, but even a minimal amount of solution addition to the suppression pool significantly delays the need for mixing in additional solution. In accordance with Plant

Procedures, injection would typically be expected within the first few minutes of an event that results in fuel damage comparable to that necessary for core radioactivity releases assumed in the DBA LOCA.

15.6.6 Feedwater Line Break

The Feedwater Line Break event is not analyzed for reload cores.

15.6.6.1 Identification of Causes and Frequency Classification

The postulation of a break in the feedwater line, representing the largest liquid process line outside the containment, provides the envelope evaluation relative to this type of occurrence. The break is assumed to be instantaneous, circumferential, and downstream of the outermost isolation valve. A feedwater line break is assumed without the cause being identified. The subject piping is designed to high quality, strict engineering codes and standards, and to severe seismic and environmental requirements. The event is categorized as a limiting fault.

A more limiting event from a core performance evaluation stand-point (feedwater line break inside containment) has been quantitatively analyzed in Section 6.3.

15.6.6.2 Sequence of Events and Systems Operation

The sequence of events and the approximate time for the feedwater line break outside containment is as follows:

Time (sec)	Event
0	One feedwater line breaks.
0+	Feedwater line check valves isolate the reactor from the break.
<30	Reactor Scram occurs at L3. L2 vessel water level trip initiates RCIC and HPCS. RPT also occurs at L2.
N120	The safety/relief valves open and close to maintain the reactor vessel pressure at approximately 1100 psig.
1 to 2 hours	Normal reactor cooldown procedure established.

It is assumed that the normally operating plant instrument and control channels are functioning. Credit is taken for the actuation of the reactor isolation system and ECCS systems. The reactor protection system (safety/relief valves, ECCS, and control rod drive) and plant protection system (RHR heat exchangers) are assumed to function properly to assure a safe shutdown. The RCIC and HPCS systems are assumed to operate normally, as are the ESF systems. Because of an additional steam flow induced process measurement error in the Level 3 scram, the timing values following Low water level scram based on the L3 Analytical Limits are slightly different. However, as described in Reference 19, the impact of the change is not significant.

The feedwater line break outside the containment is a special case of the general loss-of-coolant accident break spectrum considered in detail in Section 6.3. The general single-failure analysis for loss-of-coolant accidents is discussed in detail in Subsection 6.3.3.3. For the feedwater line break outside the containment, since the break is isolatable, either the RCIC or the HPCS can provide adequate flow to the vessel to maintain core cooling and prevent fuel rod cladding failure. A single failure of either the HPCS or the RCIC would still provide sufficient flow to keep the core covered with water. See Section 6.3 for detailed description of analysis.

Because automatic actuation and operation of the ECCS is a system design basis, no operator action is required for this accident. However, by procedural requirements the operator will perform the following monitoring actions, which are shown here for information only.

a. Determine that line break has occurred and evacuate the affected area of the turbine building.

b. Ensure that the reactor is shut down and that RCIC and/or HPCS are operating normally.

c. Implement site radiation incident procedures.

d. Shut down the feedwater system if possible and deenergize any electrical equipment which may be damaged by water from the feedwater system in the turbine building.

e. Continue to monitor reactor water level and the performance of the ECCS systems while the radiation incident procedure is being implemented; initiate normal reactor cooldown.

f. When the reactor pressure has decreased below 150 psi, initiate RHR in the shutdown cooling mode to continue cooling down the reactor.

Estimated elapsed time is 3-4 hours.

15.6.6.3 Core and System Performance

The accident is postulated to occur with the input parameters and initial conditions as given in Table 6.3-2 of the FSAR. The feedwater line break outside the containment is less limiting than either the steamline breaks outside the containment (analysis presented in Sections 6.3 and/or 15.6.4 of the FSAR) or the feedwater line break inside the containment (analysis presented in Sections 6.3.3 and 15.6.5 of the FSAR). It certainly is far less limiting than the design-basis accident (the recirculation line break analysis presented in Subsections 6.3.3 and 15.6.5 of the FSAR).

The reactor vessel is isolated on low low-water level (L1), and the RCIC and HPCS together or singly restore the reactor water level to the normal elevation. The fuel is covered throughout the transient, and there are no pressure or temperature transients sufficient to cause fuel damage.

Results

For the worst operating condition prior to the postulated break and using realistic assumptions, the total integrated mass of coolant leaving the break is 788,000 pounds, of which 165,000 pounds flashes to steam, with an assumed iodine carryover of 100%. Of the activity remaining in the unflashed liquid, 1% is assumed to become airborne. Normally all feedwater reaching the break location will have passed through condensate demineralizers which have a 99.9% iodine-removal efficiency. However, as a result of the increased feedwater flow caused by the break, differential pressure across the demineralizers is assumed to initiate flow through the demineralizer bypass line. This bypass line then carries 50% of the total flow, resulting in an effective iodine removal efficiency for all flow of 49.95%.

Taking credit for 50% removal due to holdup, decay, and plateout during transport through the turbine building, the release of activity to the environment is presented in Table 15.6-20. The release is assumed to take place within 2 hours of the occurrence of the break.

There is no fuel damage as a consequence of this accident. In addition, an insignificant quantity of activity (compared to that existing in the main condenser hotwell prior to occurrence of the break) is released from the contained piping system prior to isolation closure.

The iodine concentration in the main condenser hotwell is consistent with an off-gas release rate of 100,000 Ci/sec at 30 minutes delay and is 0.02 (2% carryover) times the concentration in the reactor coolant. Noble gas activity in the condensate is negligible, since the air ejectors remove practically all noble gas from the condenser.

The transport pathway to the environment consists of liquid release from the break, which is carried over to the turbine building atmosphere due to flashing and partitioning and is subsequently released with only particulate filtration through the turbine building ventilation system into the plant vent stack.

15.6.6.4 Barrier Performance

Accidents that result in the release of radioactive materials outside the containment are the results of postulated breaches in the reactor coolant pressure boundary or the steam power-conversion system boundary. A break spectrum analysis for the complete range of reactor conditions indicates that the limiting fault event for breaks outside the containment is a complete severance of one of the main steamlines as described in Subsection 15.6.4. For effects on containment structures, the feedwater system piping break is less severe than the main steamline break.

Results of analysis for this event can be found in Subsection 6.2.3 or Subsection 6.2.4.

15.6.6.5 Radiological Consequences

A design-basis evaluation of this accident was not made because no NRC guidelines exist for this case. However, those parameters of significance in evaluating the consequences of this event are presented in Table 15.6-19.

The realistic analysis is based on an engineered but still conservative assessment of this accident. The specific models and assumptions and the program used for computer evaluation are described in Reference 1. A schematic diagram of the leakage path for this accident is shown in Figure 15.6-6.

The calculated exposures for the realistic analysis are a small fraction of 10 CFR 100 guidelines, as shown in Table 15.6-21.

Based on Reference 15, the radiological consequences for 105% power uprate are estimated to increase 6% due to the increase in feedwater flowrate. The design basis normal operation coolant source term is not impacted by power uprate. A feedwater line break does not result in fuel damage. The calculated exposures are still a small fraction of 10 CFR 100 guidelines.

15.6.7 References

1. F. J. Moody, "Maximum Two-Phase Vessel Blowdown From Pipes," ASME Paper Number 65-WA/HT-1, March 15, 1969.

2. D. Nguyen, "Realistic Accident Analysis - The RELAC Code and User's Guide," October 1977.

3. F. J. Brutschy et. al., "Behavior of Iodine in Reactor Water During Plant Shutdown and Startup," NEDO-105, August 1972.

4. P. P. Stancavage and E. J. Morgan, "Conservative Radiological Accident Evaluation - The CONACOL Code," NEDO-21143, March 1976.

5. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, (Unit 1 & 2: Rev. 20).

6. GE Document NEDC-31510P, "LaSalle County Station Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analyses," dated December 1987.

7. GE Document NEDC-31858P, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems" submitted to the NRC by BWROG letter dated October 4, 1993. (*)

8. ComEd letter dated August 28, 1995 to NRC, "LaSalle County Nuclear Power Station Units 1 and 2 Application for Amendment of Facility Operating Licenses NPF-11 and NPF-18, Appendix A, Technical Specifications, and Exemption to Appendix J of 10CFR50 Regarding Elimination of MSIV Leakage Control System and Increased MSIV Leakage Limits NRC Docket Nos. 50-373 and 50-374". (*)

9. M. David Lynch letter to D.L. Farrar, dated April 5, 1996, "Issuance of Amendments" (TAC Nos. M93597 and M93598). This letter transmitted Amendment No. 112 to facility operating license NPF-11, and Amendment No. 97 to facility operating license NPF-18, which reflect deletion of the main steam isolation valve (MSIV) leakage control system (LCS), and raising of the allowable leakage rate of each main steamline. (*)

10. Sargent & Lundy Report EMD-068078, Rev. 2, 08/09/95, "Report of Walkdown to Verify Seismic Adequacy of Main Steam Drain Line and Condenser for Use as the Alternate MSIV Leakage Treatment System."

11. LSCS Design Analysis L-003068, Rev. 2, "Re-analysis of Loss of Coolant Accident (LOCA) Using Alternative Source Terms".

12. Donna M. Skay (USNRC) letter to Oliver Kingsley (ComEd), dated May 13, 1998, "Issuance of Amendments". This letter transmitted Amendment No. 126 to Facility Operating License NPF-11, and Amendment No. 111 to Facility Operating License NPF-18, which reflect revisions to Control Room and Auxiliary Electric Equipment Room ventilation filter testing.

13. Deleted.

14. Deleted.

15. NDIT LAS-ENDIT-H041, Upgrade 1, Power Uprate Project S&L Task No. 21b, "Radiological/Chapter 15 Accidents."

16. NDIT LAS-ENDIT H042, S&L Task 21g, "Radiological / Control Room and TSC Habitability."

17. Deleted.

18. Deleted.

19. “BWR Owners Group Evaluation of Steam Flow Induced Error (SFIE) Impact on the L3 Setpoint Analytic Limit” GEH-NE-0000-0077-4603-R1, October 2008.

20. Deleted.

21. Global Nuclear Fuel, "The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance" Technical Bases-NEDC-33256P-A, Qualification-NEDC-33257P-A, and Application Methodology-NEDC-33258P-A, September 2010.

(*) – U2 MSIV-LCS deleted, U1 MSIV-LCS abandoned-in-place

TABLE 15.6-1

INSTRUMENT LINE BREAK ACCIDENT - PARAMETERS TO BE TABULATED FOR POSTULATED ACCIDENT ANALYSES

	CONSERVATIVE (NRC) ASSUMPTIONS	REALISTIC (CONSERVATIVE ENGINEERING) ASSUMPTIONS
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power level corresponding to 102% rated Core Thermal Power**	NA	3359 MWt
B. Burnup	NA	NA
C. Fuel damaged	NA	None
D. Release of activity by nuclide	NA	Table 15.6.2-2
E. Iodine fractions		
(1) Organic	NA	0
(2) Elemental		1
(3) Particulate		0
F. Reactor coolant activity before the accident	NA	Subsection 15.6.4.5
II. Data and assumptions used to estimate activity released		
A. Primary containment leak rate (%/day)	NA	NA
B. Secondary containment leak rate (%/day)	NA	100
C. Valve movement times	NA	NA
D. Adsorption and filtration efficiencies		
(1) Organic iodine	NA	95
(2) Elemental iodine	NA	95
(3) Particulate iodine	NA	95
(4) Particulate fission products	NA	NA
E. Recirculation system parameters		
(1) Flow rate	NA	NA
(2) Mixing efficiency	NA	NA
(3) Filter efficiency	NA	NA
F. Containment spray parameters (flow rate, drop size, etc.)	NA	NA
G. Containment volumes	NA	NA
H. All other pertinent data and assumptions	NA	None
III. Dispersion Data		
A. Boundary and LPZ distance (meters)	NA	509/6400
B. χ/Q'S for time intervals of		
(1) 0-2 hr - EAB*/LPZ	NA	4.0(-7)/1.5(-7)
(2) 2-8 hr - LPZ	NA	1.5(-7)
(3) 8-24 hr - LPZ	NA	3.3(-8)
(4) 1-4 days - LPZ	NA	1.4(-8)
(5) 4-30 days - LPZ	NA	1.1(-8)
IV. Dose Data		
A. Method of dose calculation	NA	Reference 2
B. Dose conversion assumptions	NA	Reference 2
C. Peak activity concentrations in secondary containment	NA	Table 15.6.2-2
D. Doses	NA	Table 15.6.2-4

* Maximum χ/Q occurs at 4500 meters from the release point (realistic boundary). The χ/Q at the EAB is 5.0 (-13) sec/m^3.

** Reference 15

TABLE 15.6-2

INSTRUMENT LINE FAILURE

ACTIVITY AIRBORNE IN THE SECONDARY CONTAINMENT (CURIES)

(Realistic Analysis)

ISOTOPE	10 MINUTES	1 HOUR	2 HOURS	8 HOURS	1 DAY	4 DAYS	30 DAYS
I-131	1.30E-03	4.03E 01	6.48E 01	8.75E 01	4.24E 01	1.62E 00	9.08E-13
I-132	1.17E-02	5.40E 01	7.23E 01	2.21E 01	8.83E-02	1.37E-12	0.
I-133	8.86E-03	9.37E 01	1.48E 02	1.71E 02	5.16E 01	2.31E-01	0.
I-134	2.26E-02	7.66E 01	7.96E 01	2.62E 00	4.22E-06	0.	0.
I-135	1.29E-02	8.69E 01	1.31E 02	1.06E 02	1.01E 01	2.56E-04	0.
TOTAL	5.74E-02	3.51E 02	4.96E 02	3.90E 02	1.04E 02	1.85E 00	9.08E-13

TABLE 15.6-3

INSTRUMENT LINE FAILURE

ACTIVITY RELEASED TO THE ENVIRONMENT (CURIES)

(Realistic Analysis)

ISOTOPE	10 MINUTES	1 HOUR	2 HOURS	8 HOURS	1 DAY	4 DAYS	30 DAYS
I-131	5.58E-04	3.37E-02	1.43E-01	1.30E 00	3.38E 00	5.25E 00	5.33E 00
I-132	5.07E-03	5.10E-02	1.83E-01	9.19E-01	1.05E 00	1.05E 00	1.05E 00
I-133	3.81E-03	8.10E-02	3.33E-01	2.80E 00	6.12E 00	7.55E 00	7.56E 00
I-134	9.91E-03	7.88E-02	2.42E-01	7.37E-01	7.43E-01	7.43E-01	7.43E-01
I-135	5.55E-03	7.78E-02	3.06E-01	2.18E 00	3.54E 00	3.68E 00	3.69E 00
TOTAL	2.49E-02	3.22E-01	1.21E 00	7.93E 00	1.48E 01	1.83E 01	1.84E 01

TABLE 15.6-4

INSTRUMENT LINE BREAK
RADIOLOGICAL EFFECTS

		10 CFR 100		CONSERVATIVE	
		(2 hours)	(30 days)	(2 hours)	(30 days)
A. Exclusion Boundary (509 meters)	Whole Body Dose (rem)	25	N/A	1.8E-07	--
	Thyroid Dose (rem)	300	N/A	5 5E-05	--
B. Point of Interest (915 meters)	Whole Body Dose (rem)	N/A	N/A	--	--
	Thyroid Dose (rem)	N/A	N/A	--	--
C. Low Population Zone (6400 meters)	Whole Body Dose (rem)	N/A	25	N/A	4.0E-07
	Thyroid Dose (rem)	N/A	300	N/A	2 1E-04

TABLE 15.6-5

STEAMLINE BREAK ACCIDENT - PARAMETERS TO BE TABULATED FOR POSTULATED ACCIDENT ANALYSES

	CONSERVATIVE (NRC) ASSUMPTIONS	REALISTIC (CONSERVATIVE ENGINEERING) ASSUMPTIONS
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power level corresponding to 102% rated Core Thermal Power**	3458 MWt	3559 MWt
B. Burnup	NA	NA
C. Fuel damaged	None	None
D. Release of activity by nuclide	Table 15.6.4-2	Table 15.6.4-3
E. Iodine fractions		
(1) Organic	0	0
(2) Elemental	1	1
(3) Particulate	0	0
F. Reactor coolant activity before the accident	Subsection 15.6.4.5	Subsection 15.6.4.5
II. Data and assumptions used to estimate activity released		
A. Containment leak rate (%/day)	NA	NA
B. Secondary containment leak rate (%/day)	NA	NA
C. Isolation valve closure time (sec)	5.5	5.5
D. Adsorption and filtration efficiencies		
(1) Organic iodine	NA	NA
(2) Elemental iodine	NA	NA
(3) Particulate iodine	NA	NA
(4) Particulate fission products	NA	NA
E. Recirculation system parameters	NA	NA
(1) Flow rate	NA	NA
(2) Mixing efficiency	NA	NA
(3) Filter efficiency	NA	NA
F. Containment spray parameters (flow rate, drop size, etc.)	NA	NA
G. Containment volumes	NA	NA
H. All other pertinent data and assumptions	None	None
III. Dispersion Data		
A. Boundary and LPZ distance (meters)	509/6400	509/6400
B. χ/Q's for		
(1) Total dose - EAB*/LPZ	6.7(-4)/6.7(-5)	1.7(-7)/5.8(-8)
IV. Dose Data		
A. Method of dose calculation	Regulatory Guide 1.5	Reference 2
B. Dose conversion assumptions	Regulatory Guide 1.5	Reference 2
C. Peak activity concentrations in containment	NA	NA
D. Doses	Table 15.6.4-4	Table 15.6.4-4

* Maximum χ/Q occurs at 6400 meters from release point (realistic boundary). The χ/Q at the EAB is 6.1 (-25) sec/m^3.
** Reference 15

TABLE 15.6-6

STEAMLINE BREAK ACCIDENT
ACTIVITY RELEASED TO THE ENVIRONMENT (curies)
(Design (NRC) Basis)

ISOTOPE	CURIES
I-131	1.527E 00
I-132	1.410E 01
I-133	1.046E 01
I-134	2.819E 01
I-135	1.527E 01
TOTAL HALOGENS	6.954E 01
KR-83M	6.950E-02
KR-85M	1.218E-01
KR-85	4.752E-04
KR-87	3.794E-01
KR-88	3.891E-01
KR-89	1.619E 00
XE-131M	3.883E-04
XE-133M	5.806E-03
XE-133	1.626E-01
XE-135M	4.759E-01
XE-135	4.388E-01
XE-137	2.138E 00
XE-138	1.619E 00
TOTAL NOBLE GASES	7.419E 00

TABLE 15.6-7

STEAMLINE BREAK ACCIDENT

ACTIVITY RELEASED TO-THE ENVIRONMENT (curies)

(Realistic Analysis)

ISOTOPE	ACTIVITY
I-131	1.2E-01
I-132	1.1E 00
I-133	8.1E-01
I-134	2.2E 00
I-135	1.2E 00
TOTAL	5.4E 00
KR-83M	2.3E-02
KR-85M	4.1E-02
KR-85	1.6E-04
KR-87	1.2E-01
KR-88	1.3E-01
KR-89	5.4E-01
XE-131M	1.3E-04
XE-133M	1.7E-03
XE-133	5 4E-02
XE-135M	1.6E-01
XE-135	1.4E-01
XE-137	7.1E-01
XE-138	5.4E-01
TOTAL	2.5E 00

TABLE 15.6-8

STEAMLINE BREAK
RADIOLOGICAL EFFECTS

		10 CFR 100		NRC		REALISTIC	
		(2 hours)	(30 days)	(2 hours)	(30 days)	(2 hours)	(30 days)
A. Exclusion Boundary (509 meters)	Whole Body Dose (rem)	25	N/A	3.54E-2	--	5.4E-07	--
	Thyroid Dose (rem)	300	N/A	3.63E-4	--	4.5E-05	--
B. Point of Interest (915 meters)	Whole Body Dose (rem)	N/A	N/A	1.34E-02	--	--	--
	Thyroid Dose (rem)	N/A	N/A	1.31E-00	--	--	--
C. Low Population Zone (6400 meters)	Whole Body Dose (rem)	N/A	25	N/A	1.50E-5	N/A	1.8E-07
	Thyroid Dose (rem)	N/A	300	N/A	1.46E-3	N/A	1.5E-05

TABLE 15.6-9
(SHEET 1 OF 1)

LOSS-OF-COOLANT ACCIDENT - PARAMETERS
TO BE TABULATED FOR POSTULATED ACCIDENT ANALYSES

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TABLE 15.6-10

LOSS-OF-COOLANT ACCIDENT

PRIMARY CONTAINMENT ACTIVITY (Curies)

(Design (NRC) Basis)

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TABLE 15.6-11

LOSS-OF-COOLANT ACCIDENT

ACTIVITY RELEASED TO ENVIRONS (curies)

(Design (NRC) Basis)

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TABLE 15.6-12

LOSS-OF-COOLANT ACCIDENT
(DESIGN-BASIS ANALYSIS)
RADIOLOGICAL EFFECTS

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TABLE 15.6-13

LOSS-OF-COOLANT ACCIDENT

ACTIVITY AIRBORNE IN THE CONTAINMENT (curies)

(Realistic Analysis)

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TABLE 15.6-14

LOSS-OF-COOLANT ACCIDENT

ACTIVITY AIRBORNE IN THE REACTOR BUILDING (curies)

(Realistic Analysis)

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TABLE 15.6-15

LOSS-OF-COOLANT ACCIDENT

ACTIVITY RELEASED TO THE ENVIRONMENT (curies)

(Realistic Analysis)

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TABLE 15.6-16

LOSS OF COOLANT ACCIDENT

(REALISTIC ANALYSIS)

RADIOLOGICAL EFFECTS

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TABLE 15.6-17

LOSS-OF-COOLANT ACCIDENT
RADIOLOGICAL EFFECTS

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TABLE 15.6-18

LOSS-OF-COOLANT ACCIDENT (LOCA)

CONTROL ROOM DOSES
(Design (NRC) Basis)

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TABLE 15.6-19

FEEDWATER LINE BREAK ACCIDENT - PARAMETERS TO BE TABULATED FOR POSTULATED ACCIDENT ANALYSES

	NRC ASSUMPTIONS	CONSERVATIVE ASSUMPTIONS
I. Data and assumptions used to estimate radioactive sc from postulated accidents		
A. Power level corresponding to 102% rated Core Thermal Power**	NA	3559 MWt
B. Burnup	NA	NA
C. Fuel damaged	NA	None
D. Release of activity by nuclide	NA	Table 15.6.6-2
E. Iodine fractions		
(1) Organic	NA	0
(2) Elemental	NA	1
(3) Particulate	NA	0
F. Reactor coolant activity before the accident	NA	Subsection 15.6.6.5
II. Data and assumptions used to estimate activity released		
A. Containment leak rate (%/day)	NA	NA
B. Secondary containment leak rate (%/day)	NA	NA
C. Isolation valve closure time (sec)	NA	30
D. Adsorption and filtration efficiencies		
(1) Organic iodine	NA	NA
(2) Elemental iodine	NA	NA
(3) Particulate iodine	NA	NA
(4) Particulate fission products	NA	NA
E. Recirculation system parameters		
(1) Flow rate	NA	NA
(2) Mixing efficiency	NA	NA
(3) Filter efficiency	NA	NA
F. Containment spray parameters (flow rate, drop size, etc.)	NA	NA
G. Containment volumes	NA	NA
H. All other pertinent data and assumptions	NA	None
III. Dispersion Data		
A. Boundary and LPZ distance (meters)	NA	509/6400
B. χ /Q's for time intervals of		
(1) 0-2 hr - EAB*/LPZ	NA	1.7(-7)/5.8(-8)
IV. Dose Data		
A. Method of dose calculation	NA	Reference 1
B. Dose conversion assumptions	NA	Reference 1
C. Peak activity concentrations in containment	NA	NA
D. Doses	NA	Table 15.6.6-3

* Maximum χ/Q occurs at 6400 meters from release point (realistic boundary). The χ/Q at the EAB is 6.1 (-25) sec/m^3
** Reference 15

TABLE 15.6-20

FEEDWATER LINE BREAK

ACTIVITY RELEASED FROM THE BREAK (curies)

(Realistic Analysis)

ISOTOPE	ACTIVITY
I-131	2.1E-03
I-132	2.0E-02
I-133	1.4E-02
I-134	3.9E-02
I-135	2.1E-02
TOTAL	9.6E-02

NOTE: The radiological consequences for power uprate (105% of 3323 MWt) are estimated (Reference 15) to increase by a factor of 6%. See Section 15.6.6.5.

TABLE 15.6-21

FEEDWATER LINE BREAK
RADIOLOGICAL EFFECTS (PUFF RELEASE)

		10 CFR 100		CONSERVATIVE	
		(2 hours)	(30 days)	(2 hours)	(30 days)
A. Exclusion Boundary (509 meters)	Whole Body Dose (rem)	25	N/A	8.1E-09	--
	Thyroid Dose (rem)	300	N/A	7.9E-07	--
B. Point of Interest (915 meters)	Whole Body Dose (rem)	N/A	N/A	--	--
	Thyroid Dose (rem)	N/A	N/A	--	--
C. Low Population Zone (6400 meters)	Whole Body Dose (rem)	N/A	25	N/A	2.7E-09
	Thyroid Dose (rem)	N/A	300	N/A	2.7E-07

NOTE: The radiological consequences for power uprate (105% of 3323 MWt) are estimated (Reference 15) to increase by a factor of 6%. See Section 15.6.6.5.

TABLE 15.6-22

PARAMETERS USED IN ACCIDENT ASSESSMENT

INTENTIONALLY DELETED

TABLE 15.6-23

LOCA SOURCE TERMS: 15% CORE INVENTORY OF IODINE, 100% CORE INVENTORY OF NOBLE GAS*

INTENTIONALLY DELETED

TABLE 15.6-24

ATMOSPHERIC DISPERSION FACTORS

INTENTIONALLY DELETED

TABLE 15.6-25

EXCLUSION AREA BOUNDARY AND LOW POPULATION ZONE DOSES WITH SIEMENS SOURCE, REM

INTENTIONALLY DELETED

TABLE 15.6-26

TOTAL POST-LOCA CONTROL ROOM AND AEER
DOSES WITH SIEMENS SOURCE, REM

INTENTIONALLY DELETED

TABLE 15.6-27
(Sheet 1 of 2)

AST LOCA Analysis Parameters

Parameter	AST Value
Power Level	3559 MWt
Primary containment volume - Drywell free volume - Wetwell airspace volume	 229,538 ft^3 164,800 ft^3
Suppression pool water volume (pre-LOCA)	128,800 ft^3 (minimum) 131,900 ft^3 (maximum)
Primary containment leak rate	1.0% per day
Secondary containment bypass	None except for MSIV leakage
SGT system filter efficiency (HEPA and Charcoal)	99% after drawdown (with a 0.5% bypass leakage)
Reactor Building drawdown time	15 minutes
MSIV leakage rates	400 scfh total 200 scfh single line
ECCS leakage rate into secondary containment	5 gallons per minute
Emergency makeup filter unit flow	4000 ± 10% cfm
CR recirculation filter flow	18,000 cfm
CR recirculation filter bypass	900 cfm
CR outside air unfiltered inleakage after makeup filter	55.2 cfm
CR outside air unfiltered inleakage into low pressure ductwork before recirculation filter	2,400 cfm
CR outside air unfiltered inleakage rate after recirculation filter	50 cfm
CR intake filter efficiency: Charcoal HEPA	 90% 99%
CR recirculation filter charcoal filter efficiency	70%

TABLE 15.6-27
(Sheet 2 of 2)

AST LOCA Analysis Parameters

Parameter	AST Value
CR HVAC system activation times after LOCA signal - makeup filter - recirculation filter	 t = 20 minutes t = 4 hours
CR occupancy requirements	0-24 hrs: 1.0 1-4 days: 0.6 4-30 days: 0.4
AEER occupancy	Only mission occupancy is required to start the Hydrogen Recombiner system fan for containment mixing. Worst-case time was assumed, including drawdown time when SGT system filtration is not credited. Conservative total mission time is 30 minutes. This mission was assumed to be performed by an operator not assigned full-time to the CR.
AEER outside air unfiltered inleakage	100,000 cfm (artificial value, used to prevent credit for delay in activity intake)
AEER filtration system consideration	No credit for protection by the makeup filter or the AEER recirculation filter (prevents credit for delay in activity intake).
AEER volume	68,800 ft^3
EAB and LPZ - χ/Q's	See Section 2.3.4.a for new AST χ/Q's
CR/AEER - χ/Q's	See Section 2.3.4.a for new AST χ/Q's

TABLE 15.6-28
(Sheet 1 of 2)

AST LOCA Source Terms

Nuclide	Ci/MWt
Co-58	1.529E+02
Co-60	1.830E+02
Kr-85	3.946E+02
Kr-85m	8.313E+03
Kr-87	1.633E+04
Kr-88	2.303E+04
Rb-86	6.518E+01
Sr-89	2.798E+04
Sr-90	3.178E+03
Sr-91	3.801E+04
Sr-92	4.017E+04
Y-90	3.272E+03
Y-91	3.448E+04
Y-92	4.029E+04
Y-93	4.526E+04
Zr-95	4.489E+04
Zr-97	4.657E+04
Nb-95	4.512E+04
Mo-99	5.078E+04
Tc-99m	4.447E+04
Ru-103	4.242E+04
Ru-105	2.966E+04
Ru-106	1.760E+04
Rh-105	2.796E+04
Sb-127	2.944E+03
Sb-129	8.725E+03
Te-127	2.918E+03
Te-127m	3.909E+02
Te-129	8.584E+03
Te-129m	1.279E+03
Te-131m	3.892E+03
Te-132	3.829E+04
I-131	2.695E+04
I-132	3.889E+04
I-133	5.556E+04
I-134	6.165E+04
I-135	5.192E+04
Xe-133	5.491E+04
Xe-135	2.228E+04

TABLE 15.6-28
(Sheet 2 of 2)

AST LOCA Source Terms

Nuclide	Ci/MWt
Cs-134	7.280E+03
Cs-136	2.027E+03
Cs-137	4.538E+03
Ba-139	5.084E+04
Ba-140	4.896E+04
La-140	5.019E+04
La-141	4.640E+04
La-142	4.532E+04
Ce-141	4.492E+04
Ce-143	4.427E+04
Ce-144	3.596E+04
Pr-143	4.293E+04
Nd-147	1.838E+04
Np-239	5.540E+05
Pu-238	1.796E+02
Pu-239	1.207E+01
Pu-240	1.308E+01
Pu-241	6.257E+03
Am-241	9.797E+00
Cm-242	2.388E+03
Cm-244	2.602E+02

TABLE 15.6-29

AST LOCA Dose Summary

Location			
EAB (REM TEDE)	**LPZ (REM TEDE)**	**CR (REM TEDE)**	**LOCA Dose Contributor**
2.43	0.22	1.76	Primary containment leakage unfiltered for 15 minutes and SGT system filtered thereafter (i.e., 100% of La)
0.05	0.04	2.37	MSIV leakage
0.11	0.01	0.10	ECCS leakage in secondary containment
N/A	N/A	0.04	Gamma shine to CR general area
2.59	**0.27**	**4.27**	**Total Calculated Value**
25	**25**	**5**	**Regulatory Limits**

15.7 RADIOACTIVE RELEASE FROM SUBSYSTEMS AND COMPONENTS

15.7.1 Radioactive Gas Waste System Leak or Failure

Radioactive Gas Waste System Leaks or Failures are not analyzed for reload cores.

This information is based on pre-power uprate normal operating design basis source terms. The existing normal operating design basis source terms are bounding for power uprate. Therefore, 105% power uprate (i.e., 105% of 3323 MWt) has no impact on the Section 15.7.1 analyses (Reference 12).

The NRC Standard Review Plan (SRP) NUREG-0800, Revision 1 dated July 1981 (formerly NUREG 075/087) deleted this section from the SRP.

The following radioactive gas waste system components were examined under severe failure mode conditions for effects on the plant safety profile:

a. main condenser off-gas treatment system failure,

b. failure of air ejector lines, and

c. malfunction of main turbine gland sealing system.

15.7.1.1 Main Condenser Gas Treatment System Failure

15.7.1.1.1 Identification of Causes and Frequency Classification

Those events which could conceivably cause a gross failure of a charcoal adsorber tank, a prefilter vessel, or a holdup pipeline in the off-gas treatment system are:

a. a seismic occurrence - greater than design basis;

b. a hydrogen detonation which ruptures the system pressure boundary;

c. a fire in the filter assemblies; and

d. failure of adjacent equipment, which could subsequently compromise off-gas equipment.

A seismic occurrence is considered to be the most probable and severe event which the system is designed to accommodate. Nevertheless, the seismic failure is the only conceivable event which could cause significant hardware damage that might

result in a radiological release. The equipment and piping are designed to contain any hydrogen-oxygen detonation which has a reasonable probability of occurring. A detonation is not considered as a possible failure mode. The decay heat on the filters is insignificant and cannot serve as an ignition source for the filters. It can be easily accommodated inherently by the system and certainly by the available air flows. The system is reasonably isolated from other systems or components which could cause any serious interaction or failure.

Thus the only credible event which could result in the release of significant activity to the environment is an earthquake that causes building damage and subsequent rupture of off-gas components from falling building debris.

Even though the off-gas system is designed to uniform building code seismic requirements, an event more severe than the design requirements is arbitrarily assumed to occur, resulting in the failure of the off-gas system.

Failure of the off-gas treatment system is postulated to occur with the frequency of a limiting fault.

The design basis, description, and performance evaluation of the subject system is given in Section 11.3.

15.7.1.1.2 Sequence of Events and Systems Operation

Gross failure of the off-gas system results in the activation of area radiation alarms which alert the plant personnel. When the station vent stack radiation monitor reaches its limit, the off- gas system is manually isolated from the condenser, resulting in a high backpressure on the condenser and a reactor scram. The postulated failures which can release radioactivity into their respective building atmospheres include:

a. Adsorber tanks and prefilter vessels release to the off-gas building.

b. Delay line pipe failure is assumed to release radioactivity from the inlet end into the turbine building.

The sequence of events following equipment failure is as follows:

Approximate Elapsed Time	Events
0 sec	Event begins - system fails
0 sec	Noble gases are released
< 1 min	Area radiation alarms alert plant personnel

< 1 min	Operator actions begin with: (a) initiation of appropriate system isolations (b) manual scram actuation (c) assurance of reactor shutdown cooling.

Normal operator actions assure evacuation of the affected radiation areas and assure isolation of the turbine building via its HVAC dampers. The turbine building HVAC flow is then exhausted through the station vent stack. Only 2 minutes is needed to accomplish these operator actions.

In analyzing the postulated off-gas system failure, no credit is taken for the operation of plant and reactor protection systems, or of engineered safety features. Credit is taken for functioning of normally operating plant instruments and controls and other systems only to the following extent:

a. capability to detect the failure itself - indicated by an alarmed increase in radioactivity levels seen by area radiation monitoring system, by an alarmed loss of flow in the off-gas system, or by an alarmed increase in activity at the station vent stack;

b. capability to isolate the off-gas system and shut down the reactor; and

c. operational indicators and annunciators in the main control room.

The seismic event, which is assumed to occur beyond the present plant design basis for non-safety equipment, will cause the tripping of turbine or will lead to a load rejection. This initiates a scram and negates the need to postulate operator actions for a reactor shutdown.

15.7.1.1.3 Core and System Performance

The postulated failure results in a system isolation necessitating reactor shutdown because of loss of vacuum in the main condenser. This transient has been analyzed in Subsection 15.2.5. Otherwise, this auxiliary system does not directly affect the reactor core nor the power cycle systems but is coupled only through operator alarms in the control room.

The iodine activity leaving the off-gas recombiner has been assumed to be entirely retained in the first charcoal tank. Thus failure of this tank results in the highest potential iodine release. Iodine absorbs strongly to charcoal. A conservative

evaluation leads to an assumption of the release of 1% of the iodine in the first charcoal tank.

Of primary importance in determining the potential release is the inventory of radioactive products in the equipment pieces before failure. The fractional releases from the equipment, after normal operation, are indicated in Table 15.7-2. These release fractions were used for analysis purposes for both the design- basis analysis and the realistic analysis. Table 15.7-3 lists the isotope inventories contained in the various components of the off-gas system.

The transport pathway to the environment consists of a release of fission products from the failed equipment through the building ventilation system to the station vent stack.

15.7.1.1.4 Barrier Performance

The postulated failure is the rupture of the off-gas system pressure boundary. The events described here occur outside containment, hence do not involve barrier integrity. No credit is taken for performance of secondary barriers, except to the extent inherent in the assumed equipment release fractions discussed herein.

15.7.1.1.5 Radiological Consequences

Separate radiological analyses are provided for the "design basis" case and for the "realistic" case as follows:

a. The "design basis analysis" uses conservative assumptions considered acceptable to the NRC for determining adequacy of plant design to meet 10 CFR 100 guidelines.

b. The "realistic analysis" uses engineering inputs and field experience (actual meteorological data for instance), to determine the radiological consequences.

Both analyses assume the following equipment characteristics with respect to the retention or passage of radioactive solid daughter products prior to initiation of the seismic event that causes failure of the off-gas equipment:

a. Off-gas condenser and water separator - 100% retained and continuously washed out with condensate.

b. Delay line pipe - 60% retained and continuously washed out with condensate.

c. Cooler condenser and moisture separator - 100% retained and continuously washed out with condensate.

d. Prefilter - 100% retained, element changed annually.

e. Charcoal adsorbers - 100% retained.

f. After filter - 100% retained, element changed annually.

Components not listed are assumed to have zero retention of solid daughter products.

The differences in system performance in the postulated accidents are due to differences in the magnitude of the source term for the equipment and the assumptions relative to release fractions. The same analytical techniques were used for these separate analyses.

Table 15.7-1 lists the parameters used in the analyses.

Design-Basis Analysis

There are no specific NRC guidelines for this analysis. Nevertheless, an evaluation has been performed to provide comparative results for this event.

The activity in the off-gas system is based on the following initial conditions: 2 scfm air inleakage, and 100,000 μCi/sec noble gas release after 30 minute delay for a period of 11 months, followed by 1 month of 350,000 μCi/sec at 30 minutes.

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

a. Charcoal adsorber tanks - these tanks are assumed to fail circumferentially from falling concrete that tears 50% of the circumference, or that creates a shearing load resulting in the same size break in the tank. These failures would result in no more than 10% to 15% of the charcoal (carbon) being displaced from the vessel. Iodine is strongly bonded to the charcoal and would not be expected to be removed by exposure to the air. Nevertheless, to be conservative, the assumption is made that 1% of the iodine activity contained in the adsorber tank is released to the vault where the tanks are located. Additionally, the conservative assumption is made that 1% of the solid daughter products retained in the charcoal is also released.

Measurements made at KRB indicate that off-gas is about 30% richer in Kr than air. Therefore, if this carbon is exposed to air, it will eventually reach equilibrium with the noble gases in the air. However, the first few inches of carbon will blanket the underlying carbon from the air. A 10% loss of noble gas activity from a failed vessel is conservative because of the small fraction of carbon exposed to the air.

b. Prefilters - Because of the design features of the prefilter vessel (approximately 24-inch diameter, 4-foot height, 350 psig design pressure, 1/2-inch wall thickness and collapsible filter media), a failure mechanism cannot be postulated that will result in emission of filter media or daughter products from this vessel. However, to illustrate the consequences of a radioactivity loss from this vessel, 1% release of particulate activity is assumed.

c. Delay line pipe - Pipe rupture and depressurization of the pipe is considered. For the design-basis analysis, 100% of the noble gases and all of the remaining solid daughters after the 60% washout are assumed to be released.

d. Piping - It is assumed that the seismic event causing the pipe failure is accompanied by a reactor isolation that stops steam flow to the steam jet air ejectors. Therefore, the resulting release from failed piping is not significant compared to those failures previously considered.

Results

The calculated exposure for the design-basis analysis is tabulated in Table 15.7-4. The doses are well within the 10 CFR 100 guidelines.

A 105% power uprate to 3489 MWt has no impact on either the gaseous radwaste system or the design basis radioactive gas source term. Therefore, the results presented in this analysis are not impacted by power uprate (Reference 12).

Realistic Analysis

The realistic analysis is based on an engineered but still conservative assessment of this event. The specific models, assumptions, and the program used for computer evaluation are described in Reference 1. Specific values of parameters used in the evaluation are presented in Table 15.7-1.

The activity in the off-gas system is based on the following initial conditions:

a. 18.5 scfm air inleakage, and

b. 100,000 μCi/sec noble gas after 30-minute delay.

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

a. Charcoal adsorber tanks - The same tank failure mechanisms are assumed as those used in the design-basis analysis. The release fractions for radionuclides are the same, except for the solid daughters. No logical mechanism exists for any of the solid daughter products formed and retained within the micropore structure of the carbon to be released. Hence, no such release is assumed for the realistic analysis.

b. Prefilters - Because of the design features of the prefilter vessel, approximately 24-inch diameter, 4-foot height, 350-psig design pressure, 1/2-inch wall thickness, and collapsible filter media, a failure mechanism cannot be postulated that will result in emission of filter media or daughter products from this vessel. However, to illustrate the consequences of a radioactivity loss from this vessel, 1% release of particulate activity is assumed.

c. Delay line pipe - Pipe rupture and depressurization of the pipe is considered. Normally, the pipe will operate at less than 16 psia and depressurize to 14.7 psia. The possible loss of solid daughters and noble gases is conservatively taken as 20%. The model used assumes retention and washout of 60% of the particulate daughters for the calculation of the holdup pipe inventory.

d. Piping - It is assumed that the seismic event causing the pipe failure is accompanied by a reactor isolation, stopping steam flow to the steam jet air ejectors. Therefore, the resulting release from failed piping is not significant compared to those failures previously considered.

The release of activity to the environment is determined by applying the release fractions in Table 15.7-2 to the inventories shown in Table 15.7-3. The calculated exposures are presented in Table 15.7-4.

The only credible failure that could result in loss of carbon from the vessels is the failure of the concrete structure surrounding the vessel. A circumferential failure of the vessel could result from concrete falling on the vessel in either of two ways:

a. Bending Load - The vessel supported in the center and loaded on each end. This could result in a tear around 50% of the circumference.

b. Shearing Load - The vessel being supported and loaded near the same point from above.

In either case, no more than 10-15% of the carbon would be displaced from the vessel. Iodine is strongly bonded to the charcoal and would not be expected to be removed by exposure to the air. However, a conservative assumption is made, that 1% of the iodine activity contained in the absorber tanks is released to the vault containing the off-gas equipment.

Measurements made at KRB indicate that off-gas is about 30% richer in Kr than air. Therefore, if this carbon is exposed to air, it will eventually reach equilibrium with the noble gases in the air. However, the first few inches of carbon will blanket the underlying carbon from the air. A 10% loss of noble gas activity from a failed vessel is conservative because of the small fraction of carbon exposed to the air.

There is no reason to believe that any of the solid daughter products formed and retained within the micropore structure of the carbon will be released. Hence no such release is assumed for the realistic analysis.

Results

The calculated exposures for the realistic analysis are presented in Table 15.7-4. Note that station 2-year observations were used for the X/Q's used for the realistic analysis.

A 105% power uprate to 3489 MWt has no impact on either the gaseous radwaste system or the design basis radioactive gas source term. Therefore, the results presented in this analysis are not impacted by power uprate (Reference 12).

15.7.1.2 Failure of Air Ejector Lines

15.7.1.2.1 Identification of Causes and Frequency Classification

Failure of the air ejector inlet line upstream of its isolation valve to the condenser is arbitrarily assumed to occur with the frequency of a limiting fault. This line is designed to contain an explosion, therefore an explosion is not considered as a possible cause of failure.

15.7.1.2.2 Sequence of Events and Systems Operation

It is assumed that the line leading to the steam jet air ejector fails near the condenser. This results in activity normally processed by the off-gas treatment system being discharged directly to the turbine building and subsequently through the ventilation system to the environment. This failure results in a loss-of-flow signal to the off-gas system, thus closing the condenser isolation valve downstream of the assumed break.

The operator initiates a normal shutdown of the reactor within 30 minutes to reduce the gaseous activity being discharged. A loss of condenser vacuum will result in a turbine trip and reactor shutdown.

15.7.1.2.3 Core and System Performance

This auxiliary system does not directly affect the reactor core nor the power cycle systems but is coupled only through operator alarms in the control room. Failure of the air ejector lines has no applicable effect on NSSS safety performance.

15.7.1.2.4 Barrier Analysis

This failure occurs outside the containment, hence it does not involve barrier integrity.

15.7.1.2.5 Radiological Consequences

Specific regulatory guide calculation method and assumptions are not available, therefore only the realistic basis analysis is provided. The specific models, assumptions, and computer program are discussed in Reference 1. Table 15.7-5 itemizes those parametric values applicable to the realistic analysis.

The radiological exposures have been evaluated for the meteorological conditions defined in Subsection 15.0.5. There is no fuel damage as a consequence of this event, therefore the only activity released to the environment is that associated with the fluid processed by the off-gas system. This activity is discharged to the turbine building ventilation system and subsequently to the environment via the station vent stack. Because operator action is assumed to occur within 30 minutes, an uncontrolled release to the environment for this time period has been assumed. No credit is taken for plateout of iodine in the turbine building.

The activity released is shown in Table 15.7-6, and the calculated exposures are given in Table 15.7-7.

A 105% power uprate to 3489 MWt has no impact on either the gaseous radwaste system or the design basis radioactive gas source term. Therefore, the results presented in this analysis are not impacted by power uprate (Reference 12).

15.7.1.3 Malfunction of Turbine Gland Sealing System

15.7.1.3.1 Identification of Causes and Frequency Classification

Failure of various components of the turbine gland sealing system can lead to system malfunction.

The source for sealing steam for the turbine gland seals is a separate low-radioactivity steam drum on an evaporator unit that is heated by main steam. The leakage from the turbine gland sealing system is therefore low-radioactivity steam for normally anticipated leaks. Those leaks resulting from component malfunctions which invalidate or

partially compromise this source of low-radioactivity sealing steam are discussed below. This event is categorized as a limiting fault.

15.7.1.3.2 Sequence of Events and Systems Operation

Loss of Sealing Steam

The instrumentation on the steam seal header detects low sealing steam pressure. The steam seal evaporator can be bypassed manually, and the main steam supply to the steam seal header can be connected. Assuming all sources of sealing steam are lost during the postulated malfunction, the following events will take place. In the high-pressure turbine glands, high-pressure steam will filter through the glands into the steam seal exhaust header. In the low-pressure turbine glands, air will be drawn into the sealing glands. The action of cool air quenching the hot turbine shaft will cause excessive shaft vibration.

Water Induction to Turbine Shaft Seal Glands

During normal operation, the level control system on the steam seal evaporator guards against a high water level by alarm. Automatic control then closes off the condensate quality water feedline, which causes the water level in the steam seal evaporators to lower. In case the high water level situation is not alleviated, the motor-operated valve on the SSE condensate header side of the evaporators will be closed by the high level control. This prohibits water intrusion into the turbine glands and the turbine casing.

Loss of Vacuum in the Gland Steam Condenser

During normal operation, noncondensibles are removed from the gland steam condenser by one of two gland steam condenser blowers. In the event this blower malfunctions, the backup blower is manually started to take over the gas removal requirements. Assuming loss of both blowers, vacuum will be lost in the gland steam condenser. The pressure in the gland steam exhaust header will increase to greater than atmospheric, thus causing sealing steam to leak to the turbine building through the turbine glands.

Loss of Cooling Medium for Gland Steam Condenser

During normal operation, main condensate provides the cooling medium for the gland steam condenser. In the event this cooling water capacity is lost, the exhausted sealing steam will not be condensed. This results in the same situation as loss of vacuum described above and causes a pressure buildup in the exhaust header of the sealing steam system with consequent leakage of sealing steam.

Loss of Sealing Steam to Control, Intercept, and Stop Valves

During normal operation, the stems of the intercept valves, stop valves, and control valves are sealed by steam from the turbine gland sealing system. In the event of a sealing steam malfunction, the valves will leak main steam and reheated steam.

Description of Operator Action

No operator action except to manually start the redundant gland steam condenser blower or an orderly shutdown is required for any of the previous malfunctions in the turbine gland sealing system.

15.7.1.3.3 Core and System Performance

The plant is operating at full power prior to failure of some component of the normal operation gland sealing system. There are three means of providing heated steam to the steam seal evaporators. Main steam is used by the evaporators during startup and low load operation. An extraction steamline provides the evaporators with heating steam during normal and high-load operation. There is also a supply line from an auxiliary boiler which can provide steam to the evaporators during testing and startup.

In the event of the occurrence of loss of sealing steam, the action of cool air being drawn into the sealing glands and the resultant shaft vibration will cause tripping of the turbine-generator by the excessive shaft vibration trip. This tripping mechanism is independent of operator action and, consequently, will produce a rapid and safe shutdown of the turbine-generator.

The input parameters and initial conditions for this event are listed in Subsection 15.0.2.

There are no cladding perforations as a result of this event and no fission product release occurs.

15.7.1.3.4 Barrier Performance

The event does not involve the barrier since the release occurs outside of containment.

15.7.1.3.5 Radiological Consequences

The amount of steam released to the turbine building due to the above hypothesized events would be small due to the close clearances in the turbine shaft sealing glands. Any release would be well below the release used for the failure of the air ejector lines (Subsection 15.7.1.2).

The radiological effects would be inconsequential since the releases would be well below the releases used for the failure of the air ejector lines.

A 105% power uprate to 3489 MWt has no impact on either the gaseous radwaste system or the design basis radioactive gas source term. Therefore, the results presented in this analysis are not impacted by power uprate (Reference 12).

15.7.2 Radioactive Liquid Waste System Leak

Radioactive Liquid Waste System Leaks are not analyzed for reload cores.

15.7.2.1 Miscellaneous Small Releases Outside Containment

Releases which could occur from piping failures outside the containment include the feedwater system piping break (Subsection 15.6.6) and the main steamline break (Subsection 15.6.4) accidents. The analysis of these events provides doses which might occur for such a classification of piping failure events.

Other releases which could occur outside containment include small spills and leaks of radioactive materials inside structures housing process equipment. Conservative values for leakage have been assumed and evaluated in Chapter 11.0 under routine plant releases. The offsite dose that results from any small spill which could occur outside containment would be negligible in comparison to the dose resulting from the previous postulated leakages of 15.6.

15.7.3 Postulated Radioactive Releases Due to Liquid Radwaste Tank Failure

Liquid Radwaste Tank Failures are not analyzed for reload area.

15.7.3.1 Identification of Causes and Frequency Classification

An unspecified event causes the complete release of the average radioactivity inventory in the tank containing the largest quantities of significant radionuclides in the liquid radwaste system. This is one of the concentrator waste tanks in the radwaste building. The airborne radioactivity released during the accident passes directly to the environment via the station vent stack.

The radioactive liquid releases seep through the radwaste building foundation and move through the soil toward the cooling lake.

Postulated events that could cause release of the radioactive inventory of the concentrator waste tank are cracks in the vessels and operator error. The possibility of small cracks and consequent low-level release rates receives primary consideration in system and component design. The concentrates waste tank is designed to operate at atmospheric pressure and 200° F maximum temperature so

the possibility of failure is considered small. A liquid radwaste release caused by operator error is also considered a remote possibility. Operating techniques and administrative procedures emphasize detailed system and equipment operating instruction. A positive action interlock system is also provided to prevent

inadvertent opening of a drain valve. Should a release of liquid radioactive wastes occur, floor drain sump pumps in the floor of the radwaste building will receive a high water level alarm, activate automatically, and remove the spilled liquid.

Much of the exposition concerning the remote likelihood of a leakage or malfunction accident of the concentrates waste tank applies equally to a complete release accident. The probability of a complete rupture or complete malfunction accident is, however, considered even lower.

Although not analyzed for the requirements of Seismic Category I equipment, the liquid radwaste tanks are constructed in accordance with sound engineering principles. Therefore, simultaneous failure of all the tanks is not considered credible. This accident is expected to occur with the frequency of a limiting fault.

Note: Although the concentrator waste tanks have been abandoned -in- place (UFSAR Section 11.2.2.5.5), the use of the assumed tank contents for this analysis remains bounding for all other liquid radwaste tanks.

15.7.3.2 Sequence of Events and Systems Operation

The sequence of events expected to occur is as follows:

Sequence of Events	Elapsed Time
1. Event begins -- failure occurs	0
2. Area radiation alarms alert plant personnel	1 minute
3. Operator actions begin	5 minutes

The rupture of a concentrates waste tank would leave little recourse to the operator. No method of recontaining the gaseous phase discharge is available, however, isolation of the radwaste area would minimize the results. High radiation alarms both in the radwaste ventilation exhaust and in the radwaste area would alert the operator to the failure.

Normal isolation of the radwaste area ventilation is actuated upon initiation of the above alarms. However, no credit for any operator action or for ventilation isolation has been taken in evaluating this event.

15.7.3.3 Core and System Performance

The failure of this liquid radwaste system component does not directly affect either the core or the nuclear steam supply system (NSSS). It will, of course, lead to decoupling of NSSS with the subject system.

The analytical methods and associated assumptions which are used to evaluate the consequences of the accident are considered to provide a conservative assessment of the consequences.

The liquid radwaste tank failure is evaluated in accordance with the following assumptions and conditions:

a. One hundred percent of the shielding design-basis inventory of a concentrates waste tank is released into a concentrated waste tank cubicle.

b. The partition factor for radioiodine from water to air is 0.001.

c. The airborne radioactivity released into a concentrates waste tank cubicle passes to the environment via the station vent stack.

d. The (X/Q) values are from Subsection 15.0.5.

e. The concentrated waste tank cubicle is designed to retain the entire contents of the tank.

f. The cubicle floor has no floor drains.

g. The cubicle has a steel liner which is capable of holding the entire contents of one tank.

h. The liquid waste is unable to seep out of the cubicle.

i. Eighty percent of the liquid inventory of one concentrated waste tank is released to the cubicle.

j. The hydraulic gradient is toward the lake.

k. The shielding design-basis inventory given in Table 11.2-5 is dissolved in a sufficient quantity of liquid to produce 5000 gallons of concentrated waste.

l. No credit has been taken for dilution in groundwater or for dispersion.

m. Additional parameters are given in Subsection 2.4.12.

Results

The conservative assessment of the liquid radwaste tank failure leads to consideration of iodine partitioning from a spill which would be drained in a rapid manner, thereby minimizing the release of iodine to the air.

The radwaste building exhaust is filtered through HEPA filters for which no reduction in released iodine has been assumed. Table 15.7-8 presents the parametric values used in the conservative analysis and Table 15.7-10 lists the radionuclide activity release to the environment.

Because of the design features incorporated in LSCS, i.e., as the liquid radwaste tanks are each located in vented and drained cells, the failure of a liquid radwaste tank will produce virtually no additive effects above doses from normal operation.

The most conservative liquid effluence model would involve no groundwater dilution or dispersion. The model relies only on the soil permeability, the soil porosity, and the hydraulic gradient to determine the liquid concentrations that will reach the lake. These parameters are used to derive the travel time from the radwaste building to the lake.

If the concentrated waste volume remains constant, 600 years would be required for the radioisotope concentration to become less than the limits specified in 10 CFR 20, Appendix B, Table II, Column 2. Table 5.7-11 gives maximum liquid effluent concentrations versus time for the assumptions given above and in Subsection 2.4.12.

The liner in the concentrate tank cubicle will prevent any liquid waste from reaching the groundwater or the lake. Water sampling (see LSCS-ER (OLS), Chapter 6.0) has measured background beta radiation levels between 4 pCi/liter and 35 pCi/liter. Since no concentrated waste liquid can reach the lake, the activity in the lake water will fall between these values. All other radwaste tanks are located below the level of the lake surface. No liquid radwaste from these is released to the environment (see Subsection 2.4.12).

15.7.3.4 Barrier Performance

This event does not involve the barrier since the release occurs outside of containment.

15.7.3.5 Radiological Consequences

Realistic Analysis

The radiological effects are based on a short-term release to the atmosphere using the meteorological parameters presented in Subsection 15.0.5. The whole-body dose results from the gamma radiation emitted by the iodine. The doses from airborne releases are presented in Table 15.7-9.

The liquid releases due to the rupture of one concentrated waste tank will not have any adverse effects on the cooling lake or the Illinois River. No radioactivity is added to these bodies of water.

The radwaste equipment's radioactive inventory is a combination of the collection volume, the source stream(s) supply rates, and the activity in the source streams. For liquid radwaste following the 5% power uprate, all these parameters remain unchanged (Reference 12). Therefore, the results of this analysis are not impacted by power uprate.

15.7.4 Fuel Handling Accident

15.7.4.1 Identification of Causes and Frequency Classification

The fuel handling accident is assumed to occur as a consequence of a failure of the fuel assembly lifting mechanism resulting in the dropping of a raised fuel assembly onto other fuel bundles in the core. A variety of events which qualify for the class of accidents termed "fuel handling accidents" has been investigated. The accident which produces the largest number of failed spent fuel rods (including consideration of the drop of a fuel bundle onto the Unit 2 consolidated fuel storage pool) is the drop of a spent fuel bundle onto the reactor core when the reactor vessel head is off. This accident is expected to occur with the frequency of a limiting fault.

15.7.4.2 Sequence of Events and Systems Operation

The most severe fuel handling accident from the radiological viewpoint is the dropping of a fuel assembly onto the top of the core. The sequence of events is as follows:

Event	Approximate Elapsed Time
1. Fuel assembly is being handled by refueling equipment. The assembly drops onto the top of the core.	0

2.	Some of the fuel rods in both the dropped assembly and reactor core are damaged, resulting in the release of gaseous fission products to the reactor coolant and eventually to the reactor building atmosphere.	0

3.	The reactor building ventilation radiation monitoring system alarms to alert plant personnel, isolates the ventilation system, and starts operation of the SGTS.	<1 minute
4.	Operator actions begin.	<5 minutes

Normally, operating plant instrumentation and controls are assumed to function, although credit is taken only for the isolation of the normal ventilation system and the operation of the standby gas treatment system. Operation of other plant or reactor protection systems or ESF systems is not expected.

The automatic ventilation isolation system, which includes: a) the radiation monitoring detectors, b) the isolation valves, and c) the SGTS is designed to single-failure criteria and safety requirements.

Refer to Sections 7.6 and 9.4.

15.7.4.3 Core and System Performance

The analytical methods and associated assumptions used to evaluate the consequences of this accident are considered to provide a realistic yet conservative assessment of the consequences.

The kinetic energy acquired by a falling fuel assembly may be dissipated in one or more impacts. To estimate the expected number of failed fuel rods in each impact, an energy approach is used.

The fuel assembly is expected to impact on the reactor core at a small angle from the vertical, possibly inducing a bending mode of failure on the fuel rods of the dropped assembly. It is assumed that each fuel rod resists the imposed bending load by a couple consisting of two equal, opposite concentrated forces.

Therefore, fuel rods are expected to absorb little energy prior to failure as a result of bending. Actual bending tests with concentrated point-loads show that each fuel rod absorbs approximately l ft-lb prior to cladding failure. Each rod that fails as a result of gross compression distortion is expected to absorb approximately 250 ft-lb before cladding failure (based on 1% uniform plastic deformation of the rods). The energy of the dropped assembly is conservatively assumed to be absorbed by only the cladding and other core structures. Because a fuel assembly consists of 72% fuel, 11% cladding, and 17% other structural material by weight, the assumption that no energy is absorbed by the fuel material results in considerable conservatism in the mass-energy calculations that follow.

The energy absorption on successive impacts is estimated by considering a plastic impact. Conservation of momentum under a plastic impact shows that the fractional kinetic energy absorbed during impact is:

$$KE = \left[\left(1 - \frac{M}{M_1 + M_2}\right)\right] \qquad (15.7-1)$$

where M_1 is the impacting mass and M_2 is the struck mass.

The assumptions used in the analysis of this accident are listed below:

a. The fuel assembly is dropped from the maximum height allowed by the fuel handling equipment, which is less than 30 feet.

b. The entire amount of potential energy, referenced to the top of the reactor core, is available for application to the fuel assemblies involved in the accident. This assumption neglects the dissipation of some of the mechanical energy of the falling fuel assembly in the water above the core and requires the complete detachment of the assembly from the fuel hoisting equipment. This is only possible if the fuel assembly handle, the fuel grapple, or both grapple cables break.

c. None of the energy associated with the dropped fuel assembly is absorbed by the fuel material (uranium dioxide).

Energy Available

Dropping a fuel assembly onto the reactor core from the maximum height allowed by the refueling equipment, less than 30 feet, results in a maximum impact velocity of 40 ft/sec.

The kinetic energy acquired by the falling fuel assembly is less than 17,300 ft-lb and is dissipated in one or more impacts.

Energy Loss Per Impact

Based on the fuel geometry in the reactor core, four fuel assemblies are struck by the impacting assembly. The fractional energy loss on the first impact is approximately 80%.

The second impact is expected to be less direct. The broad side of the dropped assembly impacts approximately 24 more fuel assemblies, so that after the second impact only 136 ft-lb (approximately 1% of the original kinetic energy) is available for a third impact. Because a single fuel rod is capable of absorbing 250 ft-lb in

compression before cladding failure, it is unlikely that any fuel rod will fail on a third impact.

If the dropped fuel assembly strikes only one or two fuel assemblies on the first impact, the energy absorption by the core support structure results in approximately the same energy dissipation on the first impact as in the case where four fuel assemblies are struck. The energy relations on the second and third impacts remain approximately the same as in the original case. Thus, the calculated energy dissipation is as follows:

first impact 80%

second impact 19%

third impact 1% (no cladding failures)

Fuel Rod Failures

For the initial core, the first impact dissipates 0.80 x 17,300 or 13,800 ft-lb of energy. It is assumed that 50% of this energy is absorbed by the dropped fuel assembly and that the remaining 50% is absorbed by the struck fuel assemblies in the core. Because the fuel rods of the dropped fuel assembly are susceptible to the bending mode of failure and because 1 ft-lb of energy is sufficient to cause cladding failure as a result of bending, all 62 rods of the dropped fuel assembly are assumed to fail. Because the 8 tie-rods of each struck fuel assembly are more susceptible to bending failure than the other 54 fuel rods, it is assumed that they fail on the first impact. Thus 4 x 8 = 32 tie-rods (total in 4 assemblies) are assumed to fail.

Because the remaining fuel rods of the struck assemblies are held rigidly in place in the core, they are susceptible only to the compression mode of failure. To cause cladding failure of one fuel rod as a result of compression, 250 ft-lb of energy is required. To cause failure of all the remaining rods of the 4 struck assemblies, 250 x 54 x 4 or 54,000 ft-lb of energy would have to be absorbed in cladding alone. Thus, it is clear that not all the remaining fuel rods of the struck assemblies can fail on the first impact. The number of fuel rod failures caused by compression is computed as follows:

$$\frac{0.5 \times 13{,}800 \times \dfrac{11}{11+17}}{250} = 11 \qquad (15.7\text{ - }2)$$

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

dropped assembly 62 rods (bending)

struck assemblies 32 tie-rods (bending)

struck assemblies 11 rods (compression)

105 failed rods

Because of the less severe nature of the second impact and the distorted shape of the dropped fuel assembly, it is assumed that in only 2 of the 24 struck assemblies are the tie-rods subjected to bending failure. Thus 2 x 8 = 16 tie-rods are assumed to fail. The number of fuel rod failures caused by compression on the second impact is computed as follows:

$$\frac{\frac{0.19}{2} \times 17{,}300 \times \frac{11}{11+17}}{250} = 3 \qquad (15.7\text{ - }3)$$

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

struck assemblies 16 tie-rods (bending)

struck assemblies 3 rods (compression)

19 failed rods

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

first impact 105 rods

second impact 19 rods

third impact 0 rods

124 total failed rods

The above evaluation, the FSAR Base Case, was based on the GE 8x8 fuel design and considered a drop distance of 30 feet but did not consider the impact of the grapple mast and head. The current grapple mast and head weigh 619 pounds. Alternative fuel designs have been loaded in the core since the GE 8x8 design:GE9 (8x8): ATRIUM-9B (9x9): ATRIUM-10 (10x10); GE14 (10x10); and GNF2 (10x10). The number of fuel pin failures resulting from the Refuel Accident for these fuel designs are summarized below and are being retained for comparison even though some of the fuel designs are no longer in use:

	GE 8x8 (FSAR Base Case)	ATRIUM-9B	ATRIUM-10	GE14	GNF2
Fuel Assembly Drop Distance	30 ft.	34 ft.	34 ft.	34 ft.	34 ft.
Grapple Mast and Head Weight	Not considered	620 lbs	620 lbs	619 lbs	619 lbs
Pins Failed (First impact)	105	116	140	-	-
Pins Failed (Second impact)	19	15	16	-	-
Total Pins Failed	124	131	156	172	172
Reference		6	15	2	23

Reference 22 confirms that Reference 15 remains applicable for ATRIYM-10 fuel at MUR conditions and that Reference 21 remains applicable for ATRIUM 10XM LTA at MUR conditions.

15.7.4.4 Barrier Performance

The reactor coolant pressure boundary and the containment are assumed to be open. All radioactivity is released through the SGTS to the environment. The transport of fission products from the reactor building is discussed below.

15.7.4.5 Radiological Consequences

AST Analysis

The AST analysis (Reference 3) is based on engineering inputs from RG 1.183, which yield a conservative assessment of this accident. Specific input parameters used in this evaluation are shown in Table 15.7-21. The leakage path is the same as that used in the previous non-AST design-basis evaluation, Figure 15.7-1.

RG 1.183 is the basis for the AST evaluations. Concerning the FHA, this AST guidance has smaller gap fractions and a larger pool decontamination factor (DF).

Fission product release estimates for the evaluation of this fuel handling accident use the following assumptions:

- The reactor fuel has an average irradiation time of 1000 days at 3559 MWt up to 24 hours prior to the accident. This assumption results in an equilibrium fission product concentration at the time the reactor is shut down. Longer operating histories do not increase the concentration of fission products of concern. The 24-hour decay time allows time to shut down the reactor, depressurize it, remove the reactor vessel head, and remove the reactor internals above the core. It is not expected that these operations could be accomplished in less than 24 hours, and they would normally require approximately 48 hours.

- This analysis is applicable to fuel whose burnup and power limits are bounded by those specified in RG 1.183, footnote 11, except for ATRIUM-10 partial length rods that were evaluated for operation above 62 GWD/MTU in Unit 1 Cycle 16 and were shown to be bounded by full length rods that comply with footnote limits. This allows application of the gap activity fractions for Non-Loss of Coolant Accident (LOCA) events per Table 3 of RG 1.183, which are as follows:

  5% of the noble gases (excluding Kr-85)
  10% of the Kr-85
  5% of the iodine inventory (excluding I-131)
  8% of the I-131
  12% of the Alkali metal inventory

- Because of the negligible particulate activity available for release in the fuel rod plena, none of these solid fission products is assumed to be released.

- It is assumed that GE14 fuel is used and that 172 fuel rods fail as can be seen in the following table.

Bundle Type	Fuel Type	Assumed Pins-in Bundle	Failed Pins	Damaged Core Fraction	Radial-Peaking Factor (PF)	Damaged Core Fraction with PF
GE-Various	8x8	62	124	0.002618	1.5	0.003927
FANP Atrium-9B	9x9	72	131	0.002381	1.5	0.003572
Atrium-10	10x10	91	156	0.002244	1.7	0.003815
GE11&GE13	9x9	74	140	0.002476	1.5	0.003714
GE12&GE14*	10x10	87.33	172	0.002578	1.7	0.004382
GNF2	10x10	85.6	172	0.002630	1.6	0.004208
* Bounding Assembly type, with Radial Peaking Factor commensurate with full core application						

- The SGTS filter efficiency for HEPA and Charcoal is credited as 99% (with a 0.5% bypass leakage).

- Control Room Emergency Makeup Filter efficiency credited for HEPA is 99% and for Charcoal 90%.

- Recirculation Filtration unit has an effective credit of 70% (not including bypass leakage).

The following assumptions and conditions are used in calculating the release of activity to the environment:

- All of the noble gases released to the fuel pool become airborne in the secondary containment (reactor building).

- The decontamination factors for iodine elemental and organic species are 500 and 1, respectively, when the depth of water above the fuel is 23 feet or greater, therefore giving an effective decontamination factor of 200.

- The ventilation rate from the secondary containment to the environment through the SGTS is 0.1 volume change per hour. This assumption results in 77.678% of the available activity to be released during the drawdown period, 98.889% by the end of the fumigation period, and 99.9994% (effectively all) to be released with 2 hours. The SGTS filter efficiency of 99.0% with a 0.5% bypass leakage allowance is credited only after a 15 minute drawdown period.

The results for this analysis are shown in Table 15.7-22.

15.7.5 Spent Fuel Cask Drop Accident

The Spent Fuel Cask Drop accident is not considered a credible "design basis accident," because the reactor building overhead crane meets the single-failure proof criteria of ASME NOG-1-2004, NUREG-0554 and NUREG-0612, Appendix C. The main hoist is classified as a Type I main hoist per ASME NOG-1 – single-failure proof for all identified critical loads. The overhead crane is designed to ensure that a single credible failure of the crane system will not result in the loss of the capability of the system to safely retain a load. A spent fuel cask will follow the restricted critical "L" path shown in Figure 9.1-6.

[Historical Information]

The following discussion reflects the results of the original 100-Ton cask drop evaluation. This evaluation was completed prior to the reactor building overhead crane being upgraded to single-failure-proof.

The Spent Fuel Cask Drop Accident is not analyzed for reload cores.

For analysis of spent fuel cask drops, the 100-ton spent fuel cask as described in Subsection 9.1.4.2 is used.

The cask drop analysis (32 irradiated fuel bundles) makes the assumption that the fuel is normally cooled for a minimum of 360 days prior to cask loading and has assumed maximum values for irradiation time for a given fuel type. For shipments involving a small quantity of irradiated fuel rods or one or two fuel assemblies (generally for Research & Development or root cause investigation into performance issues), a shorter cooling period and/or longer period of irradiation than assumed for the large cask drop analysis are permissible provided the following are verified:

a. The fuel rods meet the shipping cask requirements for minimum cooling time, for maximum decay heat limitations, and maximum fuel rod irradiation time in addition to other cask specific requirements.

b. The source term inventory in the contained fuel rods (those isotopes which are pertinent to the offsite dose consequences of a cask drop as specified in Section 15.7.5) are bounded by the cask drop accident source term and, hence, the offsite dose consequences of the large scale cask drop analysis remains bounding.

Credible cask drop accidents have been broken down into two categories: those in which the cask is assumed to fall less than 30 feet, where there are no radiological consequences because the cask is able to withstand a free drop through a distance of 30 feet without rupturing; and those cask drops greater than 30 feet, where conservative assumptions are made to model postaccident conditions because there is no accurate way of predicting the cask conditions after such a drop.

15.7.5.1 Identification of Causes and Frequency Classification

Postulation of a cask drop accident resulting from a failure of the reactor building crane would require the failure of a component in the main hoist drive train, the upper or lower load block, the cable, or the crane hook. Due to the safety margins involved in the design of these components, the initial load tests on the crane hook and the crane and the inservice inspection programs, it is considered extremely unlikely that such an accident could happen. Therefore, this accident is categorized as an event with the frequency of a limiting fault.

The only other cask failure mechanism possible is an in-transit event. Although accidents with railroad and truck type vehicles are not uncommon, transportation restrictions and transport vehicle designs also justify the expected frequency of a limiting fault for this event. Recent input tests at Sandia National Laboratories indicate that integrity of shipping casks designed to 10 CFR 71 Appendix B criteria is not compromised during in-transit collisions or upsets.

For spent fuel cask drops of greater than 30 feet, only two drop cases are of any consequence. One results from a reactor building crane failure when the cask is over the cask storage well; this sets up a maximum drop of approximately 43 feet.

The other results when the cask is over the equipment hatch, where the maximum drop is approximately 133 feet. An analysis of the structural effects on the building from such a drop are included in Reference 5. The LaSalle Station design was specifically modified to accept the more severe fuel cask drop accident in the reactor building.

Spent fuel cask drops of less than 30 feet can be postulated for the refueling period when the cask is supported by the reactor building crane or for a cask transport vehicle accident. During refueling, permissive operations are constrained by the critical L-path control on the crane system as described in Subsection 9.1.4.1.2. The maximum cask drop height under this path control is about 10 inches. No drop will result in the spent fuel cask tipping over or structurally damaging the building while under critical path control. Although contingent upon both cask and vehicle design, cask drops from a transport vehicle are less than 30 feet and thus yield trivial results even though this event is categorized as a limiting fault.

This event is categorized as a limiting fault.

15.7.5.2 Sequence of Events and Systems Operation

The sequence of events and approximate elapsed time before the event occurs for each accident previously described are as follows:

Event Description - The spent fuel cask is dropped 43 feet into the cask storage well. The cask head is assumed to be latched to the cask.

Sequence of Events	Approximate Elapsed Time
1. Event begins - cask drops to bottom of cask well	0
2. Area is evacuated until radiological consequences are assessed	< 5 minutes
3. Spent fuel pool exhaust radiation monitor alarm and initiation of standby gas treatment system	< 5 minutes
4. Assessment of radiological consequences, structural damage to building, and cask cooling requirements	1 hour
5. Cleanup and repair operations begin	Dictated by damage assessment

Event Description - The spent fuel cask is dropped 133 feet down through the equipment hatches. A detailed analysis of this drop is presented in Reference 5.

Sequence of Events	Approximate Elapsed Time
1. Event begins - cask drops to elevation 673 of the reactor building	0

2.	Area is evacuated until radiological consequences are assessed	< 5 minutes
3.	Reactor building exhaust radiation monitor alarm and initiation of standby gas treatment system.	< 5 minutes
4.	Assessment of radiological consequences, structural damage to building and cask, and cask cooling requirements	1 hour
5.	Cleanup and repair operations begin	dictated by damage assessment

Event Description - The spent fuel cask is dropped either 10 inches into the decontamination pit or 6 inches onto the refueling floor.

	Sequence of Events	Approximate Elapsed Time
1.	Event begins - cask drops onto refueling floor	0
2.	Cask damage and crane malfunction are assessed	5 minutes
3.	Auxiliary cooling is provided to the cask if required	< 1 hour
4.	Cask is unloaded and inspected	24 hours

Event Description - The spent fuel cask is dropped from its transport vehicle.

	Sequence of Events	Approximate Elapsed Time
1.	Event begins - cask falls from transport	0

2.	Transport vehicle operator stops vehicle	≤ 1 minute
3.	Cask is loaded back on transport vehicle for continuation of journey	24 hours

15.7.5.3 Core and Systems Performance

This event does not have a direct effect on the core and does not affect the continued normal operation of the reactor system.

Damage to facility safety-related equipment, which is possible only in the case of the 133-foot drop described previously, has been outlined in Reference 5.

The reactor building ventilation and the standby gas treatment systems are expected to function normally. This allows automatic isolation of the reactor building and gives assurance that any release of radioactive gases associated with a cask drop will be filtered and treated through the standby gas treatment system.

15.7.5.4 Barrier Performance

This event occurs outside the primary containment, therefore this section is not directly applicable.

The only fission products expected to be released from the cask would be those contained in the external shield coolant annulus in the event at loss-of-cask coolant. For the purpose of analysis, the assumption is made that 1000 Ci of noble gas is released from the cask.

The cask handling area is served by the SGTS. All releases within this structure are capable of being filter-decayed and released through the elevated station vent stack.

The integrity of the secondary containment boundary (reactor building) is maintained for the spent fuel cask drop accident. Therefore only an elevated release of radioactive gases is postulated.

15.7.5.5 Radiological Consequences

A fully loaded spent fuel cask is arbitrarily assumed to drop 133 feet while being lowered through the facility hatches to a waiting flatcar inside the reactor building. The cask impacts a yielding surface thus resulting in compromised cask integrity with a postulated release of a fraction of the radioactivity content of all rods to the secondary containment atmosphere. This gap activity is assumed to be immediately

exhausted out the station vent stack. Credit taken for charcoal adsorption of radioactivity released through the SGTS is assumed to be 95% for radioiodine.

The input parameters and initial conditions are as follows:

a. The cask is loaded to a maximum of 32 fuel assemblies; this is equivalent to 4.2% of the total core inventory. An operating period of 1300 days was assumed for design-burnup condition (38.3 MWD/MTU core average exposure) for the GE analysis, 1953 days was assumed for the FANP ATRIUM-9 analysis, and 2078 days was assumed for the FANP ATRIUM-10 analysis.

b. The cask drop event occurs at 360 days after shutdown, which is the earliest normal time interval before spent fuel elements are loaded into the cask.

c. It is conservatively assumed that all of the gap activity from the 32 fuel assemblies is released to the atmosphere of the secondary containment.

d. The released gap activity consists of 10% of the noble gases other than Kr-85, 30% of the Kr-85, and 10% (12% was used for ATRIUM-10) of the total radioiodine in the damaged fuel rods at the time of the event.

Table 15.7-19 shows the fission product inventory in the core, in the cask for 32 fuel assemblies, and in the fuel rod gaps which is assumed to escape the fuel casks. Peak core activity was assumed with a radial peaking factor of 1.5 applied for the GE and FANP ATRIUM-9 fuel assemblies in the cask when it was dropped. A radial peaking factor of 1.7 is applied to ATRIUM-10 fuel assemblies in the cask when it was dropped.

The doses shown in Table 15.7-20 for the entire cloud passage are well below the limits applicable to an individual receptor at the exclusion area boundary following this postulated event. Doses at the LPZ boundary are also well below the appropriate limits. The negligible thyroid dose estimates result from the extremely small release of radioiodines from the fuel that has decayed for approximately 1 year.

The fission product inventory increases in proportion to the power level. Therefore, a 105% core licensed power uprate to 3489 MWt increases the dose consequences from a refueling cask drop accident by 5% (Reference 12). As seen in Table 15.7-17, this is still a small fraction of the 10 CFR 100 requirements.

As outlined in Reference 9, the reload analyses performed for L1C09 and L2C08 bound the extended burnup to support 24-month fuel cycles. These analyses included, in part, the fuel handling accident, cask drop accident, and the control rod drop accident. The dose consequences for the fuel handling and control rod drop accidents remain bounded by the corresponding Updated Final Safety Analysis Report (UFSAR) Chapter 15 analyses previously performed for General Electric Company (GE) supplied fuel. The dose consequences for the cask drop accident increased by an insignificant amount.

Below is the listing of the parameters, from References 9, 13 and 14, that are the basis for the high exposure ATRIUM-9B Source Term used in the L1C09 and L2C08 analyses for ATRIUM-9B fuel. The units given below are Megawatt-days (MWd) per Metric Ton of Uranium (MTU) and Megawatts Thermal (MWt).

<u>Spent Fuel Cask Drop Accident: 50,000 MWd/MTU; 3323 MWt core power; 1953 core residence days</u>

Below is the listing of the parameters, from Reference 19, that is the basis for the high exposure ATRIUM-10 Source Term used in the analyses for ATRIUM-10 fuel.

Reference 22 confirms that the Reference 19 evaluation remains applicable for ATRIUM-10 fuel at MUR conditions and that the Reference 21 evaluation remains applicable for ATRIUM 10XM LTA at MUR conditions.

<u>Spent Fuel Cask Drop Accident: 60,000 MWd/Mtu; 3910 MWt Core power; 2078 Core residence days.</u>

[End of Historical Information]

15.7.6 References

1. D. Nguyen, "Realistic Accident Analysis - the RELAC Code and User's Guide," NEDO-21142, October 1977.

2. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, (Unit 1: Rev. 22, Unit 2: Rev. 20).

3. LSCS Design Analysis L-003067, Rev. 2, "Re-analysis of Fuel Handling Accident (FHA) Using Alternative Source Terms".

4. Deleted.

5. "Supplemental Crane Design Features and Fuel Cask Drop Analysis," Special Report 2, LaSalle County Station PSAR, April 1974.

6. FANP document, “LaSalle Fuel Handling Accident for ATRIUM-9B Fuel,” EMF-96-171(P), Revision 2, Framatome – ANP, March 2001.

7. SPC letter, JHR : 96 : 399, “Re-Transmittal of Response to comment 25 LaSalle Cask Drop Accident with Table,” J.H. Riddle to R.J. Chin, October 10, 1996.

8. Letter from R. M. Krich, Commonwealth Edison (ComEd) Company, to U.S. NRC, “Request for License Amendment for Power Uprate Operation,” dated July 14, 1999, with Attachment E: General Electric Nuclear Energy, Licensing Topical Report NEDC-32701P, Revision 2, "Power Uprate Safety Analysis Report for LaSalle County Station, Units 1 and 2," dated July 1999 (Proprietary).

9. Letter from C. G. Pardee, Commonwealth Edison (ComEd) Company, to U.S. NRC, "Supplement to the License Amendment Request for Power Uprate Operation," dated 04/14/2000.

10. Letter from D. M. Skay, U.S. NRC, to O. D. Kingsley, Commonwealth Edison (ComEd) Company, "LaSalle - Issuance Of Amendments Regarding Power Uprate (TAC NOS. MA6070 AND MA6071)," dated May 9, 2000 (OL Amendments 140/125).

11. "Safety Evaluation By The Office Of Nuclear Reactor Regulation Related To Amendment No. 140 To Facility Operating License No. NPF-11 and Amendment No. 125 To Facility Operating License No. NPF-18; Commonwealth Edison Company Lasalle County Station, Units 1 And 2; Docket Nos. 50-373 And 50-374" dated May 9, 2000.

12. ENDIT H041, S&L Task 21b, "Radiological - Chapter 15 Accidents."

13. Memorandum NFM:BSA 00-028 from R. W. Tsai, Nuclear Fuel Management, to F. A. Spangenberg III, Regulatory Assurance, "Summary of Key Input Parameters for the High Exposure Siemens ATRIUM-9B Source Term," dated April 13, 2000 DG00-000413.

14. Letter JH:96:188 from J. H. Riddle (Siemens) to Dr. R. J. Chin (ComEd), "Radioactive Release Analysis Source Term Values," dated May 20, 1996 DG96-001013.

15. F-ANP Document EMF-2679(P), Revision 0, "LaSalle Fuel Handling Accident for ATRIUM-10 Fuel," dated January 2002.

16. FANP document, "Dropped Cask Accident Analysis Results for LaSalle with ATRIUM-10 Fuel", AWW:02:021, dated December 9, 2002.

17. GNF Report, "GE14 Fuel Design Cycle" Independent Analysis for LaSalle Unit 1 and Unit 2", GE-NE-0000-0026-4769-00, Rev.0 dated January 2005.

18. Letter, D.E. Garber (FANP) to F.W. Trikur (Exelon), "Radioactive Source Term Values for ATRIUM-10 Fuel at LaSalle, " DEG:01:170 dated October 22, 2001.

19. Letter, A.W. Will (FANP) to F.W. Trikur (Exelon), "Dropped Cask Accident Analysis Results for LaSalle with ATRIUM-10 Fuel," AWW:02:021 dated December 9, 2002.

20. GNF Report, "GE14 Fuel Design Cycle – Independent Analyses For LaSalle Unit 1 and Unit 2", GE-NE-0000-0026-4769-00, Rev. 0, dated January 2005.

21. AREVA document 51-9093566-000, "Radioactive Source Term Values, Fuel Handling Accident and Dropped Cask Accident Analyses at LaSalle with ATRIUM 10XM Fuel," transmitted to Exelon under letter FAB08-2573, October 30, 2008.

22. Design Analysis L-003509, Revision 0, "Evaluation of Appendix R, Station Blackout, Containment and Source Terms for LaSalle MUR Power Uprate," July 2010.

23. Design Analysis L-003696, Revision 1, "NEDC-33647P, GNF2 Fuel Design Cycle-Independent Analyses for Exelon LaSalle County Station Units 1 and 2" February 2012.

TABLE 15.7-1

GASEOUS RADWASTE SYSTEM FAILURE (OFF-GAS FAILURE)

(PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSES)

	DESIGN-BASIS (NRC) ASSUMPTIONS	REALISTIC (CONSERVATIVE ENGINEERING) ASSUMPTIONS
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power level (102% of Core Thermal Power)**	NA	3559 MWt
B. Burnup	NA	NA
C. Fuel damaged	None	None
D. Inventory of activity by nuclide	Table 15.7.1-3	Table 15.7.1-3
E. Iodine fractions		
(1) Organic	0	0
(2) Elemental	1.0	1.0
(3) Particulate	0	0
F. Reactor coolant activity before the accident	NA	NA
II. Data and assumptions used to estimate activity released		
A. Containment leak rate (%/day)	NA	NA
B. Secondary containment leak rate (%/day)	NA	NA
C. Valve movement times	NA	NA
D. Absorption and filtration efficiencies	NA	NA
(1) Organic iodine	NA	NA
(2) Elemental iodine	NA	NA
(3) Particulate iodine	NA	NA
(4) Particulate fission products	NA	NA
E. Recirculation system parameters	NA	NA
(1) Flow rate	NA	NA
(2) Mixing efficiency	NA	NA
(3) Filter efficiency	NA	NA
F. Containment spray parameters (flow rate, drop size, etc.)	NA	NA
G. Containment volumes	NA	NA
H. All other pertinent data and assumptions	None	None
III. Dispersion Data		
A. Boundary and LPZ distances (meters)	509/6400	509/6400
B. χ/Q's for EAB*/LPZ	5.7(-4) / 5.5(-5)	1.7(-7) / 5.8(-8)
IV. Dose Data		
A. Method of dose calculation		
B. Dose conversion assumptions	Reference 1	Reference 1
C. Peak activity concentrations in containment	NA	NA
D. Doses	Table 15.7.1-4	Table 15.7.1-4

* Maximum χ/Q occurs at 6400 meters from release point (realistic boundary). The χ/Q at the EAB is 6.1 (-25) sec/m^3

** Reference 12

TABLE 15.7-2

EQUIPMENT FAILURE RELEASE ASSUMPTIONS

(RELEASE FRACTIONS ASSUMED FOR DESIGN BASIS/REALISTIC ANALYSIS)

EQUIPMENT PIECE	NOBLE GASES	SOLID DAUGHTERS	RADIOIODINE
Preheater	1.00/1.00	1.00/1.00	N/A
Catalytic Recombiner	1.00/1.00	1.00/1.00	N/A
Off-gas Condenser	1.00/1.00	1.00/1.00	N/A
Water Separator	1.00/1.00	1.00/1.00	N/A
Holdup Pipe	1.00/0.20	1.00/0.20	N/A
Cooler Condenser	1.00/1.00	1.00/1.00	N/A
Moisture Separator	1.00/1.00	1.00/1.00	N/A
Reheater	1.00/1.00	1.00/1.00	N/A
Prefilter	1.00/1.00	0.01/0.01	N/A
Charcoal Adsorbers	0.10/0.10	0.01/0.0	0.01/0.01
Afterfilter	1.00/1.00	0.01/0.01	N/A

TABLE 15.7-3
(SHEET 1 OF 3)

INVENTORY ACTIVITIES FOR OFF-GAS RECHAR EQUIPMENT

(μCi)

TIMES*	PREHEATER	RECOMBINER	OFF-GAS CONDENSER	WATER SEPARATOR	HOLDUP PIPE	COOLER CONDENSER	MOISTURE SEPARATOR	REHEATER	PREFILTER	CHARCOAL VESSELS (TRAIN)	CHARCOAL VESSEL (FIRST)	AFTERFILTER
GAS RES.	8.00-1S	9.40-1S	5.00+1S	5.10 S	3.00+1M	2.97 M	6.50 S	1.45+1S	4.35+1S	2.75 H	2.06+1M	4.35+1S
KR RES.	0.	0.	0.	0.	0.	0.	0.	0.	0.	1.29+1H	1.62 H	0.
Xe	0.	0.	0.	0.	0.	0.	0.	0.	0.	9.72 D	1.22 D	0.
OPER. TIME	0.	0.	0.	0.	0.	0.	0.	0.	1.00 Y	1.00+1Y	1.00+1Y	1.00 Y
S.D. CAPTURE	0	0	100	100	60	0	0	0	100	100	100	100
S.D. WASHOUT			100	100	100				0	0	0	0
ISOTOPE												
N - 13	1.19+4	1.40+4	7.22+5	7.14+4	1.05+7	2.78+5	9.11+3	2.01+4	5.82+4	1.12+6	8.57+5	5.70-1
N - 17	0.69+3	8.83+3	5.19+4	7.04	5.24	0.	0.	0.	0.	0.	0.	0.
O - 19	1.39+6	1.60+6	4.72+7	2.20+6	1.56+7	0.	0.	0.	0.	0.	0.	0.
KR - 83M	2.77+3	3.26+3	1.73+5	1.76+4	5.66+6	5.05+5	1.83+4	4.07+4	1.22+5	2.69+7	1.22+7	1.01+3
KR - 85	1.90+1	2.24+1	1.19+3	1.21+2	4.28+4	4.24+3	1.55+2	3.45+2	1.04+3	1.12+6	1.39+5	1.05+3
KR - 85M	4.92+3	5.78+3	3.07+5	3.13+4	1.06+7	1.01+6	3.66+4	8.15+4	2.44+5	1.11+8	2.88+7	3.18+4
KR- 87	1.57+4	1.85+4	9.79+5	9.94+4	3.07+7	2.60+6	9.37+4	2.09+5	6.24+5	9.39+7	5.52+7	5.25+2
RB - 87	0.	0.	0.	0.	0.	0.	0.	0.	2.19-4	1.30-2	7.63-3	0.
KR - 88	1.61+4	1.90+4	1.01+6	1.02+5	3.40+7	3.14+6	1.14+5	2.54+5	7.60+5	2.43+8	8.36+7	3.06+4
RB - 88	4.21	1.57+1	1.74+4	1.70+2	8.55+6	1.04+6	4.26+4	9.61+4	1.10+7	2.43+8	8.36+7	3.06+4
KR - 89	9.67+4	1.13+5	5.50+6	5.07+5	2.71+7	1.88+4	4.82+2	1.04+3	2.80+3	1.64+4	1.64+4	0.
RB - 89	2.94+1	1.10+2	1.14+5	9.85+2	1.10+7	4.41+5	1.51+4	3.33+4	3.01+6	1.64+4	1.64+4	0.
SR - 89	1.24-6	1.15-5	3.17-1	2.65-4	1.63+3	2.13+2	8.02	1.80+1	1.08+7	1.64+4	1.64+4	0.
Y - 89M	0.	0.	1.23-1	1.43-5	1.59+3	2.12+2	7.97	1.78+1	1.08+7	1.64+4	1.64+4	0.
KR - 90	1.67+5	1.93+5	6.23+6	3.35+5	2.89+6	0.	0,	0.	0.	0.	0.	0.
RB - 90	2.87+2	1.06+3	7.96+5	3.70+3	1.73+6	3.47+2	8.35	1.78+1	2.78+2	0.	0.	0.
SR - 90	0.	0.	1.16-2	4.83-6	2.01	1.56-1	5.71-3	1.27-2	2.74+4	0.	0.	0.
Y - 90	0.	0.	0.	0.	4.69-3	7.56-4	2.92-5	6.55-5	2.71+4	0.	0.	0.
KR - 91	8.62+4	9.45+4	1.18+6	7.19+3	1.41+4	0.	0.	0.	0.	0.	0.	0.
RB - 91	4.17+2	1.50+3	5.11+5	2.30+2	8.49+3	2.44-6	0.	0.	0.	0.	0.	0.
SR - 91	2.23-3	2.02-2	2.36+2	8.09-3	2.83+2	1.93+1	7.05-1	1.57	5.45+3	0.	0.	0.
Y - 91	0.	0.	1.31-6	0.	3.94-3	8.70-4	3.49-5	7.86-5	5.58+3	0.	0.	0.
Y - 91M	0.	2.15-6	8.30-1	2.42-6	4.90+1	6.62	2.51-1	5.63-1	5.62+3	0.	0.	0.
KR - 92	1.41+3	1.20+3	2.81+3	1.56-5	2.68-6	0.	0.	0.	0.	0.	0.	0.
RB - 92	8.82+1	2.67+2	5.07+3	6.08-6	1.61-6	0.	0.	0.	0.	0.	0.	0.
SR - 92	1.74-3	1.42-2	1.65+1	0.	0.	0.	0.	0.	0.	0.	0.	0.
Y - 92	0.	0.	1.97-2	0.	0.	0.	0.	0.	1.06-6	0.	0.	0.
KR - 93	4.54+1	3.35+1	5.10+1	0.	0.	0.	0.	0.	0.	0.	0.	0.
RB - 93	2.22	6.47	1.21+2	0.	0.	0.	0.	0.	0.	0.	0.	0.
SR - 93	9.51-4	7.53-3	8.01	0.	0.	0.	0.	0.	0.	0.	0.	0.
Y - 93	0.	0.	3.30-3	0.	0.	0.	0.	0.	0.	0.	0.	0.
ZR - 93	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
NB - 93M	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	
KR - 94	1.20	7.77-1	8.46-1	0.	0.	0.	0.	0.	0.	0.	0.	0.
RB - 94	1.27-1	3.30-1	2.37	0.	0.	0.	0.	0.	0.	0.	0.	0.
SR- 94	3.22-4	2.37-3	9.57-1	0.	0.	0.	0.	0.	0.	0.	0.	0.
Y - 94	0.	0.	1.35-2	0.	0.	0.	0.	0.	0.	0.	0.	0.
KR - 95	7.16-6	??	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
RB - 95	3.99-6	??	??	0.	0.	0.	0.	0.	0.	0.	0.	0.
R - 95	0.	0.	7.71-6	0.	0.	0.	0.	0.	0.	0.	0.	0.

* S = second, M = minute, H = hour, D = day, Y = year

TABLE 15.7-3
(SHEET 2 OF 3)

TIMES*	PREHEATER	RECOMBINER	OFF-GAS CONDENSER	WATER SEPARATOR	HOLDUP PIPE	COOLER CONDENSER	MOISTURE SEPARATOR	REHEATER	PREFILTER	CHARCOAL VESSELS (TRAIN)	CHARCOAL VESSEL (FIRST)	AFTERFILTER
Y - 95	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
ZR- 95	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
NB - 95	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
KR - 97	7.22-4	4.66-4	5.08-4	0.	0.	0.	0.	0.	0.	0.	0.	0.
RB - 97	5.68-4	5.37-4	5.90-4	0.	0.	0.	0.	0.	0.	0.	0.	0.
SR-97	2.40-4	5.57-4	8.99-4	0.	0.	0.	0.	0.	0.	0.	0.	0.
Y - 97	3.44-5	2.29-4	1.43-3	0.	0.	0.	0.	0.	0.	0.	0.	0.
ZR - 97	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
NB - 97	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
NB - 97M	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
XE - 131M	1.21+1	1.42+1	7.56+2	7.72+1	2.72+4	2.69+3	9.82+1	2.19+2	6.57+2	9.70+6	1.53+6	3.74+2
XE - 133	6.61+3	7.76+3	4.13+5	4.21+4	1.48+7	1.47+6	5.35+4	1.19+5	3.58+5	3.95+9	8.01+8	1.02+5
XE - 133M	2.22+2	2.61+2	1.39+4	1.41+3	4.97+5	4.90+4	1.79+3	3.99+3	1.20+4	7.36+7	2.41+7	6.07+2
XE - 135	1.78+4	2.09+4	1.11+6	1.13+5	3.98+7	3.90+6	1.42+5	3.17+5	9.50+5	1.05+9	9.32+8	9.44-6
XE - 135M	2.09+4	2.45+4	1.28+6	1.28+5	2.49+7	1.11+6	3.78+4	8.37+4	2.46+5	7.56+6	7.56+6	2.06-2
CS - 135	0.	0.	0.	0.	2.06-4	3.04-5	1.23-6	2.79-6	6.39	3.16+3	2.80+3	0.
XE - 137	1.18+5	1.39+5	6.85+6	6.42+5	4.12+7	7.49+4	2.05+3	4.42+3	1.22+4	8.66+4	8.66+4	0.
CS - 137	3.45-5	1.28-4	1.37-1	1.19-3	2.65+1	2.14	7.82-2	1.74-1	3.75+5	1.78+4	1.77+4	0.
BA - 137M	0.	0.	1.02-2	9.15-6	2.26+1	2.12	7.77-2	1.73-1	3.75+5	1.78+4	1.77+4	0.
XE - 138	7.11+4	8.35+4	4.35+6	4.34+5	8.02+7	3.25+6	1.10+5	2.44+5	7.14+5	1.98+7	1.98+7	0.
CS - 138	1.02+1	3.81+1	4.18+4	3.97+2	1.53+7	1.46+6	5.53+4	1.24+5	2.46+7	1.98+7	1.98+7	0.
XE - 139	1.76+5	2.03+5	7.18+6	4.41+5	4.78+6	0.	0.	0.	0.	0.	0.	0.
CS - 139	8.73+1	3.23+2	2.72+5	1.41+3	2.54+6	4.38+4	1.42+3	3.13+3	1.73+5	0.	0.	0.
BA - 139	3.24-3	3.00-2	7.01+2	3.36-1	3.75+5	3.58+4	1.31+3	2.92+3	1.62+6	0.	0.	0.
XE - 140	1.21+5	1.36+5	2.64+6	5.50+4	1.94+5	0.	0.	0.	0.	0.	0.	0.
CS - 140	5.28+2	1.92+3	9.11+5	1.56+3	1.16+5	2.57-4	2.95-6	5.88-6	3.43-5	0.	0.	0.
BA - 140	8.87-5	8.11-4	1.23+1	1.71-3	1.23+2	8.64	3.16-1	7.04-1	7.74+4	0.	0.	0.
LA - 140	0.	0.	8.69-4	0.	4.98-1	7.33-2	2.81-3	6.31-3	7.75+4	0.	0.	0.
XE - 141	7.19+2	5.96+2	1.29+3	2.00-6	0.	0.	0.	0.	0.	0.	0.	0.
CS - 141	8.45	2.68+1	1.92+3	0.	0.	0.	0.	0.	0.	0.	0.	0.
BA - 141	1.46-3	1.22-2	3.68+1	0.	0.	0.	0.	0.	0.	0.	0.	0.
LA - 141	0.	0.	3.34-2	0.	0.	0.	0.	0.	0.	0.	0.	0.
CE - 141	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
XE - 142	2.20+1	1.58+1	2.24+1	0.	0.	0.	0.	0.	0.	0.	0.	0.
CS - 142	3.47	8.82	4.80+1	0.	0.	0.	0.	0.	0.	0.	0.	0.
BA - 142	1.06-1	7.77-3	3.01	0.	0.	0.	0.	0.	0.	0.	0.	0.
LA - 142	0.	0.	9.05-3	0.	0.	0.	0.	0.	0.	0.	0.	0.
XE - 143	4.17-1	2.63-1	2.71-1	0.	0.	0.	0.	0.	0.	0.	0.	0.
CS - 143	6.68-2	1.61-1	7.23-1	0.	0.	0.	0.	0.	0.	0.	0.	0.
BA - 143	1.10-3	7.60-3	8.81-1	0.	0.	0.	0.	0.	0.	0.	0.	0.
LA - 143	0.	3.25-6	2.45-2	0.	0.	0.	0.	0.	0.	0.	0.	0.
CE - 143	0.	0.	2.73-6	0.	0.	0.	0.	0.	0.	0.	0.	0.
PR - 143	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
XE - 144	1.55+2	1.70+2	2.22+3	1.57+1	3.26+1	0.	0.	0.	0.	0.	0.	0.
CS - 144	3.35+1	9.64+1	2.41+3	1.14+1	1.95+1	0.	0.	0.	0.	0.	0.	0.

TABLE 15.7-3
(SHEET 3 OF 3)

TIMES* ISOTOPE	PREHEATER	RECOMBINER	OFF-GAS CONDENSER	WATER SEPARATOR	HOLDUP PIPE	COOLER CONDENSER	MOISTURE SEPARATOR	REHEATER	PREFILTER	CHARCOAL VESSELS (TRAIN)	CHARCOAL VESSEL (FIRST)	AFTERFILTER
BA - 144	5.38-1	4.15	2.18+3	1.31	1.95+1	0.	0.	0.	0.	0.	0.	0.
LA - 144	1.87-3	3.46-2	7.74+2	3.24-2	1.95+1	0.	0.	0.	0.	0.	0.	0.
CE - 144	0.	0.	3.86-4	0.	9.42-4	6.54-5	2.39-6	5.33-6	7.67	0.	0.	0.
PR- 144	0.	0.	3.20-6	0.	3.80-4	4.57-5	1.71-6	3.83-6	7.67	0.	0.	0.
ND - 144	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
I - 131	0.	0.	0.	0.	0.	0.	0.	0.	0.	3.30+5	3.30+5	0.
I - 132	0.	0.	0.	0.	0.	0.	0.	0.	0.	3.50+4	3.50+4	0.
I - 133	0.	0.	0.	0.	0.	0.	0.	0.	0.	2.40+5	2.40+5	0.
I - 134	0.	0.	0.	0.	0.	0.	0.	0.	0.	2.60+4	2.60+4	0.
I - 135	0.	0.	0.	0.	0.	0.	0.	0.	0.	1.10+5	1.10+5	0.
TOTAL	2.34+6	2.69+6	8.98+7	5.24+6	3.83+8	2.04+7	7.35+5	1.64+6	6.70+7	5.85+9	2.07+9	1.98+5
GAS (N + O)	1.41+6	1.62+6	4.79+7	2.27+6	2.61+7	2.78+5	9.11+3	2.01+4	5.82+4	1.12+6	8.57+5	5.70-1
GAS (XR + XE)	9.24+5	1.06+6	3.92+7	2.96+6	3.18+8	1.71+7	6.11+5	1.36+6	4.05+6	5.58+9	1.97+9	1.68+5
B. D.	1.50+3	5.38+3	2.86+6	8.47+3	3.97+7	3.02+6	1.16+5	2.59+5	6.29+7	2.63+8	1.04+8	3.06+4
KR GAS	3.91+5	4.49+5	1.54+7	1.10+6	1.11+8	7.28+6	2.63+5	5.86+5	1.75+6	4.76+8	1.80+8	6.49+4
KE GAS	5.32+5	6.16+5	2.38+7	1.86+6	2.07+8	9.85+6	3.47+5	7.72+5	2.29+6	5.11+9	1.79+9	1.03+5

TABLE 15.7-4

GASEOUS RADWASTE SYSTEM FAILURE

RADIOLOGICAL EFFECTS

		10 CFR 100		DESIGN BASIS		REALISTIC	
		(2 hours)	(30 days)	(2 hours)	(30 days)	(2 hours)	(30 days)
A. Exclusion Boundary (509 meters)	Whole Body Dose (rem)	25	N/A	3.3E-01	--	--	N/A
	a) Holdup Pipe	--	--	--	--	5.0E-06	--
	b) Charcoal Train	--	--	--	--	1.0E-06	--
	c) Prefilter	--	--	--	--	--	--
	Inhalation Dose (rem)	300	N/A	4.6E-08	--	--	--
	a) Holdup Pipe (Bone)	--	--	--	--	2.2E-08	N/A
	b) Charcoal Train (Thyroid)	--	--	--	--	9.8E-07	--
	c) Prefilter (Lung)	--	--	--	--	2.8E-06	--
B. Point of Interest (915 meters)	Whole Body Dose (rem)	N/A	N/A	--	--	--	--
	Inhalation Dose (rem)	N/A	N/A	--	--	--	--
C. Low Population Zone (6400 meters)	Whole Body Dose (rem)	N/A	25	--	3.2E-02	--	N/A
	a) Holdup Pipe	--	--	--	--	1.2E-05	--
	b) Charcoal Train	--	--	--	--	2.5E-05	--
	c) Prefilter	--	--	--	--	--	--
	Inhalation Dose (rem)	N/A	300	--	4.4E-09	--	--
	a) Holdup Pipe (Bone)	--	--	--	--	6.6E-06	N/A
	b) Charcoal Train (Thyroid)	--	--	--	--	2.4E-06	--
	c) Prefilter (Lung)	--	--	--	--	5.3E-08	--

TABLE 15.7-5

FAILURE OF AIR EJECTOR LINES, PARAMETERS

TO BE TABULATED FOR POSTULATED ACCIDENT ANALYSES

	DESIGN-BASIS (NRC) ASSUMPTIONS	REALISTIC (CONSERVATIVE) ENGINEERING ASSUMPTIONS
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power level (102% of Core Thermal Power)**	NA	3559 MWt
B. Burnup	NA	NA
C. Fuel damaged	NA	None
D. Release of activity by nuclide	NA	Table 15.7.1-6
E. Iodine fractions	NA	
(1) Organic	NA	0
(2) Elemental	NA	1
(3) Particulate	NA	0
F. Reactor coolant activity before the accident	NA	NA
II. Data and assumptions used to estimate activity released		
A. Containment leak rate (%/day)	NA	NA
B. Secondary containment leak rate (%/day)	NA	NA
C. Valve movement times	NA	NA
D. Adsorption and filtration efficiencies	NA	NA
(1) Organic iodine	NA	NA
(2) Elemental iodine	NA	NA
(3) Particulate iodine	NA	NA
(4) Particulate fission products	NA	NA
E. Recirculation system parameters	NA	NA
(1) Flow rate	NA	NA
(2) Mixing efficiency	NA	NA
(3) Filter efficiency	NA	NA
F. Containment spray parameters (flow rate, drop size, etc.)	NA	NA
G. Containment volumes	NA	NA
H. All other pertinent data and assumptions	NA	None
III. Dispersion Data		
A. Boundary and LPZ distances (meters)	NA	509/6400
B. χ/Q's for EAB*/LPZ	NA	1.7(-7)/5.8(-8)
IV. Dose Data		
A. Method of dose calculation	NA	Reference 1
B. Dose conversion assumptions	NA	Reference 1
C. Peak activity concentrations in containment	NA	NA
D. Doses	NA	Table 15.7.1-7

* Maximum χ/Q occurs at 6400 meters from release point (realistic boundary). The χ/Q at the EAB is 6.1 (-25) sec/m^3.

** Reference 12

TABLE 15.7-6

FAILURE OF AIR EJECTOR LINES

ACTIVITY RELEASED TO THE ENVIRONMENT (curies)

(Realistic Analysis)

ISOTOPE	ACTIVITY
I-131	6.4E-03
I-132	5.9E-02
I-133	4.3E-02
I-134	1.2E-01
I-135	6.4E-02
TOTAL	2.9E-01
KR-83M	7.6E 00
KR-85M	1.3E 01
KR-85	5.2E-02
KR-87	4.1E 01
KR-88	4.2E 01
KR-89	1.8E 02
XE-131M	4.2E-02
XE-133M	6.3E-01
XE-133	1.8E 01
XE-135M	5.2E 01
XE-135	4.8E 01
XE-137	2.3E 02
XE-138	1.8E 02
TOTAL	8.1E 02

TABLE 15.7-7

FAILURE OF AIR EJECTOR LINES
RADIOLOGICAL EFFECTS

		10 CFR 100		REALISTIC	
		(2 hours)	(30 days)	(2 hours)	(30 days)
A. Exclusion Boundary (509 meters)	Whole Body Dose (rem)	25	N/A	3.0E-05	--
	Thyroid Dose (rem)	300	N/A	2.4E-06	--
B. Point of Interest (915 meters)	Whole Body Dose (rem)	N/A	N/A	--	--
	Thyroid Dose (rem)	N/A	N/A	--	--
C. Low Population Zone (6400 meters)	Whole Body Dose (rem)	N/A	25	N/A	1.0E-05
	Thyroid Dose (rem)	N/A	300	N/A	8.1E-07

TABLE 15.7-8

LIQUID RADWASTE TANK FAILURE - PARAMETERS TO BE TABULATED FOR POSTULATED ACCIDENT ANALYSES

	NRC ASSUMPTIONS	REALISTIC (CONSERVATIVE ENGINEERING) ASSUMPTIONS
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power Level	NA	NA
B. Burnup	NA	NA
C. Fuel damaged	NA	NA
D. Release of activity by nuclide	NA	Table 15.7.3-3
E. Iodine fractions	NA	
(1) Organic	NA	0
(2) Elemental	NA	1
(3) Particulate	NA	0
F. Reactor coolant activity before the accident	NA	NA
II. Data and assumptions used to estimate activity released		
A. Containment leak rate (%/day)	NA	NA
B. Secondary containment leak rate (%/day)	NA	NA
C. Valve movement times	NA	NA
D. Adsorption and filtration efficiencies	NA	NA
(1) Organic Iodine	NA	NA
(2) Elemental Iodine	NA	NA
(3) Particulate Iodine	NA	NA
(4) Particulate fission products	NA	NA
E. Recirculation system parameters	NA	NA
(1) Flow rate	NA	NA
(2) Mixing Efficiency	NA	NA
(3) Filter Efficiency	NA	NA
F. Containment spray parameters (flow rate, drop size, etc.)	NA	NA
G. Containment volumes	NA	NA
H. All other pertinent data and assumptions	NA	None
III. Dispersion Data		
A. Boundary and LPZ distances (meters)	NA	509/6400
B. χ/Q's for EAB*/LPZ	NA	1.7(-7)/5.8(-8)
IV. Dose Data		
A. Method of dose calculation	NA	Regulatory Guide 1.3
B. Dose conversion	NA	Regulatory Guide 1.3
C. Peak activity concentrations in containment	NA	NA
D. Doses	NA	Table 15.7.3-2

* Maximum χ/Q occurs at 6400 meters from release point (realistic boundary). The χ/Q at the EAB is 6.1 (-25) sec/m^3.

TABLE 15.7-9

LIQUID RADWASTE TANK FAILURE
RADIOLOGICAL EFFECTS

		10 CFR 100		REALISTIC	
		(2 hours)	(30 days)	(2 hours)	(30 days)
A. Exclusion Boundary (509 meters)	Whole Body Dose (rem)	25	N/A	1.3E-08	--
	Thyroid Dose (rem)	300	N/A	5.0E-05	--
B. Point of Interest (915 meters)	Whole Body Dose (rem)	N/A	N/A	--	--
	Thyroid Dose (rem)	N/A	N/A	--	--
C. Low Population Zone (6400 meters)	Whole Body Dose (rem)	N/A	25	N/A	4.3E-09
	Thyroid Dose (rem)	N/A	300	N/A	1.7E-05

TABLE 15.7-10

LIQUID RADWASTE TANK FAILURE

ACTIVITY RELEASE TO ENVIRONMENT (Curies)

{Realistic Analysis)

RADIONUCLIDE	LIQUID CONCENTRATES WASTE TANK INVENTORY (Curies)	2-HOUR ACTIVITY RELEASED TO ATMOSPHERE (Curies)	TOTAL ACTIVITY RELEASED TO ATMOSPHERE (Curies)
I-131	555.5	5.56×10^{-1}	5.56×10^{-1}
I-132	20.49	2.01×10^{-2}	2.01×10^{-2}
I-133	46.36	4.63×10^{-2}	4.63×10^{-2}
I-134	.1396	1.32×10^{-4}	1.32×10^{-4}
I-135	6.602	6.55×10^{-3}	6.55×10^{-3}

TABLE 15.7-11

CONCENTRATES WASTE CONCENTRATIONS VS. TIME*

INVENTORY	ISOTOPES	0 SECONDS	65 DAYS	600 YEARS
1. Shield design-basis	Sr-90	1.52+6	1.52+6	8.8-1
	Iodine	1.0 +8	3.7 +5	0.0
	Others	1.3 +6	4.9 +5	2.3-4
2. Maximum shutdown	Sr-90	2.75+5	2.74+5	1.6-1
	Iodine	4.7 +7	1.7 +5	0.0
	Others	3.8 +5	1.1 +5	4.4-3*
3. Maximum normal operation	Sr-90	3.78+5	3.77+5	2.2-1
	Iodine	1.8 +7	6.1 +4	0.0
	Others	3.5 +5	1.2 +2	6.0-3**

*The concentrations are given in units of maximum permissible concentrations (MPC's).
Dilution in groundwater and dispersion were not considered.

**Cesium-137

TABLE 15.7-12
(Sheets 1 and 2)

FUEL HANDLING ACCIDENT PARAMETERS
TO BE TABULATED FOR POSTULATED ACCIDENT ANALYSES

INTENTIONALLY DELETED

TABLE 15.7-13

. FUEL HANDLING ACCIDENT ACTIVITY AIRBORNE IN REACTOR BUILDING (curies)*

(Design (NRC) Basis)

INTENTIONALLY DELETED

TABLE 15.7-14

FUEL HANDLING ACCIDENT ACTIVITY RELEASED TO ENVIRONS (curies)*

(Design (NRC) Basis)

INTENTIONALLY DELETED

TABLE 15.7-15

FUEL HANDLING ACCIDENT

ACTIVITY AIRBORNE IN THE REACTOR BUILIDNG (CURIES)*

(Realistic Analysis)

INTENTIONALLY DELETED

TABLE 15.7-16

FUEL HANDLING ACCIDENT

ACTIVITY RELEASED TO THE ENVIRONMENT (CURIES)*

(Realistic Analysis)

INTENTIONALLY DELETED

TABLE 15.7-17

FUEL HANDLING ACCIDENT
RADIOLOGICAL EFFECTS*

INTENTIONALLY DELETED

TABLE 15.7-18

REFUELING CASK DROP ACCIDENT

TABULATED FOR POSTULATED ACCIDENT ANALYSES

	DESIGN-BASIS CONSERVATIVE (NRC) ASSUMPTIONS	CONSERVATIVE ENGINEERING ASSUMPTIONS
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power level	NA	3631 MWt (GE) 3489 MWt (FANP)**
B. Radial peaking factor	NA	1.5 (1.7 for Atrium 10)
C. Fuel damaged	NA	32 assemblies
D. Release of activity by nuclide	NA	Table 15.7.5-2
E. Iodine fractions	NA	
(1) Organic		0
(2) Elemental		1
(3) Particulate		0
F. Reactor coolant activity before the accident	NA	NA
II. Data and assumptions used to estimate activity released		
A. Primary containment leak rate (%/day)	NA	NA
B. Secondary containment leak rate (%/day)	NA	NA
C. Valve movement times	NA	NA
D. Adsorption and filtration efficiencies	NA	
(1) Organic iodine		95%
(2) Elemental iodine		95%
(3) Particulate iodine		95%
(4) Particulate fission products		NA
E. Recirculation system parameters	NA	NA
(1) Flow rate (cfm)		
(2) Mixing efficiency		
(3) Filter efficiency		
F. Containment spray parameters (flow rate, drop size, etc.)	NA	NA
G. Containment volumes	NA	NA
H. All other pertinent data and assumptions	None	None
III. Dispersion Data		
A. Boundary and EAB*/LPZ distances (m)	NA	509/6400
B. χ/Q's	NA	4.0(-7)/1.5(-7)
IV. Dose Data		
A. Method of dose calculation	NA	NA
B. Dose conversion assumptions	NA	NA
C. Peak activity concentrations in containment	NA	NA
D. Doses	NA	Table 15.7.5-3

* Maximum χ/Q occurs at 4500 meters from release point (realistic boundary). The χ/Q at the EAB is 5.0 (-13) sec/m^3.
** Reference 12

TABLE 15.7-19

FUEL FISSION PRODUCT INVENTORY*

360 DAYS AFTER SHUTDOWN

NOBLE GASES AND IODINES

ISOTOPE	FULL CORE INVENTORY (Curies)	CASK INVENTORY FOR 32 BUNDLES (Curies)	TOTAL GAP INVENTORY FOR RELEASE (Curies)
GE Analysis (Reference 17)			
Kr-85	1.3E+06	8.3E+04	2.5E+04
Xe-131m	1.9E-03	1.2E-04	1.2E-05
I-129	4.2E+00	2.7E-01	2.7E-02
I-131	2.1E-06	1.3E-07	1.3E-08
FANP ATRIUM-9 Analysis (Reference 7)			
Kr-85	2.78E+06	1.17E+05	3.51E+04
Xe-131m	2.48E-03	1.04E-04	1.04E-05
I-129	1.10E+01	4.61E-01	4.61E-02
I-131	5.18E-06	2.16E-07	2.16E-08
FANP ATRIUM-10 Analysis (Reference 16)			
Kr-85	3.68E+06	1.54E+05	4.63E+04
Xe-131m	3.39E-03	1.42E-04	1.42E-05
I-129	1.57E+01	6.59E-01	7.91E-02
I-131	7.02E-06	2.74E-07	3.53E-08

+ GE inventory reflects power uprate conditions of 3489 MWt (105% of 3323 MWt). Power uprate is estimated to increase FANP inventories by 5%.

TABLE 15.7-20

FUEL CASK DROP ACCIDENT RADIOLOGICAL EFFECTS*

		10 CFR 100		GE INVENTORY		FANP ATRIUM-9 INVENTORY	FANP ATRIUM-10 INVENTORY
		(2 hours)	(30 days)	(2 hours)	(30 days)	(2 hours)	
A. Exclusion Boundary (509 meters)	Whole Body Dose (rem)	6	N/A	5.5E-6	---	7.36E-6	6.13E-6
	Thyroid Dose (rem)	75	N/A	Negligible	---	1.27E-6	3.18E-6
B. Point of Interest (915 meters)	Whole Body Dose (rem)	N/A	N/A	--	--	--	--
	Thyroid Dose (rem)	N/A	N/A	--	--	--	--
C. Low Population Zone (6400 meters)	Whole Body Dose (rem)	N/A	6	N/A	2.1E-6	--	--
	Thyroid Dose (rem)	N/A	75	N/A	Negligible	--	--

* Power uprate to 3489 MWt (105% of 3323 MWt) is estimated (Reference 12) to increase the FANP source term by 5%. The GE inventory already reflects the power uprate values.

[1] Per section II, "Acceptance Criteria", Standard Review Plan 15.7.5, NUREG-0800

TABLE 15.7-21
(Sheet 1 of 2)

FHA PARAMETERS
AST ANALYSIS

Parameter	AST Value
Core Power Level	3559 MWt
Fuel assembly configuration and properties	10x10 in a 87.33 fuel pin bundle and 172 pins damaged
Radial Peaking Factor	1.7
Allowable fuel burnup and non-LOCA gap fractions	Table 3 of RG 1.183. Fuel burnup will not exceed 62 GWD/MTU, except for ATRIUM-10 partial length rods that were evaluated for operation above 62 GWD/MTU in Unit 1 Cycle 16 and were shown to be bounded by full length rods that comply with footnote limits. Linear heat generation rate (LHGR) for fuel >54 GWD/MTU will not exceed 6.3 KW/ft.
FHA radionuclide inventory	The 60 isotopes forming the standard RADTRAD library, with decay to 24 hours.
Underwater Decontamination Factor	Noble Gases: 1 Particulate (cesium and rubidium): infinity Iodine: 200 (conservative value for the limiting case of a drop over the reactor well)
Dose conversion factors	Federal Guidance Reports 11 and 12
Secondary containment automatic isolation and filtration	Credited
CR volume	117,400 ft3
Reactor Building normal ventilation	Artificially set at an air change rate of 0.1 per minute
CR release point basis	Main stack (elevated) through SGT system (after 15 minute drawdown period of ground level release).

TABLE 15.7-21
(Sheet 2 of 2)

FHA PARAMETERS
AST ANALYSIS

Parameter	AST Value
CR Release Point Dispersion Factors 0-0.25 hr period (drawdown) 0.25 -2 hr period 2-8 hr period	 Plant Vent Stack 6.83E-04 sec/m3 Ground 1.17E-05 sec/m3 Elevated 1.00E-36 sec/m3 Elevated
EAB release point basis	Main stack (elevated) through SGT system (after 15 minute drawdown period of ground level release and one-half hour of fumigation).Distance to EAB = 423 meters
EAB Release Point Dispersion Factors 0-0.25 hr period (drawdown) 0.25 - 0.75 hr period (fumigation) 0.75 -2 hr period	 Plant Vent Stack 6.63E-04 sec/m3 Ground 8.80E-05 sec/m3 Elevated 2.74E-06 sec/m3 Elevated
LPZ release point basis	Main stack (elevated) through SGT system (after 15 minute drawdown period of ground level release and one-half hour of fumigation). Distance to LPZ = 6400 meters
LPZ Release Point Dispersion Factors 0-0.25 hr period (drawdown) 0.25 - 0.75 hr period (fumigation) 0.75 -2 hr period 2- 8 hr period 8 - 24 hr period	 Plant Vent Stack 2.65E-05 sec/m3 Ground 1.05E-05 sec/m3 Elevated 1.77E-06 sec/m3 Elevated 8.34E-07 sec/m3 Elevated 5.72E-07 sec/m3 Elevated

TABLE 15.7-22

FHA DOSE SUMMARY

Location	**Limits (REM TEDE)**	**Dose (REM TEDE)**
EAB	6.3	1.45
LPZ	6.3	0.059
CR	5.0	1.14

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

The ATWS event is not analyzed for reload cores.

Prior to the issuance of the ATWS Rule (Reference 7) for the mitigation of the consequences of postulated anticipated transients without scram (ATWS), a potential concern about the possibility of common mode failure of the scram system was noted in ACRS reports for various reactors. General Electric examined the BWR reactor protection system and concluded that no credible common mode failure could prevent a scram signal from reaching the scram actuators when scram is called for. This study is documented in Reference 1. The same type of equipment is used in the actuation and drive mechanism for each control rod. However, each of the 185 control rods has its own separate scram actuator and rod drive equipment. Therefore, the probability that the control rod drives or scram actuators for every rod would all fail at exactly the same time is extremely low. This coupled with the technical specification requirements for periodic testing provides a very high degree of assurance that any failure mode which might affect the scram function for all control rods would be detected long before the failure had affected enough of the rods to prevent reactor shutdown in the event scram was called for. Thus, the occurrence of a common mode failure which would cause complete failure to scram, or even failure to insert enough rods to shut down the reactor to the hot standby condition, is an extremely unlikely event.

General Electric believed that the complete failure of the BWR scram system due to common mode failure is of such extremely low probability that no change in BWR design to account for the event is warranted. However, at the request of the AEC in 1971, a study was performed to evaluate the consequences of an undefined failure of the scram protection system. GE reported on those features which could mitigate the effects of a reactor shutdown from ATWS without a control rod drive scram. Based on its evaluation of the nature of ATWS, General Electric established appropriate criteria and provided an analysis of the event, including a suggested design change which enabled the then current BWR product line to meet the criteria. This analysis was reported in Reference 2.

As was indicated in Reference 2, tripping the recirculation pumps at the start of an ATWS event prevents the violation of any of the criteria when the reactor is shut down shortly after the beginning of the ATWS event. This shutdown can be achieved by manual insertion of control rods. The following criteria established in NEDO 10349 (Reference 2) are satisfied: radiological releases are less than 10 CFR 100 limitations; there is no failure of the reactor coolant pressure boundary; the core is in a long-term coolable geometry; and containment integrity is maintained.

References 3, 4, and 5 addressed the regulatory staff position published via WASH-1270. This NEDO-20626 document discusses design modifications which would mitigate ATWS consequences in Class 1.B plants.

In September 1976, GE submitted the reliability evaluation for RPT and an alternate rod insertion system (ARI), which identified the simple modifications to BWR scram systems that would increase their reliability by several orders of magnitude. Edison committed to install the RPT plus ARI via letter of September 29, 1976, Edison to the NRC (Lee to Hanauer).

NUREG-0460 was issued by NRC in late 1978 with recommendations for modifications to specific categories of BWR plants. GE's generic response was solicited by a February 15, 1979, request from Dr. Mattson to Dr. Sherwood. That proprietary response, as provided in May 1979, covered Alternate 3 specifically (RPT + ARI + Autoboron) as requested, but also concluded that Alternate 2 (RPT + ARI) adequately met the NRC's criteria and did it in a cost-effective way, whereas the NRC's Alternate 3 recommended solution was not justified by value assessment. "Implementation of Alternate 2 of NUREG-0460 as defined by the Staff is all that would be necessary to make the ATWS risk acceptable based on the staff's own statement that the present likelihood of severe consequences arising from an ATWS event is acceptably small and presently there is no undue risk to the public from ATWS.". The value of implementing any modifications beyond Alternate 2 is diminishingly small and non-economic.

A plant-unique evaluation was performed for the LaSalle plant to demonstrate the adequacy of the RPT and ARI design for preventing and mitigating ATWS events (reference 6). The limiting transient events analyzed thus confirm the prior conclusion that RPT and ARI are adequate design modifications for the ATWS hypothesis. Nevertheless, in response to 10CFR50.62 (C)(4), LSCS has incorporated design changes in each Unit, to mitigate the consequences of certain ATWS events. The rule requires that each BWR have a Standby Liquid Control System which a minimum flow capacity and boron content equivalent in control capacity to 86 gpm of 13 weight percent sodium pentaborate solution. To meet this requirement the SBLC system will use a boron 10 enriched sodium pentaborate. The concentration and temperature requirements will remain exactly as before. The LaSalle County Station plant design currently contains protection against failure to scram via recirculation pump trip on high reactor dome pressure or on low-low reactor water level using one-out-of-two taken twice logic.

An additional steam flow induced process measurement error in the Level 3 scram was evaluated by GE in Reference 14 for ATWS event and it was concluded that it is not affected by change in L3 analytical limit (AL) as there is no L3 function directly credited by the ATWS events. However since there is no scram there is bypass steam flow in the annular region outside the dryer, which causes an induced error in the L2 trip. The L2 function is not directly credited by the ATWS events.

An ATWS analysis was performed by GEH to support the 2012 transition to the GNF2 fuel design. The results of the plant-specific ATWS analysis (Reference 16) for LaSalle based on an all GNF2 core indicate that the deployment of GNF2 does not adversely affect the plant's ability to meet the ATWS acceptance criteria. Additionally, this conclusion is applicable for future LaSalle cycles with a mixed core of GNF2 (GNF) and ATRIUM-10 (AREVA). Therefore, the analyses of record are still applicable.

For the limiting ATWS events, the scenario involves pressurization due to the MSIV closure. The reactor isolation leads to a recirculation pump trip (ATWS RPT) very early in the transient and the trip is usually reached at about the same time the MSIVs are full closed. The ATWS RPT rapidly reduces power and steaming rate and is the key feature that reduces the steaming rate to be within the capacity of the Safety / Relief Valves. The post RPT power level is on the order of 50 to 55% power and by the time the level is near the Level 2 AL, the power and steaming rate is below 50%. With the reactor steaming rate reduced to 50%, the error will be significantly reduced, and its effect with be approximately ¼ of the effect at rated conditions. This would reduce the error of 6 inches, for example, at rated power to about 1.5 inches at these conditions. Since the LaSalle RCIC/HPCS initiation at this water level for ATWS is not critical to the event mitigation, this error and delay to L2 is considered insignificant. A small delay for the RCIC / HPCS initiation would be slightly beneficial as the water level would be lower during a portion of the transient and would result in a reduced reactor power and reduced steaming rate to the suppression pool. The long-term mitigation of these events involves controlling water level to low levels in the vessel. Again the small error at these conditions (< 2 inches) is insignificant for water level control and power generation compared to the analysis. Based on above discussion, the analysis of record in Reference 13 for a mixed core of GE and AREVA fuel types remains applicable with respect to steam flow induced process measurement error.

Non-limiting ATWS events that may initiate the Level 2 ATWS-RPT or other L2 functions for ATWS would also be affected by L3 analytical limit error. An example would be the LOFW event. This event would result in recirculation runback associated with the loss of flow and low level (e.g., level 4). This would reduce the power and steaming rate. The power would also reduce due to the reduced subcooling associated with the loss of feedwater flow. The combined effect would reduce the error to approximately half of the condition at rated power (based on an estimated power and steaming rate reduced to 70% prior to level 2). As events that trip ATWS-RPT on low level are power and pressure reduction events, they do not challenge the ATWS acceptance criteria and therefore a low level ATWS RPT delay due to L3 scram error is not significant for compliance to the ATWS acceptance criteria. Therefore, the expected steam flow induced error (approximately half of the error at rated conditions) will have no significant affect on the power and pressure events, and these events will remain far from limiting.

15.8.1 References

1. "An Analysis of Functional Common Mode Failures in GE BWR Protection and Control Instrumentation," NEDO-10l89, July 1979.

2. L. A. Michelotti, "Analysis of Anticipated Transients Without Scram," NEDO-10349, March 1971.

3. L. B. Claassen and E.C. Eckert, "Studies of BWR Designs for Mitigation of Anticipated Transients Without Scrams," NEDO-20626, October 1974.

4. L. B. Claassen and E. C. Eckert, "Studies of BWR Designs for Mitigation of Anticipated Transients Without Scrams, Amendment 1," NEDO-20626-1, June 1975.

5. L. W. Baysinger, E. C. Eckert, and D. G. Weis, "Studies of BWR Designs for Mitigation of Anticipated Transients Without Scrams, Amendment 2," NEDO-20626-2, July 1975.

6. "Evaluation of Anticipated Transients without Scram for LaSalle County Station Unit 2" (1983), QUAD-1-83-007.

7. ATWS Rule - 10CF50.62 Reduction of Risk from Anticipated Transient Without Scram (ATWS) events for Light-Water-Cooled Nuclear Power Plants.

8. Deleted

9. Deleted

10. Deleted

11. Power Uprate Project Task 902, "Anticipated Transient Without Scram," GE-NE-A1300384-25-01, Revision 1, June 2000.

12. Deleted

13. GE-Document NE-0000-0026-4769-00, Revision 0, "GE14 Fuel Design Cycle – Independent Analyses for LaSalle Unit 1 and Unit 2," January 2005.

14. “BWR Owners Group Evaluation of Steam Flow Induced Error (SFIE) Impact on the L3 Setpoint Analytic Limit” GEH-NE-0000-0077-4603-R1.

15. Deleted

16. Design Analysis L-003696, Revision 1, "NEDC-33647P, GNF2 Fuel Design Cycle- Independent Analyses for Exelon LaSalle County Station Units 1 and 2" February 2012.

15.9 Loss of All Alternating Current Power (Station Blackout)

The Loss of all Alternating Current Power (Station Blackout) (SBO) is not analyzed for reload cores.

15.9.1 Identification of Causes and Frequency Classification

The Station Blackout Rule (Reference 1), requires that each light- water-cooled nuclear power plant licensed to operate must be able to withstand for a specified duration and recover from a station blackout. This event is not given a frequency classification based on event frequency categories, but is required to be analyzed per the Station Blackout Rule, 10 CFR 50.63.

Station Blackout occurs as a result of a Loss Of Off-site Power (LOOP) in conjunction with a loss of on-site AC power, failure of Diesel Generators 0 and 1A or 2A. Diesel Generators 1B and 2B are assumed to be available to support the operation of the HPCS system during the Blackout, but are not classified as "Alternate AC" power sources, because Division 3 does not supply power to safe shutdown loads. Therefore, even though Diesel Generators 1B and 2B are available, LaSalle coping analysis uses the AC-independent approach.

15.9.2 Sequence of Events and System Operation

a. The first 50 seconds of a Station Blackout are the same as for the Loss of A-C Power event in section 15.2.6. The initial conditions for the analysis of a Station Blackout are included in Table 15.9-1. Station Blackout is assumed for analysis to occur on both units. Immediately prior to the postulated Station Blackout event, the reactor and supporting systems are within normal operating ranges for pressure, temperature, and water level. All plant equipment is either normally operating or available from the standby state.

   The Division 3 Diesel Generators are assumed to operate normally during the Station Blackout, allowing the HPCS systems to supply make-up water to the reactor vessel from the suppression pool. Also, RCIC systems are assumed to operate normally during the Station Blackout.

The Station Blackout coping duration of 4 hours is based on:

a. Plant offsite AC power design characteristic Group "P1",

b. Emergency AC (EAC) power configuration Group "D",

c. Target Emergency Diesel Generator (EDG) reliability of 0.975.

15.9.3 Station Blackout Coping Capability

15.9.3.1 Condensate Inventory for Decay Heat Removal

LaSalle's station blackout coping method uses RCIC or HPCS to provide makeup water for core cooling. The HPCS system normally takes suction from the suppression pool, and RCIC suction automatically transfers to the suppression pool on low condensate storage tank level. Decay heat is removed by discharge of steam through the Safety/Relief Valves into the suppression pool, where the steam is condensed. As a result, gradual heatup of the suppression pool is expected.

Per analysis the suppression pool water inventory is sufficient to make up for decay heat removal requirements and expected leakage during a four hour station blackout. The suppression pool temperature will remain below the heat capacity temperature limits while providing this water, if cooldown is limited to 20°F/hr. (Reference 13).

The condensate inventory analysis was performed to determine if the suppression pool contains sufficient water for a four-hour SBO with the reactor at cooldown (Using either RCIC or HPCS cooling system). The analysis (Reference 22) assumes:

a. Water is not available from the cycled condensate tanks.

b. Losses in suppression pool inventory are associated with changes (an increase) to reactor vessel inventory, T.S. leakage to the drywell, RR pump seal leakage, and RCIC pump turbine gland seal leakage. These inventories do not return to the suppression pool.

c. The Reactor Coolant System leakage is 61 GPM.

The total amount of water needed for decay heat removal, cooldown, and leakage is 142,703 gallons, HPCS mode, and 140,349 gallons, RCIC mode, of which 101,921 gallons (HPCS) and 99,447 gallons (RCIC) is returned to the pool.

With applied margin, the calculated suppression pool water level at the end of a four hour SBO (plus 15-min recovery period) is 698.04 feet (a drop of 1.5 feet), for either HPCS or RCIC operation. NPSH calculations for the SBO coping period use an analytical suppression pool water level of 696.88 ft. This value includes additional applied margin beyond the minimum computed level above (Reference 22).

The Reactor Coolant System inventory analysis demonstrates that the RCIC system is capable of maintaining the water level above the top of active fuel (Reference 22) assumes that:

a. The reactor is operating at full power of 3559 MWt, dome pressure of 1040 psia, and normal water level at time of initiation. (The power level and dome pressure are analytical values that include uncertainty above the plant operating limits to allow for instrument uncertainty).

b. The reactor scrams at event initiation.

c. The MSIVs are fully closed in 5 seconds.

d. The ANSI/ANS 5.1-1979 decay heat correlation is used.

e. RCIC initiates automatically when water level drops to Level 2.

f. Operator action to control the maximum depressurization cooldown to a rate of 20° F/hr is assumed following RCIC startup.

Evaluations performed by GE for LaSalle indicate that the introduction of GNF2 fuel designs (Reference 23) does not affect the existing decay analysis design basis and, as a result, is not expected to affect this aspect of the station blackout coping capability.

AREVA documents their disposition for the ATRIUM-10 reload fuel and the ATRIUM 10XM LTA at MUR conditions in Reference 21.

15.9.3.2 Class 1E Battery Capacity

The 125 Vdc (Divisions 1 and 2) and 250 Vdc Class 1E batteries are sized to provide SBO loads for 4 hours. A calculation was performed to ensure that these batteries have sufficient capacity (i.e.) a minimum remaining margin of 5% to meet the station blackout loads for four hours assuming that loads not needed to cope with a station blackout are shed. The required loads include power restoration from either the emergency ac power supplies or the preferred power source. The loads that need to be shed are listed in Table 15.9-2. The shedding of these loads is proceduralized (Reference 5 and 6).

The methodology used to determine battery capacity is as follows (Reference 6):

The existing 125 Vdc and 250 Vdc computer models were used for this SBO calculation because they include the battery manufacturer's characteristics required in determining battery sizing, all dc loads (with their various characteristics such as inrush and continuous current, when the load is energized, and the duration of the load), and they calculate the required number of positive plates and the battery capacity remaining. The loads for SBO were verified to be energized for the entire four-hour SBO plus recovery, unless the load was shed. The exceptions to this are excitation cubicles, fire protection sirens, and the TIP panel loads which are verified to be energized for fifteen minutes or less after the inception of SBO.

The design margin for this study was assumed as 1.0. The maximum temperature was set at the Technical Specification limits. The aging factor may be adjusted (less than 1.25) to maintain a minimum 5% remaining margin. Appropriate battery performance procedure(s) require verification that the batteries have a minimum capacity consistent with the aging factor used in the station blackout battery sizing calculation. The temperature factor, design margin and aging factor are all incorporated into the computer models. Note, battery sizing is calculated using the methodology of IEEE-485.

15.9.3.3 Compressed Air

There is no AC power to station air compressors during SBO, however, instrument nitrogen is required for the relief mode operation of the main steamline SRVs. The Automatic Depressurization System (ADS) valves (7 of the SRV's) have existing backup nitrogen bottle banks that have been analyzed to ensure they are sufficient to support SRV actuations for the four-hour coping duration. Manual opening and closing of individual ADS valves, to depressurize the reactor in a controlled manner, requires sending an operator to the Auxiliary Electric Equipment Room (AEER). Emergency lighting and communications already exist to gain access to the AEER and to utilize the required controls. Control of the ADS valves from the AEER can be established within 20 minutes. During this time, the Low Low Set function of five of the ADS valves will automatically control reactor pressure in the normal operating range. The evaluation of the nitrogen bottle bank is based on the opening of the individual SRVs as necessary to maintain an average 20°F/hr. cooldown over the four-hour coping period (Reference 13). Additionally, the ADS valves (all 7 at once) can be manually initiated by either of two divisions of DC logic from the Control Room. As a backup, the mechanical safety mode of SRV operation is available independent of nitrogen bottles to control pressure (Reference 5).

15.9.3.4 Effects of the Loss of HVAC

The areas of concern at LaSalle Station due to the loss of ventilation were chosen from rooms that based on documented engineering judgement (1) contained SBO response equipment, (2) have substantial heat generation terms and (3) lack normal heat removal systems due to the blackout. The Main Control Room, Auxiliary Electric Equipment Rooms (AEER), and the RCIC room satisfy these criteria. Areas immediately adjacent to these areas of concern were considered, as well as the floors immediately above and below, in determining heat flow and heat contributions. Additionally, a temperature transient analysis was performed on both the Drywell and Suppression Pool to determine the maximum expected temperature and equipment operability.

The Main Steam Tunnel was considered for the temperature heat up analysis but a review revealed that it did not contain SSD equipment credited for SBO, nor RCIC isolation temperature instrumentation.

The RCIC pipe tunnel contains ambient and differential temperature instrumentation for steam leak detection. However, the RCIC turbine isolation valves which are affected by this logic (1(2)E51-F008, 1(2)E51-F063, and 1(2)E51-F076) are AC powered and AC controlled. These valves are maintained in the position required for proper operation of the RCIC system when the system is lined up in the standby condition, that is, 1(2)E51-F008 and 1(2)E51-F063 are open and 1(2)E51-F076 is closed. Thus, during an SBO event the loss of HVAC is not an isolation concern for RCIC operation as these valves remain in their required positions on a loss of AC power.

The HPCS diesel is available to power the HPCS pump and its associated systems during a station blackout. This source powers ventilation in the HPCS rooms. Since ventilation will be provided if the HPCS is used during a station blackout, equipment operability is established and no heatup analysis is required in this area (Reference 6).

The Main Control Room, AEER, and RCIC temperature transient calculations are based on evaluation which included a surveillance review of summertime room temperatures. The RCIC room temperature calculations also included the maximum calculated steam leakage from the RCIC turbine gland seals at a conservative turbine backpressure of 50.0 psig (Reference 13). The initial room temperatures and heat loads used in the calculations necessitated procedural requirements to open panel doors in both the Main Control Room and AEER during this event. The access doors to the AEER and Main Control Room are not required to be open during a SBO for cooling purposes. NUMARC 87-00 (Reference 3) provides guidelines for determining "Reasonable Assurance of Operability," (RAO). The initial temperatures assumed, final temperatures at the end of a four hour SBO, and the RAO justification for the Main Control Room, AEER areas, and RCIC rooms are provided in Table 15.9.3.

The control room, AEER, and RCIC room temperatures are monitored daily. If the initial temperatures assumed in the SBO analyses are exceeded, then appropriate action should be taken to investigate the problem and resolve it in a timely manner.

15.9.3.5 Primary Containment Calculations

The initial conditions for the primary containment calculations are:

Suppression Chamber water temperature:	105 °F
Drywell Air temperature:	135 °F
Reactor Recirculation pump seal leakage per pump:	18 gpm
Technical Specification Reactor coolant system leakage rate (excluding RR pump seal leakage):	25 gpm

The calculations for Suppression Chamber water temperature assume an average reactor pressure vessel depressurization rate of 20°F/hr (Reference 13). The slower the depressurization rate, the slower the heatup of the suppression chamber water. The water temperature after 4 hours and 15 minutes with no cooling is 200°F

with HPCS supplying RPV inventory makeup during SBO and 196°F with RCIC supplying RPV inventory makeup during SBO. The cooldown rate is administratively controlled.

There were calculations performed for two different major assumptions for determination of drywell air temperature at the end of a 4 hour SBO. The analysis assumed that the reactor coolant system remains at saturated conditions of 1020 psia and the corresponding saturation temperature. Also, all the primary system leakage (61 gpm) is assumed to stay in the drywell atmosphere, with no atmosphere transfer to the suppression chamber. This results in a final drywell air temperature of 315°F, assuming the reactor is not depressurized during a 4-hour station blackout. Should the reactor be depressurized and normal venting of the drywell to suppression pool occur, this temperature will be lower (Reference 6 and 8).

Therefore, the Drywell maximum air temperature decreases as reactor depressurization rate increases. Thus, there is a trade-off between having lower drywell air temperature by increasing reactor depressurization rate versus a lower final suppression pool temperature by a slower reactor depressurization rate. However, the analyses bound depressurization rates from 0 °F/hr up to and including an average of 20°F/hr (Reference 13).

The equipment qualification curve for the drywell is a step function with the following temperature limits:

340 °F from 0 to 3 hours
320 °F from 3 to 6 hours
250 °F from 6 to 24 hours

Thus, the EQ equipment inside the drywell is designed to operate at 320 °F for 6 hours and envelopes the SBO drywell temperatures (Reference 8).

15.9.3.6 Containment Isolation

The list of containment isolation valves in UFSAR Table 6.2-21 were reviewed to ensure that the isolation functions and position indication can be provided during an SBO event. Position indication is considered acceptable if it includes local mechanical indication, DC powered indication (including AC indicators powered from inverters) and HPCS DG powered indication. The acceptable means of valve closure include manual operation, air operation (including air operated valves that are mechanically closed on loss of air), DC powered operation and HPCS DG

powered operation. As recommended in NUMARC 87-00 the following criteria were used to exclude valves from consideration:

1. valves normally locked closed during operation;
2. valves that fail closed on loss of AC power or air;
3. check valves;
4. valves in non-radioactive closed-loop systems not expected to be breached in a station blackout (with the exception of lines that communicate directly with the containment atmosphere); and,
5. all valves less than 3-inch nominal diameter.

Since independent valve failures are not assumed to occur during a station blackout, a valve in line with an excluded valve was also excluded from consideration. In addition, valves which continue to be powered and operable during a station blackout do not require manual operation capability. Table 5-1 of Reference 6 lists the valves reviewed and their exclusion justification. When multiple valves are in line with one penetration, all the valves are listed but only one valve would need to be closed. Table 15.9-4 lists the valves that would require operator action and verification of closure.

Full containment isolation is not expected to be necessary as a result of a station blackout. However, Regulatory Guide 1.155 requires reactors to have the ability to maintain "containment integrity" in station blackout conditions should this be necessary for other reasons. Such other reasons could include requirements to close certain valves following a loss of offsite power, loss of decay heat removal capability, or other casualties affecting the reactor coolant system (Reference 6).

15.9.3.7 Recovery from a Station Blackout

The SBO analyses for suppression pool temperature assume that suppression pool cooling is established ≤ 15 minutes following a SBO. The suppression pool temperature 4 hours and 15 minutes after a SBO occurs is 200°F when using the HPCS system and 196°F when using the RCIC system for decay heat removal and reactor coolant inventory (Reference 13). These temperatures are within the environmental qualification temperature rating for HPCS materials, the RCIC pump materials, and RHR materials (Reference 8).

Per Reference 16, the Net Positive Suction Head (NPSH) requirements versus the Net Positive Suction Head Available at the end of a 4 hour SBO plus a 15 minute allowance to start an RHR pump in the Suppression Pool Cooling mode are as follows:

1. RCIC Pump NPSH Available (at pump inlet centerline) is 27.25 feet.

2. RCIC Pump NPSH Required (at pump inlet centerline) is 20.5 feet.

3. HPCS Pump NPSH Available (at pump inlet centerline) is 16.7 feet.

4. HPCS Pump NPSH Required (at pump inlet centerline) is 6.0 feet.

5. RHR Pump NPSH Available (at pump inlet centerline) is 16.45 feet.

6. RHR Pump NPSH Required (at pump inlet centerline) is 15.0 feet.

The RCIC turbine backpressure was determined based on the worst case suppression pool water levels, suppression chamber pressure and RCIC turbine exhaust flow following the SBO. The calculated maximum RCIC turbine backpressure is 36.4 psig (Reference 13). This pressure is below the RCIC turbine backpressure trip setpoint of 43.7 psig (Reference 8 and 11).

15.9.4 Quality Assurance

A QA program meeting the requirements of Regulatory Guide 1.155 Appendices A and B has been applied to cover non-safety related equipment needed for coping with a station blackout that were not already covered by existing QA requirements in Appendices B or R of 10 CFR 50 (References 5 and 8).

15.9.5 References

1. 10 CFR 50.63, Loss of All Alternating Current Power.

2. Regulatory Guide 1.155, Rev. 0, Dated June 1988; Station Blackout.

3. NUMARC 87-00, Rev. 1, Dated August, 1991; Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors.

4. Letter dated April 17, 1989 from M.H. Richter to Dr. T.E. Murley, Director of the Office of Nuclear Reactor Regulation; Response to Station Blackout Rule, including NRC Docket Nos. 50-373 and 50-374.

5. Letter dated June 22, 1990 from M.H. Richter to Dr. T.E. Murley, Director of the of Nuclear Reactor Regulation; LaSalle County Station Units 1 and 2, Revised Office Response to Station Blackout Rule, NRC Docket Nos. 50-373 and 50-374.

6. Letter dated September 23, 1991 from P.L. Piet to the Office of Nuclear Reactor Regulation; LaSalle County Station Units 1 and 2,

Supplemental Response to Station Blackout Rule, NRC Docket Nos. 50-373 and 50-374.

7. Letter dated March 6, 1992 from B.L. Siegel, NRR Project Manager; Safety Evaluation of the LaSalle County Station Response to the Station Blackout Rule (TAC Nos. M68559 and M68560).

8. Letter dated May 15, 1992 from J.N. Shields to Dr. T.E. Murley, Director of the Office of Nuclear Reactor Regulation; LaSalle County Station Units 1 and 2, Response to Safety Evaluation on the Station Blackout Rule, NRC Docket Nos. 50-373 and 50-374.

9. Letter dated July 17, 1992 from B.L. Siegel, NRR Project Manager; Safety Evaluation Related to Station Blackout Analysis, LaSalle County Station, Units 1 and 2 (TAC Nos. M68559 and M68560.

10. Chron# 300100 dated March 28, 1994, LaSalle County Station Units 1 and 2 Station Blackout Analysis Review of S&L Calculations ATD-0117 and 3C7-0390-011.

11. Chron #302940 dated September 30, 1994, LaSalle County Station, Units 1 and 2, Approval of RCIC Turbine Exhaust High Pressure Trip Setpoint Change.

12. Calculation L-001260, ECCS & RCIC Suppression Pool Suction Strainer Head Loss for a 50% Plugged Strainer.

13. NDIT LAS-ENDIT-1255, Upgrade 3, “Station Blackout”.

14. NEDO-32701P, Rev. 2, Power Uprate Safety Analysis Report for LaSalle County Station Units 1 and 2, July 1999.

15. Deleted.

16. Calculation L-002540, "NPSH Margin for HPCS, RHR, and RCIC Pumps, Backpressure for RCIC Turbine."

17. Calculation ATD-0351, RCIC Pump Room Temperature Transient Following Station Blackout With Gland Seal Leakage.

18. Deleted.

19. Calculation (Design Analysis) 3C7-0290-001, Main Control Room Temperature Transient Following Station Blackout.

20. Calculation (Design Analysis) 3C7-0289-001, Aux Electric Equipment Room Temperature Transient Following Station Blackout.

21. Design Analysis L-003509, Revision 0, "Evaluation of Appendix R, Station Blackout, Containment and Source Terms for LaSalle MUR Power Uprate," July 2010.

22. Calculation (Design Analysis) 3C7-0189-001, Station Blackout Condensate Inventory Coping Assessment.

23. Section 3.0, GNF Document NEDC-33647P, Revision 2, “GNF2 Fuel Design Cycle-Independent Analyses for Exelon LaSalle County Station Units 1 and 2,” dated February 2012.

TABLE 15.9-1

INPUT PARAMETERS AND INITIAL CONDITIONS FOR ANALYSIS OF STATION BLACKOUT

1.	Thermal power level, MWt Analysis value *	3489 (100% rated thermal power)
2.	Operation History *	ANSI/ANS 5.1-1979
3.	Suppression Pool	
	a. temperature	105°F
	b. minimum water volume	128,800 ft^3
4.	Primary system leakage:	
	a. Tech Spec allowed leakage	25 gpm
	b. Reactor Recirc Pump seal leakage	18 gpm/pump
	Total leakage assumed:	61 gpm
5.	Drywell atmosphere temperature	135°F
6.	ADS bottle bank initial pressure for Low Low Set SRV actuation	500 psig
7.	Dominant Areas of Concern for loss of ventilation effects	
	a. Auxiliary Electric Equipment	90°F
	Rooms	90°F
	b. Control Room	124°F
	c. RCIC Rooms	

*Reference 13

TABLE 15.9-2

LOAD SHEDDING FOR STATION BLACKOUT

(Sheet 1 of 2)

BATTERY	LOAD	LOCATION OF FEEDER BREAKER – BREAKER NUMBER
Unit 1, 125-Vdc Div. 1 (1DC07E)	Reactor Bldg. Ltg. Cab. 140 LFMG Aux. LFMG Aux. Relay Panel Turbine Bldg. Ltg. Cab. 141 Radwaste Cont. Panel 0PL01J Radwaste Annunciator 1A Control FW Pump Turbine 1A Control	125-Vdc Dist. Pnl. 111Y (1DC11E), Compartment 4A-11 125-Vdc Dist. Pnl. 111Y (1DC11E), Compartment 4A-14 125-Vdc Dist. Pnl. 111X (1DC10E), Compartment 1A-12 125-Vdc Dist. Pnl. 111X (1DC10E), Compartment 1A-8 125-Vdc Dist. Pnl. 111X (1DC10E), Compartment 1A-4 125-Vdc Dist. Pnl. 111X (1DC10E), Compartment 1A-2
Unit 1, 125-Vdc Div. 2 (1DC14E)	Reactor Bldg. Ltg. Cab. 142 Reactor PNL Recirc. Sys. FW Control Panel FW Pump Turbine 1B Control Hyd. & Stator Cooling Cabinet Turbine Bldg. Ltg. Cab. 143	125-Vdc Dist. Pnl. 112Y (1DC13E), Compartment 4A-13 125-Vdc Dist. Pnl. 112Y (1DC13E), Compartment 4A-9 125-Vdc Dist. Pnl. 112X (1DC12E), Compartment 1A-2 125-Vdc Dist. Pnl. 112X (1DC12E), Compartment 1A-1 125-Vdc Dist. Pnl. 112X (1DC12E), Compartment 1A-4 125-Vdc Dist. Pnl. 112X (1DC12E), Compartment 1A-19
Unit 1, 125-Vdc Div. 3	None	---
Unit 1, 250-Vdc (1DC01E)	Emergency Bearing Oil Pump Emergency Seal Oil Pump Uninterruptible Power Supply (1IP01E)	250-Vdc MCC 121X (1DC05E) – 3A 250-Vdc MCC 121X (1DC05E) – 3C 250-Vdc MCC 121Y (1DC06E) – 2A
Unit 2, 125-Vdc Div. 1 (2DC07E)	Reactor Bldg. Ltg. Cab. 240 LFMG Aux. Relay Panel Turbine Bldg. Ltg. Cab. 241 FW Pump Turbine 2A Control	125-Vdc Dist. Pnl. 211Y (2DC11E), Compartment 4A-11 125-Vdc Dist. Pnl. 211Y (2DC11E), Compartment 4A-14 125-Vdc Dist. Pnl. 211X (2DC10E), Compartment 1A-12 125-Vdc Dist. Pnl. 211X (2DC10E), Compartment 1A-2

TABLE 15.9-2

LOAD SHEDDING FOR STATION BLACKOUT

(Sheet 2 of 2)

BATTERY	LOAD	LOCATION OF FEEDER BREAKER – BREAKER NUMBER
Unit 2, 125-Vdc Div. 2 (2DC14E)	Reactor Bldg. Ltg. Cab. 242 LFMG Aux. Relay Panel FW Control Panel FW Pump Turbine 2B Control Hyd. & Stator Cooling Cabinet Turbine Bldg. Ltg. Cab. 243	125-Vdc Dist. Pnl. 212Y (2DC13E), Compartment 4A-13 125-Vdc Dist. Pnl. 212Y (2DC13E), Compartment 4A-9 125-Vdc Dist. Pnl. 212X (2DC12E), Compartment 1A-2 125-Vdc Dist. Pnl. 212X (2DC12E), Compartment 1A-1 125-Vdc Dist. Pnl. 212X (2DC12E), Compartment 1A-4 125-Vdc Dist. Pnl. 212X (2DC12E), Compartment 1A-19
Unit 2, 125-Vdc Div. 3	None	---
Unit 2, 250-Vdc (2DC01E)	Emergency Bearing Oil Pump Emergency Seal Oil Pump Uninterruptible Power Supply (2IP01E)	250-Vdc MCC 221X (2DC05E) – 3A 250-Vdc MCC 221X (2DC05E) – 3C 250-Vdc MCC 221Y (2DC06E) – 2A
NOTE: All loads are shed within 30 minutes after SBO begins, with the exception of the 250-Vdc MCC 121X and MCC 221X Loads which are shed within 180 minutes after SBO begins.		

TABLE 15.9-3

AREAS OF CONCERN FROM THELOSS OF HVAC DUE TO STATION BLACKOUT

AREA	INITIAL TEMP.	FINAL TEMP.	RAO JUSTIFICATION
Control Room	90°F	116°F Note (1)	Less than 120°F (open panel doors)
U1 South AEER	90°F	110°F Note (2)	Less than 120°F (open panel doors)
U1 North AEER	90°F	119.7°F Note (2)	Less than 120°F (open panel doors)
U2 SE AEER	90°F	119.7°F Note (2)	Less than 120°F (open panel doors)
U2 NW AEER	90°F	108.6°F Note (2)	Less than 120°F (open panel doors)
RCIC Rooms	124.0°F	157.4°F* Note (3)	Less than the design temp. of 212°F for 6 hours

(1) The calculated maximum peak and final temperature at the end of the four-hour SBO event are less than 116°F (Ref. 19).

(2) These values assume the doors between the U1 South and North AEER subrooms, and the U2 SE and NW AEER subrooms, are closed. With the doors between the subrooms open, the temperature in the U1 South and U2 NW subrooms could increase and the temperature in the U1 North and U2 SE subrooms could decrease. However, the peak final temperature at the end of the four-hour SBO event of 119.7 F will not be exceeded (Ref. 20).

(3) Maximum RCIC room temperature during station blackout. This temperature is reached at the end of the four-hour SBO event plus a 15 minute allowance (Reference 17).

TABLE 15.9-4

CONTAINMENT ISOLATION VALVES FOR WHICH POSITION INDICATION AND MANUAL OPERATION CAPABILITY SHOULD BE PROVIDED DURING STATION BLACKOUT

CONTAINMENT PENETRATION DESCRIPTION	SIZE (INCHES)	VALVE NUMBER	VALVE TYPE	NOTES
RHRS Shutdown Suction	20 20	1&2E12-F009 1&2E12-F008	Motor Operated Gate Motor Operated Gate	E12-F008 is in line with E12-F009
System to RCIC System	10 4	1&2E51-F063 1&2E51-F008	Motor Operated Gate Motor Operated Gate	E51-F063 is in line with E51-F008
RHRS/Containment Spray	16 16	1&2E12-F017A, B 1&2E12-F016A, B	Motor Operated Gate Motor Operated Gate	E12-F016 is in line with E12-F017
Main steam Drains	3 3	1&2B21-F016 1&2B21-F019	Motor Operated Gate Motor Operated Gate	B21-F016 is in line with B21-F019
Combustible Gas Control Drywell Suction	4 4	1&2HG001A, B 1&2HG002A, B	Motor Operated Gate Motor Operated Gate	HG-001 is in line with HG-002
Reactor Cleanup	6 6	1&2G33-F001 1&2G33-F004	Motor Operated Gate Motor Operated Gate	G33-F001 is in line with G33-F004
LPCS Suc. From Sup. Pool	24	1&2E21-F001	Motor Operated Gate	
RHR Suc. fr. Sup. Pool	24	1&2E12-F004A	Motor Operated Gate	
RHR Suc. fr. Sup. Pool	24	1(2)E12-F004C	Motor Operated Gate	
RHR Suc. fr. Sup. Pool	24	1&2E12-F004B	Motor Operated Gate	
RHR to Sup. Pool Spray	24	1&2E12-F027A, B	Motor Operated Gate	
LPCS Minimum Flow	14 4	1&2E21-F012 1&2E21-F011	Motor Operated Globe Motor Operated Gate	
RHR Minimum Flow and Test Lines	18 18 8 4	1&2E12-F024A, B 1&2E12-F021 1&2E12-F064 A, B, C 1&2E12-F011A, B	Motor Operated Globe Motor Operated Globe Motor Operated Gate Motor Operated Gate	
Combustible Gas Control Return	6 6	1&2HG005A, B 1&2HG006A, B	Motor Operated Gate Motor Operated Gate	HG005 is in line with HG006
Vacuum Breaker	24	1&2PC003A, B, C, D	Butterfly	Normally manually operated
NOTE: When two or more valves are in line, manual operability need be provided for only one of the valves. Valves 1(2)E51-F064 have been replaced by spectacle Flanges 1(2)E51-D324.				

15.10 Thermal Hydraulic Instability Transient

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

15.10.1 Identification of Causes and Frequency Classification

Events such as Reactor Recirculation (RR) pump trips and runbacks, turbine/generator runbacks, loss of feedwater heating, and RR flow controller failures can result in unplanned entry into the high power and low flow region of the power to flow map. Under these conditions, axially varying moderator density in the fuel channels can cause flux oscillations that increase in amplitude. Without manual or automatic suppression, such oscillations can cause the MCPR Safety Limit to be exceeded (Reference 15.10.6.1).

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

Anticipated stability-related neutron flux oscillations are those instabilities that result from normal operating conditions, including conditions resulting from anticipated operational occurrences. This category of events is equivalent to the standard terminology for the analysis of events of moderate frequency (Reference 15.10.6.2).

15.10.2 Sequence of Events and System Operation

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

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

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

C.1 If the OPRM is not operable, the operator manually scrams the reactor upon recognition of the instability.

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

15.10.3 Core and System Performance

The OPRM system contains 4 LPRMs per OPRM cell (using the Bockstanz-Lehmann LPRM assignment methodology described in Reference 15.10.6.3) and requires 1 LPRM input for the cell to be operable. The amplitude setpoint for oscillation magnitude and the number of confirmation counts are specified for the analysis. Since core thermal hydraulic instability is characterized by a consistent period for the oscillations, the OPRM logic includes a check for a set number of consecutive counts as well as a magnitude.

For Unit 1, the specific stability OPRM Amplitude Setpoint and Successive Confirmation Count (SCC) Setpoint are confirmed for each cycle using the Option III Solution with GS3 methodology described in Reference 15.10.6.5.

For Unit 2, the specified system setpoints are used to determine the hot channel oscillation magnitude. This information is used, along with empirical data applicable to the fuel in the core, to determine the fractional change of CPR (delta CPR/IMCPR, where IMCPR is initial MCPR).

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

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

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

The Final MCPR (FMCPR) is determined using the IMCPR and CPR/IMCPR data (Reference 15.10.6.3).

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

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

Alternatively, the Operating Limit MCPR may be changed, or the LPRM assignment scheme may be modified.

The above is confirmed for each cycle as part of the reload analysis.

15.10.4 Barrier Performance

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

15.10.5 Radiological Consequences

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

15.10.6 References

15.10.6.1 USNRC Generic Letter 94-02, "Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal Hydraulic Instabilities in Boiling Water Reactors," dated July 11, 1994.

15.10.6.2 GE Document NEDO-31960-A, Supplement 1, "BWR Owner's Group Long-Term Stability Solutions Licensing Methodology (Supplement 1)," dated November 1995.

15.10.6.3 GE Document NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload applications," dated August 1996.

15.10.6.4 Deleted

15.10.6.5 NEDE-33766P-A, Revision 1, Licensing Topical Report, "GEH Simplified Stability Solution (GS3)," March 2015.

15.A Cycle Specific Safety Analyses

The most current safety analyses information can be found in cycle specific transient and reload documents.

The 1999 LaSalle County Station Power Uprate Project included re-evaluating a broad set of most limiting transient events at the power uprate conditions for a representative cycle in Reference 14.. The basis for the selection of the transient events for re-analysis is documented in NEDC-31897P-A, LTR, General Guidelines for GE BWR Power Uprate, May 1992, Appendix E where it stipulates, "Analysis will be performed for the limiting transient events. This includes all events that establish the core thermal operating limits and the events that show bounding conformance to the other transient protection criteria (e.g., ASME overpressure limits)." These transient events, re-analyzed with power uprate conditions from 3323 MWth to 3489 MWth core thermal power, are documented in Section 15.B and Reference 14.

15.A.1 Approach to Safety Analysis

This safety analysis evaluates the ability of the plant to operate within the regulatory guidelines without undue risk to the public health and safety. For each cycle, the station must provide a new safety analysis and request license changes as applicable, based on the new safety analysis. Sections 15.1 through 15.8 describe analyses for a variety of events that were required to be performed on the initial core. The events were classified as either transients or accidents. Transients and accidents are defined in Subsections 15.0.1.a and 15.0.1.b, respectively. Only a few of these events, however, would result in a significant reduction of MCPR or an increase in transient LHGR. Consequently, only the limiting transients must be analyzed for each cycle. These limiting events are discussed in Subsections 15.A.2 and 15.A.3. GE's Disposition of why the other events are not limiting are provided in the event documentation of Sections 15.1 through 15.8.

15.A.2 Categories of Safety Events

The events analyzed in Sections 15.1 through 15.8 were categorized by their initiating cause. The eight categories are described in Section 15.0.2. For the reload analysis, the limiting events are categorized as either Pressurization or Non-pressurization events. The limiting pressurization events include Feedwater Controller Failure and Generator Load Reject without bypass. Limiting non-pressurization events include loss of Feedwater Heater and rod withdrawal error.

15.A.3 Judgment of Nonacceptable Safety Results

The Feedwater Controller Failure event is classified as a Moderate Frequency event. The Loss of Feedwater Heater and Rod Withdrawal Error events are considered Infrequent events. However, due to a lack of sufficient data, these events are conservatively analyzed as Moderate Frequency events. The Generator Load Reject Without Bypass event is analyzed as an Infrequent event.

The guidelines for determining unacceptable safety results for Moderate Frequency and Infrequent events are described in Subsection 15.0.3.

15.A.4 Method of Analysis

The MCPR operating limits provide protection against fuel failure from overheating of the fuel due to inadequate cooling. The MCPR operating limit ensures that the MCPR safety limit, which corresponds to the onset of boiling transition, is not violated during the transient. Power dependent MCPR ($MCPR_p$) limits are required in order to provide the necessary protection with the implementation of Partial ARTS (Reference 9). Flow dependent MCPR ($MCPR_f$) limits provide protection against fuel failures resulting from slow core flow excursions. The implementation of the MCPR operating limit requires that it be set equal to the most restrictive of the full power and flow MCPR limit or the applicable $MCPR_f$ or $MCPR_p$ limits. Power dependent MCPR limits may be provided as multipliers (K_p). The K_p are applied as multipliers on the rated power MCPR limits.

LHGR limits provide protection against fuel failures due to overstraining of the cladding during steady-state operation. These limits also ensure that the LHGR during an anticipated operational occurrence initiated at rated conditions does not exceed the transient LHGR limit. To ensure that the LHGR does not exceed the transient LHGR limit during postulated transient events from reduced power or flow conditions, adjustments to the steady-state LHGR limits are necessary. This is accomplished by applying power and flow dependent multipliers ($LHGRFAC_p$ and $LHGRFAC_f$ respectively) to the steady-state LHGR limits.

Analyses are performed to ensure that the vessel and steam dome pressure safety limits are not exceeded during postulated transients in accordance with the requirements of the ASME Boiler and Pressure Vessel Code.

Cycle specific analyses are performed to ensure that plant operation with the cycle specific core design meets the safety requirements. This is accomplished by establishing or confirming operating limits on cycle-specific basis. Operating limits for the Minimum Critical Power Ratio (MCPR) and Linear Heat Generation Rate limit (LHGR) are established based on the results of the safety analyses for the transient events for the operating cycle to ensure that the safety limits are met.

Operating limits for each cycle are provided in the reload licensing documentation to support operation in each of the Extended Operating Domain (EOD) and Equipment Out of Service (EOOS) conditions. The operating limits include Kp/MCPRp and $MCPR_f$, $LHGRFAC_p$ and $LHGRFAC_f$, and LHGR limits. The $MCPR_f$ and $MCPR_p$ are applicable to both GE and AREVA fuel. The Kp limits apply to both GE and AREVA fuel as MCPR multipliers. The $LHGRFAC_p$ and $LHGRFAC_f$ limits apply to both GE and AREVA fuel LHGRs as LHGR multipliers.

15.A.4.1 Sequence of Events and Systems Operations

Each transient discussed in Subsection 15.A.3 is discussed and evaluated in terms of:

(1) a step-by-step sequence of events from initiation to final stabilized condition;
(2) the extent to which normally operating plant instrumentation and controls are assumed to function;
(3) the extent to which plant and reactor protection systems are required to function;
(4) the credit taken for the functioning of normally operating plant systems;
(5) the operation of engineered safety systems that is required.

15.A.4.2 Analysis Basis

15.A.4.2.1 Input Parameters and Initial Conditions for Analyzed Events

The analyses described in this section are for the typical core arrangement. The present fuel loading scheme for each unit can be found in the Reload Licensing Package (referenced in Reference 8). In general, the analyzed events have numerical input parameters and initial state conditions as specified in Table 15.A-1.

15.A.4.2.2 Initial Power/Flow Operating Constants

15.A.4.2.2.1 Standard Operating Domain

The initial power and flow statepoints for performance of the transient analyses are selected using the allowed pressure range such that thermal margins (as applicable) are obtained. Because the thermal limits are power and flow dependant, in most cases analyses are performed over a range of power and flow statepoints within the analyzed power/flow map. EOD/EOSS combinations are specified each cycle as necessary in the COLR.

Any other constraint which may truncate the bounded power/flow map must be observed, such as the recirculation valve and pump cavitation regions, the licensed power limit and other restrictions based on pressure and thermal margin criteria.

A typical power/flow map is shown in Figure 15.0-1. A more detailed discussion of this curve can be found in Subsection 15.0.4.

15.A.4.2.2.2 Extended Operating Domain

15.A.4.2.2.2.1 GE Analysis

To permit improved power ascension capability to full power within the design bases, the operating envelope is modified to include an expanded operating region bounded by the 108% APRM rod block line (0.58W + 50%), the rated power line and the rated rod line. The region of operation above the rated rod line is known as the Maximum Extended Load Line Limit Analysis (MELLA) region. In addition, the use of increased core flow (ICF) above 100% rated core flow can provide even greater operational flexibility in reaching and maintaining full power during the cycle and can extend the operating cycle at rated power. Final Feedwater Temperature Reduction (FFWTR) is a cycle extension technique. By reducing Feedwater Temperature at the End of Cycle, more moderation of neutrons will increase the power capability and extend the operating cycle. FFWTR can be coupled with any Equipment Out-of-Service.

The limiting operational transients analyzed at 100% power, 100% core flow (100P/100F) were reevaluated to ensure that operation above the rated rod line and within the ICF region was acceptable. Nuclear transient data for 100% power, 82.8% core flow (100P/82.8F) and 100% power, 105% core flow (100P/105F) at the end-of-cycle exposure were developed. Although some parts of the analysis allow 108% rated core flow, 105% was selected as the procedural limit to provide consistency with the jet pump fatigue analysis. Where reduced feedwater temperature might affect the analysis, cases were run with a 100 degree F reduction in temperature, as well as the normal temperature. This assures that the analysis is good for cycle extension using FFWTR. For a detailed description of the results of these analyses, see Reference 8, and the references listed therein.

15.A.5 Evaluation of Results

As discussed earlier, all of the transient/accident descriptions and analyses for initial cores are given in Sections 15.1 through 15.8. The descriptions of the events were for the initial cycle. However, the descriptions for the limiting transients are very similar to the analyses performed on a cycle specific basis. A detailed description of the analyzed event can be obtained in the appropriate Subsections of Sections 15.1 through 15.8 or in Reference 1.

15.A.5.1 Core and System Performance

The plant response to the Generator Load Reject Without Bypass (LRNBP), Turbine Trip Without Bypass (TTNBP), Feedwater Controller Failure (FWCF) and Main Steam Isolation Valve (MSIV) closure, in the Extended Operating Domain (see Section 15.A.4.2.2.2) and in normal operation, is shown in the Reload Licensing Package (referenced in Reference 8). A limiting Rod Withdrawal Error (RWE) rod pattern is used each cycle for the RWE event.

15.A.5.1.1 Minimum Critical Power Ratio Operating Limit Determination

Section 4.4 describes the various fuel failure mechanisms. An acceptable criterion was determined to be that 99.9% of the fuel rods in the core would not be expected to experience boiling transition (Reference 2). This criterion is met by demonstrating that transients and accidents do not result in a minimum critical power ratio (MCPR) less than the Safety Limit MCPR for LaSalle Units 1 and 2.

15.A.5.1.1.1 GE Analysis

A plant-unique MCPR operating limit (OLMCPR) is established to ensure that the MCPR Safety Limit is not exceeded for the limiting transient events. This operating requirement is obtained by addition of the absolute, maximum ΔCPR value for the most limiting transient from rated conditions and extended operating domain conditions postulated to occur at the plant to the MCPR Safety Limit. The transient analysis results, including the ΔCPR values, are given in Reference 8 for each event. The ΔCPR value for each event, when added to the MCPR Safety Limit, yields the event-based MCPR, except for those events whose ΔCPR is calculated using ODYN. For events whose ΔCPR is determined by ODYN (all rapid pressurization events), the event-based MCPR is determined in conjunction with NRC-additive correction factors, the ΔCPR, and the MCPR Safety Limit. The MCPR limits given in Reference 8 are either the event-based MCPR values, or conservatively bound those values.

The OLMCPR is the maximum locus of values from the event MCPR's calculated with the above method. The limiting transient may vary from cycle to cycle. It is also possible for the limiting transient for a given cycle to vary versus the average scram time. Operation at or above the OLMCPR will assure that the MCPR safety limit is not violated during any postulated transient.

For TRACG analyses, OLMCPR is calculated differently than ODYN analyses. For a description of the TRACG process, see References 19 and 20.

In addition, analyses are performed to allow operation with certain equipment out-of-service (EOOS). Reference 8 provides a comprehensive listing of allowed

combinations of these EOOS. Detailed descriptions of these EOOS are found in the documents referenced in Reference 8, and select EOOS are also described later in this section.

In these modes of operation, thermal limit penalties are required for various EOOS combinations. These more restrictive thermal limits are listed in Reference 8.

Thermal limits, such as the OLMCPR previously described, are adjusted for offrated power and flow conditions within the analyzed power/flow map. This is to provide the additional margin necessary to ensure the fuel safety limits are not violated from a transient initiated at offrated conditions, as such conditions may result in more severe consequences (i.e., delta-CPRs) for some events. Reference 8 provides the power and flow dependent thermal limits.

15.A.5.1.2 TCV(s) Slow Closure

15.A.5.1.2.1 GE Analysis

Analyses referenced in Reference 8 provide for continued operation with one or more Turbine Control Valves (TCV) closing slower than normal due to its corresponding fast acting solenoid not being functional. The analysis assumed a generator load reject event (LRNBP) at rated power, 105% core flow, a TCV slow closure time within the range listed in Table 15.A-1, and failure of the other RPS channel to receive a signal from the TCV Relay Emergency Trip System (RETS). In this condition, only a half scram would occur (from the other RPS channel) and the reactor would then scram due to high neutron flux instead of TCV position. In addition, the EOC-RPT system would not be initiated since it would not receive a RETS signal.

This sequence of events varies at off-rated conditions, and assumes from one to four TCVs closing slow, within the range of slow closure times listed in Table 15.A-1, such that the inputs used are bounding at the applicable power/flow condition. The thermal limits applicable to this condition, and the analysis results, may be found in Reference 8.

15.A.5.1.3 Rod Drop Accident (RDA)

For GE methods, the RDA analysis may be performed generically or cycle specifically. Either analysis provides assurance that the 280 cal/gm enthalpy deposition limit will not be violated. This provides confidence on 95/95 level that the Technical Specification limit will not be violated in the unlikely event of the postulated Design Basis RDA. If the 280 cal/gm design limit is not exceeded, fuel failure should not occur. Consequently, the fission product barrier (fuel cladding) will remain intact. For a more detailed discussion of the RDA see Subsection 15.4.9.

15.A.5.1.4 Vessel Overpressurization Analysis

As a conservative approach to the vessel overpressurization analysis, a main steamline isolation valve (MSIV) closure with flux scram was postulated. Cycle-specific results are available in the Supplemental Reload Licensing Report.

15.A.5.1.4.1 GE Analysis

The cycle-specific overpressurization analysis includes one safety/relief valve out-of-service. See Section 15.A.5.1.6 for further information pertaining to the RVOOS analysis.

15.A.5.1.5 Loss of Coolant Accident (LOCA) Analysis

GE has reanalyzed the LaSalle units with an improved ECCS analysis code package called SAFER/GESTR-LOCA. GESTR-LOCA is a variation of the GESTR-MECHANICAL fuel mechanical design code and is used to calculate the initial fuel assembly stored energy at the beginning of the LOCA event. In the ECCS-LOCA analysis for GNF2 fuel, the SAFER/GESTR-LOCA code has been replaced by the SAFER/PRIME-LOCA code to capture the physical phenomenon of fuel pellet conductivity degradation with pellet exposure. SAFER is the combination of previous GE ECCS codes SAFE and REFLOOD along with some modeling improvement. CHASTE is also used for fuel heatup calculations. For a more detailed discussion of the LOCA analysis see Subsections 6.2.1.1.3, 6.3.3.7, 15.6.5 and G.3.1.2.1 and References 2 and 18.

15.A.5.1.6 Relief Valve Out of Service (RVOOS)

The analysis considers the effects of the relief function of a safety/relief valve out of service on the power dependent MCPR and LHGR limits. The cycle specific ASME overpressurization analyses will explicitly include the safety function of a safety/relief valve out of service. The effect on the LOCA will be addressed in the LOCA analyses. (See section 15.A.5.1.4)

15.A.5.1.6.1 GE Analysis

The analysis (referenced in Reference 8) considers the effects of the relief function of a safety/relief valve out of service on the LOCA, plant transients, and analyzed equipment Out-of-Service. The analysis concludes that one RVOOS has no effect on either the LOCA or transients, or Equipment Out-of-Service for the analyzed plant conditions. Therefore, no MCPR or MAPLHGR penalty is required for operation with one RVOOS. For a detailed description of the analysis results, see Reference 8.

15.A.5.1.7 Feedwater Heaters Out of Service (FWHOOS)

Analyses were performed to justify operation in the MELLLA or ICF Regions with a 100 degree F reduction in feedwater temperature. The FWHOOS analysis supports a contingency operating mode allowing continued operation with reduced feedwater temperature over a full fuel cycle.

Transient events have been analyzed with FWHOOS. With reduced feedwater temperature the LRNBP and TTNBP event will be less severe because of the reduced core steaming rate and lower initial void fraction. The FWCF event typically becomes more severe with FWHOOS. Reference 8 (and its associated references) provides the results of the analyses that support operation with FWHOOS. FWHOOS is allowed in combination with various other EOOS conditions. Reference 8 provides a comprehensive listing of those combinations.

The FWCF event will continue to be analyzed on a cycle specific basis with an assumed 100 degree F feedwater temperature reduction.

15.A.5.1.8 Main Turbine Bypass System Out of Service

The Main Turbine Bypass System is not assumed for the LOCA analyses. Therefore, the system out of service does not affect the analyses or their resulting PCTs.

The analysis provides for continued operation with the entire Main Turbine Bypass System inoperable.

In this situation, operation may continue provided the appropriate thermal limits, determined from Reference 8 are utilized. Additional information may be found in Reference 8.

15.A.5.1.8.1 GE Analysis

The limiting pressurization transient for LaSalle may be the LRNBP or TTNBP. Since the analysis of this transient assumed that the Main Turbine Bypass System is unavailable, the results are applicable whether or not the Main Turbine Bypass System is out-of-service. The FWCF event may also be limiting since the FWCF event normally takes credit for the Main Turbine Bypass System and in this case cannot, it is analyzed assuming the bypass system is inoperable.

The thermal limit penalties to be applied in this OOS condition are given in Reference 8. Results of the analyses for this OOS condition may also be found in Reference 8.

15.A.5.1.9 Recirculation Pump Trip Out of Service

The Recirculation Pump Trip (RPT) system reduces the severity of the turbine trip and load reject transients by running back the recirculation pumps. To support operation with RPT out of service, the limiting transient is analyzed without RPT.

The thermal limit penalties to be applied in this OOS condition are given in Reference 8. Results of the analyses for this OOS condition may also be found in Reference 8.

The RPT is not assumed for the LOCA analyses. Therefore, the system out of service does not affect the analyses or their resulting PCTs.

The analysis provides for continued operation with the RPT system inoperable, provided the appropriate thermal limits are utilized. Additional information may be found in Reference 8.

15.A.5.1.10 Turbine Control Valve Slow Closure in Combination with Feedwater Heaters Out of Service

15.A.5.1.10.1 GE Analysis

GE analyses allow continued plant operation with TCV slow closure in combination with Feedwater Heaters Out of Service (FWHOOS).

15.A.5.1.11 Single Loop Operation

Refer to UFSAR Section 6.B "Recirculation System Single-Loop Operation".

15.A.5.1.12 OPRM Confirmation

OPRM system is provided to detect and suppress thermal-hydraulic instabilities that can occur in BWRs. See Section 15.10 for an analysis of the transient.

15.A.5.1.13 Analyzed Equipment Out-of-Service

Reference 8 contains the allowed modes of operation with combinations of equipment out-of- service. The Base Case mode of operation also includes some of the non-limiting out-of-services as detailed in Reference 8. Reference 8 also provides the thermal limit adjustments required for these modes of operation.

15.A.5.2 Radiological Consequences

There are no radiological consequences for the transient events as no radioactive material is released from the fuel. The calculated exposures for the LOCA analyses are well within the guidelines of 10CFR100.

15.A.6 References

1. General Electric Report NEDE-24011-P-A, "General Electric Standard Application for Reactor fuel (GESTAR II)," (Unit 1: Rev. 22, Unit 2: Rev. 20).

2. NEDC-23785-2-P, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident," Volumes I, II, and III.

3. GE document NEDE-31455, “Extended Operating Domain and Equipment Out of Source for LaSalle County Station Units 1 and 2,” dated March 1990, with addenda.

4. Deleted

5. GE document 6E-NE-187-62-1191, “Equipment Out of Service in the Increased Core Flow Domain for LaSalle County Station Units 1 and 2”, January 1992, Rev. 1.

6. Evaluation of a Postulated Slow Turbine Control Valve Closure Event, LaSalle County Station, Unit 1 and 2, 6E-NE-187-13-0792, February 1993.

7. GE document GENE-637-016-0693, DRF-J11-02096, Analysis of EFPC coast with load following for LaSalle 1 and 2, dated June, 1993.

8. LaSalle Administrative Technical Requirements.

9. General Electric Report NEDC-31531P, “ARTS Improvement Program Analysis for LaSalle County Stations Units 1 and 2," December 1993 and Supplement 1, June 1998, Removal of Direct Scram Bypassed Limits.

10. Deleted

11. “LaSalle Units 1 and 2 Operating Limits with Multiple Equipment Out-of Service (E005)”, NFS letter R. W. Tsai to D. A. Henry (ComEd), April 6, 1995.

12. Deleted

13. Deleted

14. Power Uprate Project Task 900, “Transient Analyses,” GE-NE-A1300384-08 Revision 1, September 1999.

15. Deleted

16. Deleted

17. Deleted

18. Global Nuclear Fuel, "The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance" Technical Bases-NEDC-33256P-A, Qualification-NEDC-33257P-A, and Application Methodology-NEDC-33258P-A, September 2010.

19. "Migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients," NEDE-32906P Supplement 3-A, Revision 1, April 2010.

20. "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," NEDE-32906P-A, Revision 3, September 2006.

TABLE 15.A-1
(SHEET 1 OF 3)

INPUT PARAMETERS AND INITIAL CONDITIONS FOR
ANALYSIS OF TRANSIENTS AND ACCIDENT FOR UNITS 1 AND 2

1. Thermal power level, MWt	
Analysis value	*
2. Steam flow, lb per hr	*
3. Core flow, lb per hr	*
4. Feedwater flow rate, lb per sec	*
5. Feedwater temperature, °F	*
6. Vessel dome pressure, psig	*
7. Vessel core pressure, psig	*
8. Turbine bypass capacity, %NBR	23.2
9. Core coolant inlet enthalpy,	
Btu per lb	*
10. Turbine inlet pressure, psig	*
11. Fuel lattice	10 x 10
12. Core average gap conductance,	
Btu/sec-ft^2 - °F	*
13. Core leakage flow, %	*
14. Required MCPR operating limit	Reference 8
15. MCPR Safety Limit	See appropriate Technical Specification for MCPR Safety Limit
16. Core average rated void fraction, %	*
17. Scram reactivity	Reference 8
18. Rod motion during scram	

Control Fraction (%)	0	5	20	50	90
Scram Time (sec)	0.20	0.490	0.90	2.0	3.5

TABLE 15.A-1
(SHEET 2 OF 3)

19. Jet pump ratios, M	*
20. Safety/Relief valve capacity, pounds per hour at 1150 psig , per SRV	862400.0
Manufacturer	Crosby
Quantity Installed	13 ** (Unit 1) 13 (Unit 2
21. Relief function delay ***, seconds	0.4
22. Relief function response, seconds	0.15
23. Analytical values for safety/ relief valves	
Safety function, psig	1185, 1210, 1221, 1231, 1241
Relief function, psig	1091, 1101, 1111, 1121, 1131
24. Number of valve groupings simulated	
Safety function, No.	5 **
Relief function, No.	5 **
25. High flux trip, % NBR	
Analysis setpoint % NBR	124.2
26. High-pressure scram setpoint, psig	1071
27. Vessel level, inches above steam dryer skirt bottom (instrument zero 527.5)	
Level 8 - (L8) Analytical Value	60.2
Level 4 - (L4) Analytical Value	31.5
Level 3 - (L3) Analytical Value	1.2
Level 2 - (L2) Analytical Valve	-97.9
Level 1 - (L1) Analytical Valve	-161.5

TABLE 15.A-1
(SHEET 3 OF 3)

28. APRM thermal trip	
Setpoint, %NBR	119.50
29. RPT delay, seconds	0.190
30. RPT inertia time constant (t), sec*	6
31. TCV slow closure time (allowed range), seconds from full open	2.70-7.75

* Varies depending on power/flow statepoint of analysis, or assumed reactor pressure.

** A lower, more conservative number may be assumed in some analyses.

*** Excludes pressure sensor delay.

$$*t = \frac{2\pi}{g} \frac{J_o}{T_o} \frac{n}{}$$

where t = inertia time constant

J_o = pump motor inertia,
n = rated pump speed,
g = gravitational constant;
T_o = pump electrical torque

15.B POWER UPRATE RE-ANALYSIS

This section documents the analyses performed to support power uprate to 3489 MWt. This section is retained for its historical information only.

15.B.1. Introduction

15.B.1.1 LaSalle Power Uprate Overview

The LaSalle Power Uprate Project encompasses the evaluations and licensing process for implementation of a 105% core thermal power uprate at LaSalle County Station, Units 1 & 2 (LaSalle). Within the scope of the LaSalle Power Uprate Project, GE Nuclear Energy (GE) performed selected power uprate evaluations to support a ComEd submittal for an amended Operating License (OL). The GE safety evaluations provide the technical justification for reactor operation at 3489 MW_t and 1020 psia reactor dome pressure, consistent with the plant design and licensing basis of record as of January 1, 1999. These evaluations consider the impact of operating the Nuclear Steam Supply System (NSSS), as well as other plant safety-related or important-to-safety systems, in its design configuration and condition at 105% of the current licensed core thermal power level.

All GE evaluations provided in support of the LaSalle submittal for an amended OL, including the evaluation documented in this section, are performed in accordance with the Generic Guidelines for General Electric Boiling Water Reactor Power Uprate, NEDC-31897P-A (Reference 1). Where applicable, GE references the Generic Evaluations of General Electric Boiling Water Reactor Power Uprate, NEDC-31984P and its Supplements (Reference 2), to support conclusions and results for the LaSalle power uprate. The overall conclusions of the GE safety evaluations are provided to ComEd in a power uprate Safety Analysis Report (SAR) for reference in requesting the amended plant OL. The Reference 59 evaluation documented in this transient analysis section provides the basis for the LaSalle power uprate SAR section (Reference 7). The LaSalle 105% Power Uprate SAR is the basis for the NRC review and Safety Evaluation on the acceptability of uprated power operations at LaSalle. Documentation of the Power Uprate analyses and supplemental information are provided in References 1 through 115. Issuance of the power uprate License Amendments 140/125 (Unit 1/Unit 2 respectively) was granted via Reference 116, and the NRC Safety Evaluation (SE) of power uprate is Reference 117. ComEd comments on the NRC SE were provided in Reference 118.

In addition, the LaSalle Power Uprate Project includes a GE assessment of the LaSalle Turbine-Generator (T-G) and Balance of Plant (BOP)systems and components operating at the 105% power uprate condition. Refer to References 53 and 54.

15.B.1.2 Task Overview and Objective

The purpose of the transient analysis task (Reference 59) is to evaluate a representative reactor transient response for LaSalle, consistent with licensed operation up to 3489 MWt (105% of the current licensed core thermal power). Plant specific transient analysis were performed at uprated conditions to show continued compliance with regulatory requirements including:

- Operating Limit Minimum Critical Power Ratio (OLMCPR)
- Thermal/mechanical transient criteria (1% plastic strain and centerline melt)
- ASME Upset Code Limit

The analysis is a re-analysis of the representative cycle, Unit 1, Reload 7 / Cycle 8 Reload Licensing at 105% thermal power uprate.

15.B.1.3 Task Evaluation Summary and Conclusions

15.B.1.3.1 Summary

The 105% thermal power uprate transient analysis showed the expected change in pressurization transient operating limits. The approximate change in OLMCPR was a 0.01 increase compared to the OLMCPR obtained in the Unit 1 Reload 7 / Cycle 8 transient analysis. Margin has been maintained to the ASME upset code limit.

15.B.1.3.2 Statement of Impact on Plant Configuration

The results of this transient analysis evaluation confirm that 105% thermal power uprate has no impact on the LaSalle plant configuration.

15.B.1.3.3 Statement of Impact on Design Operating Margins

Minimum critical power ratio (MCPR) and linear heat generation rate (LHGR) operating margins are reduced at power uprate as a result of the decrease in steady state operating MCPR and the increase in steady state operating LHGR. The operating limit MCPR (OLMCPR) actually increased slightly with power uprate. The LaSalle reactor operating domain for 105% thermal power uprate with the incorporation of MELLLA also reduces the LaSalle design operating margin. The LHGR limits are impacted by Loss of Feedwater Heating (LFWH). The impact of the LFWH of 145 °F on the rated LHGRs and the power dependent MAPLHGR factors, MAPFAC(p) for uprate showed a significant decrease in operating margin. Some of this margin could be regained by crediting the simulated thermal power

scram setpoint. The cycle specific reload licensing would determine the actual LHGR limit requirement to meet the fuel thermal limits.

The effect of the reactor operating domain, including MELLLA, on the core and other LaSalle Systems, Structures, and Components (SSCs) and design features are evaluated in the associated referenced Task Reports.

15.B.1.3.4 Implementation Requirements

The implementation of power uprate requires cycle specific transient analysis calculations and a Core Operating Limits Report (COLR) modification to incorporate the power uprate operating limits. The power uprate is accomplished by operating along extensions of rod lines on the power/flow map with no increase in maximum core flow. The cycle-specific core reload analyses will be performed with the most conservative core flow. Operating limits are established to ensure that regulatory requirements and safety limits are not exceeded for a range of postulated events as is currently the practice. The operating limit and safety limit Minimum Critical Power Ratio (MCPR) as well as the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Linear Heat Generation Rate (LHGR) limits are cycle dependent and as such will be established or confirmed at each reload.

The ability of the SLCS boron solution to achieve and maintain safe shutdown is not a direct function of core thermal power, and therefore, is not affected by the power uprate. SLCS shutdown capability is re-evaluated for each reload core.

15.B.2. Evaluation

15.B.2.1 Scope

The transient analysis task evaluates the applicable limiting transient events at the power uprate conditions. Representative plant specific transient analyses were performed to demonstrate compliance with the fuel thermal margin requirements and the ASME reactor vessel overpressure protection criteria.

15.B.2.2 Method of Evaluation

The transient events are grouped in various categories. The events in the increase in heat removal by the secondary system include loss of feedwater heater, feedwater controller failure maximum demand, pressure regulator failure in the open position, and inadvertent RHR shutdown cooling operation. Per Reference 1, the bounding events evaluated for the power uprate are the loss of feedwater heaters and the feedwater control failure events.

The events listed under the decrease in heat removal by the secondary system include pressure regulator failure closed, generator load rejection with and without operational turbine bypass, turbine trip with and without operational turbine bypass, inadvertent MSIV closure, loss of condenser vacuum, loss of A-C power, loss of feedwater flow, and failure of the RHR shutdown cooling system. The limiting events in terms of minimum critical power ratio are generator load rejection without operational bypass and turbine trip without operational bypass events. The limiting over-pressurization event is MSIV closure with flux scram. The limiting event in terms of a loss of water level is a loss of feedwater flow. The four limiting transients are evaluated for the power uprate.

The events listed as a decrease in reactor coolant system flow rate include recirculation pump trip, recirculation flow control failure with decreasing flow, recirculation pump seizure, and recirculation pump shaft break. Per reference 1, these events are not limiting for any BWR and are not evaluated for reload cores and not included in the power uprate licensing report. These events are bounded by the events listed under the decrease in heat removal by the secondary system and are therefore not analyzed.

The transient events listed as reactivity and power distribution anomalies include control rod withdrawal errors at low power, rod withdrawal error at full power, control rod mis-operation, abnormal startup of the idle recirculation pump, recirculation flow control failure with increasing flow, misplaced fuel assembly and subsequent operation, and rod drop accident. The rod drop accident and rod withdrawal error at low power are evaluated in a separate task report. The rod withdrawal error at full power is considered the bounding event for this category of transients. The remaining events are bounded by the rod withdrawal error at full power and not evaluated at power uprate. The recirculation flow control failure with increasing flow is bounded by the slow flow run-up event, which is the basis for the MCPR(f) limits. These limits were determined in Reference 3 and are validated for power uprate and MELLLA.

The increase in reactor coolant inventory transient is bounded by the feedwater controller failure (maximum demand) event. This event is evaluated for the power uprate.

The decrease in reactor coolant inventory transient in the UFSAR is an inadvertent safety/relief valve opening which is not a limiting event and is not analyzed for power uprate conditions. The other UFSAR events listed for decrease in reactor coolant inventory are loss of coolant accident, which are addressed in another power uprate task report.

The engineering computer program ODYN which simulates the dynamic behavior of a boiling water reactor (BWR) was used to analyze the transients. ISCOR and

TASC computer codes were used for the ΔCPR analysis. ISCOR performs steady state thermal-hydraulic core calculations, and TASC performs single channel transient calculations.

15.B.2.3 Inputs and Assumptions

15.B.2.3.1 Transient Conditions

The reactor transient analyses were performed at the uprate power conditions corresponding to 105% of the current licensed core thermal power or 3489 MW_t. The transient power and flow conditions considered are shown in Table 15.B-1. The standard operating domain is defined as 100% uprated power and 100% rated core flow or 100P/100F. The increase core flow (ICF) domain is defined as 100P/105F. The MELLLA domain is defined as 100P/81F. The MSIVF over-pressurization event was evaluated at 102P/105F and initial dome pressure of 1020 psig to ensure that LaSalle meets ASME criteria under a worst-case pressurization event. The transient analysis results are valid up to a 13 °F reduction in feedwater temperature from nominal. Certain equipment out of service and operational flexibility options, consistent with current license operations, were evaluated as outlined in Tables 15.B-1, 15.B-3, and 15.B-4.

The off-rated limits were reviewed to verify that the ARTS power and flow dependent limits established in References 3 and 5 remain applicable. Additional off-rated limits were determined for the turbine control valve (TCV) slow closure equipment out of service in Reference 6. These limits were also reviewed for continued application at power uprate and MELLLA conditions. Events were evaluated at End of Cycle (EOC) exposure which is the most limiting condition for operating MCPR limits for LaSalle. This is consistent with the Reference 4 analysis. The EOC scram characteristics occur when all of the control rods are fully withdrawn at the beginning of the transient.

15.B.2.3.2 Transient Input Parameters

The plant operating features used in the LaSalle transient analysis are those used in Reference 4. Table 15.B-2 lists the important input parameters used in the analysis. Comparable values from LaSalle Unit 1 Cycle 8 reload licensing calculations and from the UFSAR are also presented.

The transients are evaluated with one SRV which has the lowest pressure setpoint out-of-service. There are 12 SRVs available.

The load rejection without bypass transient (LRNBP) and turbine trip without bypass transient (TTNBP) are evaluated assuming normal feedwater temperature at power uprate conditions. The feedwater controller failure transient (FWCF) is

evaluated at a 100 °F feedwater temperature reduction (FWTR) to justify operation at reduced feedwater temperature at uprated power (3489 MWt).

MCPR values were calculated with OPTION B ΔCPR adjustment factors which take credit for conservatism in the scram speed (Reference 4).

15.B.3. Results

15.B.3.1 Evaluation Results

The transient analysis (Reference 59) includes a broad set of transients. The most limiting events were evaluated at the power uprate conditions and are summarized in Table 15.B-3. The peak neutron flux and the peak heat flux are listed for each transient evaluated. The vessel experiences a peak pressure of 1332 psig during the MSIVF event with 1 dual mode safety relief valve out of service (1SRVOOS). This is 36 psid higher than the peak vessel pressure of 1296 psig for the same event at the current licensed core thermal power with 1SRVOOS (Reference 4). The result is within the acceptable limits of the ASME vessel overpressure limits during an MSIVF event with up to 1SRVOOS.

The FWCF event at the power uprate conditions was evaluated based on the ODYN engineering computer program. The highest uncorrected ΔCPR for a FWCF is unchanged with power uprate at 0.18. Figure 15.B-4 shows the response of key system parameters for a FWCF. The fuel thermal and mechanical overpower results for a FWCF listed in Table 15.B-4 are within the acceptable limits with power uprate. The highest uncorrected ΔCPR for a LRNBP slightly increases with power uprate from 0.20 to 0.21. Figure 15.B-3 shows the response of key system parameters for a LRNBP. The fuel thermal and mechanical overpower results for a LRNBP listed in Table 15.B-4 are within the acceptable limits with power uprate. The TTNBP transient was analyzed at uprated power to ensure that it remains bounded by the LRNBP event. This is evident in Table 15.B-3 where the ΔCPR results for TTNBP are lower than the ΔCPR results for LRNBP with power uprate.

The rod withdrawal error (RWE) analysis at full power considers the limiting core conditions in response to the rod block monitor (RBM) system. The limiting core condition depends on the core and fuel design and the planned control rod positions during the cycle. The severity of the RWE is a function of the power distribution in the core. The analysis with power uprate conditions did not result in a more severe RWE transient since the same power distributions were attainable at either rated or uprated power levels. The resulting ΔCPR values for the RWE analysis with power uprate conditions changed less than 0.01 compared to the ΔCPR values determined by ComEd during the cycle 8 reload licensing analysis.

The LFWH was evaluated to confirm that it remains non-limiting at uprated conditions. The LFWH event at the power uprate conditions was evaluated based on the BWR simulator and not evaluated based on the REDY model. The uncorrected ΔCPR of 0.17 listed in Table 15.B-3 confirms it is far from limiting. The fuel thermal overpower result for a LFWH listed in Table 15.B-4 is above the acceptable limit with power uprate. This will impact the LHGR limits. The impact of the LFWH of 145 °F on the rated LHGR's and the power dependent MAPLHGR factors MAPFAC(p) for power uprate is shown in Table 15.B-5.

The ARTS basis for LaSalle as described in Reference 5 for MCPR(p), MCPR(f), and MAPFAC(f) are still applicable for power uprate and MELLLA. The results for MAPFAC(p) need to be adjusted from Reference 5 due to the worst LFWH at 145 °F. The power dependent MAPFAC(p) curve without the thermal power scram (TPS) credit is given in the top part of Table 15.B-5. If credit is taken for TPS then the power dependent MAPFAC(p) curve is given in the bottom part of Table 15.B-5. These adjustments will be determined on a cycle specific basis if the LFWH does not meet the TOP acceptance criteria.

The Loss of Feedwater Flow (LOFW) transient has not been explicitly analyzed because a generic evaluation is contained in Section 3.1 of Reference 2. During a LOFW transient and assuming an additional single failure (loss of RCIC or HPCS), reactor water level is automatically maintained above the top of the active fuel (TAF) by the RCIC or the HPCS system without any operator action required. Because of the extra decay heat from power uprate, slightly more time is required for the automatic systems to restore water level. Operator action is only needed for long-term plant shutdown once water level is restored (control of water level, reduction of pressure and initiation of RHR shutdown cooling). These sequences of events do not require any new operator actions or shorter operator response times. Therefore, the operator actions for a LOFW transient do not significantly change for power uprate.

15.B.3.2 Key Parameter Comparison

The key parameters of ΔCPR and peak vessel pressure are compared for the current licensed core thermal power and power uprate in the table below.

Parameter	Units	Current Rated Power	Power Uprate
Limiting Non-Pressurization Transient (RWE) ΔCPR	--	0.28	0.28
Limiting Pressurization Transient (LRNBP) ΔCPR	--	0.20	0.21
Limiting Transient Peak Vessel Pressure (MSIVF)	psig	1296	1332

15.B.3.3 Evaluation Conclusions

Bounding core wide transient evaluations were performed in Reference 59 for the power uprate condition based on the LaSalle County Station Unit 1 Cycle 8 core configuration. Postulated transient events involving malfunctions, failures of various equipment or a single operator error were evaluated to determine the effects of power uprate on plant safety performance response. The results of the core wide transient analyses for LaSalle at 105% of current licensed core thermal power show the margin of safety for anticipated operational occurrences remain within the applicable criteria and the maximum reactor vessel pressure does not exceed the ASME code allowable peak pressure for upset category events of 1375 psig.

The analysis shows the overall capability of LaSalle to meet all transient safety criteria for power uprate operation. The RWE remains the limiting event for the reload licensing basis events. The LRNBP event remains the most limiting in the pressurization event category for the reload licensing basis events. The over-pressurization event, MSIVF, yields a peak vessel bottom pressure in compliance with the ASME upset code limit of 1375 psig for vessel overpressure protection. The MAPFAC(P) curve was modified to assure fuel thermal and mechanical integrity at off-rated power conditions.

15.B.4. References

1. Licensing Topical Report, "Generic Guidelines for General Electric BWR Generic Power Uprate", NEDC-31897P-A, Class III, May 1992 (LTR-1).

2. Licensing Topical Report, "Generic Evaluations of General Electric BWR Power Uprate", 31984P, Class III, July 1991 (LTR-2), and Supplements 1 and 2.

3. "ARTS Improvement Program Analysis for LaSalle County Station Units 1 and 2", NEDC-31531, December 1993.

4. "Supplemental Reload Licensing Report for LaSalle County Station Unit 1, Reload 7, Cycle 8", 24A5180, Revision 1, Class 1, May 1998.

5. "ARTS Improvement Program Analysis LaSalle County Nuclear Station, Units 1 and 2 Removal of Direct Scram Bypassed Limits", NEDC-31351P, Revision 1, Supplement 1, DRF L12-00850-00, Class 3, September 1998.

6. "Evaluation of a Postulated Slow Turbine Control Valve Closure Event for LaSalle County Station Units 1 and 2", GE-NE-187-13-0792, Revision 2, July 1998.

7. Letter from R. M. Krich (Commonwealth Edison Company) to the USNRC dated July 14, 1999, "Request for License Amendment for Power Uprate Operation" with Attachment E: General Electric Nuclear Energy, Licensing Topical Report NEDC-32701P, Revision 2, "Power Uprate Safety Analysis Report for LaSalle County Station, Units 1 and 2," dated July 1999 (Proprietary).

8. Power Uprate Project Task 100 / 101, "Power Uprate Nominal Heat Balance / Power Uprate Off-rated Heat Balance," GE-NE-A1300384-01, Revision 0, July 1999.

9. Power Uprate Project Task 200, "Core and Fuel Design," GE-NE-A1300384-09-02, Revision 1, September 1999.

10. Power Uprate Project Task 201, "Reactor Power/Flow Map," GE-NE-A1300384-07-01, Revision 1, September 1999.

11. Power Uprate Project Task 202, "Thermal Hydraulic Stability," GE-NE-A1300384-13-01, Revision 0, August 1999.

12. Power Uprate Project Task 203, "Control Rod Drive System," GE-NE-A1300384-05-01, Revision 0, August 1999.

13. Power Uprate Project Task 300, "Nuclear Boiler System," GE-NE-A1300384-28-01, Revision 0, October 1999; including Report, "Nuclear Boiler System Overpressure Protection," 22A4342, Rev. 5 / DC22A4342, Rev. 4.

14. Power Uprate Project Task 301, "RPV Fracture Toughness Evaluation," GE-NE-A1300384-23-01, Revision 0, May 1999.

15. Power Uprate Project Task 302, "Reactor Pressure Vessel Stress Report Reconciliation," GE-NE-A1300384-37, Revision 0, September 1999.

16. Power Uprate Project Task 303, " Reactor Internals," GE-NE-A1300384-32-01, Revision 0, October 1999.

17. Power Uprate Project Task 304, "Reactor Internal Pressure Differences," GE-NE-A1300384-41-01, Revision 0, July 1999.

18. Power Uprate Project Task 305i, "Reactor Vibration Evaluation - Internal," GE-NE-A1300384-46-01, Revision 0, September 1999.

19. Power Uprate Project Task 305e, "Reactor Vibration Evaluation - External," GE-NE-A1300384-46-02, Revision 0, September 1999.

20. Power Uprate Project Task 306, "Steam Dryer/Separator Performance," GE-NE-A1300384-40-01, Revision 0, July 1999.

21. Power Uprate Project Task 307, "Reactor Recirculation System," GE-NE-A1300384-38-01, Revision 1, September 1999.

22. Power Uprate Project Task 308, "Reactor Coolant Pressure Boundary Piping," GE-NE-A2200092-00-01, Revision 2, January 2000.

23. Power Uprate Project Task 309, "Reactor Core Isolation Cooling System," GE-NE-A1300384-26-01 Revision 1, August 1999.

24. Power Uprate Project Task 310, "Residual Heat Removal System," GE-NE-A1300384-12-01, Revision 0, October 1999.

25. Power Uprate Project Task 311, "Reactor Water Cleanup System," GE-NE-A1300384-33-01 Rev. 1, Revision 1, June 1999.

26. Power Uprate Project Task 312, "BOP Piping - Large (>4") Bore," GE-NE-A2200092-00-02, Revision 1, January 2000.

27. Power Uprate Project Task 400, "Containment System Response," GE-NE-A1300384-02-01 Rev. 1, Revision 1, October 1999.

28. Power Uprate Project Task 401, "Subcompartment (Annulus) Pressurization Loads," GE-NE-A1300384-06-01, Revision 0, June 1999.

29. Power Uprate Project Task 403, "MOV Disposition," GE-NE-A1300384-17-01 Rev. 1, Revision 2, April 2000.

30. Power Uprate Project Task 404, "High Pressure Core Spray System," GE-NE-A1300384-11-01, Revision 0, September 1999.

31. Power Uprate Project Task 405, "Low Pressure Core Spray System," GE-NE-A1300384-10-01, Revision 0, September 1999.

32. Power Uprate Project Task 407, "ECCS Performance," GE-NE-A1300384-39-01 Revision 1, September 1999.

33. Power Uprate Project Task 409, "Standby Gas Treatment System," GE-NE-A1300384-20-01, Revision 0, May 1999.

34. Power Uprate Project Task 411, "Post-LOCA Combustible Gas Control," GE-NE-A1300384-19-01, Revision 0, June 1999.

35. Power Uprate Project Task 500, "Neutron Monitoring System w/RBM," GE-NE-A1300384-22-01, Revision 0, September 1999.

36. Power Uprate Project Task 501, "RWM System Evaluation," GE-NE-A1300384-42-01, Revision 0, August 1999.

37. Power Uprate Project Task 504, "Feedwater Control System," GE-NE-A1300384-31-01, Revision 0, October 1999.

38. Power Uprate Project Task 506, "NSSS Trip Instrument Setpoint Assessment," GE-NE-A1300384-44-01, Revision 0, October 1999.

39. Power Uprate Project Task 507, "Non-power Cycle Instrument Setpoints," GE-NE-A1300384-60-01, Revision 0, October 1999.

40. Power Uprate Project Task 600, "Off-site Power System/Grid Stability," GE-NE-A1300384-57-01, Revision 0, October 1999.

41. Power Uprate Project Task 601, "On-site AC Electrical Power," GE-NE-A1300384-47-01, Revision 0, August 1999.

42. Power Uprate Project Task 602, "DC Power," GE-NE-A1300384-52-01, Revision 0, September 1999.

43. Power Uprate Project Task 603, "Fuel Pool Cooling and Cleanup System," GE-NE-A1300384-35-02, Revision 1, December 1999.

44. Power Uprate Project Task 604, "Station Service Water System," GE-NE-A1300384-53-01, Revision 1, October 1999.

45. Power Uprate Project Task 605, "Circulating Water System," GE-NE-A1300384-48-01, Revision 0, August 1999.

46. Power Uprate Project Task 606, "Reactor Building Closed Cooling Water System," GE-NE-A1300384-54-01, Revision 0, October 1999.

47. Power Uprate Project Task 607, "Turbine Building Closed Cooling Water System," GE-NE-A1300384-45-01, Revision 0, October 1999.

48. Power Uprate Project Task 609, "Standby Liquid Control System," GE-NE-A1300384-27-01, Revision 0, August 1999.

49. Power Uprate Project Task 610, "Power-Dependent HVAC," GE-NE-A1300384-55-01, Revision 0, August 1999.

50. Power Uprate Project Task 611, "Appendix R Fire Protection," GE-NE-A1300384-43-01, Revision 1, September 1999.

51. Power Uprate Project Task 612, "Systems not Impacted," GE-NE-A1300384-04-01, Revision 0, August 1999.

52. Power Uprate Project Task 613, "Core Standby Cooling System-Equipment Cooling Water System," GE-NE-A1300384-56-01, Revision 0, October 1999.

53. Power Uprate Project Task 700, "Turbine-Generator Final Report 5% Core Thermal Power Uprate," GE ICN A91186Q/D20933Q, Revision 1, June 1999; including GE-LPUP-310, "Errata for "T-700 Report," DRF A13-00384-00, dated October 28, 1999.

54. Power Uprate Project Task 701, "BOP Power Uprate Evaluation Report," NEDC 32622P, DRF-A13-00384-03, Revision 1, November 1999.

55. Power Uprate Project Task 702, "Condensate Polishing System," GE-NE-A1300384-59-01, Revision 0, August 1999.

56. Power Uprate Project Task 800, "Liquid and Solid Waste Management," GE-NE-A1300384-34-02, Revision 0, October 1999.

57. Power Uprate Project Task 801, "Gaseous Waste Management," GE-NE-A1300384-18-01, Revision 0, May 1999.

58. Power Uprate Project Task 802, "Radiological Source Terms," GE-NE-A1300384-21-01, Revision 0, September 1999.

59. Power Uprate Project Task 900, "Transient Analyses," GE-NE-A1300384-08 Revision 1, September 1999.

60. Power Uprate Project Task 901, "Accident Radiological Analyses," GE-NE-A1300384-49-01, Revision 0, October 1999.

61. Power Uprate Project Task 902, "Anticipated Transient Without Scram," GE-NE-A1300384-25-01, Revision 0, October 1999.

62. Power Uprate Project Task 904, "Accident Analyses (non-LOCA)," GE-NE-A1300384-36-01, Revision 0, August 1999.

63. Power Uprate Project Task 1000, "High Energy Line Break Input," GE-NE-A1300384-50-01 Revision 1, September 1999.

64. Power Uprate Project Task 1001, "Power Uprate Recirculation Pipe Whip Restraint Impact," GE-NE-A1300384-51-01, Revision 0, September 1999.

65. Power Uprate Project Task 1004, "Environmental Qualification (EQ)," GE-NE-A1300384-58-01, Revision 0, October 1999.

66. Power Uprate Project Task 1005, "Startup Test Recommendations," GE-NE-A1300384-29, Revision 0, October 1999.

67. Power Uprate Project Task 1008, "Plant Life Effects," GE-NE-A1300384-24-01, Revision 0, September 1999.

68. ENDIT 1251, S&L Task 8f, NSSS-RHR.

69. ENDIT H043, S&L Task 8j, NSSS-FW Control.

70. ENDIT 1249, S&L Task 10a, Power Conversion-MS.

71. ENDIT 1253, S&L Task 10b, Power Conversion-FW.

72. ENDIT H033, S&L Task 10f, Cooling Lake and Ultimate Heat Sink.

73. ENDIT 1256, S&L Task 10k, Power Conversion-Containment Isolation.

74. ENDIT H039, S&L Task 11b, ESF-SGTS.

75. ENDIT 1254, S&L Task 11h, ESF-ADHR System.

76. ENDIT H038, S&L Task 11i, ESF-MSIVLCS & PVLCS.

77. ENDIT H044, S&L Task 11j, ESF-Leak Detection.

78. ENDIT H046, S&L Task 13a, HVAC Systems - Non-Safety Related.

79. ENDIT H045, S&L Task 13a, HVAC Systems - Safety Related.

80. ENDIT 1248, S&L Task 14a, Radwaste Systems.

81. ENDIT H047, S&L Task 15a, I & C.

82. ENDIT H065, S&L Task 16, Process Computer

83. ENDIT H066, S&L Task 17, Plant Simulator

84. ENDIT H034, S&L Task 20a, Structural-Containment.

85. ENDIT H034, S&L Task 20b, Structural-Drywell.

86. ENDIT H034, S&L Task 20c, Structural-Primary Shield Wall (Bioshield).

87. ENDIT H034, S&L Task 20d, Structural-Auxiliary Building.

88. ENDIT H034, S&L Task 20e, Structural-Fuel Building.

89. ENDIT H034, S&L Task 20f, Structural-Turbine Building.

90. ENDIT H030, S&L Task 21a, Radiological-Source Terms.

91. ENDIT H041, S&L Task 21b, Radiological-Chapter 15 Accidents.

92. ENDIT H032, S&L Task 21c, Radiological-Normal Doses & Shielding.

93. ENDIT H029, S&L Task 21d, Radiological-Vital Areas.

94. ENDIT H040, S&L Task 21e, Radiological-PASS.

95. ENDIT H037, S&L Task 21f, Radiological-Appendix I.

96. ENDIT H042, S&L Task 21g, Radiological-CR & TSC Habitability.

97. ENDIT H036, S&L Task 21h, Radiological-Equipment Qualification.

98. ENDIT H026, S&L Task 22a, Programs-High Energy Line Break (HELB).

99. ENDIT 1255, S&L Task 22b, Programs-Station Black Out (SBO).

100. ENDIT H035, S&L Task 22c, Programs-Appendix R.

101. ENDIT H071, S&L Task 22e, Programs-ISI/IST

102. ENDIT 1252, S&L Task 22f, Programs-Appendix J.

103. ENDIT H028, S&L Task 23b, EQ Packages.

104. ENDIT H027, S&L Task 24, Environmental Impact.

105. Letter from Jeffrey A. Benjamin (ComEd) to United States Nuclear Regulatory Commission Document Control Desk, dated January 21, 2000, "Response to Request for Additional Information License Amendment Request for Power Uprate Operation" 00-002.

106. Letter from Jeffrey A. Benjamin (ComEd) to United States Nuclear Regulatory Commission Document Control Desk dated February 15, 2000, "Response to Request for Additional Information License Amendment Request for Power Uprate Operation" 00-006.

107. Letter from Jeffrey A. Benjamin (ComEd) to United States Nuclear Regulatory Commission Document Control Desk, dated February 23, 2000, "Response to Request for Additional Information License Amendment Request for Power Uprate Operation" 00-008.

108. Letter from Charles G. Pardee (ComEd) to United States Nuclear Regulatory Commission Document Control Desk, dated March 10, 2000, "Response to Request for Additional Information License Amendment Request for Power Uprate Operation" 00-040.

109. Letter from Charles G. Pardee (ComEd) to United States Nuclear Regulatory Commission Document Control Desk, dated March 24, 2000, "Response to Request for Additional Information License Amendment Request for Power Uprate Operation" 00-057.

110. Letter from Charles G. Pardee (ComEd) to United States Nuclear Regulatory Commission Document Control Desk, dated March 24, 2000, "Response to Request for Additional Information License Amendment Request for Power Uprate Operation" 00-058.

111. Letter from Charles G. Pardee (ComEd) to United States Nuclear Regulatory Commission Document Control Desk, dated March 31, 2000, "Response to Request for Additional Information License Amendment Request for Power Uprate Operation" 00-063.

112. Letter from Charles G. Pardee (ComEd) to United States Nuclear Regulatory Commission Document Control Desk, dated March 31, 2000, "Response to Request for Additional Information License Amendment Request for Power Uprate Operation" 00-064.

113. Letter from Charles G. Pardee (ComEd) to United States Nuclear Regulatory Commission Document Control Desk, dated April 7, 2000, "Response to Request for Additional Information License Amendment Request for Power Uprate Operation" 00-069.

114. Letter from C. G. Pardee, Commonwealth Edison (ComEd) Company, United States Nuclear Regulatory Commission Document Control Desk, dated April 14, 2000,"Supplement to the License Amendment Request for Power Uprate Operation" 00-119.

115. ERIN PRA/PSA Engineering Report, "Identification of Risk Implications Due to 5% Power Uprate at LaSalle," Revision 0 dated April 5, 2000.

116. Letter from D. M. Skay, U.S. NRC, to O. D. Kingsley, Commonwealth Edison (ComEd) Company, "LaSalle - Issuance Of Amendments Regarding Power Uprate (TAC Nos. MA6070 AND MA6071)," dated May 9, 2000 (OL Amendment: TS 140/125).

117. "Safety Evaluation By The Office Of Nuclear Reactor Regulation Related To Amendment No. 140 To Facility Operating License No. NPF-11 And Amendment No. 125 To Facility Operating License No. NPF-18; Commonwealth Edison Company LaSalle County Station, Units 1 And 2; Docket Nos. 50-373 and 50-374" dated May 9, 2000 (NRC SE).

118. Letter from C. G. Pardee, Commonwealth Edison (ComEd) Company, United States Nuclear Regulatory Commission Document Control Desk, dated July 12, 2000, "NRC Safety Evaluation for LaSalle County Station Unit 1 License Amendment 140 and Unit 2 License Amendment 125" 00-217.

119. Design Analysis L-003505, Revision 0, “Disposition of Events Summary for the LaSalle MUR Power Uprate,” July 2010.

TABLE 15.B-1

TRANSIENT ANALYSIS POWER / FLOW STATE POINTS AT POWER UPRATE

Transient Event (100P =3489 MWt)	100P/105F Nom FWT	100P/105 FFWTR =100F[1]	100P/81F Nom FWT	102P/105F Nom FWT[2]
LRNBP	X		X	
TTNBP	X		X	
FWCF	X	X	X	
LRNBP w/o RPT	X			
TTNBP w/o RPT	X			
FWCF w/o TBP	X	X		
MSIVF				X
LRNBP w/o RPT[3]	X			
Loss of Feedwater Heater[4]			X	

Source of information: Reference 59.

Notes:

1. Initial feedwater temperature of 326.5 °F assumes Feedwater Temperature Reduction of 100 °F from the normal feedwater temperature at 3489 MWt.

2. Overpressurization event, analyzed at a dome pressure of 1035 psia and a 2% overpower for initial power. The nominal feedwater temperature for 102% power is 428.5 °F.

3. Turbine Control Valve slow closure, only TCV #1 closes at 50% per second.

4. 145 °F worst loss of feedwater temperature.

TABLE 15.B-2

TRANSIENT ANALYSIS INITIAL PLANT CONDITIONS

Parameter	FSAR Basis *	Cycle 8 Basis	Power Uprate Basis
Rated Thermal Power (MWt)	3296	3323	3489
Analysis Power (% Rated)	104.8	100	100
Rated Core Flow (Mlb/hr)	108.4	108.5	108.5
Rated Power Core Flow Range (% of rated)	—	87 - 105	81 - 105
Rated Vessel Steam Flow and FW flow (Mlb/hr)	14.24	14.30	15.14
Analysis Steam Flow (% rated steam flow)	104	100	100
Analysis Dome Pressure (psig)	1020	986.3 **	986.3 **
Analysis Turbine Pressure (psig)	962	933.1	926.3
Normal Feedwater Temperature (°F)	420	420	426.5
Feedwater Temperature Reduction, (°F)		100	100
Steam Bypass Capacity (% rated steamflow)	25	25	23.6
Number of SRVs assumed in Analysis	18	17	12
SV Setpoint (# of valves @ psig) Analytical Limit	2 @ 1150 4 @ 1175 4 @ 1185 4 @ 1195 4 @ 1205	2 @ 1185 *** 4 @ 1210 4 @ 1221 4 @ 1231 4 @ 1241	1 @ 1210 *** 4 @ 1221 4 @ 1231 4 @ 1241
RV Setpoint (# of valves @ psig) Analytical Limit	2 @1076 4 @1086 4 @1096 4 @1106 4 @1116	2 @ 1091 *** 4 @ 1101 4 @ 1111 4 @ 1121 4 @ 1131	1 @ 1101 *** 4 @ 1111 4 @ 1121 4 @ 1131
MCPR Safety Limit	1.06	1.07	1.07
CRD Speed	Figure 15.0-2	67B	67B

Source of information: Reference 59.

* These values are for the initial core analysis.

** For MCPR calculations a conservatively lower operating pressure is assumed in the analysis. For the overpressure analysis the technical specification maximum dome pressure of 1020 psig is used in the analysis.

*** The lowest setpoint valve is assumed Out-of-Service.

TABLE 15.B-3

LASALLE POWER UPRATE TRANSIENT ANALYSIS RESULTS

Transient (a)	Initial Power /Flow (b)	Peak Neutron Flux (%NBR)	Peak Heat Flux (%NBR)	Peak Steamline Pressure (psig)	Peak Vessel Pressure(psig)	ΔCPR GE8x8NB (c)	Notes**
Equipment In Service		{ } = Cycle 8 Results					
LRNBP	100P/105F	487 {491}	119 {119}	1171 {1152}	1206 {1190}	0.21 {0.20}	d d (Ref 4)
TTNBP	100P/105F	474	118	1170	1205	0.19	d
FWCF	100P/105F	375	118	1137	1172	0.17	d
FWCF	100P/105F	350 {345}	121 {121}	1129 {1116}	1156 {1145}	0.18 {0.18}	d,g d,g (Ref 4)
MSIVF	102P/105F	493 {467}	130 {130}	1303 {1260}	1332 {1296}	—	d d (Ref 4)
LRNBP	100P/81F	338	116	1174	1202	0.15	d
TTNBP	100P/81F	307	114	1173	1201	0.14	d
FWCF	100P/81F	253	113	1139	1167	0.12	d
LFWH	100P/81F	—	—	—	—	0.17	
Equipment out of Service		{ } = Cycle 8 Results					
LRNBP	100P/105F	592 {595}	124 {124}	1174 {1154}	1216 {1200}	0.24 {0.24}	d,e d,e (Ref 4)
TTNBP	100P/105F	602	123	1172	1215	0.23	d,e
FWCF	100P/105F	474	126	1164	1200	0.23	d,f,g
LRNBP	100P/105F	426	125	1192	1225	0.26	d,e,h
FWCF	100P/105F	523	124	1170	1204	0.23	d,f

Source of information: Reference 59.

FOOTNOTES for Table 15.B-3

(a) LRNBP = Generator Load Rejection with Bypass Failure
FWCF = Feedwater Controller Failure (to maximum demand)
TTNBP= Turbine Trip with Bypass Failure
MSIVF = Main Steam Isolation Valve Closure, Flux Scram
HPCS = Inadvertent Actuation of High Pressure Core Spray
LFWH = Loss of Feedwater Heaters (145 deg F FWTR)

(b) 100P = uprate power of 3489 MWt
100F = rated core flow of 108.5Mlb/hr
105F = ICF flow point at uprated power
81F = MELLLA flow point at uprated power (87.9 Mlb/hr)

(c) ΔCPR based on initial CPR which yields MCPR = 1.07

** Notes: d = 1SRVOOS
e = RPTOOS
f = TBSOOS
g = FFWTR
h = TCV slow closure

TABLE 15.B-4

LASALLE POWER UPRATE TOP/MOP RESULTS

Transient (a)	Initial Power/Flow (b)	TOP (%)	TOP Design limit (%)	MOP (%)	MOP design limit (%)	Notes**
Equipment In Service			{} = Cycle 8 Results			
LRNBP	100P/105F	24.9 {24.9}	38.0 {38.0}	25.3 {25.2}	38.0 {38.0}	d d (Ref 4)
TTNBP	100P/105F	23.8	38.0	24.2	38.0	d
FWCF	100P/105F	20.2	37.0	20.9	39.0	d
FWCF	100P/105F	21.8 {21.9}	37.0 {37.0}	23.7 {23.6}	39.0 {39.0}	d,g d,g (Ref 4)
LRNBP	100P/81F	23.0	38.0	23.4	38.0	d
TTNBP	100P/81F	22.6	38.0	22.9	38.0	d
FWCF	100P/81F	17.8	37.0	18.2	39.0	d
LFWH	100P/81F	37.3	25.0	37.3	45.0	
Equipment Out of Service			{} = Cycle 8 Results			
LRNBP	100P/105F	30.1 {30.3}	38.0 {38.0}	30.6 {30.6}	38.0 {38.0}	d,e d,e (Ref 4)
TTNBP	100P/105F	29.9	38.0	30.4	38.0	d,e
FWCF	100P/105F	28.6	37.0	30.0	39.0	d,f,g
LRNBP	100P/105F	30.7	38.0	30.9	38.0	d,e,h
FWCF	100P/105F	27.5	37.0	28.4	39.0	d,f

Source of information: Reference 59.

FOOTNOTES for Table 15.B-4

(a) LRNBP = Generator Load Rejection with Bypass Failure
FWCF = Feedwater Controller Failure (to maximum demand)
TTNBP= Turbine Trip with Bypass Failure
MSIVF = Main Steam Isolation Valve Closure, Flux Scram
HPCS = Inadvertent Actuation of High Pressure Core Spray
LFWH = Loss of Feedwater Heaters (145 deg F FWTR)

(b) 100P = uprate power of 3489 MWt
100F = rated core flow of 108.5Mlb/hr
105F = ICF flow point at uprated power
81F = MELLLA flow point at uprated power (87.9 Mlb/hr)

** Notes: d = 1SRVOOS
e = RPTOOS
f = TBSOOS
g = FFWTR
h = TCV slow closure

TABLE 15.B-5

ARTS VERIFICATION FOR MAPFAC(P) RESULTS

Power Dependent MAPLHGR Factor without TPS Credit	
25% ≤ P < 85%	MAPFAC(p) = 0.910 + 0.00503(P-85)
85% ≤ P < 100%	MAPFAC(p) = 0.910

Power Dependent MAPLHGR Factor with TPS Credit	
25% ≤ P < 95%	MAPFAC(p) = 0.960 + 0.00503(P-95)
95% ≤ P < 100%	MAPFAC(p) = 0.960

Source of information: Reference 59.

15.C AREVA Evaluations of Measurement Uncertainty Recapture (MUR) Uprate

This section documents the analyses performed to support power uprate to 3546 MWth. This section is retained for historical information only.

15.C.1 Introduction

The LaSalle measurement uncertainty recapture (MUR) Power Uprate Project encompasses the evaluations and licensing process for implementation of a 101.65% core thermal power uprate to 3546 MWth at LaSalle County Station Unit 1 and 2 (LaSalle). Within the scope of the LaSalle Power Uprate Project, AREVA NP Inc. (AREVA) performed a set of fuel-related evaluations to support an Exelon submittal for a Licensing Amendment Request (LAR). The AREVA evaluations provide the technical justification for reactor operation at 3548 MWth (2 MWth above MUR power) and 1020 psia reactor dome pressure. Sections 15.C.2 through 15.C.10 provide the summaries of AREVA evaluations.

The increase in thermal power is based on an improved system to measure the reactor power level which results in a reduction in the reactor power level measurement uncertainty. The rated thermal power level plus uncertainty will remain unchanged. This type of application is commonly referred to as an MUR power uprate.

LaSalle Unit 1 implemented MUR during mid-Cycle 14. All the fuel operating during this cycle are manufactured by AREVA. Unit 1 has a full core of ATRIUM-10 assemblies.

LaSalle Unit 2 implemented MUR during the L2R13 refuel outage. All the fuel operating during this cycle is manufactured by AREVA. Unit 2 has a full core of ATRIUM-10 and ATRIUM 10XM assemblies.

15.C.2 Applicability of AREVA NP BWR Methods

Reference 1 reviews the applicability of the AREVA BWR methods to the MUR power uprate conditions for LaSalle. The AREVA BWR methods have been reviewed previously by AREVA and the NRC for applicability to EPU power uprate conditions (EPU power uprate is 20% of the original rated power level). The NRC has previously approved operation at EPU conditions of plants which utilize AREVA BWR methods. The MUR rated power conditions are bounded by the EPU rated power conditions.

AREVA's review (Reference 1) concluded that there are no SER restrictions on AREVA methodology that are impacted by MUR. Since the MUR core and assembly conditions for the LaSalle units are equivalent to core and assembly conditions of other plants for which the methodology was benchmarked, the AREVA methodology (including uncertainties) remains applicable for MUR conditions at the LaSalle Units.

15.C.3 Disposition of Event Summary

The objective of this disposition of events (Reference 2) was to identify the limiting events which must be analyzed to establish operating limits at LaSalle County Station to support operation with an MUR power uprate of up to 1.7%. The disposition was accomplished by reviewing the plant licensing basis and classifying each fuel related event as potentially limiting, bound by the consequences of another event or the current analysis remains applicable. The final disposition status was to identify each event as needing further analysis or not. Those events identified as needing further analysis were performed for the initial MUR cycle (Unit 1 mid Cycle 14 and Unit 2 Cycle 14).

15.C.4 Seismic/LOCA Evaluation

Reference 4 evaluated the ATRIUM-10 and ATRIUM 10XM LTA fuels for seismic/LOCA at MUR conditions. Reference 4 found that the Reference 3 analyses for ATRIUM-10 at CLTP remain applicable for ATRIUM-10 at MUR conditions. Because the ATRIUM 10XM LTAs have dynamic properties that are well within those previously used at LaSalle and nearly identical to the original GE9 assembly supported by the existing core seismic analysis, the reactor core internals and reactor pressure vessel will not be affected by the introduction of the LTAs.

Reference 4 calculated liftoff heights for ATRIUM-10 and ATRIUM 10XM LTAs under MUR conditions. The liftoff heights for both fuel types are less than the minimum lower tie plate nozzle engagement of 0.67 inches.

15.C.5 Applicability of EPG Data

AREVA provides fuel design data that is used by Exelon to prepare Emergency Procedure Guidelines (EPG). AREVA assessed the impact of the MUR power uprate on EPG data and determined that the EPG data provided in support of pre-MUR operating conditions remains applicable for operation at MUR conditions. Reference 5 provides a summary of the evaluation.

15.C.6 Lost Part Analyses

A generic lost parts analysis has been performed for the LaSalle reactors. The analysis was performed by calculating the fuel assembly ΔMCPR values as a function of percent flow blockage for ATRIUM-10, ATRIUM-9B, GE9, and GE14 fuel assemblies at BOC, peak reactivity (MOC), and EOC. Reference 6 addresses the ΔMCPR values for ATRIUM-10 fuel assemblies at an uprated core power level of up to 3548 MWth associated with the LaSalle MUR project. If a part is lost and cannot be recovered, the possible consequence on reactor operation can be evaluated using the results presented in Reference 6.

15.C.7 Appendix R, Station Blackout, Containment Analysis, and the Radiological Source Terms

Reference 7 concludes that operation with ATRIUM-10 fuel and ATRIUM 10XM LTA does not impact the MUR Appendix R, Station Blackout, and Containment analyses. The radiological source terms analyses for ATRIUM-10 fuel and ATRIUM

10XM LTAs were generated at a core power level of 3910 MWth which bounds the MUR power level.

15.C.8 Vessel Fluence

Reference 8 addresses the impact of reactor vessel fluence due to MUR. Operation at the higher MUR power results in approximately 1.7% increase in neutron fluence rates at the current CLTP. As a result the embrittlement damage will theoretically be higher than the current CLTP basis. Current bounding fluence values associated with the material limits developed in Reference 9 were found to be conservative by more than a factor of 2 by cycle-by-cycle tracking. Therefore, the effects on embrittlement will be inconsequential due to the large margin.

15.C.9 LPRM Calibration Interval

The extended LPRM calibration uncertainty analysis was performed previously for 3489 MWT for the exposure interval 2500 EFPH (2000 EFPH with 25% allowance), which corresponded to 2675 MWd/MTU. The MUR power uprate evaluation is based on an increase in the reactor rated power from 3489 to 3548 MWth. The exposure interval will increase to 3548/3489*2675=2720 MWd/MTU. Reference 10 details the evaluation.

15.C.10 ATWS Over-Pressure

The increase in power associated with MUR will increase the severity of an ATWS event. The licensing basis ATWS analysis for MUR is being performed by GEH; however, the analysis is not based on the core configuration for the initial LaSalle MUR cycle. Therefore, AREVA compared the key kinetics parameters and concluded that differences between a mixed core (L2C11) and an all ATRIUM-10 core (L1C14) do not play a significant role in the outcome of the ATWS peak pressure analysis (Reference 11).

15.C.11 References

1) Design Analysis L-003504, Revision 0, “Applicability of AREVA NP BWR Methods to Measurement Uncertainty Recapture Power Uprate Conditions for LaSalle Units 1 and 2,” July 2010.

2) Design Analysis L-003505, Revision 0, “Disposition of Event Summary for the LaSalle MUR Power Uprate,” July 2010.

3) “LaSalle ATRIUM/LOCA Evaluation,” Letter (DEG:01:150) from David Garber to F. W. Trikur, September 25, 2001.

4) Design Analysis L-003506, Revision 0, “LaSalle ATRIUM 10 and ATRIUM 10XM Seismic/LOCA MUR Evaluation,” July 2010.

5) Design Analysis L-003507, Revision 0, “Applicability of EPG Data at LaSalle MUR Operating Conditions,” July 2010.

6) Design Analysis L-003508, Revision 0, "LaSalle Lost Parts Analysis," July 2010.

7) Design Analysis L-003509, Revision 0, "Evaluation of Appendix R, Station Blackout, Containment, and Source Terms for LaSalle MUR Power Uprate," July 2010.

8) Design Analysis L-003510, Revision 0, "Vessel Fluence for LaSalle MUR Power Uprate," July 2010.

9) "LaSalle 1&2 Neutron Flux Evaluation," GENE Document GE-NE-000-0002-5244-02 (AREVA NP Document 58-5069936-00)," Revision 0, June 2002.

10) Design Analysis L-003511, Revision 0, "LPRM Calibration Interval Extension Evaluation at Measurement Uncertainty Recapture Power Uprate Conditions for LaSalle Units 1 and 2," July 2010.

11) Design Analysis L-003512, Revision 0, "ATWS Evaluation for LaSalle MUR Power Uprate," July 2010.

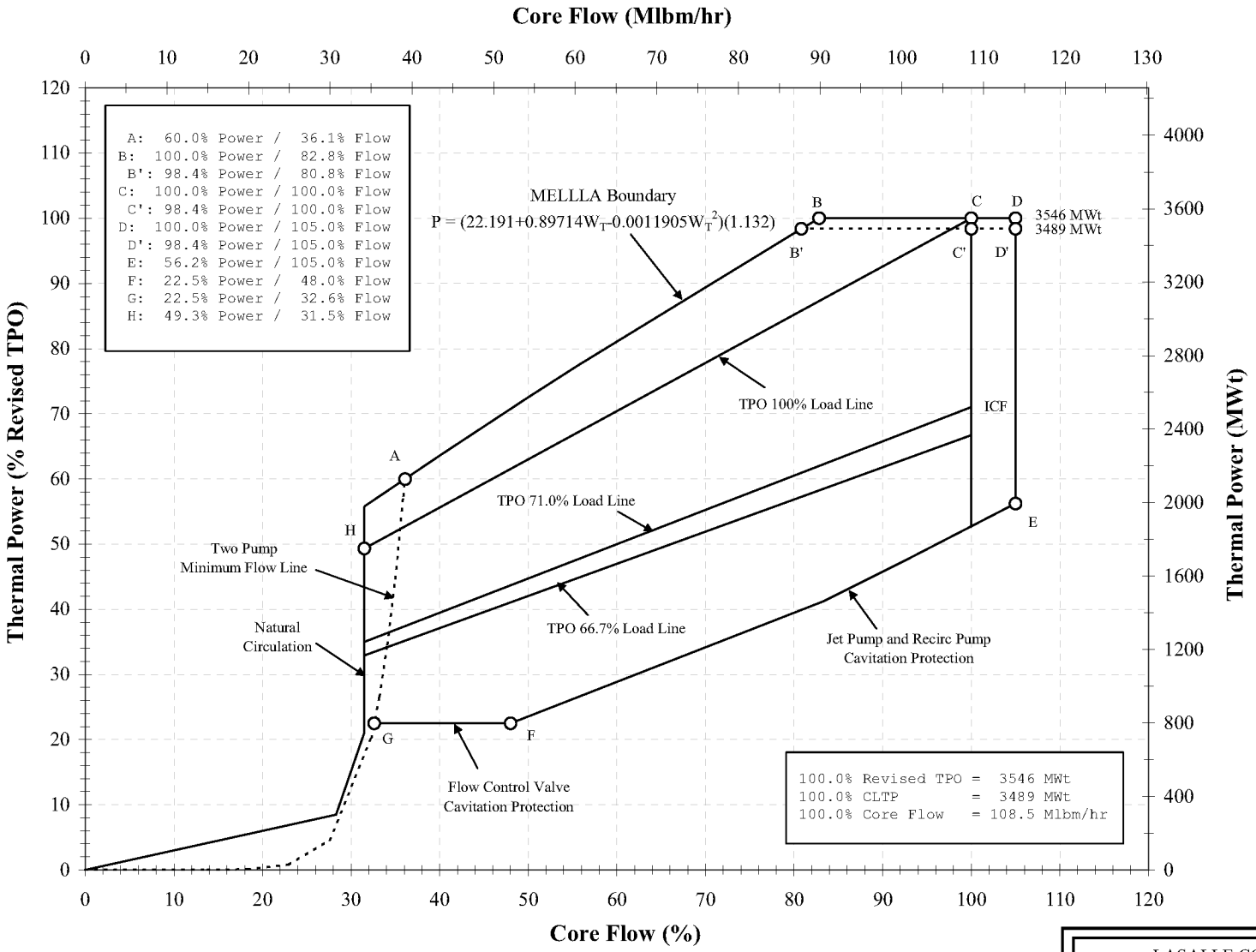

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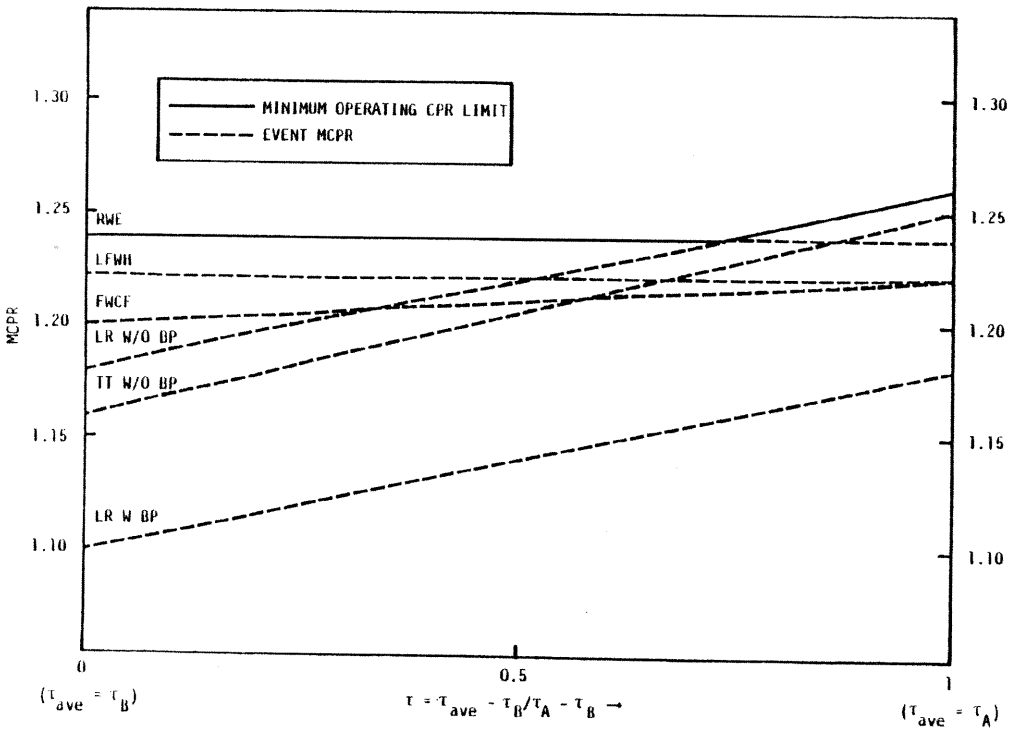
FIGURE 15.0-1

LASALLE POWER/FLOW MAP

Figure 15.0-1(a) has been intentionally deleted.



The Operating limit MCPR at rated core flow is the dominating locus of event MCPR values from Figure.
The La Salle reactors are limited by the rod withdrawal error event (RWE) at an $OLMCPR_B$ = 1.24 up to that point where the generator load reject event without bypass (LR W/O BP) intersects and dominates the RWE event. Then the $OLMCPR_B$ becomes 1.26 at the value of $\tau = \tau_A$.

Applicable definitions are as follows:

τ = measured core average scram time to notch position 39.

τ_A = Technical Specification limit on core average scram time to notch position 39, which for La Salle is 0.860 seconds.

τ_B = Maximum allowable core average scram time to notch position 39 when operating under the $OLMCPR_B$ limit of 1.24 MCPR.

$OLMCPR_A$ = Operating Limit based on ODYN Option A.

$OLMCPR_B$ = Operating Limit based on ODYN Option B.

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FIGURE 15.0-2
INITIAL CORE
MINIMUM OPERATING CPR LIMIT

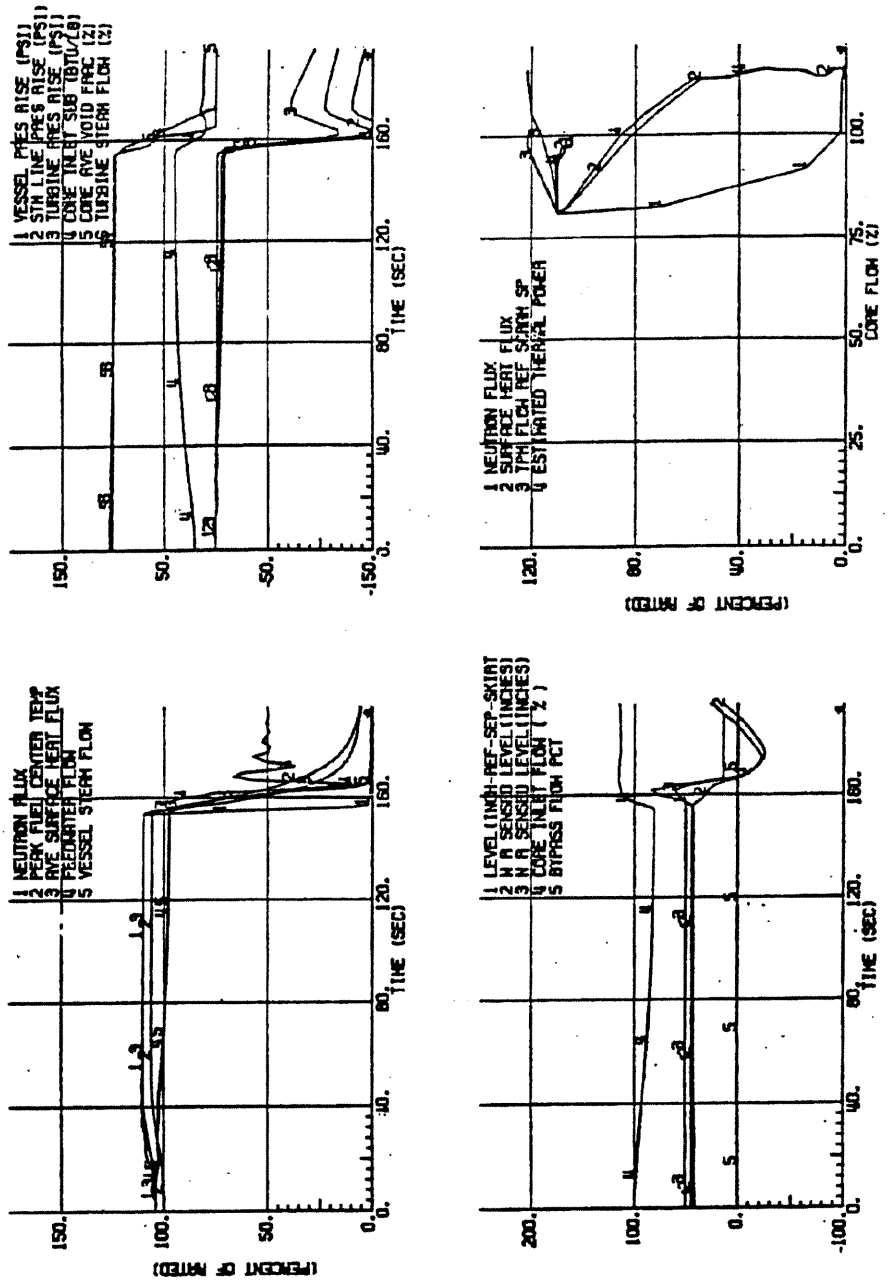

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FIGURE 15.1-1

(INITIAL CORE)

LOSS OF FEEDWATER HEATER AUTO FLOW CONTROL MODE

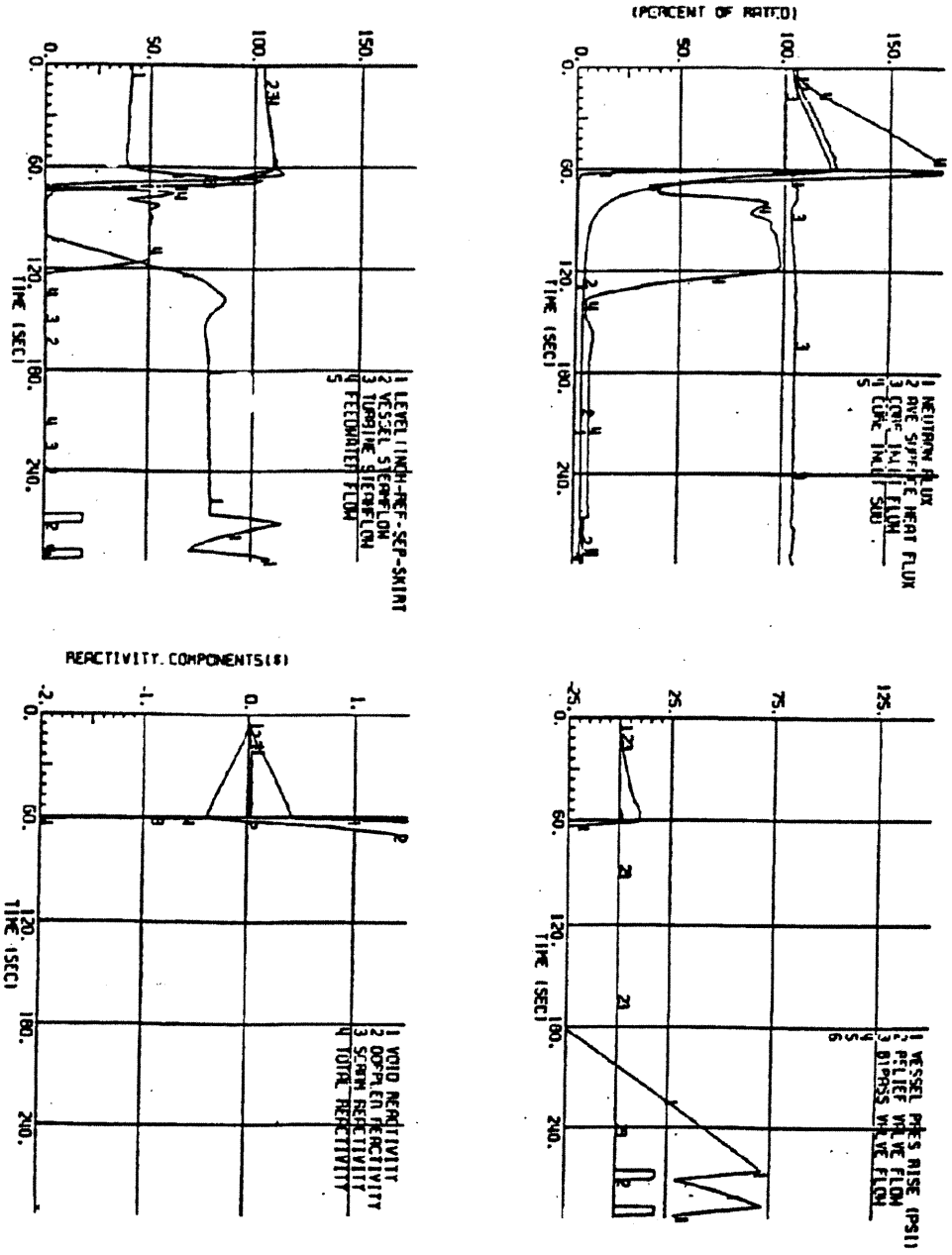

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FIGURE 15.1-2

(INITIAL CORE)
LOSS OF FEEDWATER HEATER MANUAL FLOW CONTROL

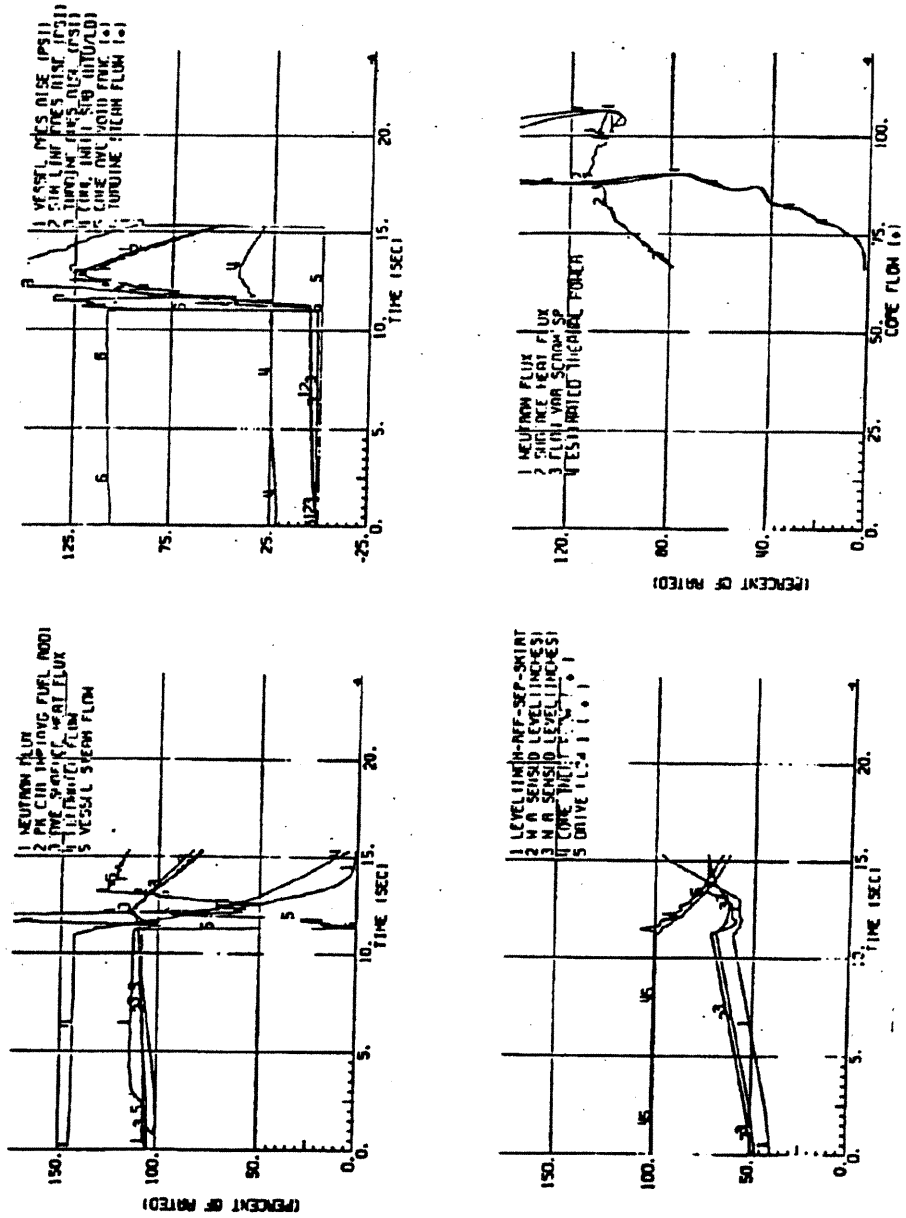

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FIGURE 15.1-3

FEEDWATER CONTROLLER FAILURE, MAXIMUM DEMAND, WITH HIGH WATER LEVEL TRIPS, 104% POWER – ODYN REANALYSIS

(INITIAL CORE RESULTS)

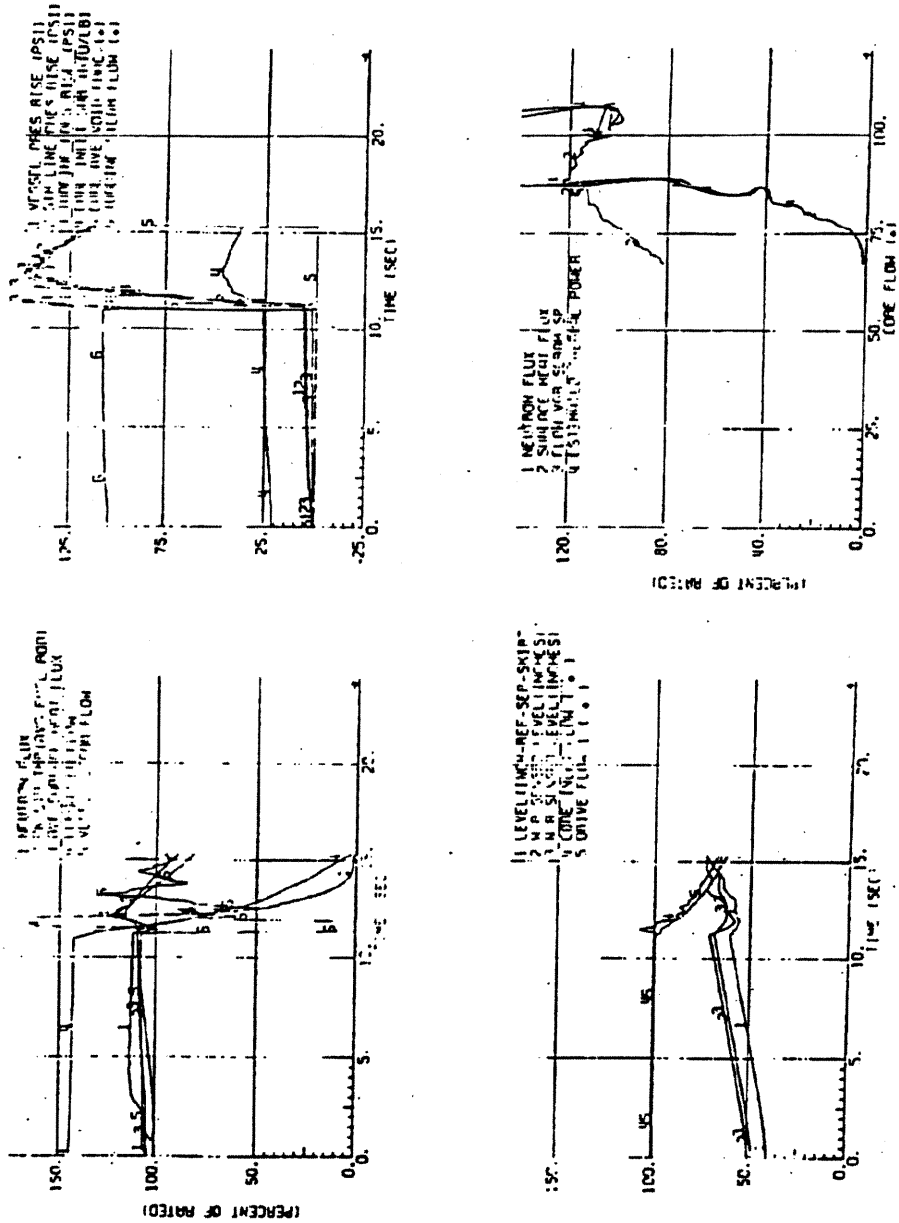

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FIGURE 15.1-4

FEEDWATER CONTROLLER FAILURE, MAXIMUM DEMAND, WITH HIGH WATER LEVEL TRIPS, 104% POWER – ODYN REANALYSIS WITHOUT BYPASS (INITIAL CORE RESULTS)

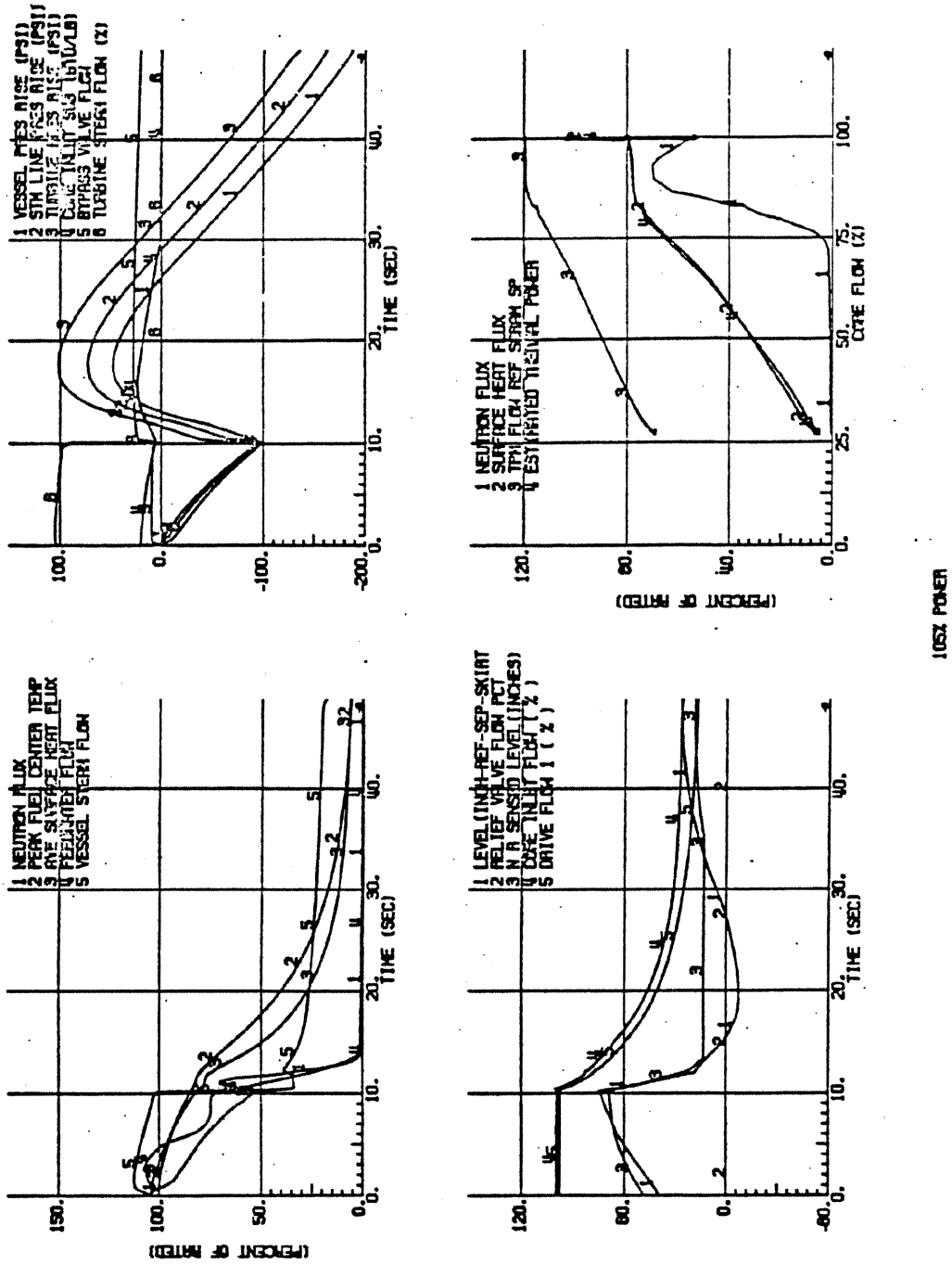

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FIGURE 15.1-5

PRESSURE REGULATOR FAILURE TO 115% DEMAND
(INITIAL CORE RESULTS)

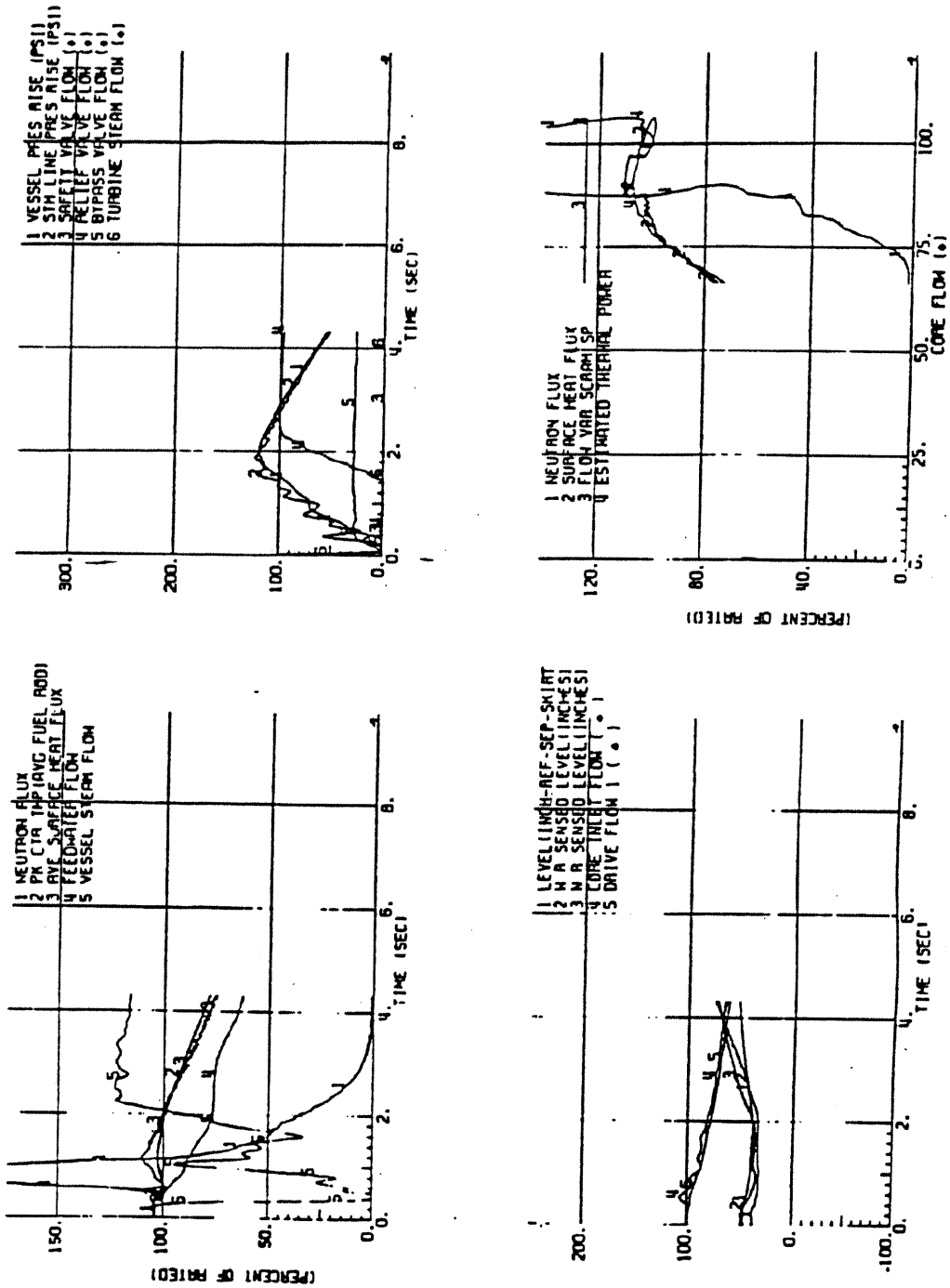

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FIGURE 15.2-1

GENERATOR LOAD REJECTION WITH BYPASS ON
104 LOAD – ODYN REANALYSIS
(INITIAL CORE RESULTS)

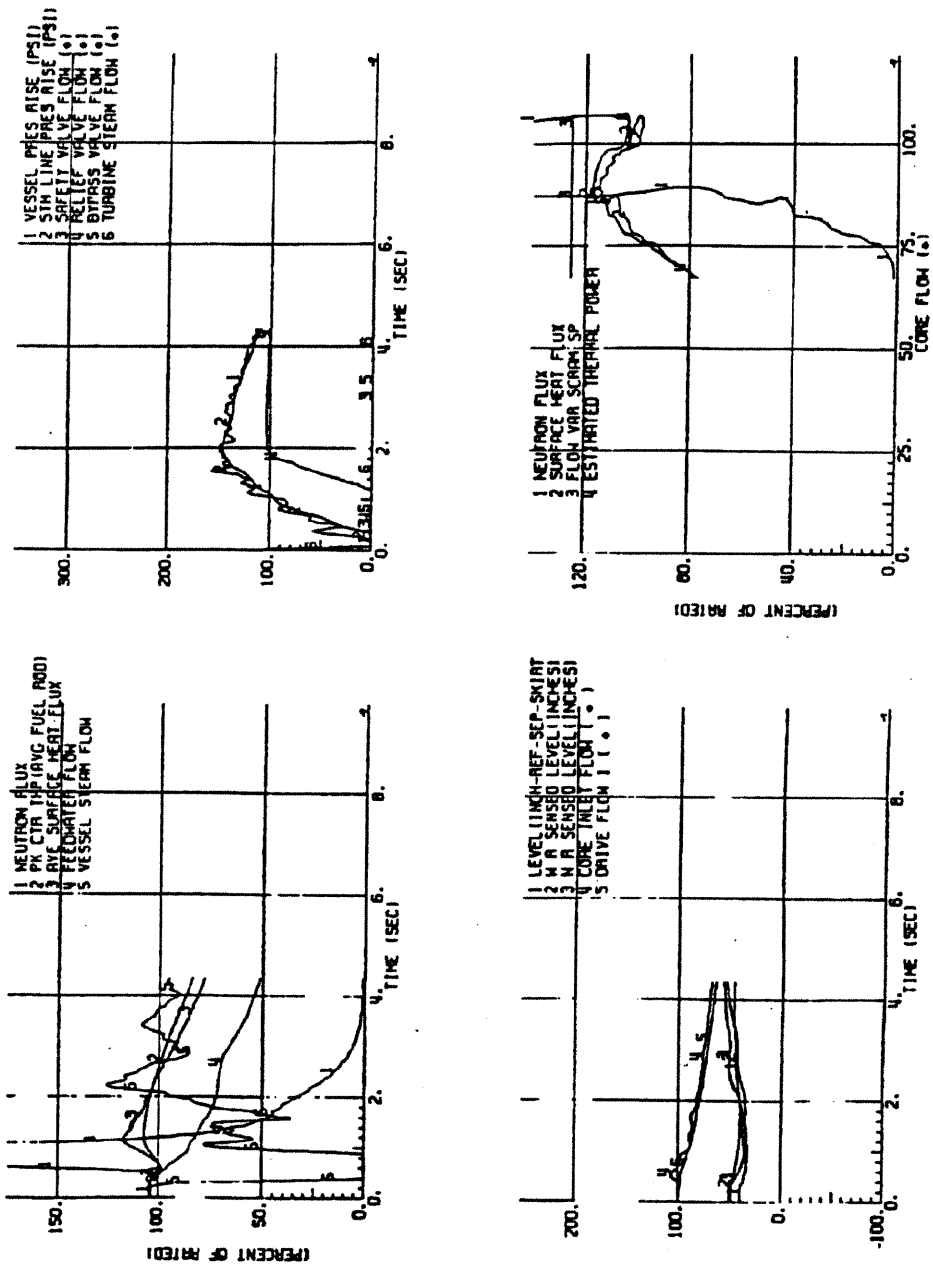

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FIGURE 15.2-2

GENERATOR LOAD REJECTION WITHOUT BYPASS 104% POWER – ODYN REANALYSIS
(INITIAL CORE RESULTS)

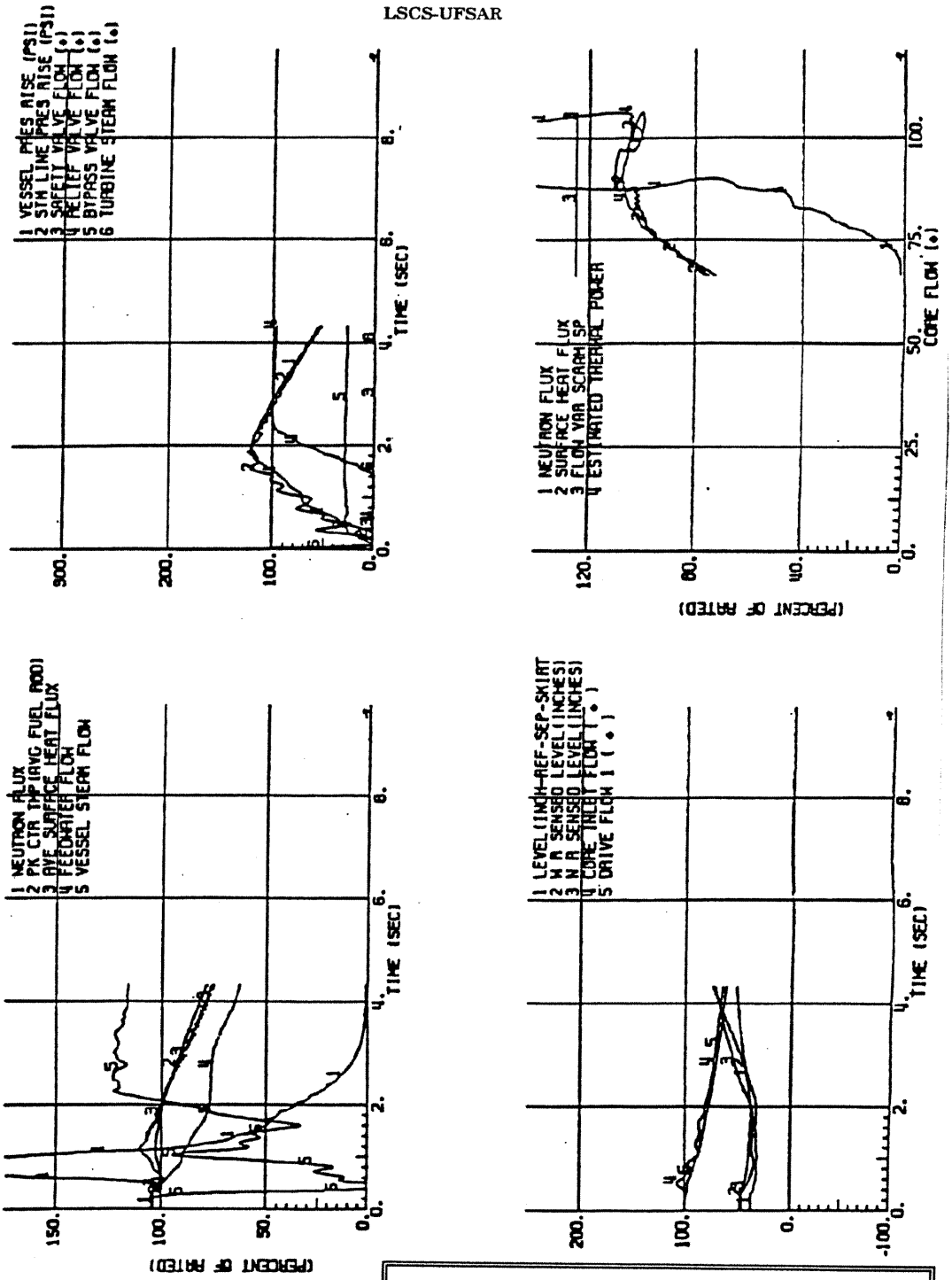

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FIGURE 15.2-3

GENERATOR LOAD REJECTION WITH BYPASS ON
104% LOAD - ODYN REANALYSIS
(INITIAL CORE RESULTS)

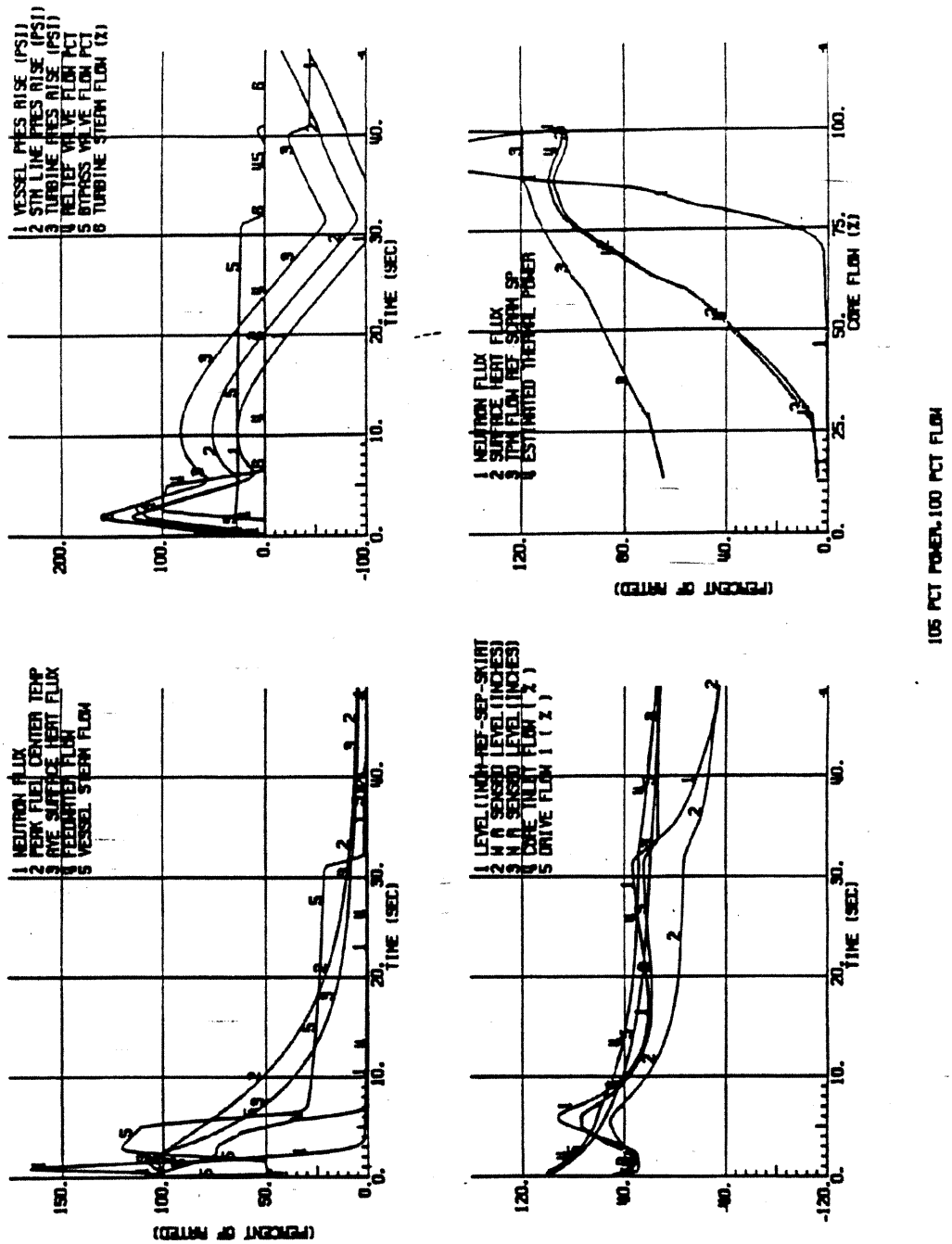

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FIGURE 15.2-4

TURBINE TRIP, TRIP SCRAM, BYPASS-ON, RV-ON, RPT-ON
(INITIAL CORE RESULS)

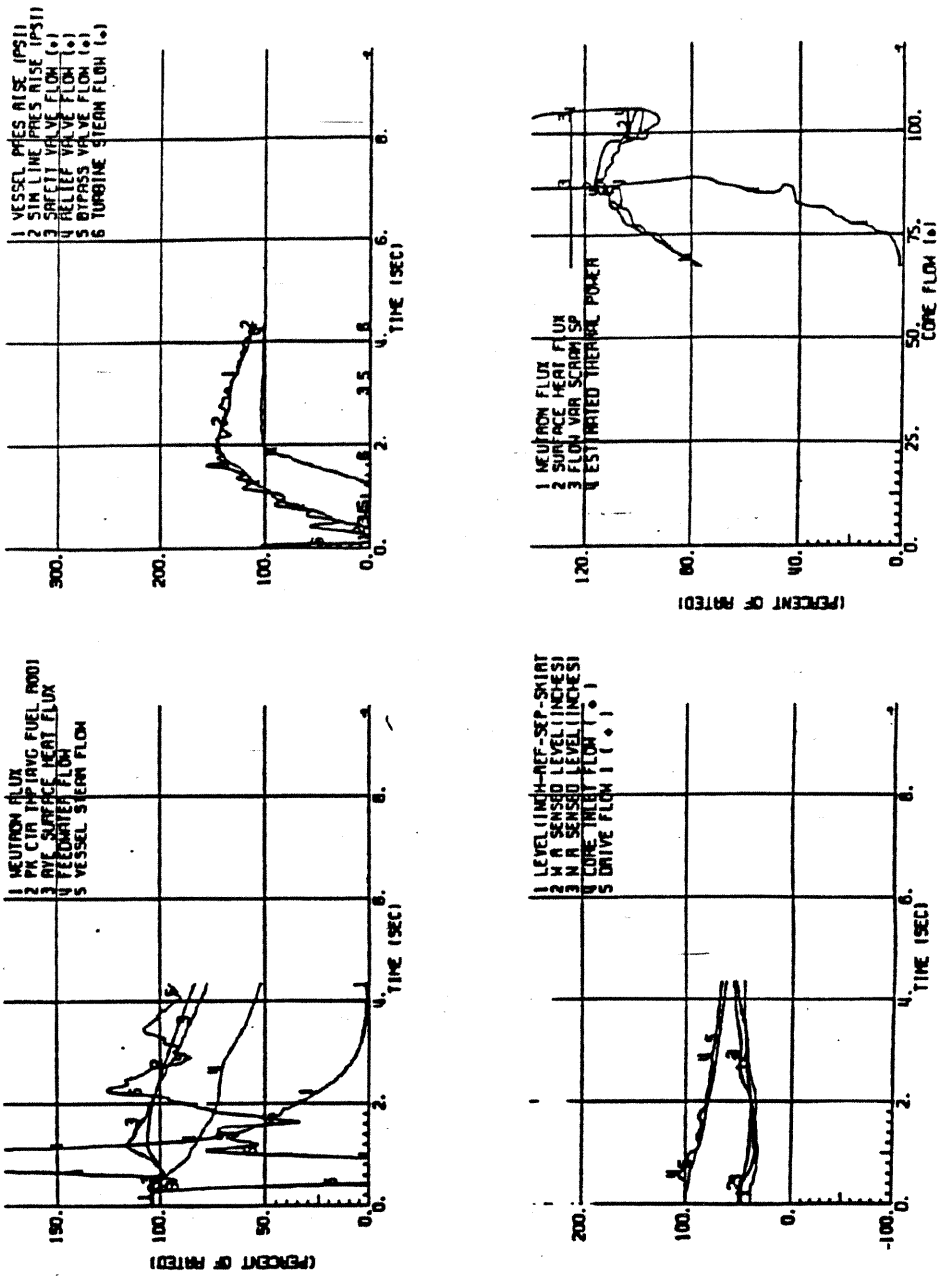

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FIGURE 15.2-5

TURBINE TRIP WITHOUT BYPASS,TRIP SCRAM, 104% POWER – ODYN REANALYSIS

(INITIAL CORE RESULTS)

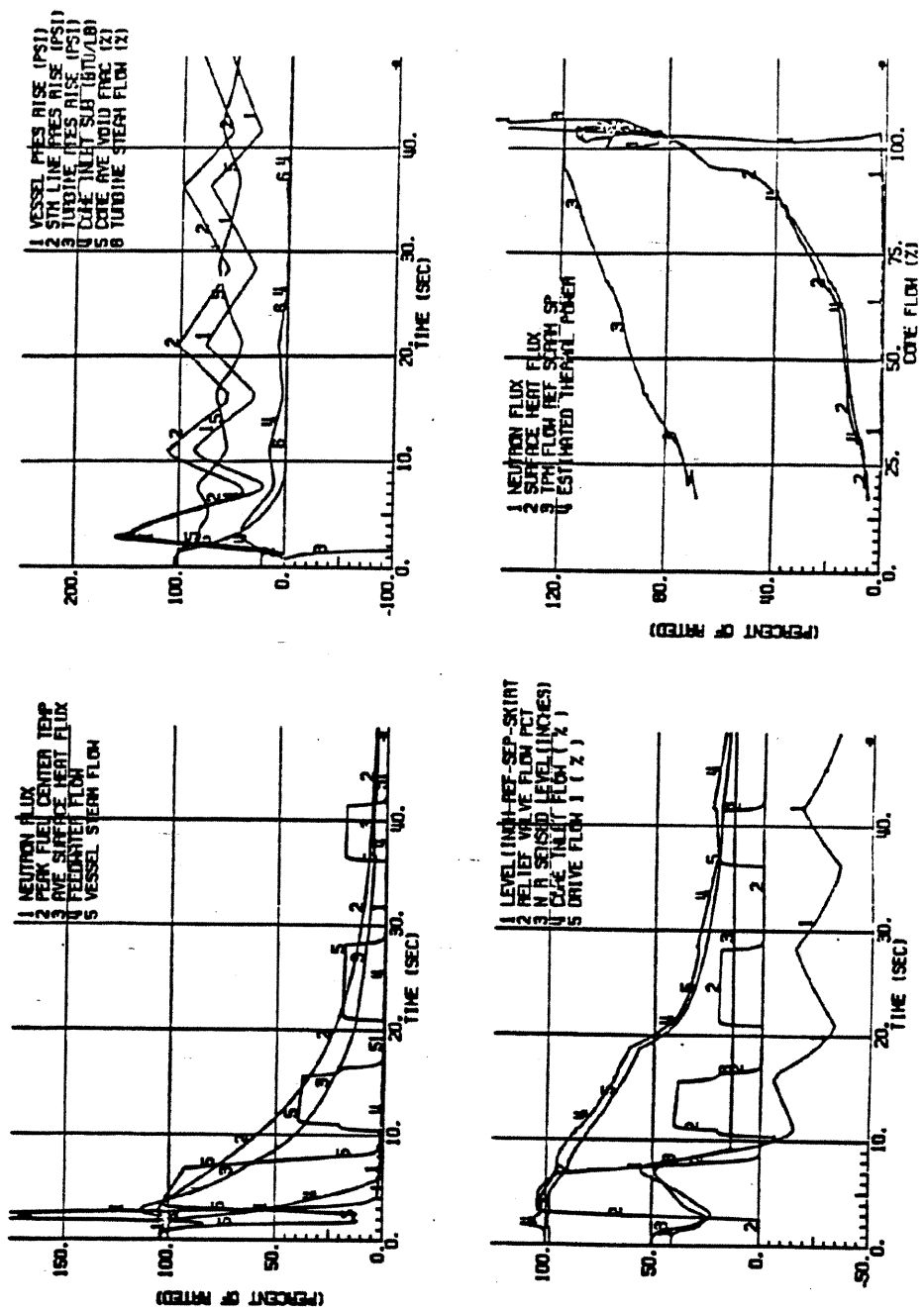

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FIGURE 15.2-6

3-SECOND CLOSURE OF ALL MSIV'S 105% POWER
(INITIAL CORE RESULTS)

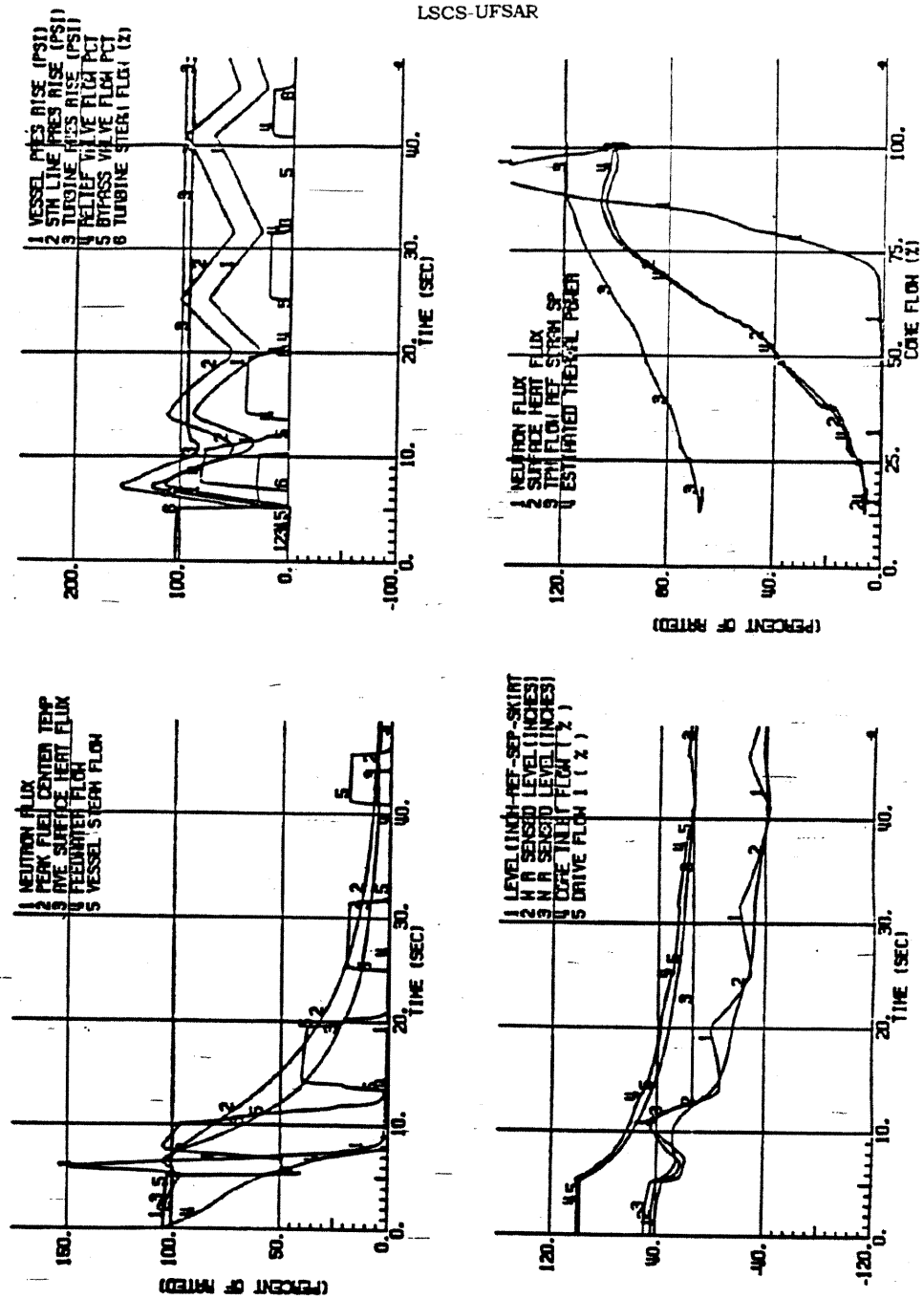

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FIGURE 15.2-7

LOSS OF CONDENSER VACUUM AT 2 INCHES PER SECOND
105% POWER, 100% FLOW
(INITIAL CORE RESULTS)

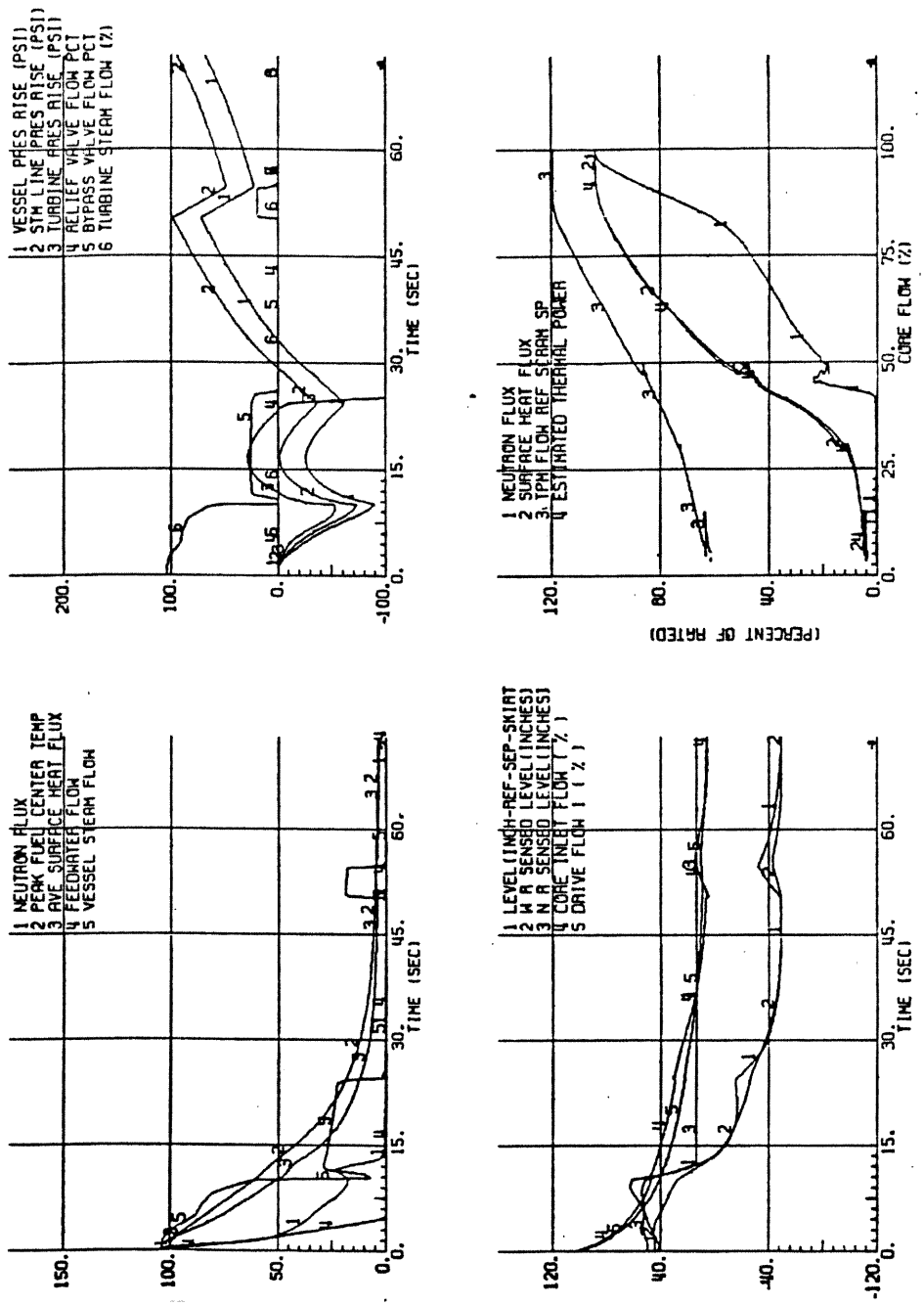

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FIGURE 15.2-8

LOSS OF AUXILIARY POWER TRANSFORMERS

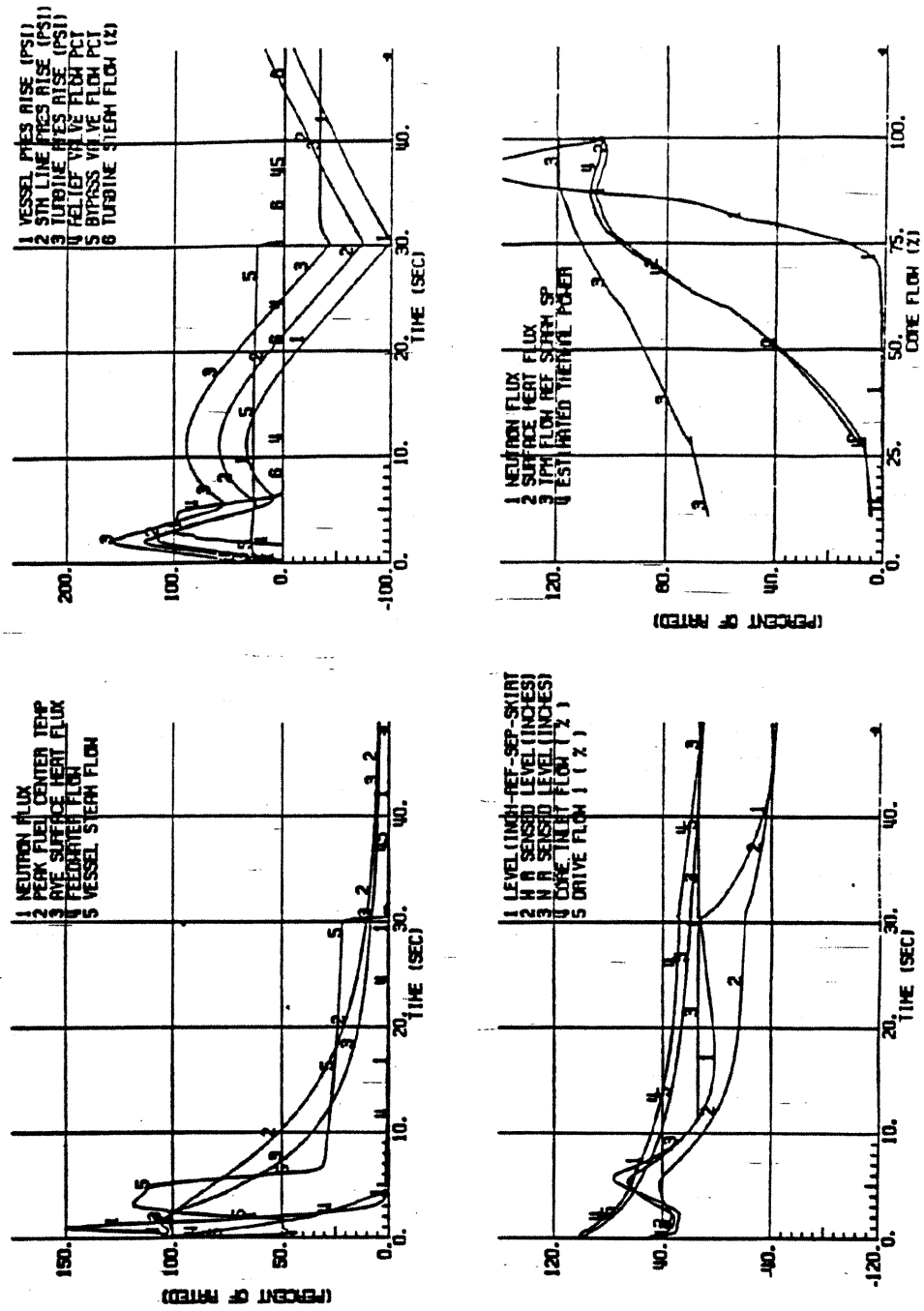

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FIGURE 15.2-9

LOSS OF ALL GRID CONNECTIONS 105% POWER, 100% FLOW

(INITIAL CORE RESULTS)

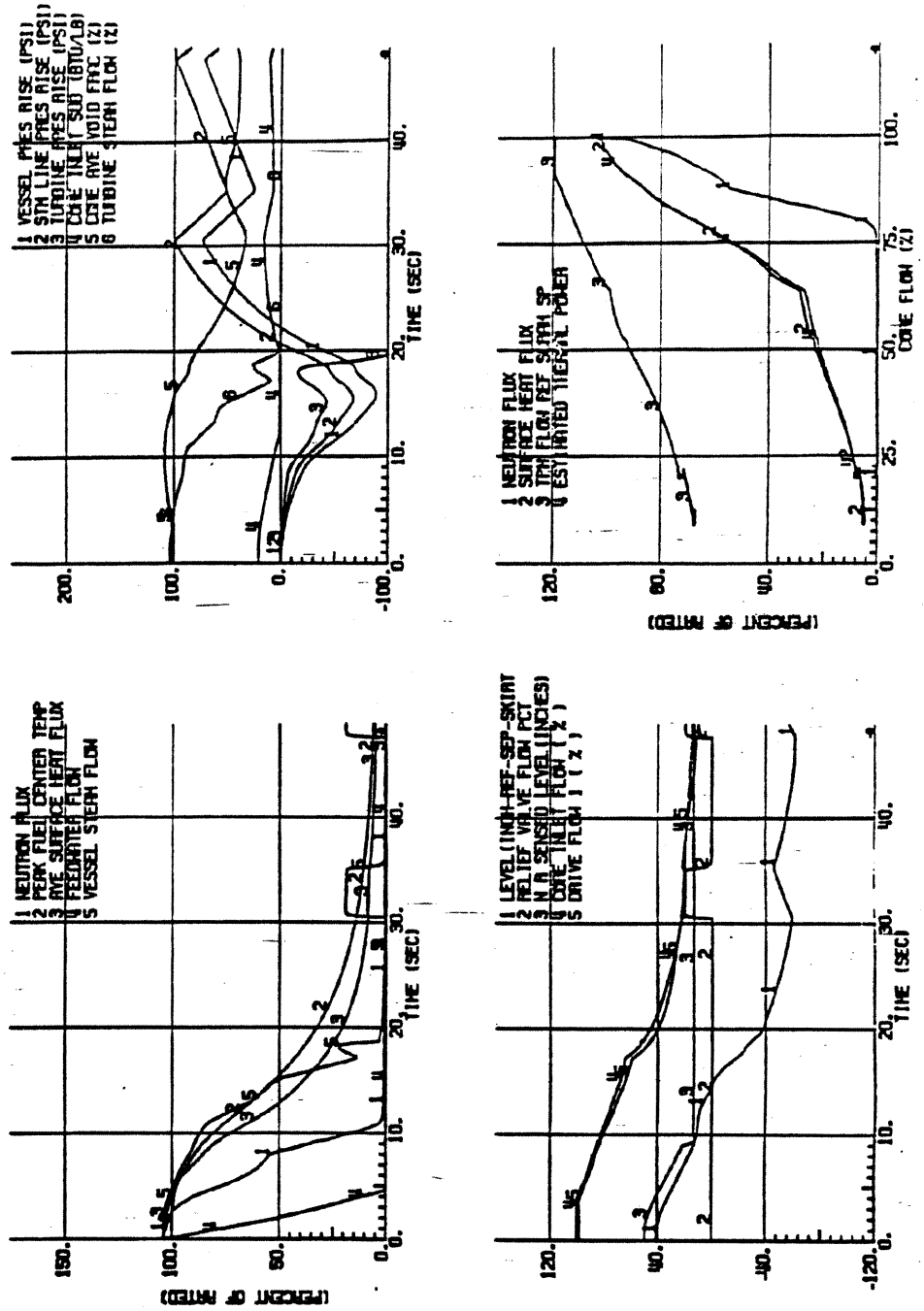

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FIGURE 15.2-10

LOSS OF ALL FEEDWATER FLOW 105% POWER, 100% FLOW

(INITIAL CORE RESULTS)

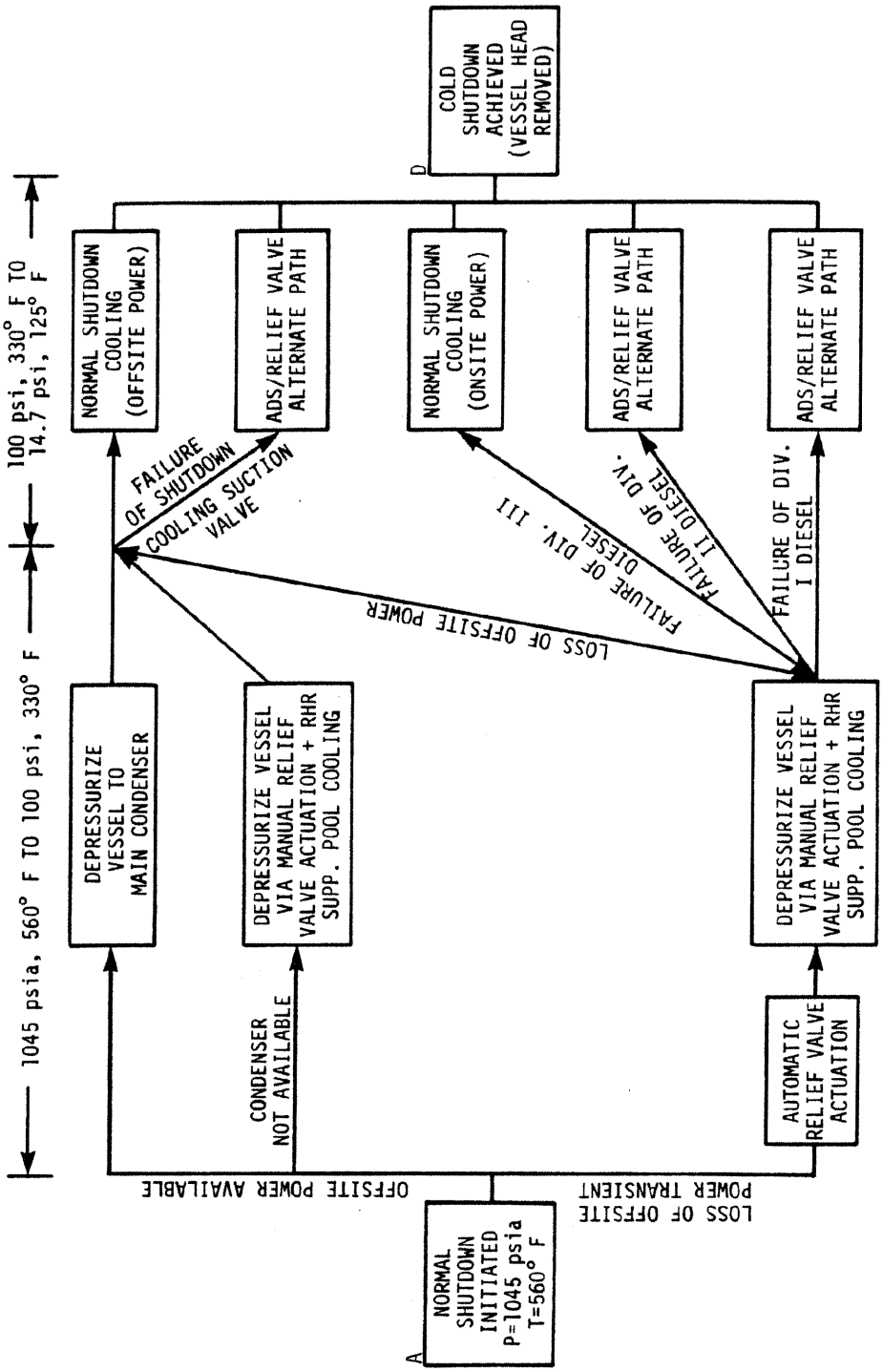

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FIGURE 15.2-11

SUMMARY OF PATHS AVAILABLE TO ACHIEVE COLD SHUTDOWN

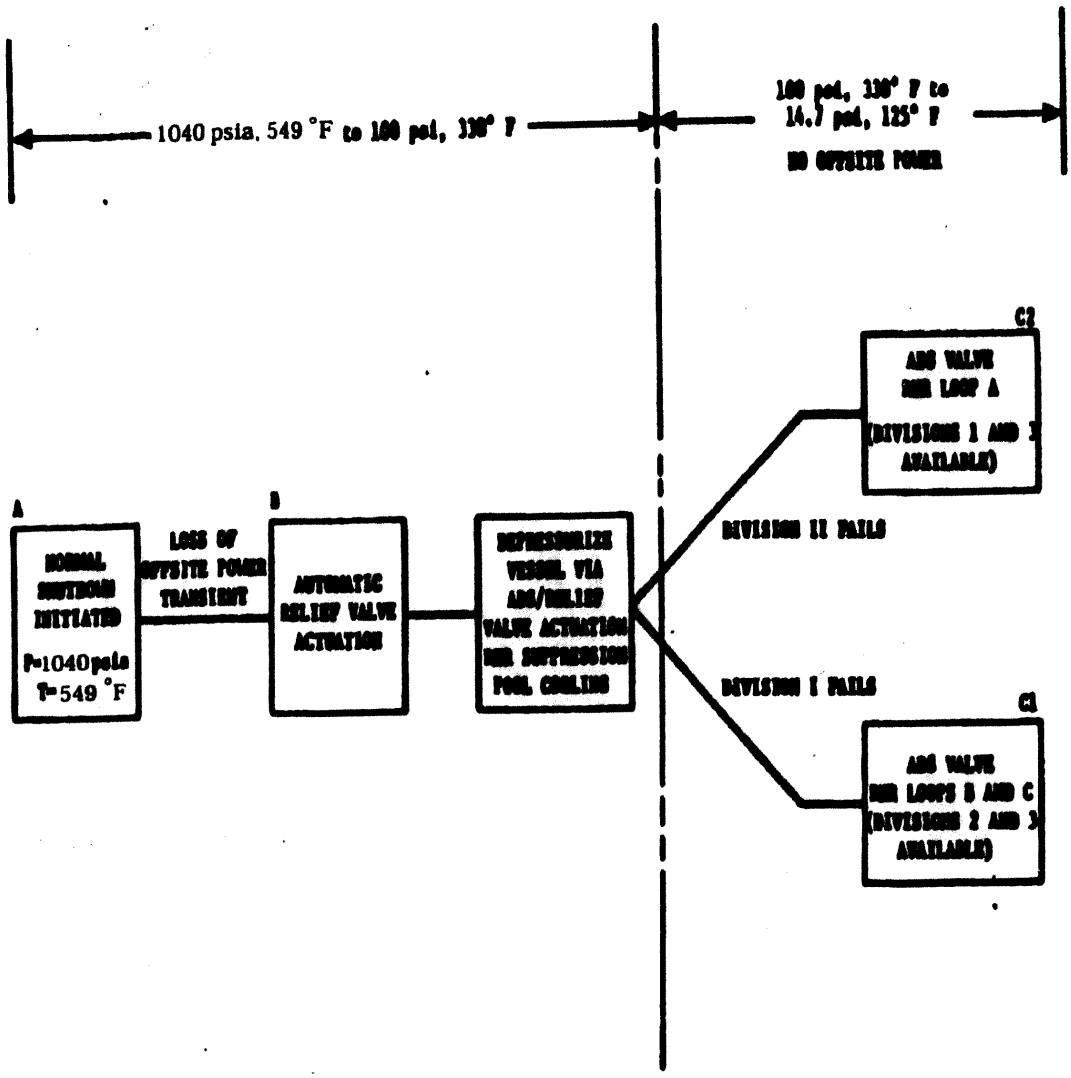

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FIGURE 15.2-12

ADS/RHR COOLING LOOPS
(Sheet 1 of 2)

Activity A
Initial pressure – 1040 psia
Initial temperature - 549° F

For purposes of this analysis, the following worst-case conditions are assumed to exist:

a. The reactor is assumed to be operating at 102% nuclear boiler rated steam flow:
b. a loss of power transient occurs (see Subsection 15.2.6); and
c. a simultaneous loss of onsite power (Division 1 or 2), which eventually results in the operator not being able to open one of the RHR shutdown cooling line suction valves.

Activity B
Initial pressure = 1040 psia
Initial system temperature = 549° F

Operator Actions
During approximately the first 30 minutes, reactor decay heat is passed to the suppression pool by the automatic operation of the reactor relief valves. Reactor water level will be returned to normal by the HPCS and RCIC system automatic operation.

After approximately 10 minutes, it is assumed one RHR heat exchanger will be placed in the suppression pool cooling mode to remove decay heat. The operator initiates depressurization of the reactor vessel to control vessel pressure. Controlled depressurization procedures consist of controlling vessel pressure and water level by using the ADS, and RCIC and HPCS systems.

When the reactor pressure approaches 100 psig, the operator would normally prepare for operation of the RHR system in the shutdown cooling mode.

Activity C1 (Division 1 fails, Division 2 available)
System pressure ~100 psi
System temperature ~330°F

Operator Actions
The operator establishes a closed cooling path as follows:

a. Three to five ADS valves (DC Division 2) are powered open;
b. Either of the following cooling paths is established:

1. Utilizing RHR loop B, water from the suppression pool is pumped through the RHR heat exchanger (where a portion of the decay heat is removed) into the reactor vessel. The cooled suppression pool water flows through the vessel (picking up a portion of the decay heat) out of the ADS valves and back to the suppression pool. This alternate cooling path is shown in Figure 15.2-13.

2. Utilizing RHR loops B and C together, water is taken from the suppression pool and pumped directly into the reactor vessel. The water passed through the vessel (picking up decay heat) and out the ADS valves returning to the suppression pool as shown in Figure 15.2-4. Suppression pool water is then cooled by operation of RHR loop B in the cooling mode (see Figure 15.2-15). In this alternate cooling path RHR loop C is used for injection and RHR loop B for cooling.

Activity C2 (Division 2 fails, Division 1 available) (Figure 15.2-16)
System pressure ~ 100 psi
System temperature ~ 330° F

Operator Actions
The operator establishes a closed cooling path as follows:

a. Three to five ADS valves (DC Division 1) are powered open;
b. Utilizing RHR loop A instead of loop B, and alternate cooling path is established as in Activity C1 Item 2 (a) above.

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FIGURE 15.2-12

ADS/RHR COOLING LOOPS (Sheet 2 of 2)

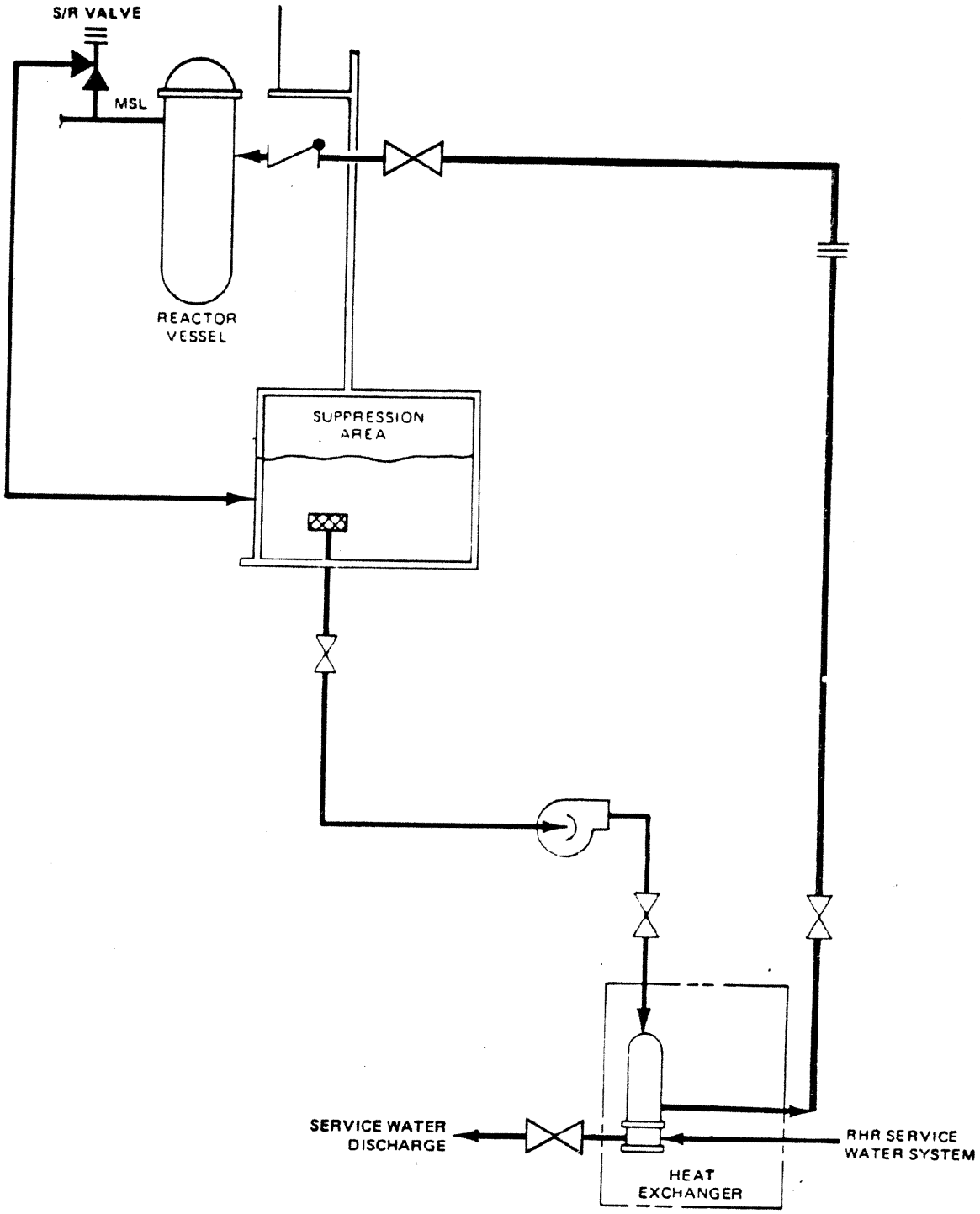

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FIGURE 15.2-13

ACTIVITY C1 ALTERNATE SHUTDOWN COOLING PATH UTILIZING RHR LOOP B

REV. 0 - APRIL 1984

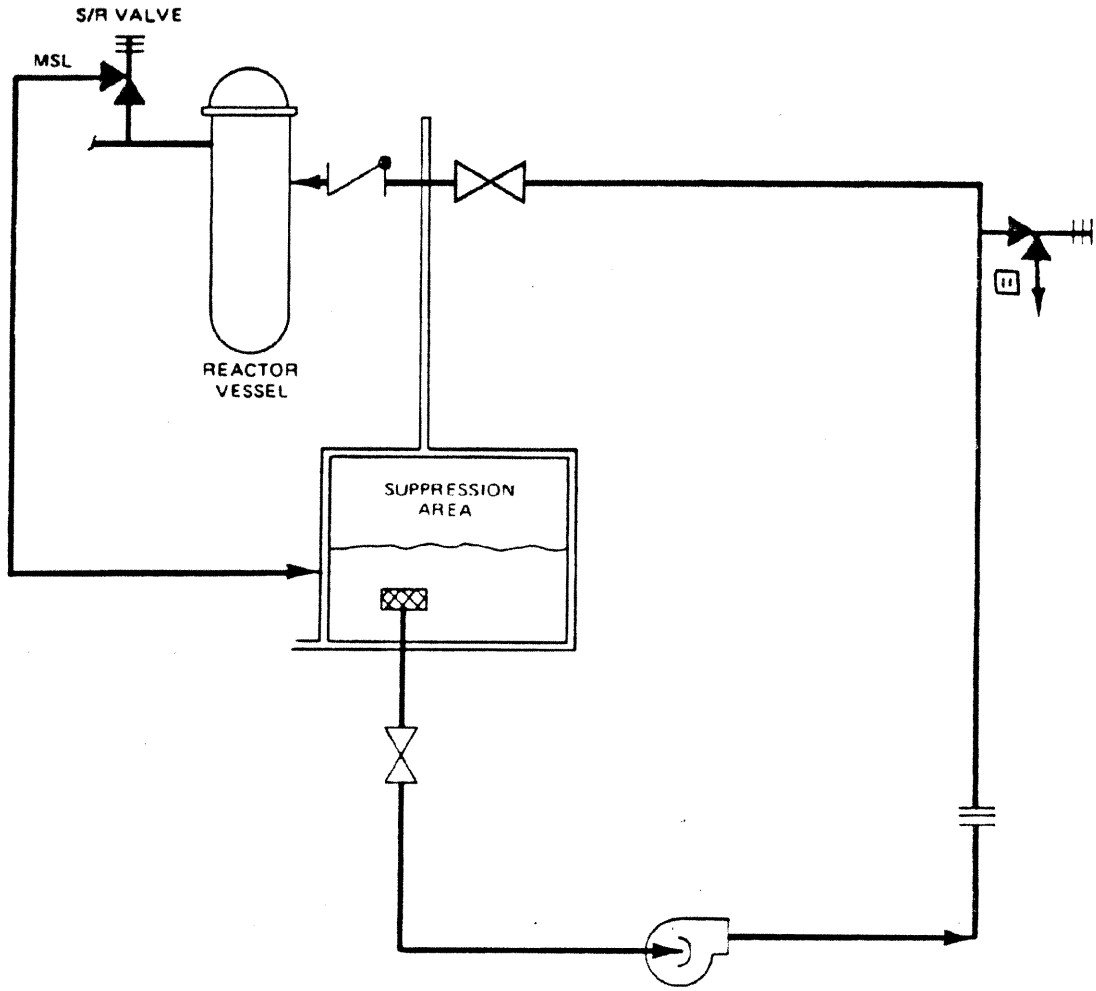

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FIGURE 15.2-14

RHR LOOP C

REV. 0 - APRIL 1984

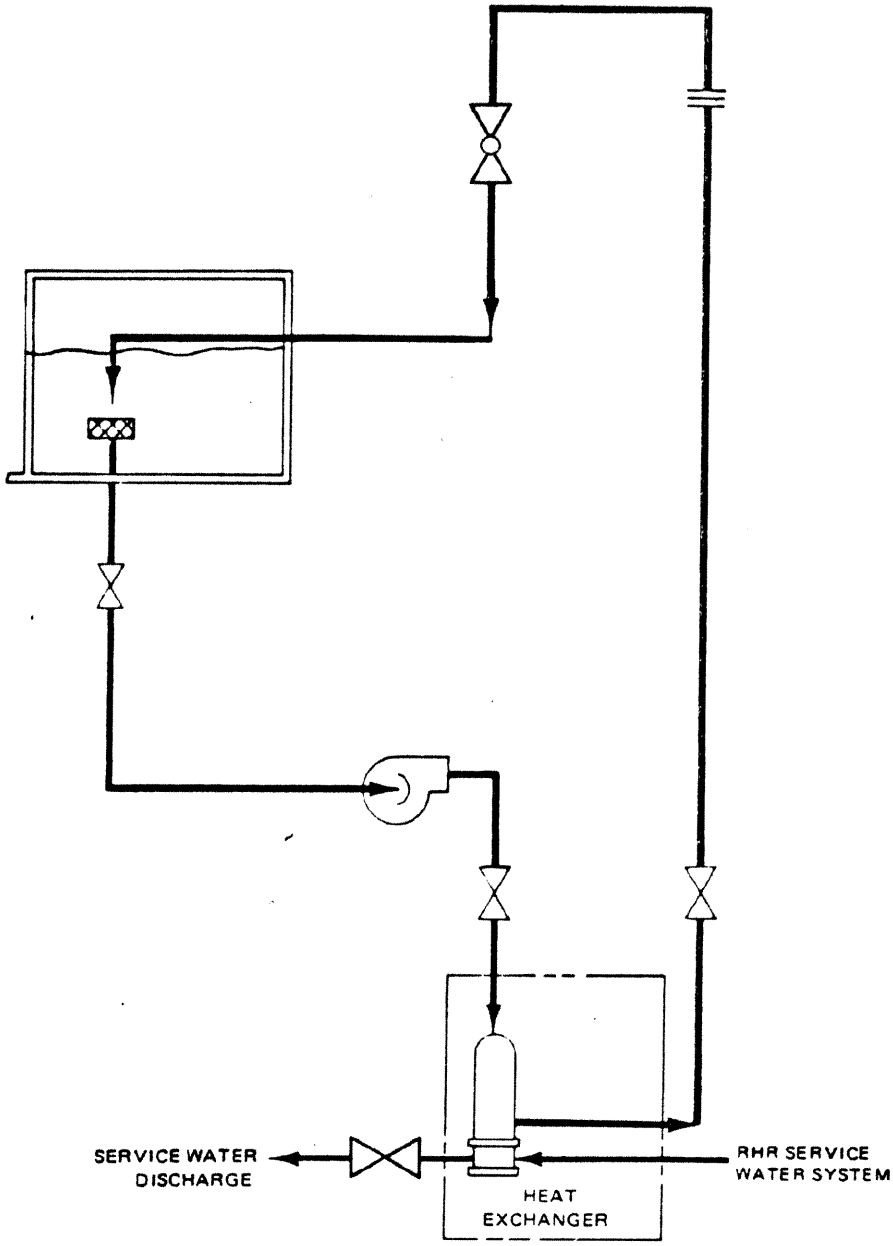

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FIGURE 15.2-15

RHR LOOP B
(SUPPRESSION POOL COOLING MODE)

REV. 0 - APRIL 1984

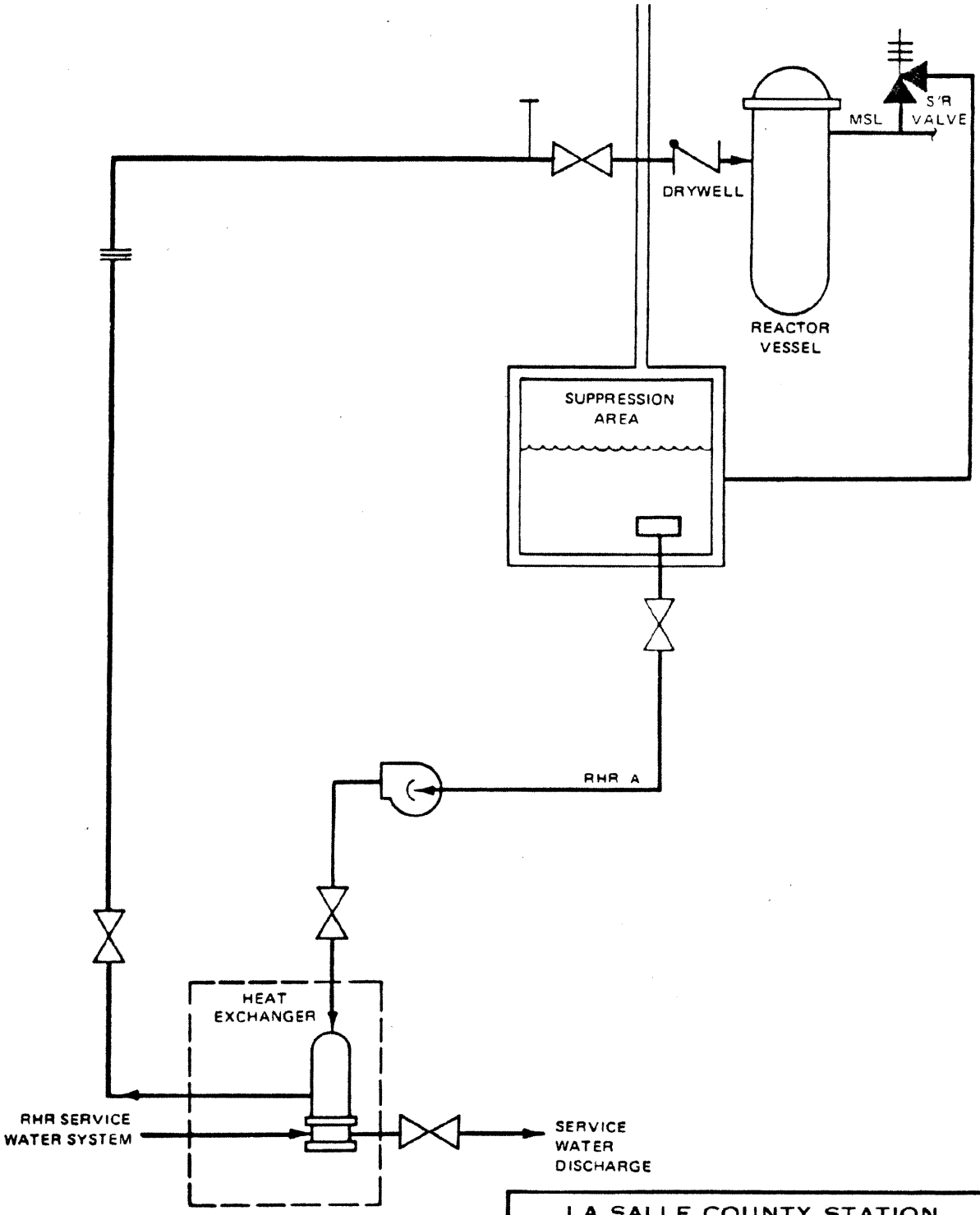

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FIGURE 15.2-16

ACTIVITY C2 ALTERNATE SHUTDOWN COOLING PATH UTILIZING RHR LOOP A

REV. 0 - APRIL 1984

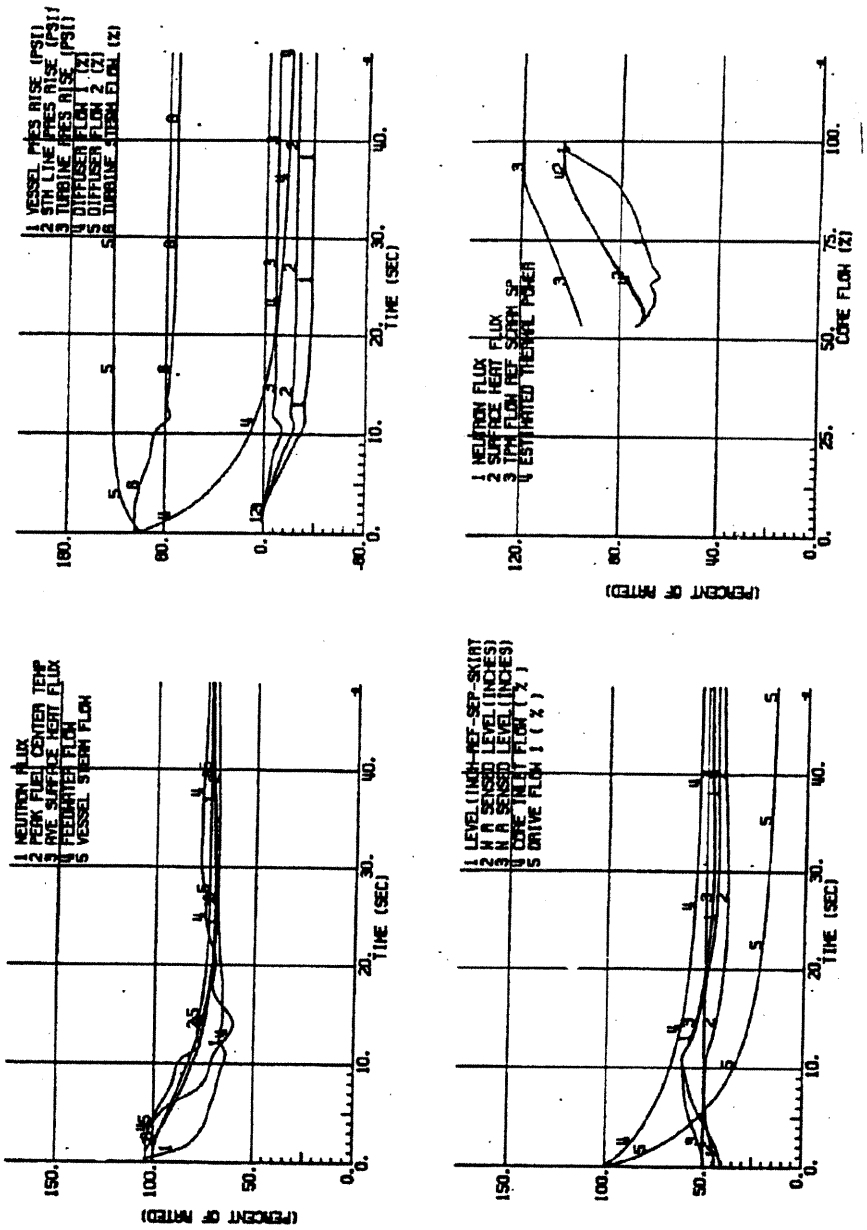

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FIGURE 15.3-1

TRIP OF ONE RECIRCULATION PUMP MOTOR – 105% POWER

(Typical)

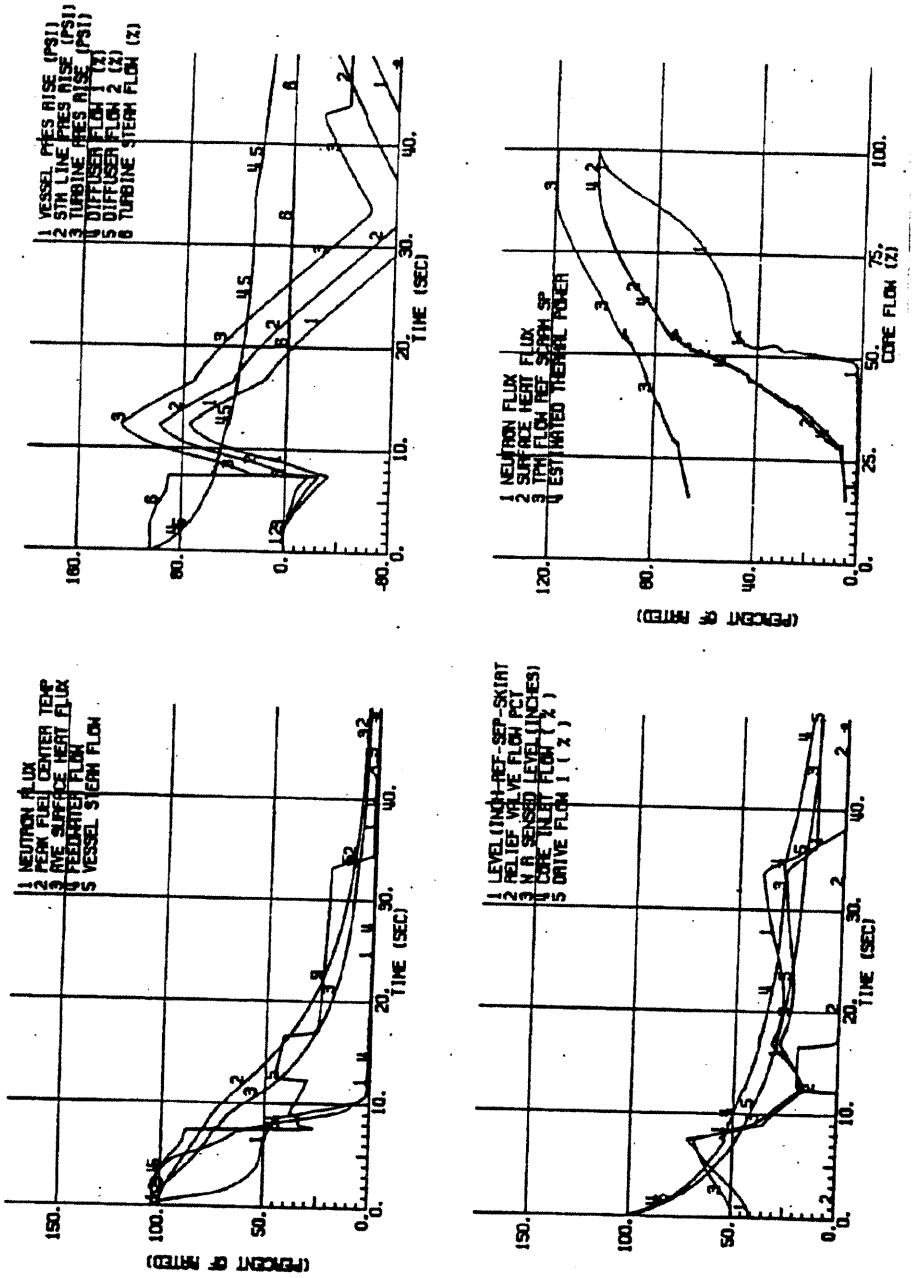

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FIGURE 15.3-2

TRIP OF BOTH RECIRCULATION PUMP MOTORS - 105% POWER

(Typical)

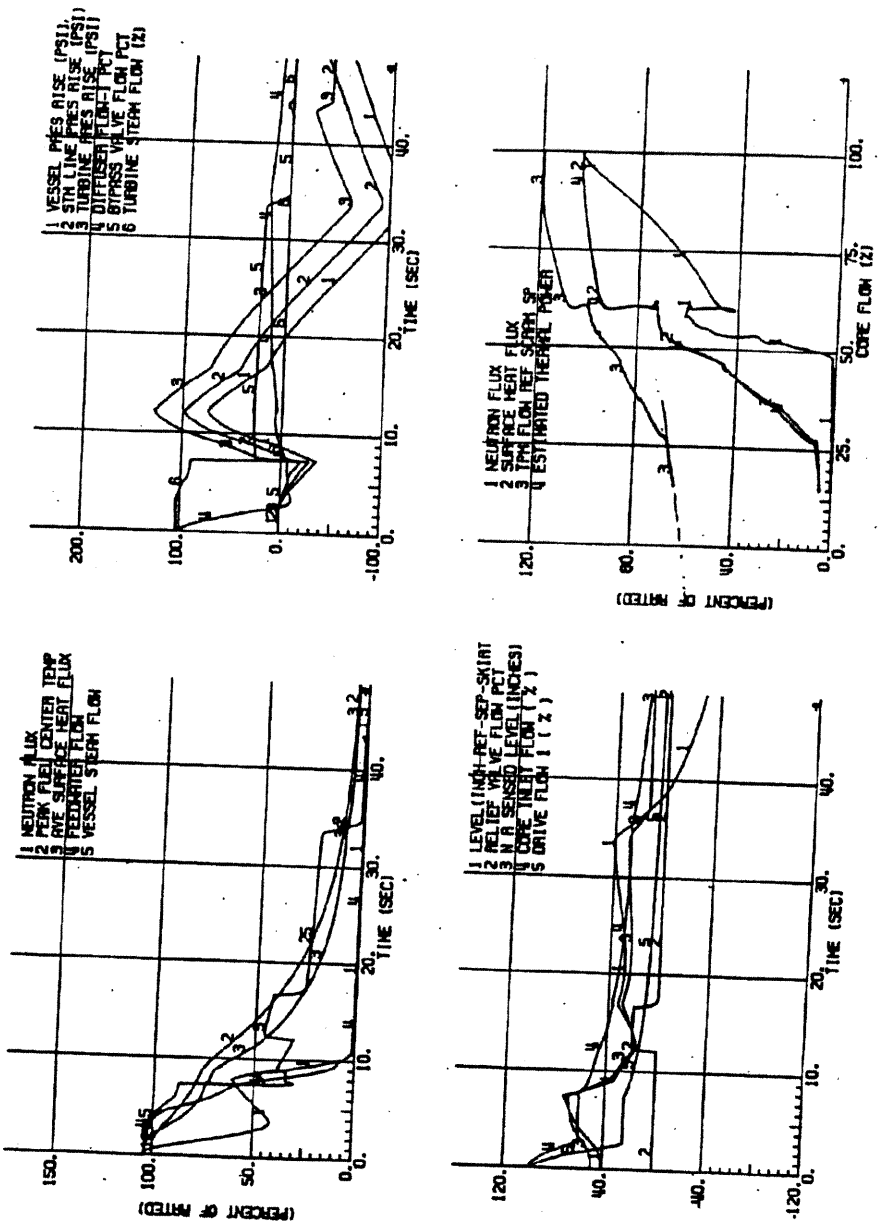

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FIGURE 15.3-3

FAST CLOSURE OF ONE RECIRCULATION FLOW CONTROL VALVE AT 30° / SEC - 105% POWER

(Typical)

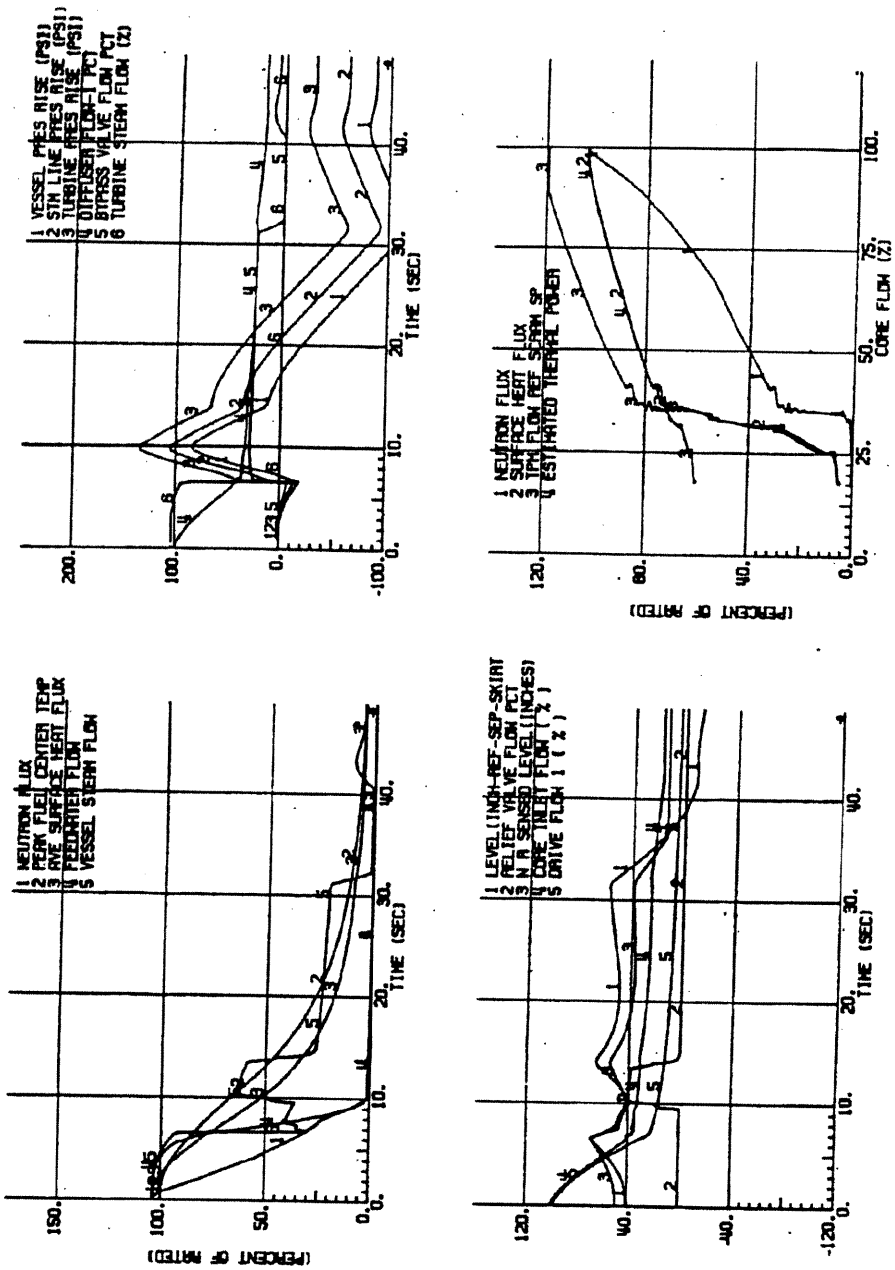

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FIGURE 15.3-4

FAST CLOSURE OF BOTH RECIRCULATION FLOW CONTROL VALVES AT 11° / SEC – 105% POWER

(Typical)

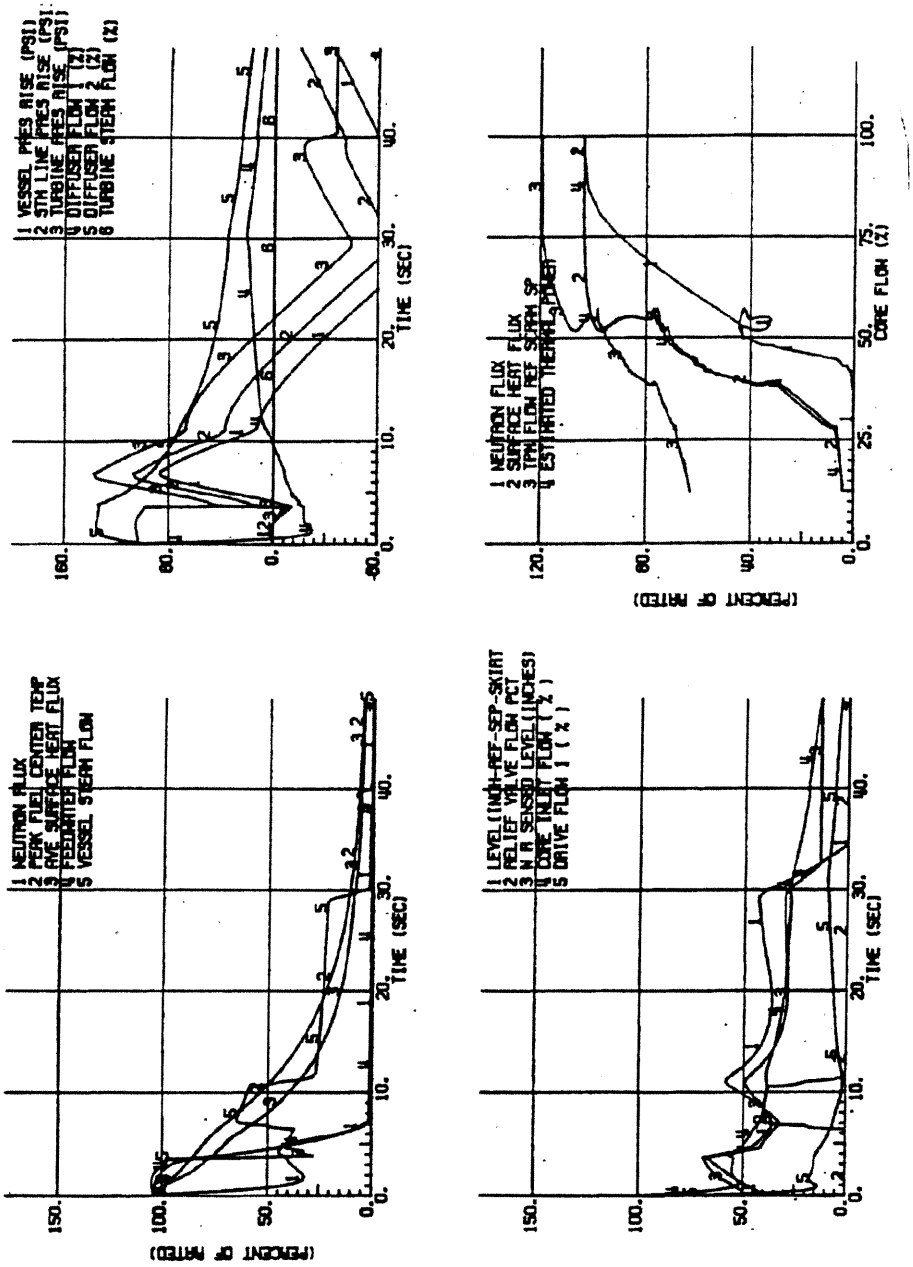

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FIGURE 15.3-5

SEIZURE OF ONE RECIRCULATION PUMP 105% POWER, 100% FLOW

(Typical)

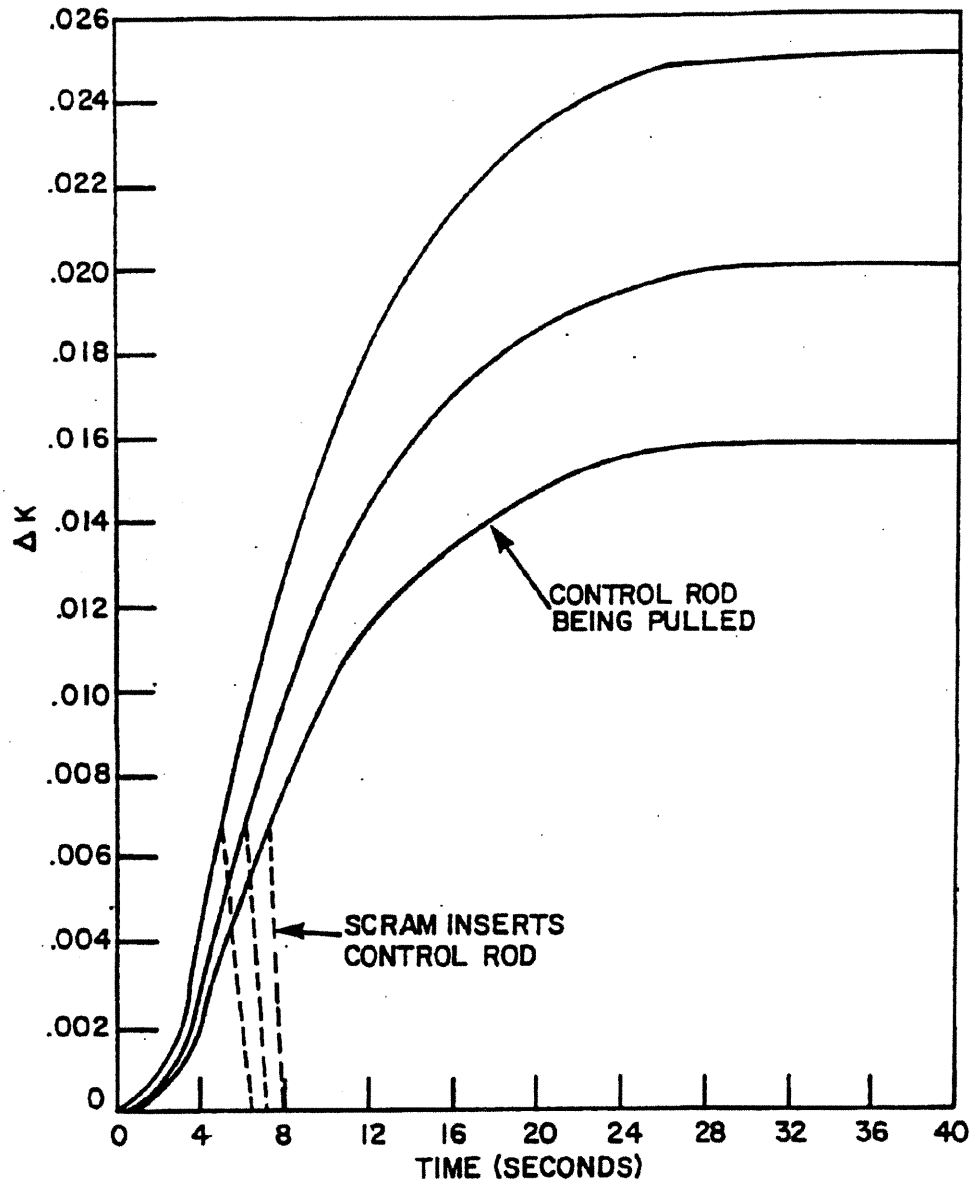

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FIGURE 15.4-1

(INITIAL CORE)
POINT KINETICS CONTROL ROD REACTIVITY INSERTION

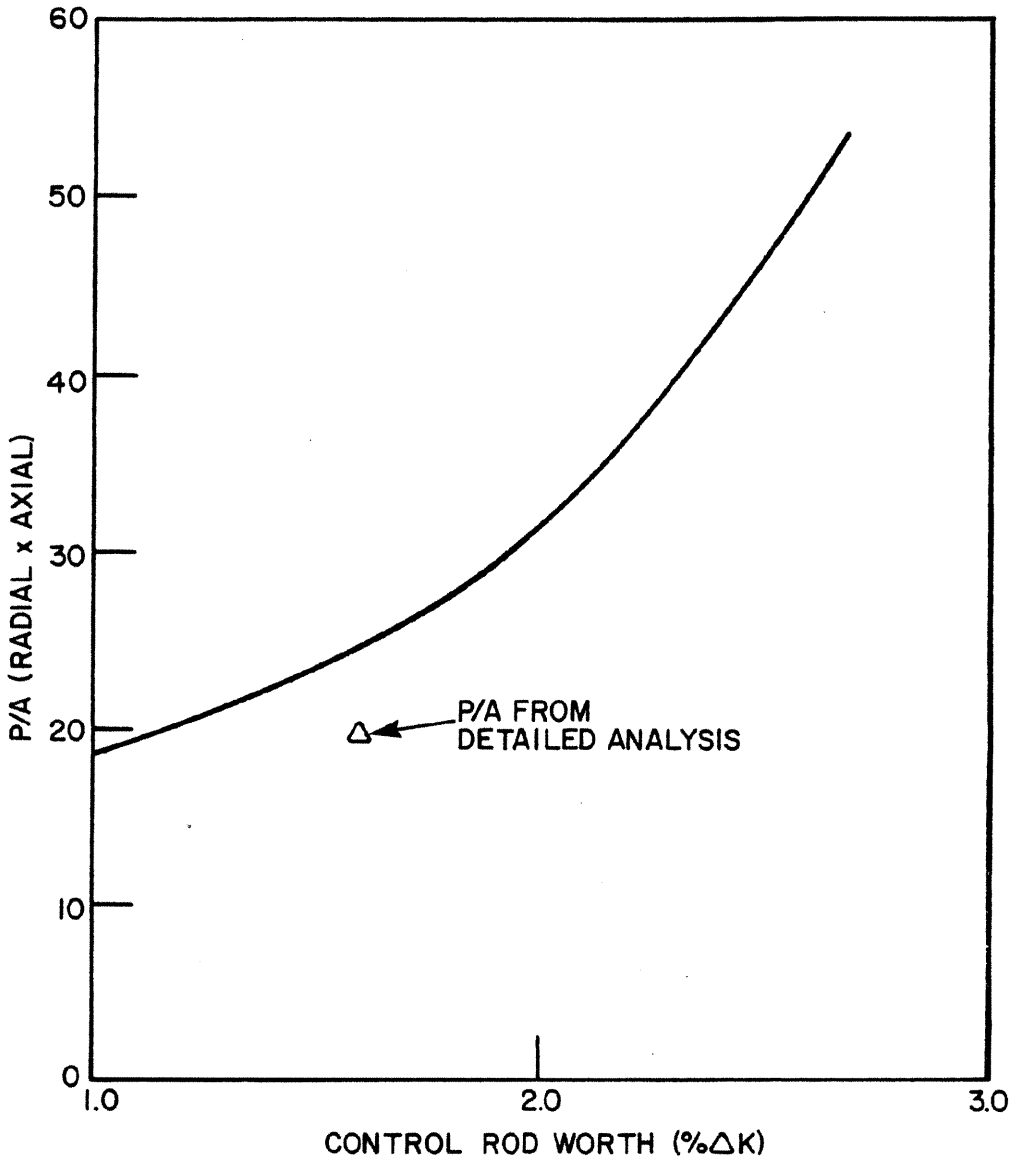

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FIGURE 15.4-2

P/A VS. ROD WORTH NEDO-10527
SUPPLEMENT 1[2] AND DETAILED ANALYSIS

REV. 0 - APRIL 1984

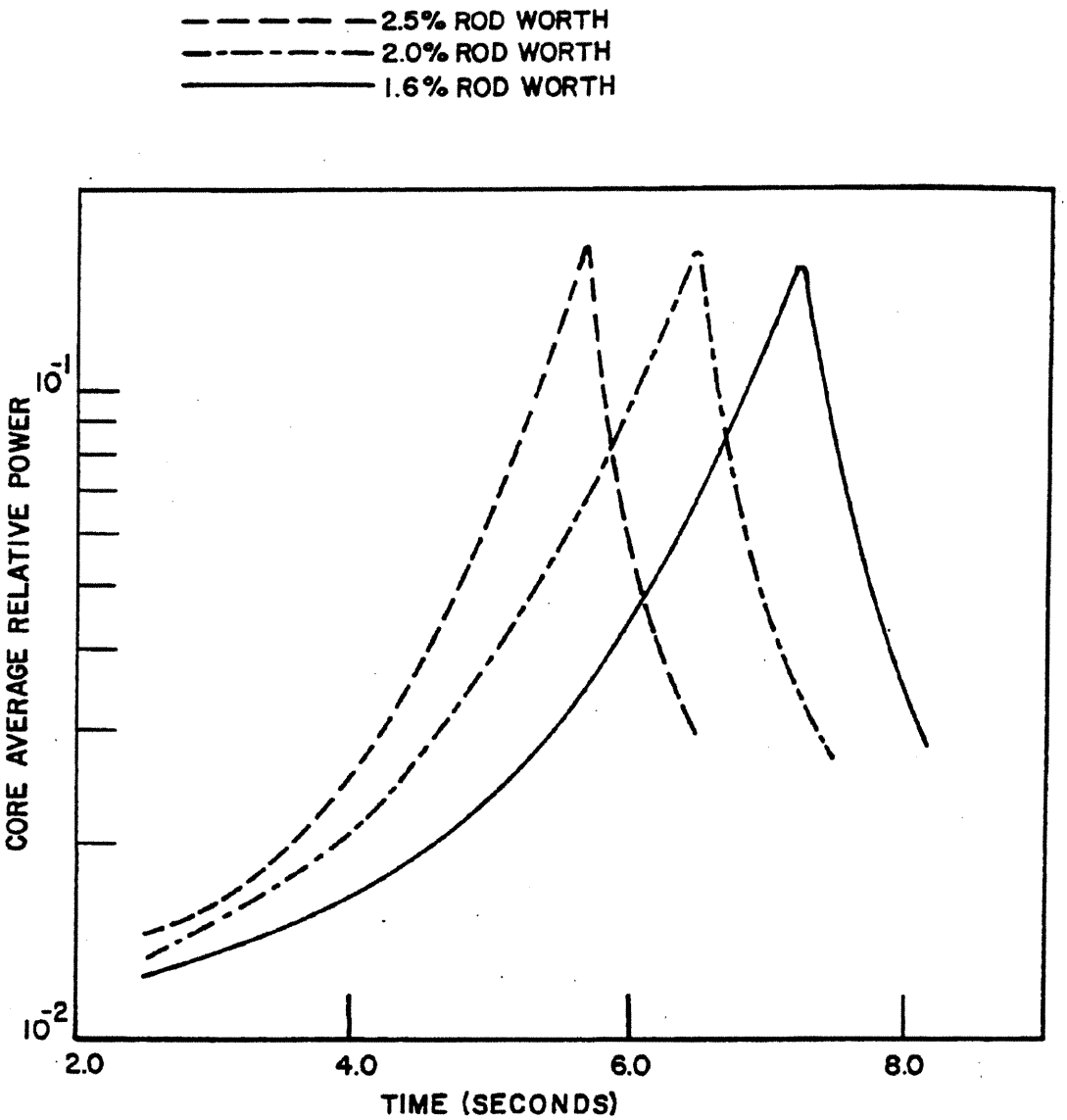

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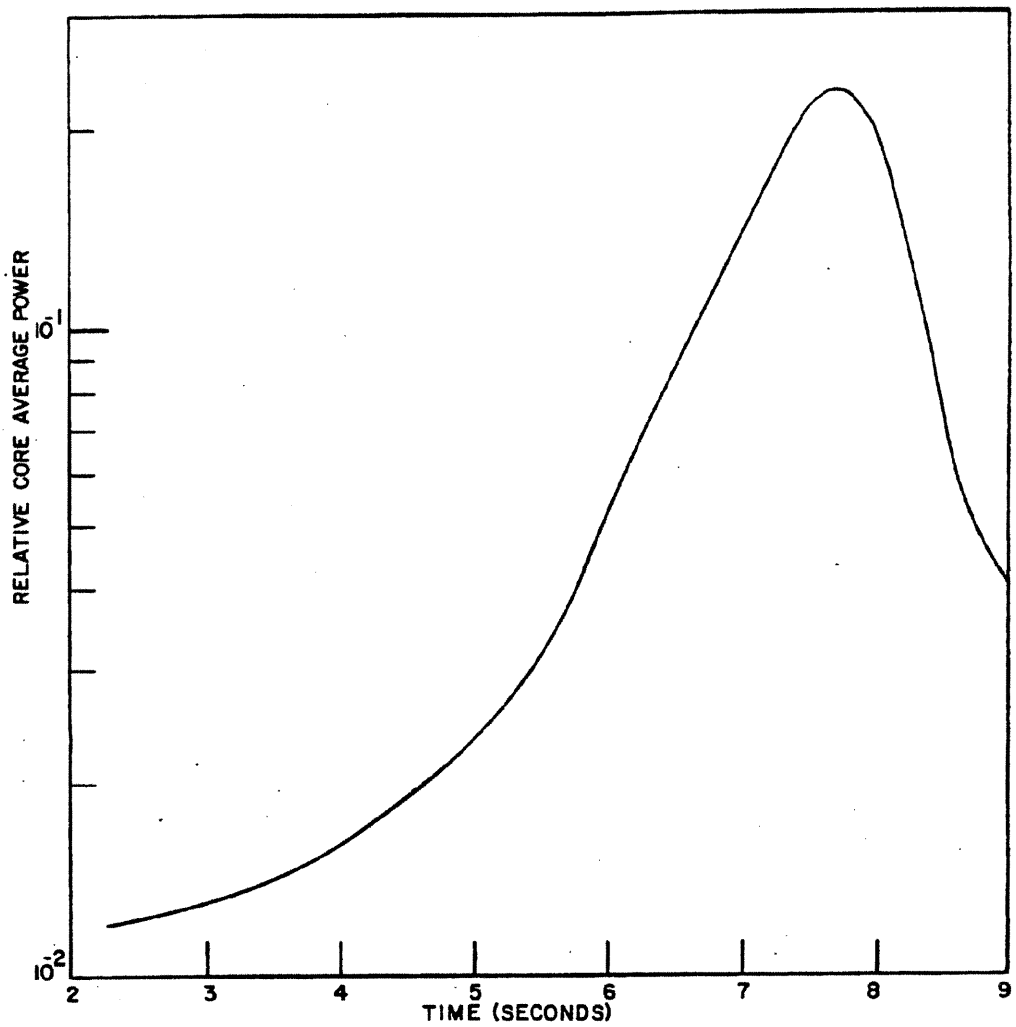
FIGURE 15.4-3

(INITIAL CORE)

CONTINUOUS RWE IN THE STARTUP RANGE CORE AVERAGE POWER VS. TIME FOR 1.6%, 2.0% and 2.5% ROD WORTH'S

(POINT MODEL KINETICS)



ASSUMPTIONS:

1. 1.6% Δk ROD
2. 0.3 fps WITHDRAWAL VELOCITY
3. IRM SCRAM FOR WORST BYPASS CONDITION
4. $P_0 = 10^{-2}$ OF RATED
5. 1967 PRODUCT LILNE ECH SPEC SCRAM RATE
6. EXPOSURE = 0.0 GWD/T

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FIGURE 15.4-4

(INITIAL CORE)
CONTINUOUS CONTROL ROD WITHDRAWAL FROM HOT STARTUP

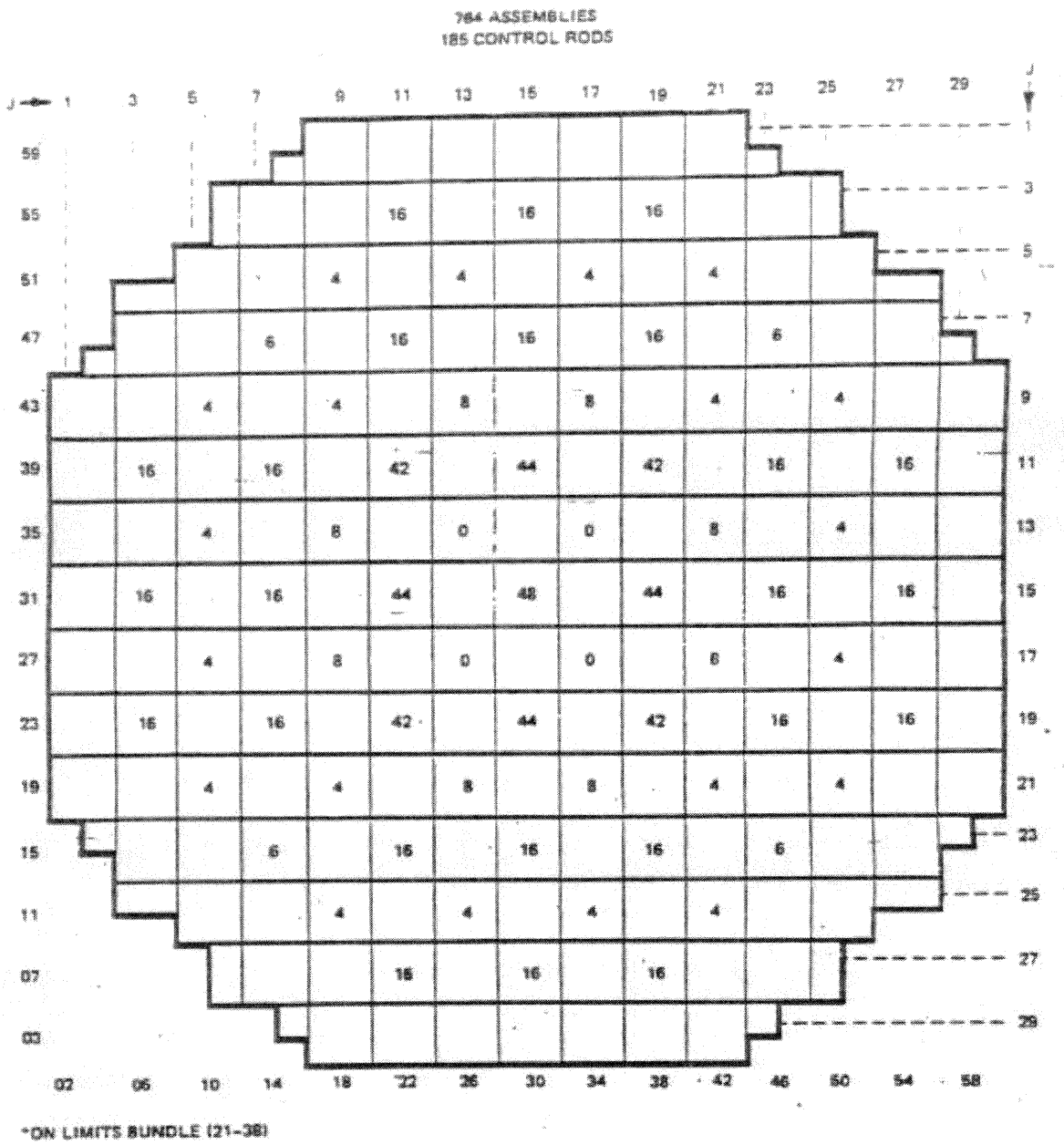

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FIGURE 15.4-5 INITIAL CYCLE ANALYSIS ON LIMITS ROD PATTERN FOR ROD (26 , 35)

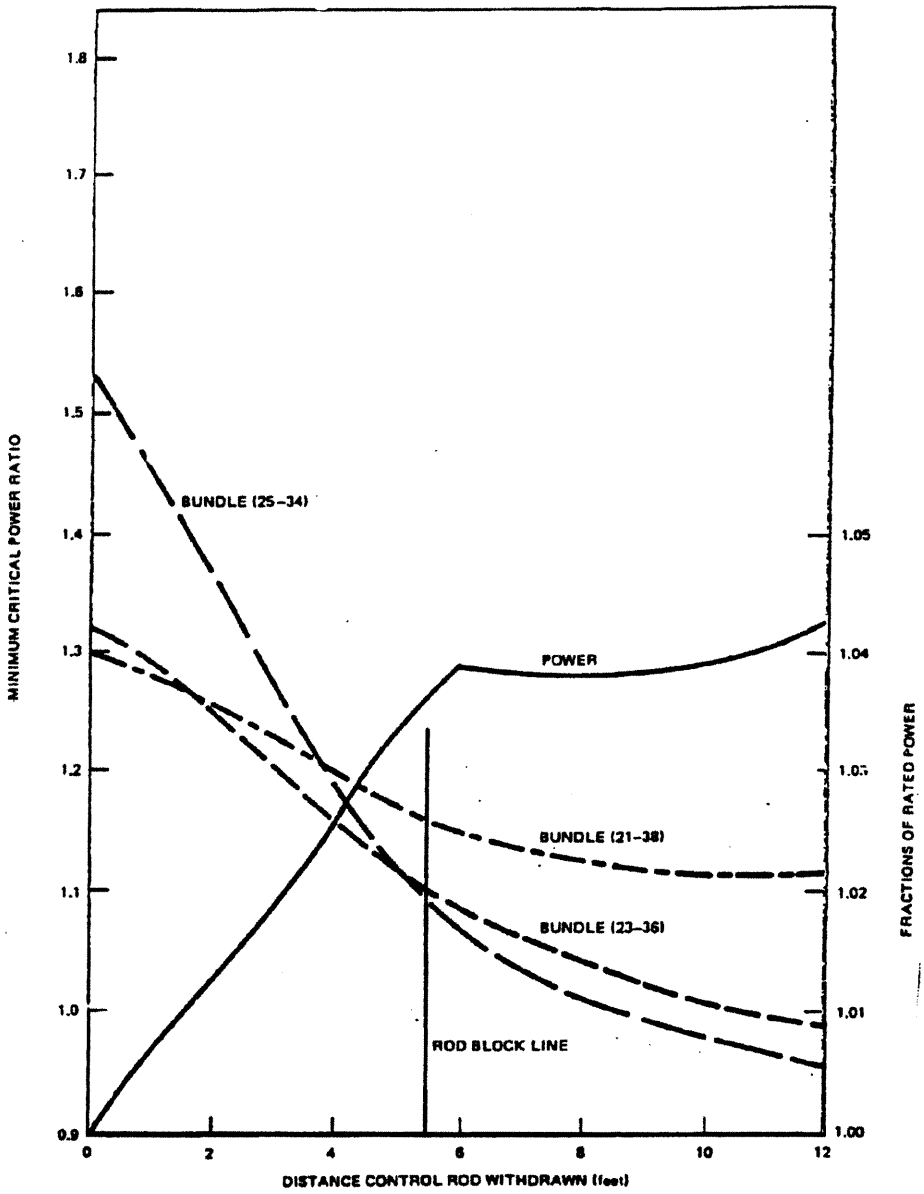

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FIGURE 15.4-6

(INITIAL CORE)
FRACTION OF RATED POWER AND MCPR VS. MCPR VS. DISTANCE ROD (26 , 35) WITHDRAWN

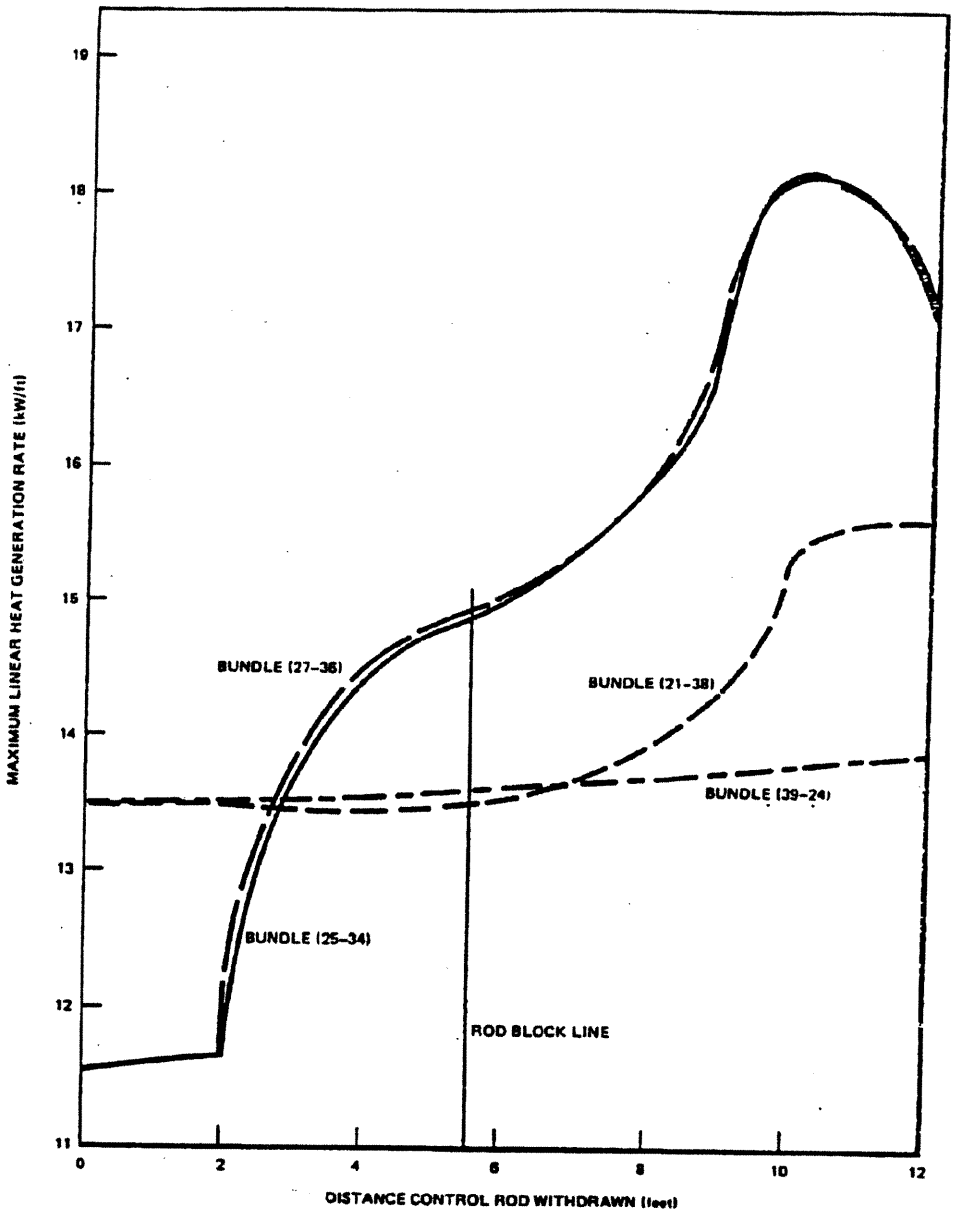

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FIGURE 15.4-7 (INITIAL CORE) MLHGR VS. DISTANCE ROD (26 , 35) WITHDRAWN

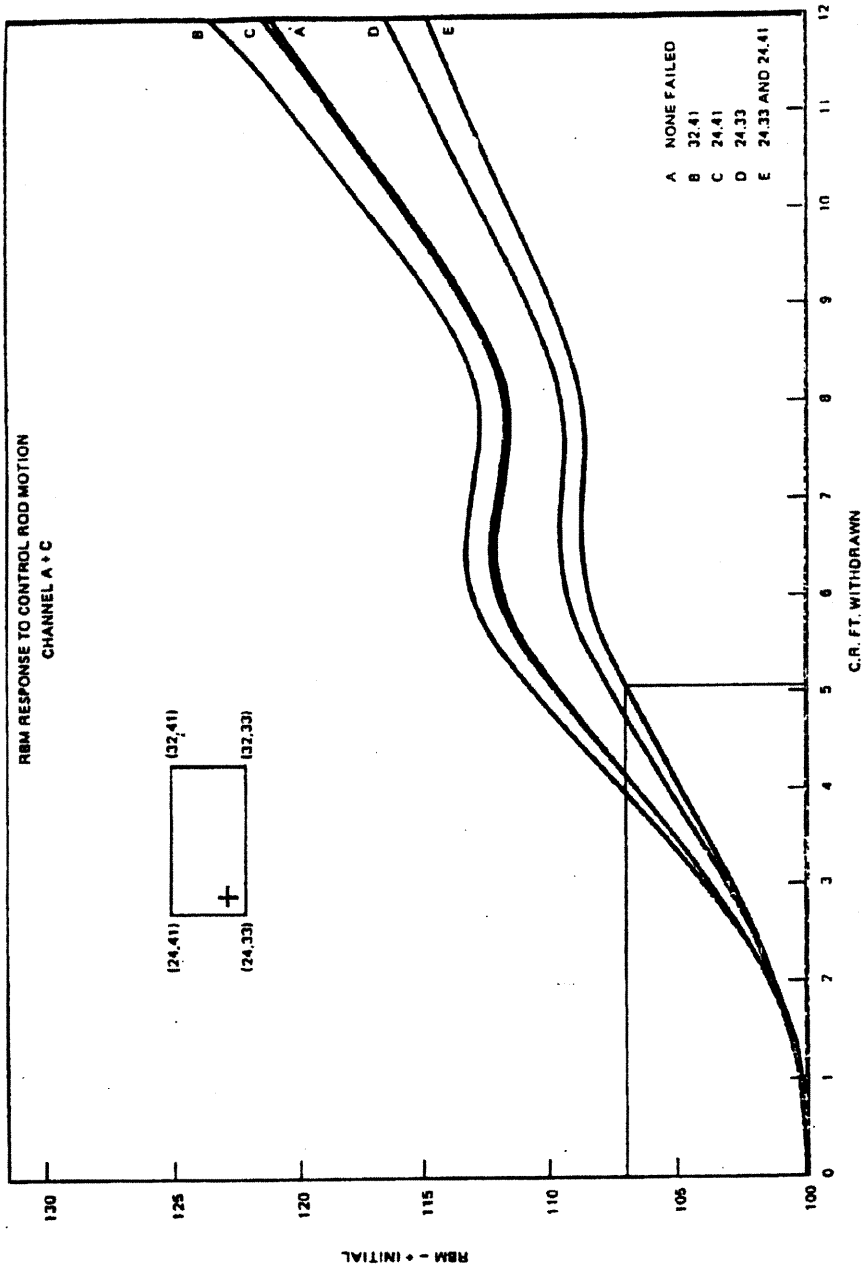

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FIGURE 15.4-8 (INITIAL CORE) RBM RESPONSE TO CONTROL ROD MOTION, CHANNEL A & C

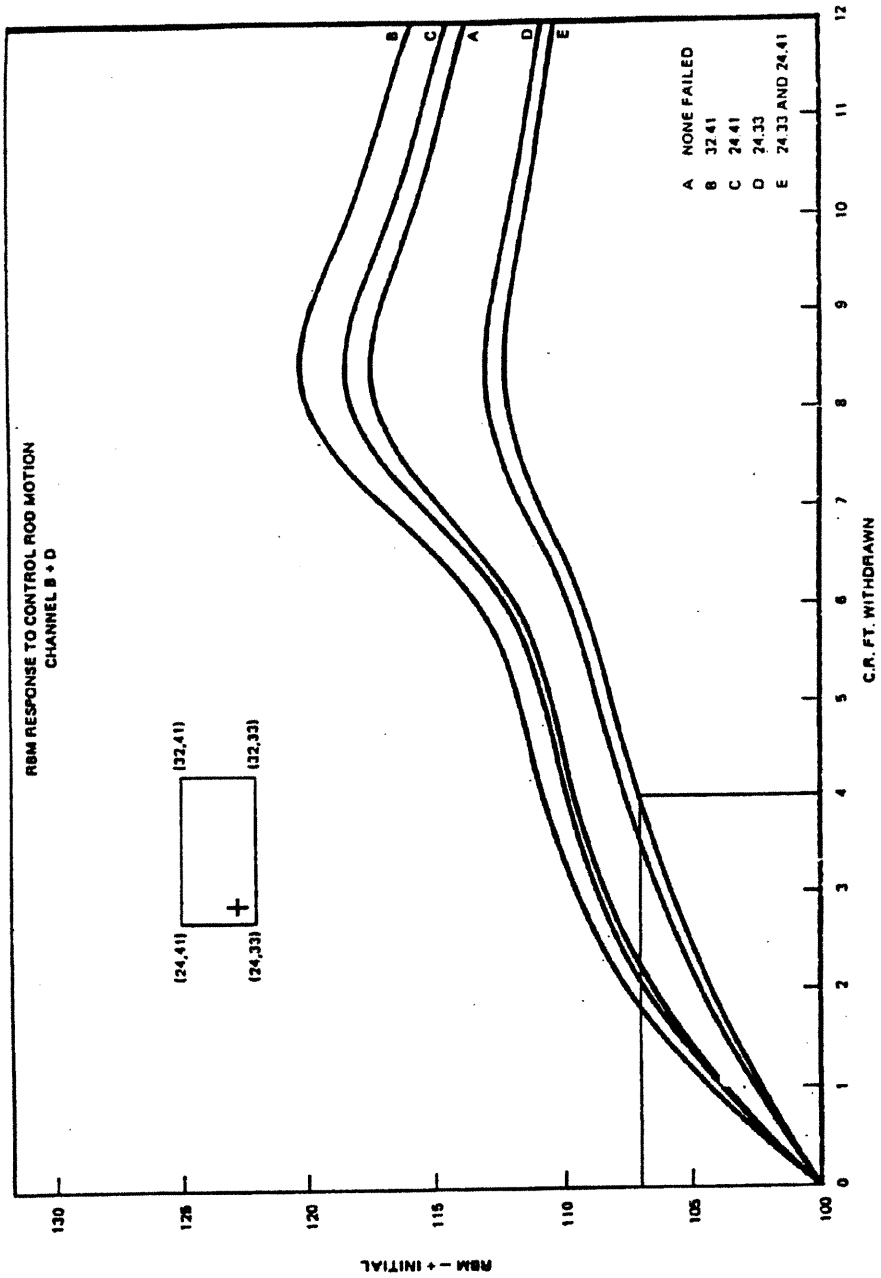

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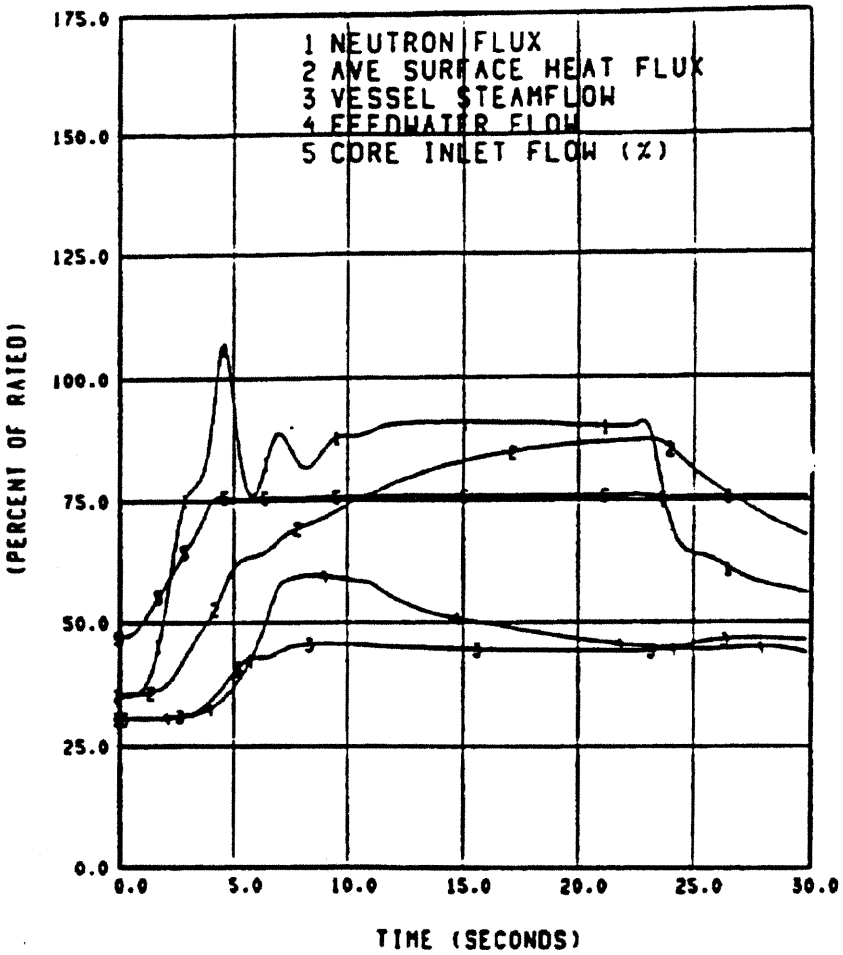
FIGURE 15.4-9 (INITIAL CORE) RBM RESPONSE TO CONTROL ROD MOTION, CHANNEL B & D

IDLE LOOP STARTUP POWER: 35.0% RATED FLOW: 47.0% RATED SNUMB: 0012A



Plots from REDYV04 Output for the Simulation of an Abnormal Idle Recirculation Loop Start-Up Event: (35% Power, 47% Core Flow).

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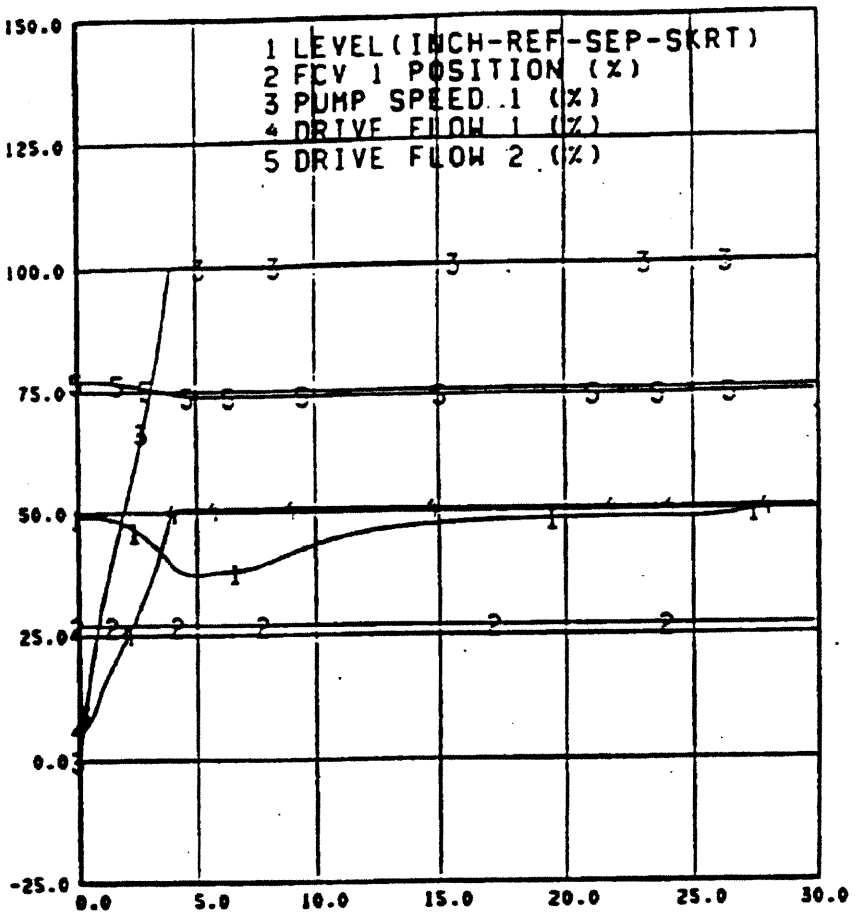
FIGURE 15.4-10a ABNORMAL IDLE RECIRCULATION LOOP START-UP EVENT – 35% POWER, 47% CORE FLOW

IDLE LOOP STARTUP POWER: 35.0% RATED FLOW: 47.0% RATED SNUMB: 0012A



TIME (SECONDS)

PLOTS from REDYV04 Output for the Simulation of an Abnormal Idle Recirculation Loop Start-up Event: (35% Power, 47% Core Flow).
Rated drive flow = 9917 lbm/sec)

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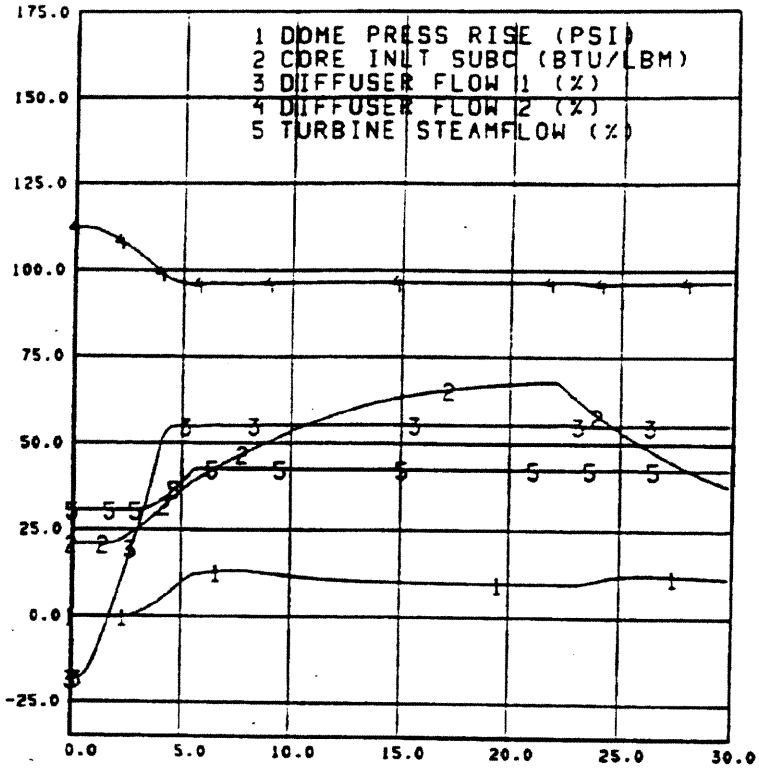
FIGURE 15.4-10b
ABNORMAL IDLE RECIRCULATION LOOP START-UP EVENT – 35% POWER, 47% CORE FLOW

IDLE LOOP STARTUP POWER: 35.0% RATED FLOW: 47.0% RATED SNUB: 0012A



Plots from REDYV04 Output for the Simulation of an Abnormal Idle Recirculation Loop Start-up Event:
(35% Power, 47% Core Flow).

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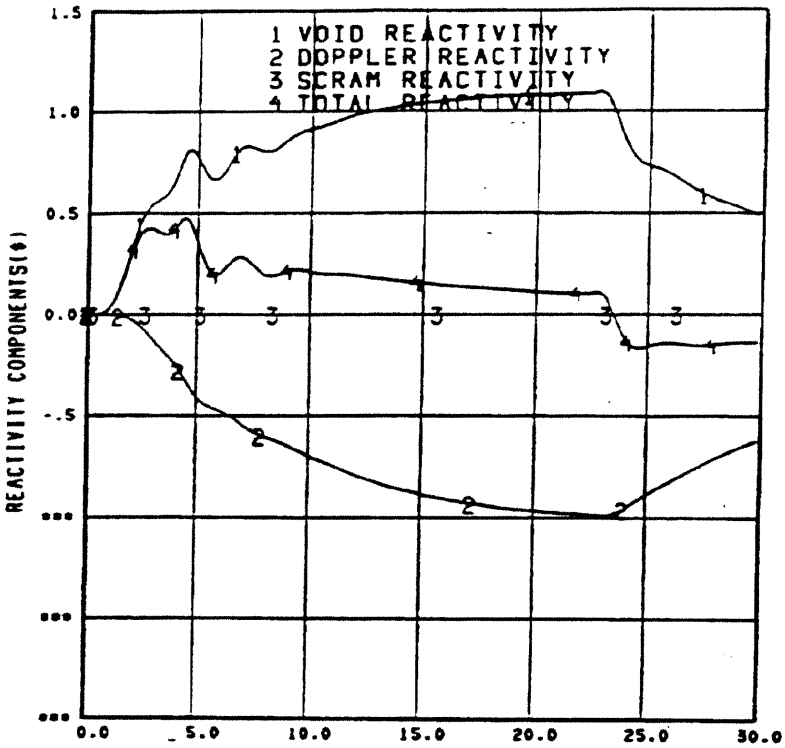
FIGURE 15.4-10c ABNORMAL IDLE RECIRCULATION LOOP START-UP EVENT-35% POWER, 47% CORE FLOW

IDLE LOOP STARTUP POWER: 35.0% RATED FLOW: 47.0% RATED SNUB: 0012A



Plots from REDYV04 Output for the Simulation of an Abnormal Idle Recirculation Loop Start-up Event: (35% Power, 47% Core Flow).

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FIGURE 15.4-10d ABNORMAL IDLE RECIRCULATION LOOP START-UP EVENT-35% POWER, 47% CORE FLOW

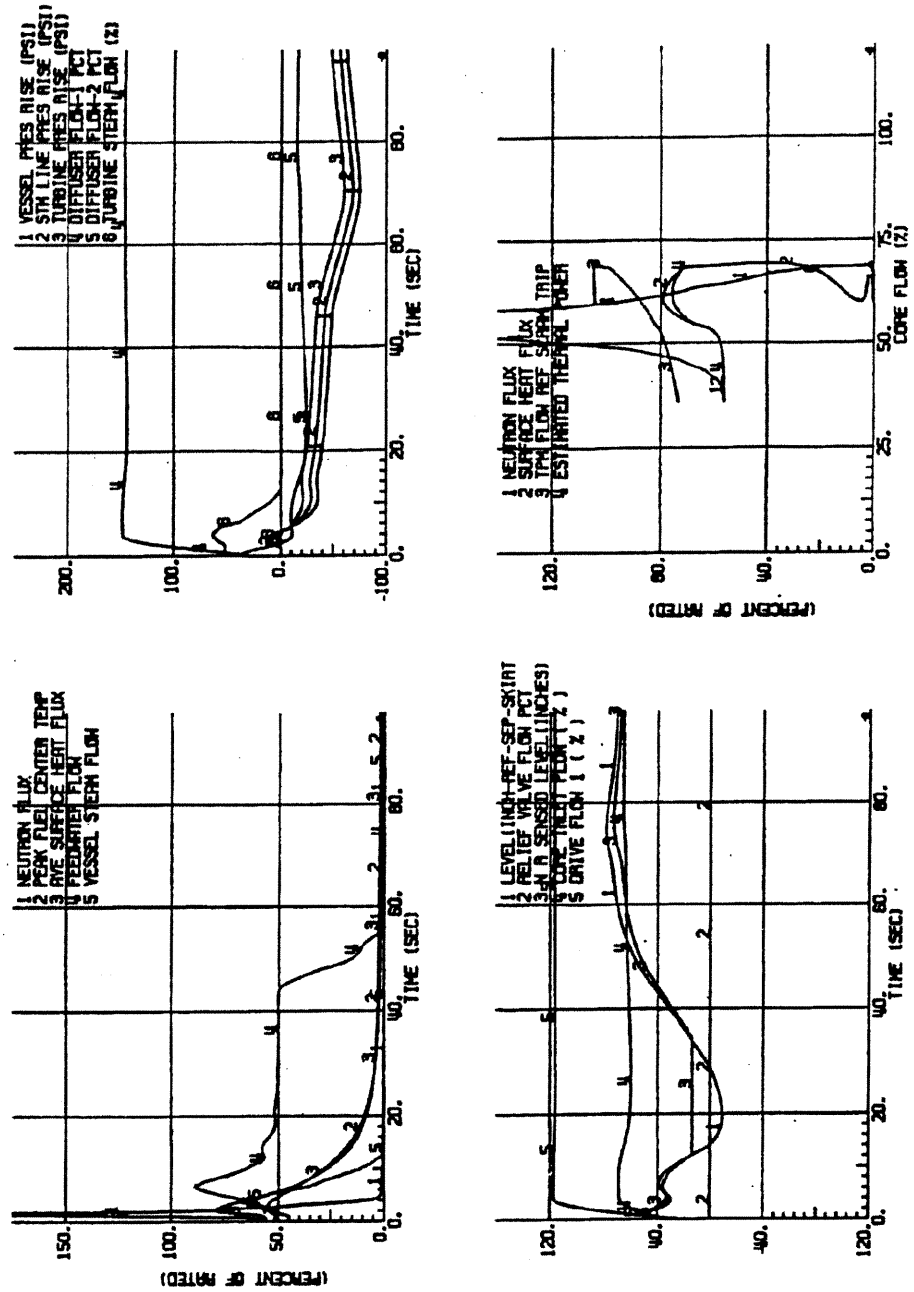

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FIGURE 15.4-11

FAST OPENING OF ONE RECIRCULATION VALVE 30%/SEC AT 58% POWER, 35% FLOW

(Typical)

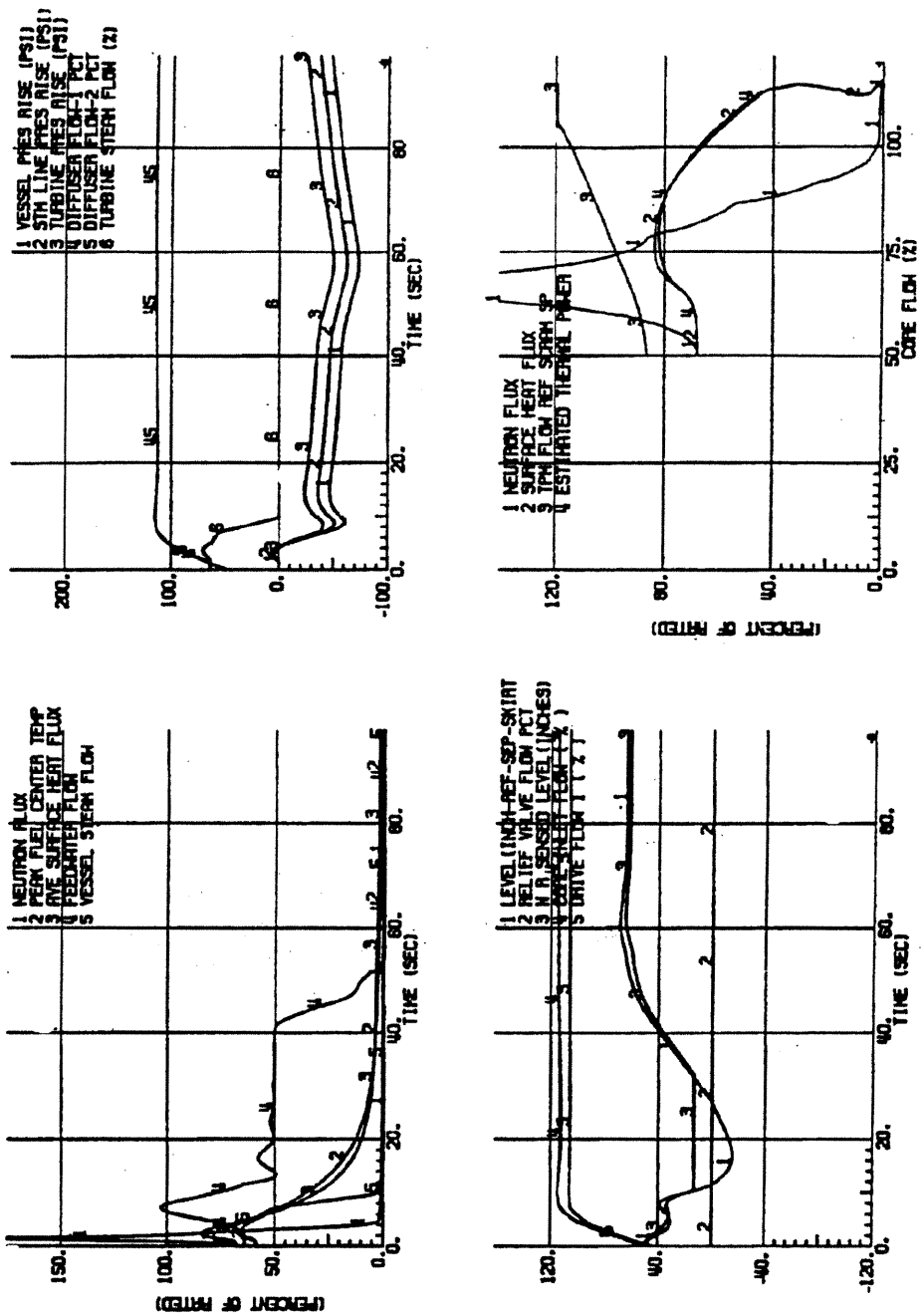

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FIGURE 15.4-12

FAST OPENING OF BOTH RECIRCULATION FLOW CONTROL VALVES 11%/SEC AT 68% POWER 50% FLOW

(Typical)

(THIS FIGURE INTENTIONALLY LEFT BLANK)

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FIGURE 15.4-13 CRITICAL ROD PATTERN AND FUEL BUNDLE EXCHANGE LOCATIONS FOR MISPLACED BUNDLE ACCIDENT 0.0 Gwd/t

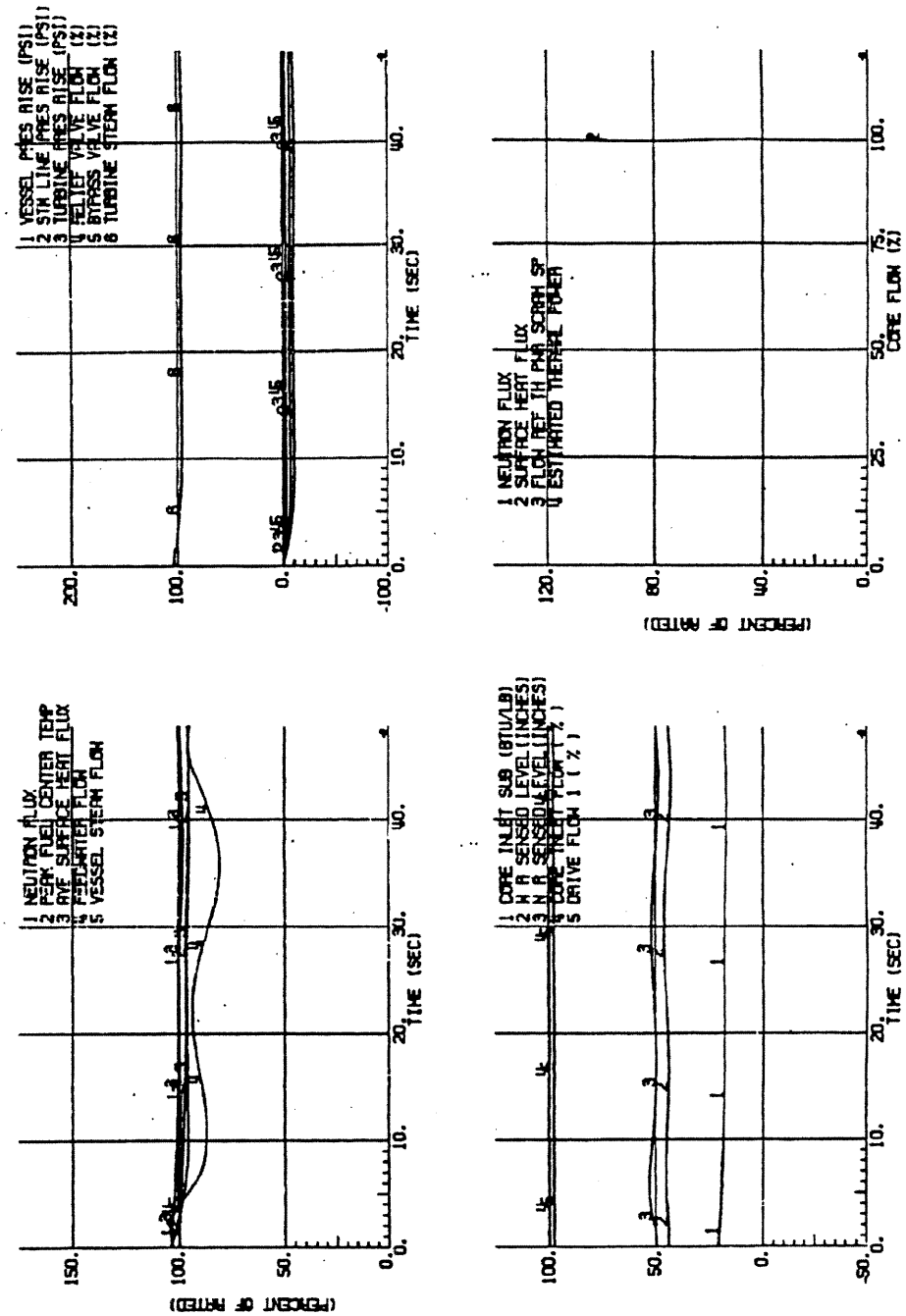

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FIGURE 15.5-1

INADVERTENT PUMP START OF HPCS PUMP
(GE, Typical)

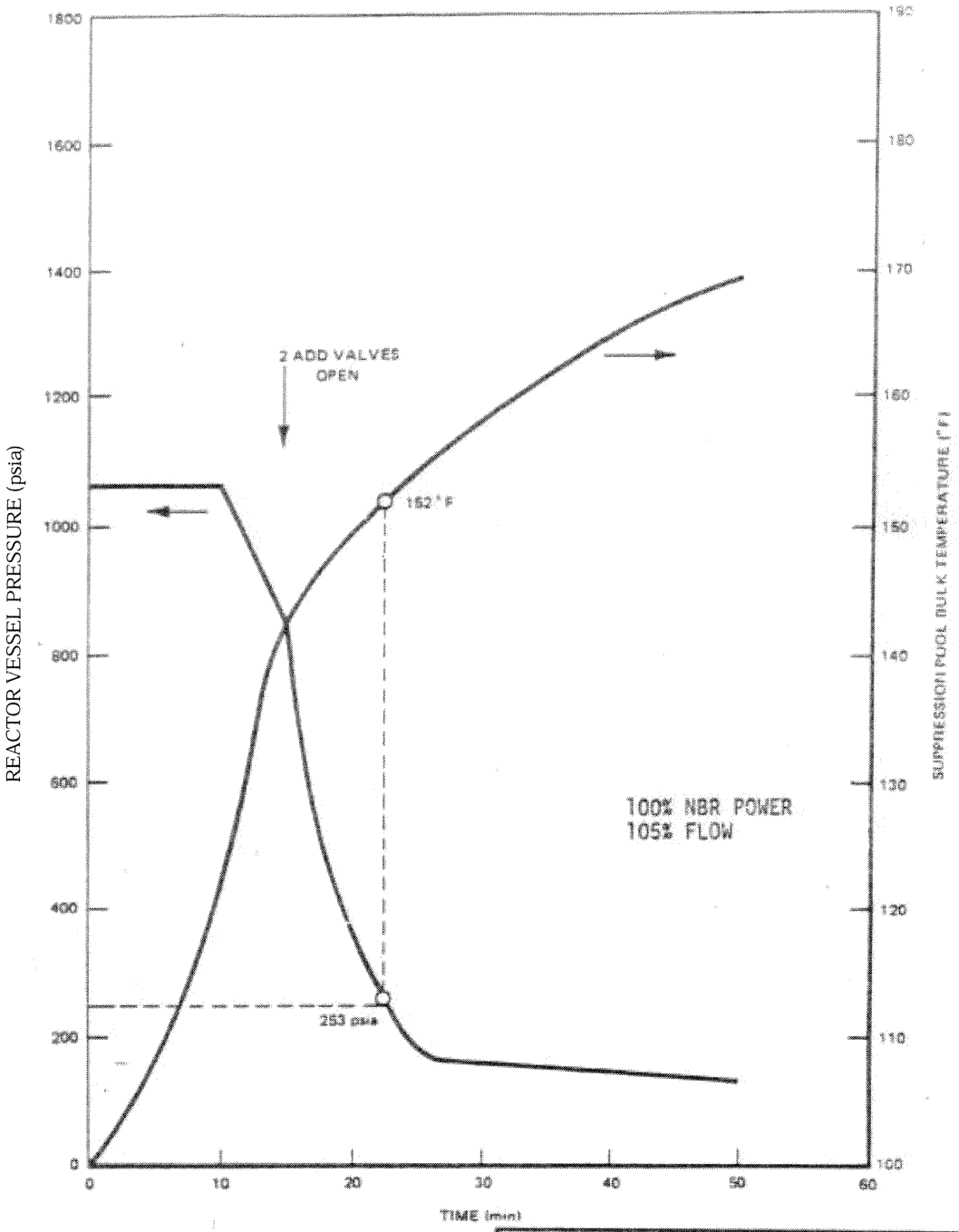

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FIGURE 15.6-1 INITIAL CYCLE ANALYSIS SUPPRESSION POOL TEMPERATURE RESPONSE STUCK OPEN RELIEF VALVE FROM POWER OPERATION

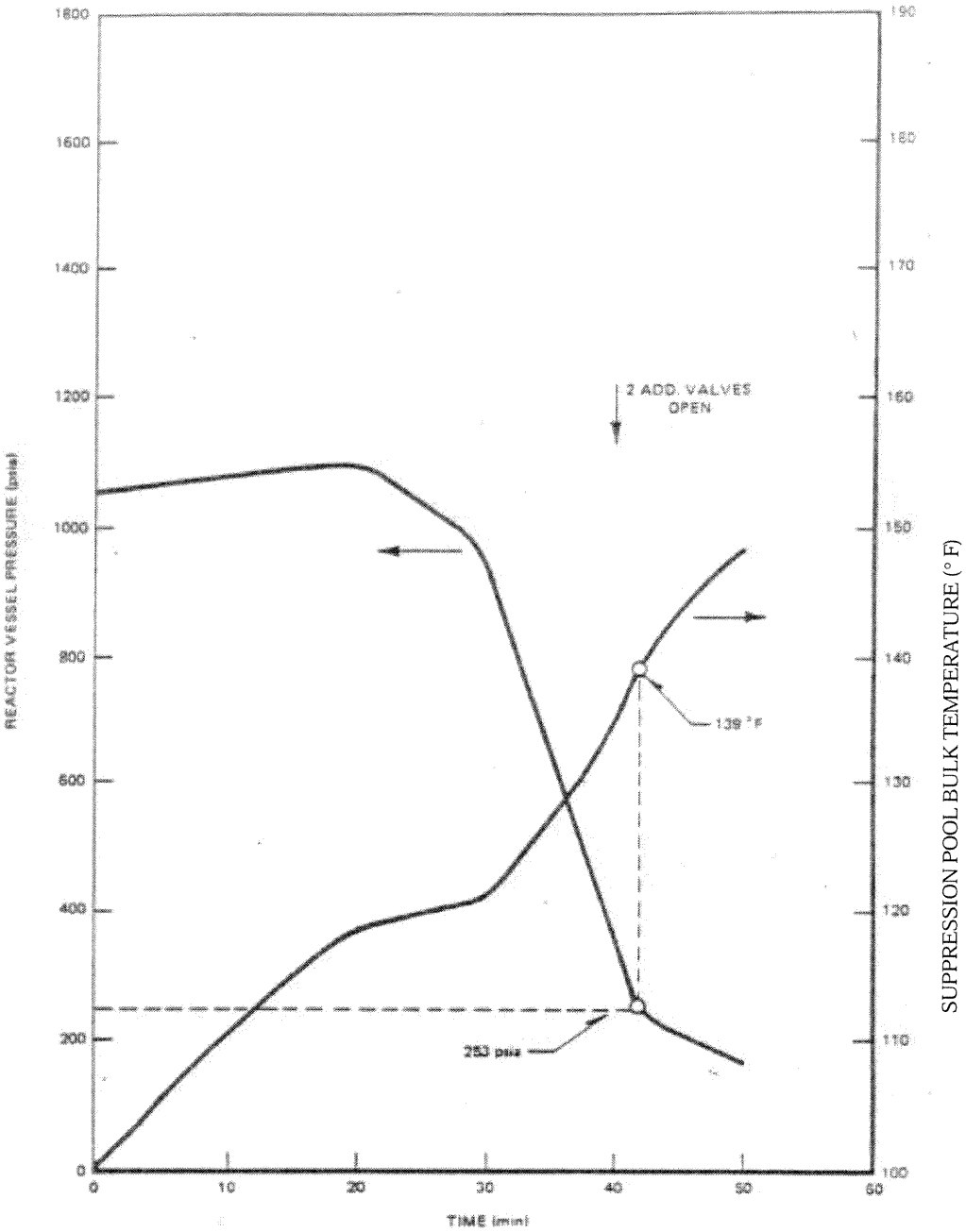

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FIGURE 15.6-2 INITIAL CYCLE ANALYSIS SUPPRESSION POOL TEMPERATURE RESPONSE STUCK OPEN RELIEF VALVE FROM HOT STANDBY

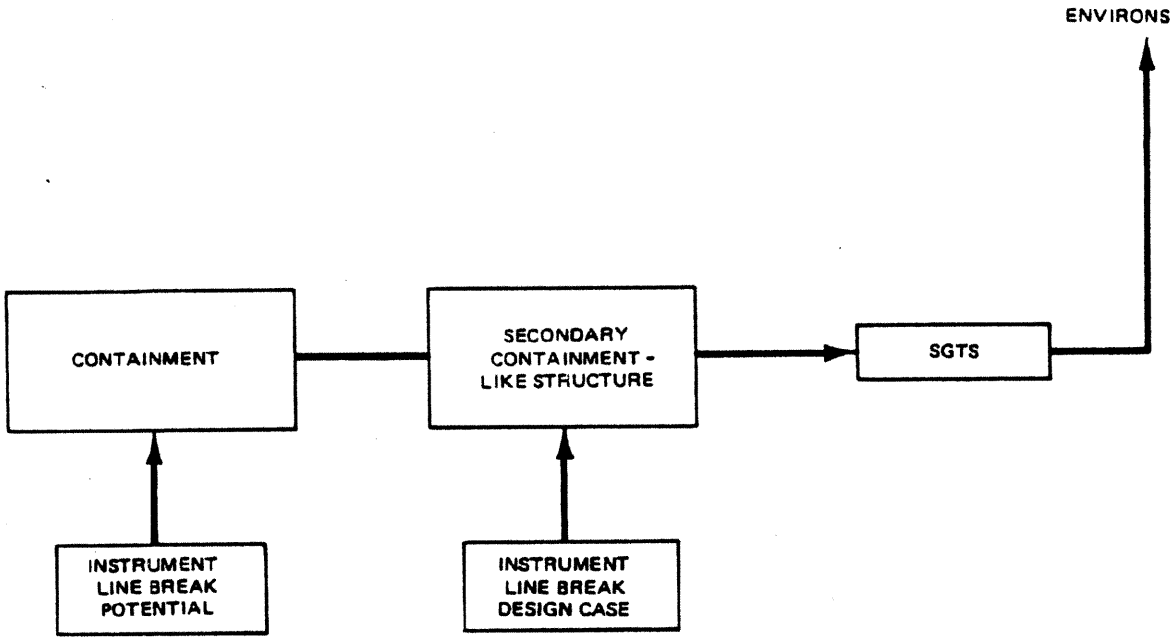

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FIGURE 15.6-3

LEAKAGE PATH FOR INSTRUMENT LINE BREAK

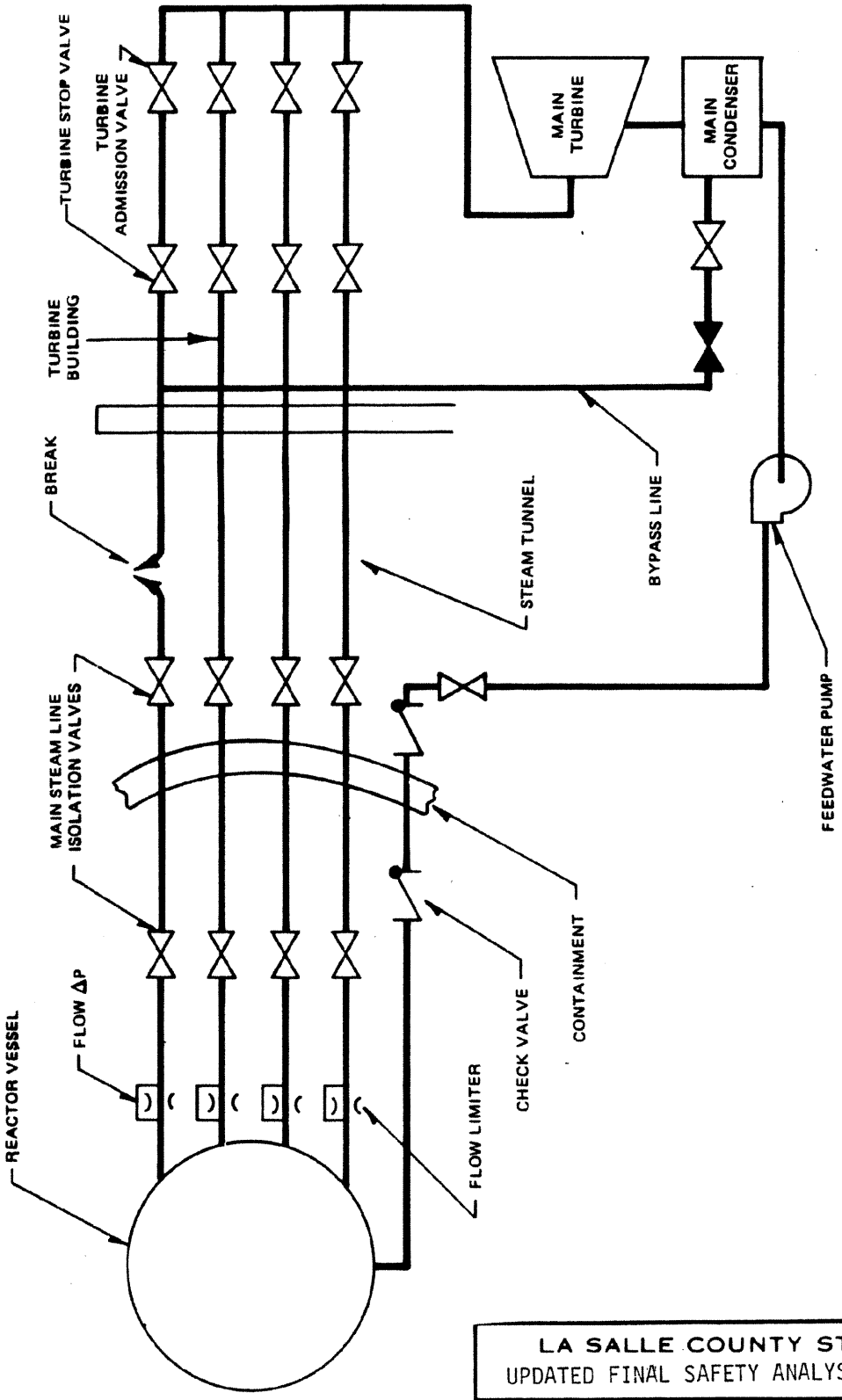

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FIGURE 15.6-4

STEAM FLOW SCHEMATIC FOR STEAM BREAK OUTSIDE CONTAINMENT

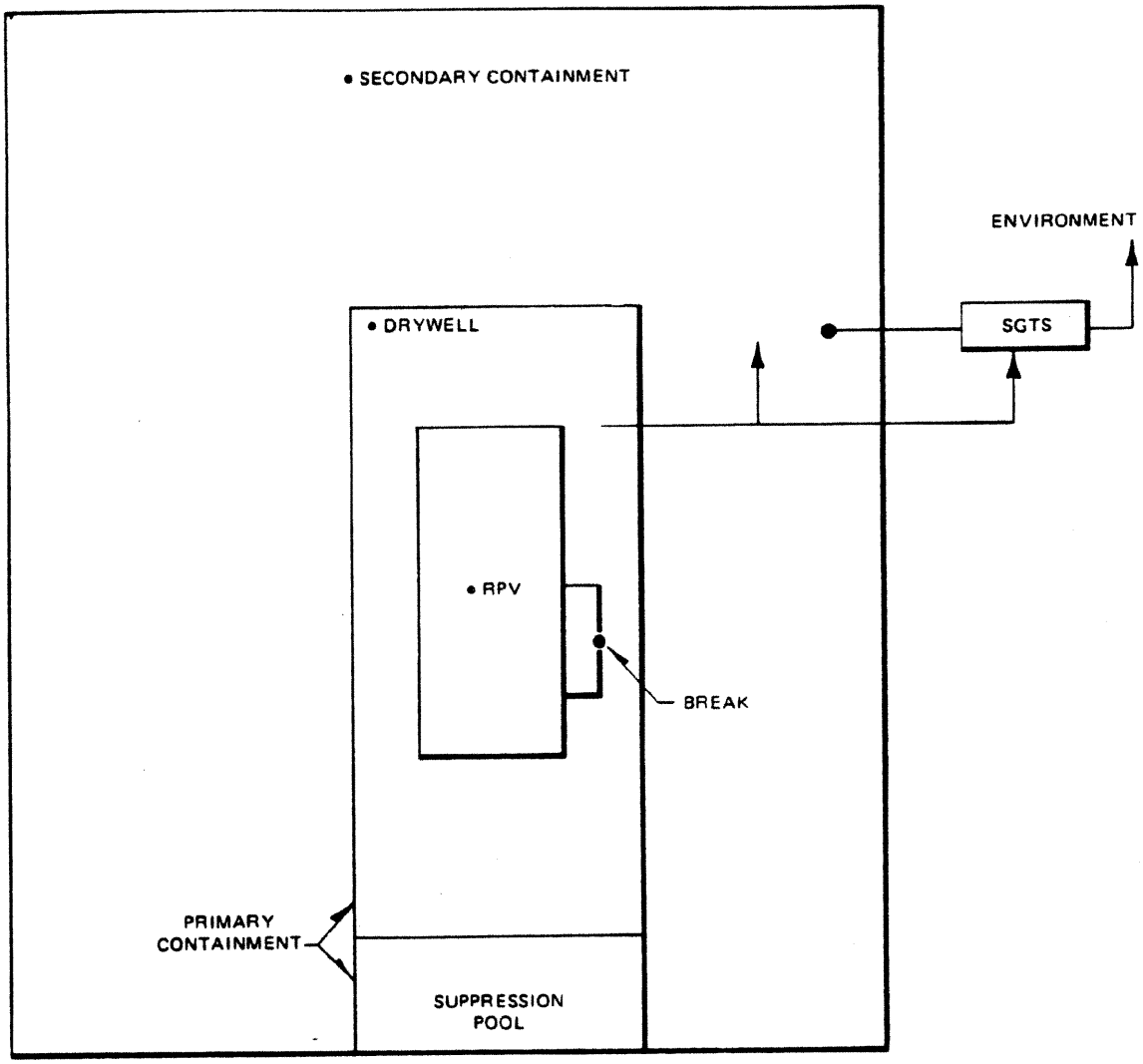

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FIGURE 15.6-5

LEAKAGE FLOW FOR LOCA

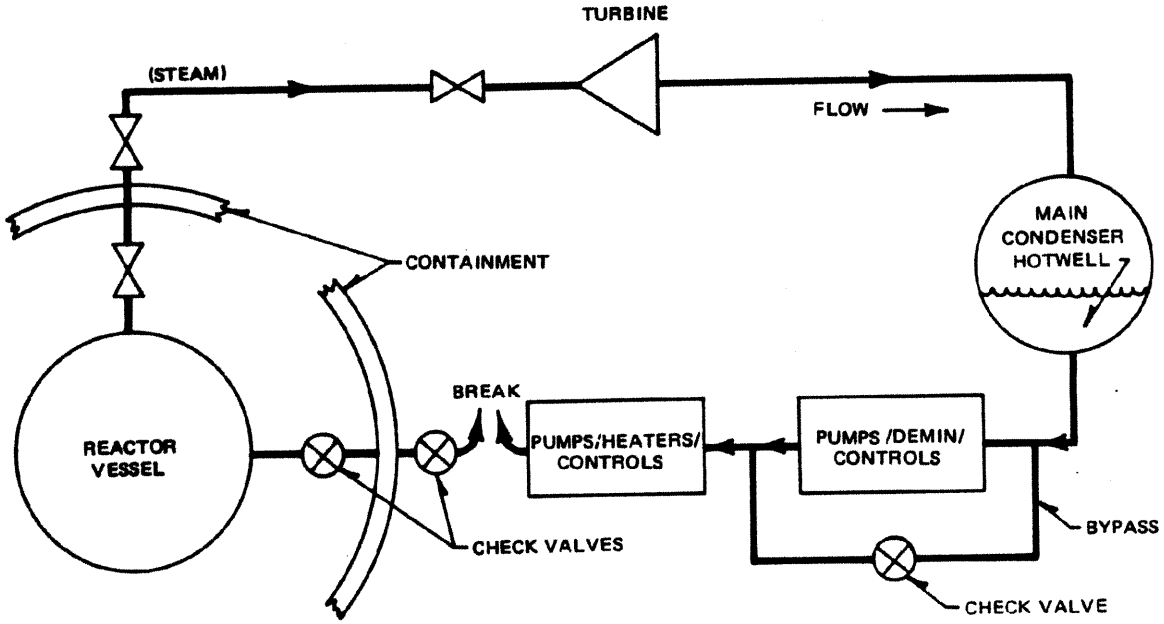

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FIGURE 15.6-6

LEAKAGE PATH FOR FEEDWATER LINE BREAK
OUTSIDE CONTAINMENT

REV. 0 - APRIL 1984

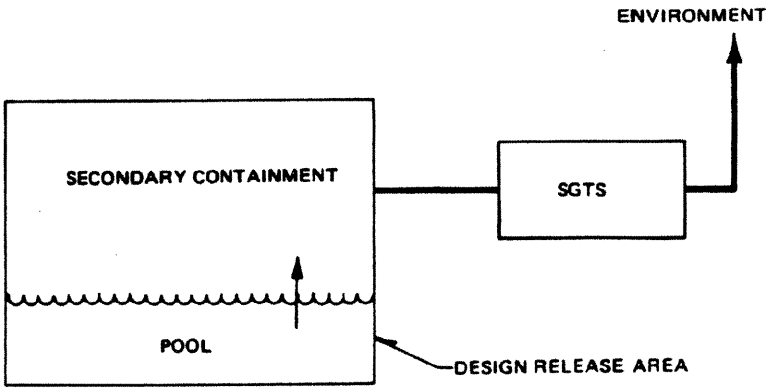

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FIGURE 15.7-1

LEAKAGE PATH FOR FUEL HANDLING ACCIDENT

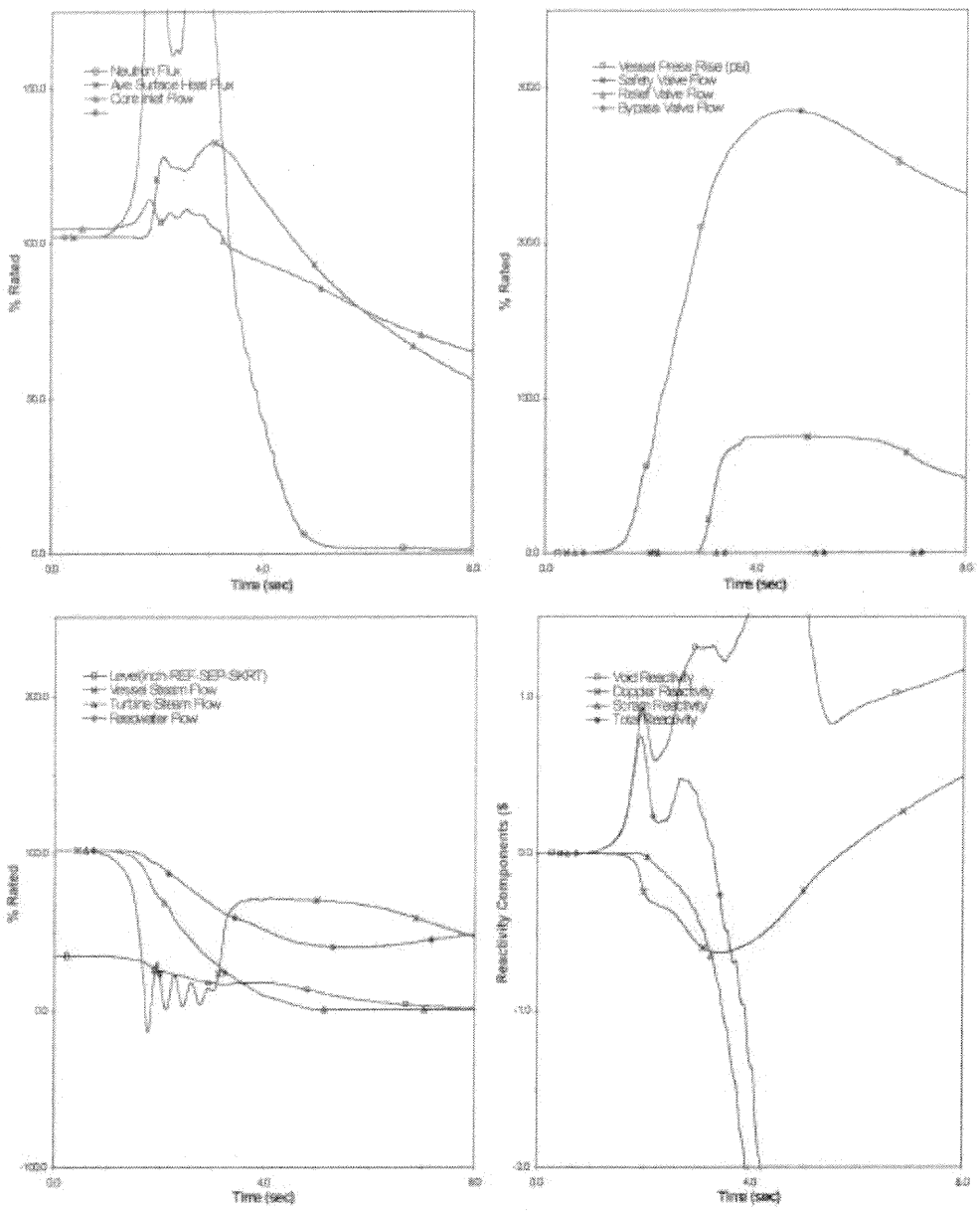

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FIGURE 15.B-1

RESPONSE TO MSIV CLOSURE WITH FLUX SCRAM
(102%uprated power, 105% core flow and
1035 psia initial dome pressure)
Source of information: Reference 59

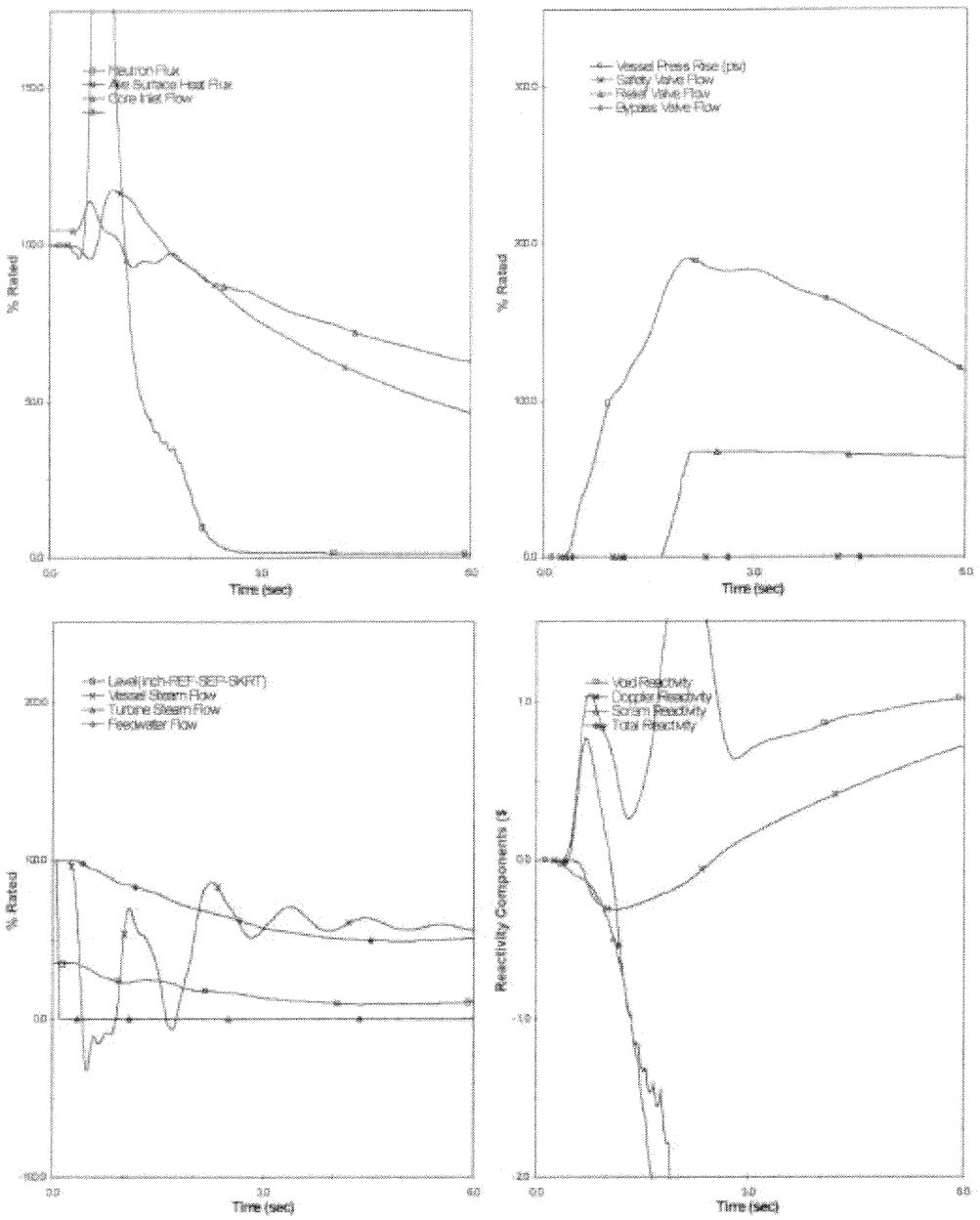

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FIGURE 15.B-2 TURBINE TRIP WITH BYPASS FAILURE (@ 100% Uprated Power, & 105% Core Flow) Source of information: Reference 59

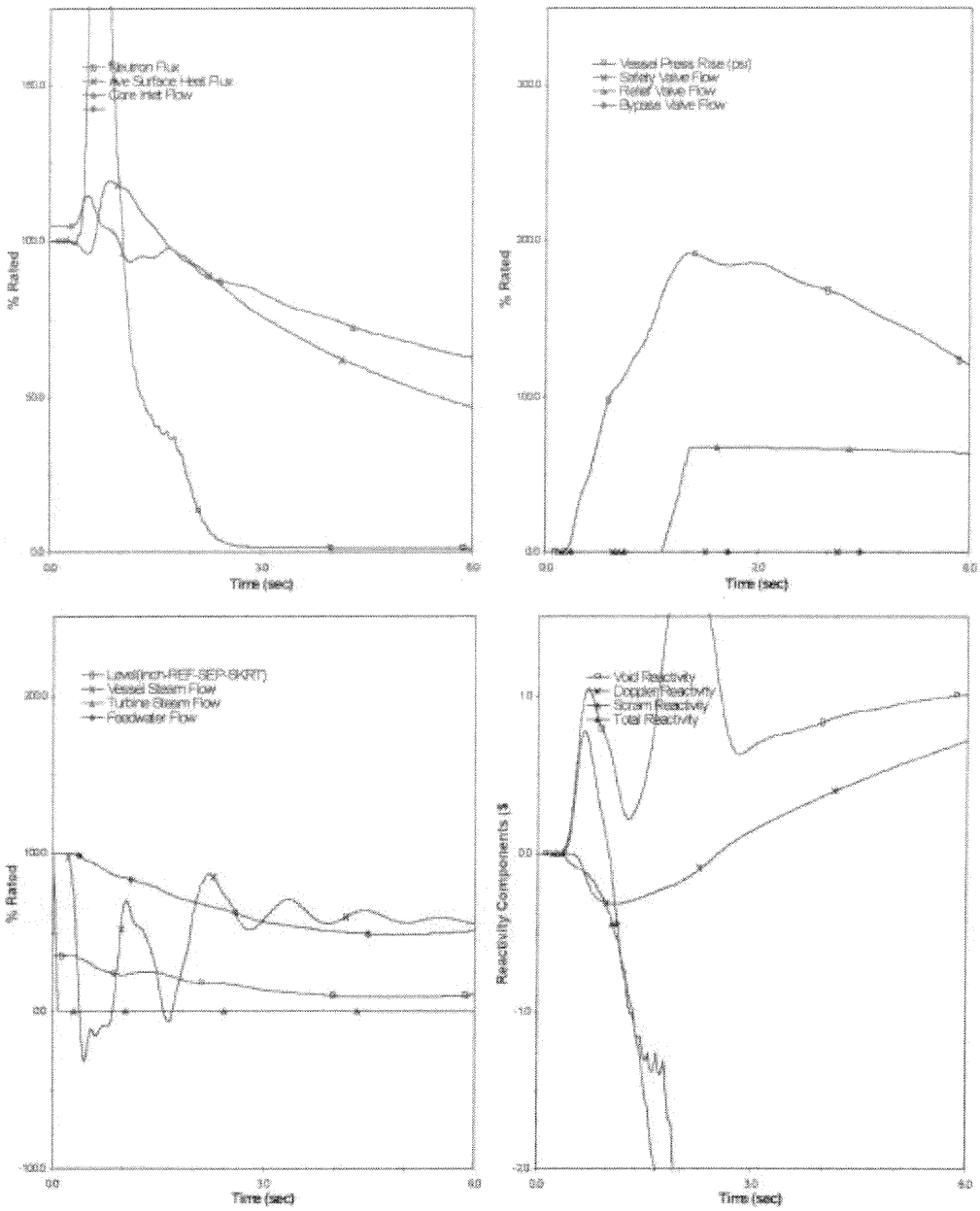

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FIGURE 15.B-3 GENERATO R LOAD REJECTION WITH BYPASS FAILURE (@ 100% Uprated Power, & 105% Core Flow) Source of information: Reference 59

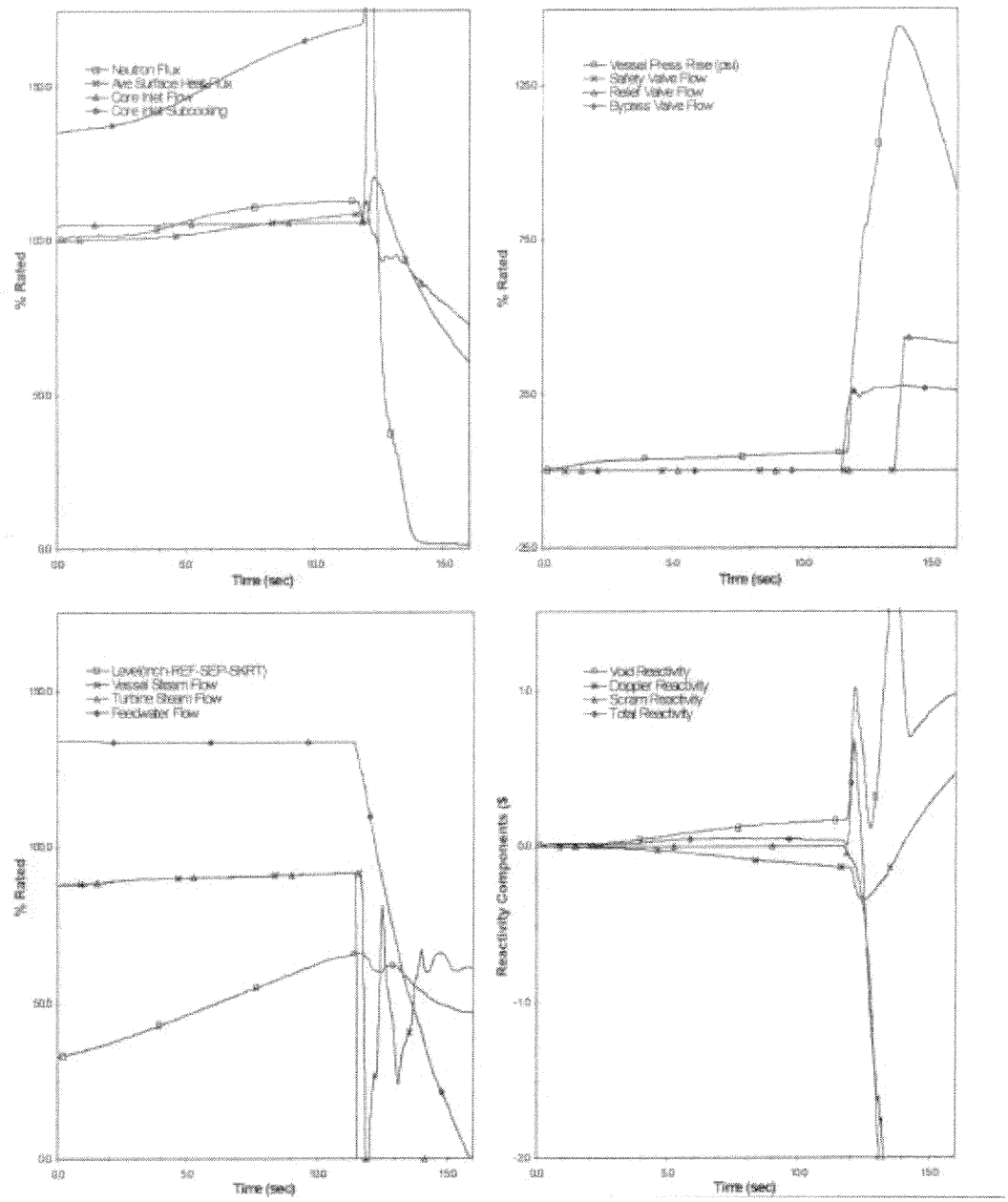

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FIGURE 15.B-4 FEEDWATER CONTROLER FAILURE – MAXIMUM DEMAND (@ 100% Uprated Power, & 105% Core Flow & 326.5 °F Feedwater Temperature) Source of information: Reference 59