SECTION 15

ACCIDENT ANALYSES

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SECTION 15

ACCIDENT ANALYSES

15.0 GENERAL

This section represents the safety analysis for the Hope Creek Generating Station. The safety analysis is evaluated on a cycle-to-cycle basis by re-evaluating the potentially limiting events. A potentially limiting event is defined as an event or accident that has the potential to affect the core operating or safety limits. The non-limiting events are not re-evaluated, since the limiting events bound the consequences of their occurrence.

The presentation of the results for the limiting events or reload events are presented in Appendix 15D. This information represents the reload licensing analysis for the current cycle. The appropriate sections of chapter 15 reference Appendix 15D.

The remaining information for this section is provided in Section A.15.0 of Reference 15.0-1 and in Reference 15.0-2.

15.0.1 References

15.0-1 "General Electric Standard Application for Reactor Fuel", including The United States Supplement, NEDE-24011-P-A and NEDE-24011-P-A-US, latest revision.

15.0-2 NEDC-33076P, Rev. 2, "Safety Analysis Report for Hope Creek Constant Pressure Power Uprate," August 2006.

TABLE 15.0-1 (Historical)

RESULTS SUMMARY OF TRANSIENT EVENTS APPLICABLE TO BWRs

Section Number	Figure Number	Description	Maximum Neutron Flux, percent NBR	Maximum Dome Pressure, psig	Maximum Vessel Pressure, psig	Maximum Steam Line Pressure, psig	Maximum Core Average Surface Heat Flux, percent of Initial	Δ CPR[2]	Frequency Category[1]	Number of Valves 1st Blowdown	Duration of Blowdown, s
15.1	-	DECREASE IN CORE COOLANT TEMPERATURE									
15.1.1	15.1-1	Loss of feedwater heater, automatic flow control	121.1	1024	1062	1011	113.9	0.12	a	0	0
15.1.1	15.1-2	Loss of feedwater heater, manual flow control[6]	125.6	1030	1069	1016	117.1	0.14	a	0	0
15.1.2	15.1-3	Feedwater controller failure, maximum demand, with bypass[6]	158.3	1153	1194	1148	108.3	0.09	a	14	5
15.1.3	15.1-4	Pressure regulator failure - open	104.3	1127	1142	1126	100.3	0.0[3]	a	0	0
15.2	-	INCREASE IN REACTOR PRESSURE									
15.2.2	15.2-1	Generator load rejection, bypass - on[4]	148.7	1154	1182	1152	100.6	0.03	a	14	4.5
15.2.2	15.2-2	Generator load rejection, bypass - off[4,6]	198.7	1178	1207	1178	105.3	0.07	b	14	>9
15.2.3	15.2-3	Turbine trip, bypass - on[4]	132.1	1153	1180	1148	100.3	0.01	a	14	5
15.2.3	15.2-4	Turbine trip, bypass - off[4,6]	180.0	1176	1206	1177	103.7	0.06	b	14	>9

TABLE 15.0-1 (Cont) (Historical)

Section Number	Figure Number	Description	Maximum Neutron Flux, percent NBR	Maximum Dome Pressure, psig	Maximum Vessel Pressure, psig	Maximum Steam Line Pressure, psig	Maximum Core Average Surface Heat Flux, percent of Initial	ΔCPR[2]	Frequency Category[1]	Number of Valves 1st Blowdown	Duration of Blowdown, s
15.2.4	15.2-5	Closure of all MSIVs[4]	104.3	1168	1207	1165	100.1	~0.0[3]	a	14	5.5
15.2.5	15.2-6	Loss of condenser vacuum	132.4	1151	1178	1144	100.3	~0.0[3]	a	14	11
15.2.6	15.2-7	Loss of all grid connections	120.6	1170	1198	1170	100.1	~0.0[3]	a	14	11
15.2.7	15.2-8	Loss of all feedwater flow	104.5	1020	1059	1007	100.1	~0.0[3]	a	0	0
15.3	-	DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE									
15.3.1	15.3-1	Trip of one recirculation pump motor	104.3	1021	1059	1008	100.1	~0.0[3]	a	0	0
15.3.1	15.3-2	Trip of both recirculation pump motors	104.3	1108	1119	1107	100.1	~0.0[3]	a	0	0
15.3.2	-	Recirculation flow control failure - decreasing flow	See text								
15.3.3	15.3-3	Seizure of one recirculation pump	104.3	1082	1044	1081	100.2	0.09	c	0	0
15.4	-	REACTIVITY AND POWER DISTRIBUTION ANOMALIES									

TABLE 15.0-1 (Cont) (Historical)

Section Number	Figure Number	Description	Maximum Neutron Flux, percent NBR	Maximum Dome Pressure, psig	Maximum Vessel Pressure, psig	Maximum Steam Line Pressure, psig	Maximum Core Average Surface Heat Flux, percent of Initial	ΔCPR[2]	Frequency Category[1]	Number of Valves 1st Blowdown	Duration of Blowdown, s
15.4.4	15.4-6	Abnormal startup of idle recirculation loop	396.3	981	998	976	146.3	(5)	a	0	0
15.4.5	15.4-7	Recirculation flow control failure - increasing flow[6]	366.5	982	1001	976	143.4	(5)	a	0	0
15.5	-	INCREASE IN REACTOR COOLANT INVENTORY									
15.5.1	15.5-1	Inadvertent HPCI pump start	118.7	1020	1059	1007	107.0	0.06	a	0	0

(1) a = Incidents of moderate frequency
b = Infrequent incidents
c = Limiting faults.
(2) ΔCPRs are based on an initial CPR that would yield an MCPR of 1.06.
(3) Estimated value.
(4) Results, not including adjustment factors, are based on end-of-cycle-one nuclear data.
(5) These events are initiated from low power, and the resultant MCPR will be well above 1.06.
(6) These events are analyzed as part of the reload licensing analysis. The results are represented in Appendix 15D. The results presented within this table remain for comparative purposes only.

TABLE 15.0-2

SUMMARY OF ACCIDENTS

Section	Title	Failed Fuel Calculated Value	Failed Fuel NRC Worst-Case Assumption
15.3.3	Seizure of One Recirculation Pump	None	
15.3.4	Recirculation Pump Shaft Break	None	
15.4.9	Rod Drop Accident (number of rods)	$\leq$850	850
15.6.2	Instrument Line Break	None	None
15.6.4	Steam System Pipe Break Outside Containment	None	None
15.6.5	LOCA Within RCPB	None	100 percent
15.6.6	Feedwater Line Break	None	None
15.7.1.1	Main Condenser Gas Treatment System Failure	NA	NA
15.7.3	Liquid Radwaste Tank Failure	NA	NA
15.7.4	Fuel Handling Accident	$\leq$124	124
15.7.5	Cask Drop Accident	None	None

TABLE 15.0-3 (Historical)

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS

For General Electric Analyzed Events
(See Appendix 15D for Reload Analysis)

1.	Thermal power level, MWt	
	Warranteed value	3293
	Analysis value	3435
2.	Steam flow, lb/h	
	Warranteed value	14.16×10^6
	Analysis value	14.87×10^6
3.	Core flow, lb/h	100×10^6
4.	Feedwater flow rate, lb/s	
	Warranteed value	3933
	Analysis value	4130
5.	Feedwater temperature °F	424.5
6.	Vessel dome pressure, psig	1020
7.	Vessel core pressure, psig	1031
8.	Turbine bypass capacity, percent NBR	25
9.	Core coolant inlet enthalphy, Btu/lb	526.6
10.	Turbine inlet pressure, psig	960
11.	Fuel lattice	c(p8 x 8r)
12.	Core average gap conductance, Btu/s-ft^2-°F	0.1744

TABLE 15.0-3 (Cont) (Historical)

13.	Core leakage flow, percent	11.27
14.	Required MCPR operating limit	See Table 15.0-4 and Figure 15.0-2
15.	MCPR safety limit	1.06
16.	Doppler coefficient (-)¢/°F	
	Analysis for power increase events[(1)]	0.2210
	Analysis for power decrease events[(1)]	0.244
17.	Void coefficient (-)¢/percent rated voids	
	Analysis data for power increase events[(1)]	8.703
	Analysis data for power decrease events[(1)]	7.874
18.	Core average rated void fraction, percent[(1)]	42
19.	Scram reactivity, $\$\Delta K$ analysis data[(1)]	Figure 15.0-1
20.	Control rod drive speed, position versus time	Figure 15.0-1
21.	Nuclear characteristics used in ODYN analysis	EOC-1 (End of cycle 1)
22.	Jet pump ratio, M	1.84
23.	Safety/relief valve capacity, percent NBR at 1121 psig	85.8
	Manufacturer	Target Rock
	Quantity installed	14

TABLE 15.0-3 (Cont) (Historical)

24.	Relief function delay, s	0.4
25.	Relief function response time constant, s	0.15
26.	Setpoints for safety/relief valves	
	Safety/relief function, psig	1121, 1131, 1141
27.	Number of valve groupings simulated	
	Safety/relief function, no.	3
28.	SRV reclosure setpoints-both modes (percent of setpoint) maximum safety limit	97
29.	High flux trip, percent NBR	
	Analysis setpoint (121 x 1.043)	126.2
30.	High pressure scram setpoint, psig	1071
31.	Vessel level trips, feet above bottom of separator skirt bottom, ft	
	Level 8 - (L8)	6.042
	Level 4 _ (L4)	3.625
	Level 3 - (L3)	1.792
	Level 2 - (L2)	-3.708
32.	APRM thermal power trip, percent NBR	
	Analysis setpoint (117 x 1.042)	122.0
33.	Recirculation pump trip delay, s	0.175

34.	Recirculation pump trip inertia time constant for analysis, s[2]	
	Maximum	4.5
	Minimum	3.0
35.	Total steamline volume, ft^3	6619
36.	Pressure setpoint of RPT, psig	1101

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(1) For transients simulated on the ODYN computer, this input is calculated by ODYN.

(2) The inertia time constant is defined by the expression:

$$t = \frac{2\pi J_o n}{g T_o}$$

where:

t = inertia time constant, s

J_O = pump motor moment of inertia, $lb\text{-}ft^2$

n = rated pump speed, rps

g = gravitational constant, ft/s^2

T_O = pump shaft torque, ft-lb.

TABLE 15.0-4

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TABLE 15.0-5

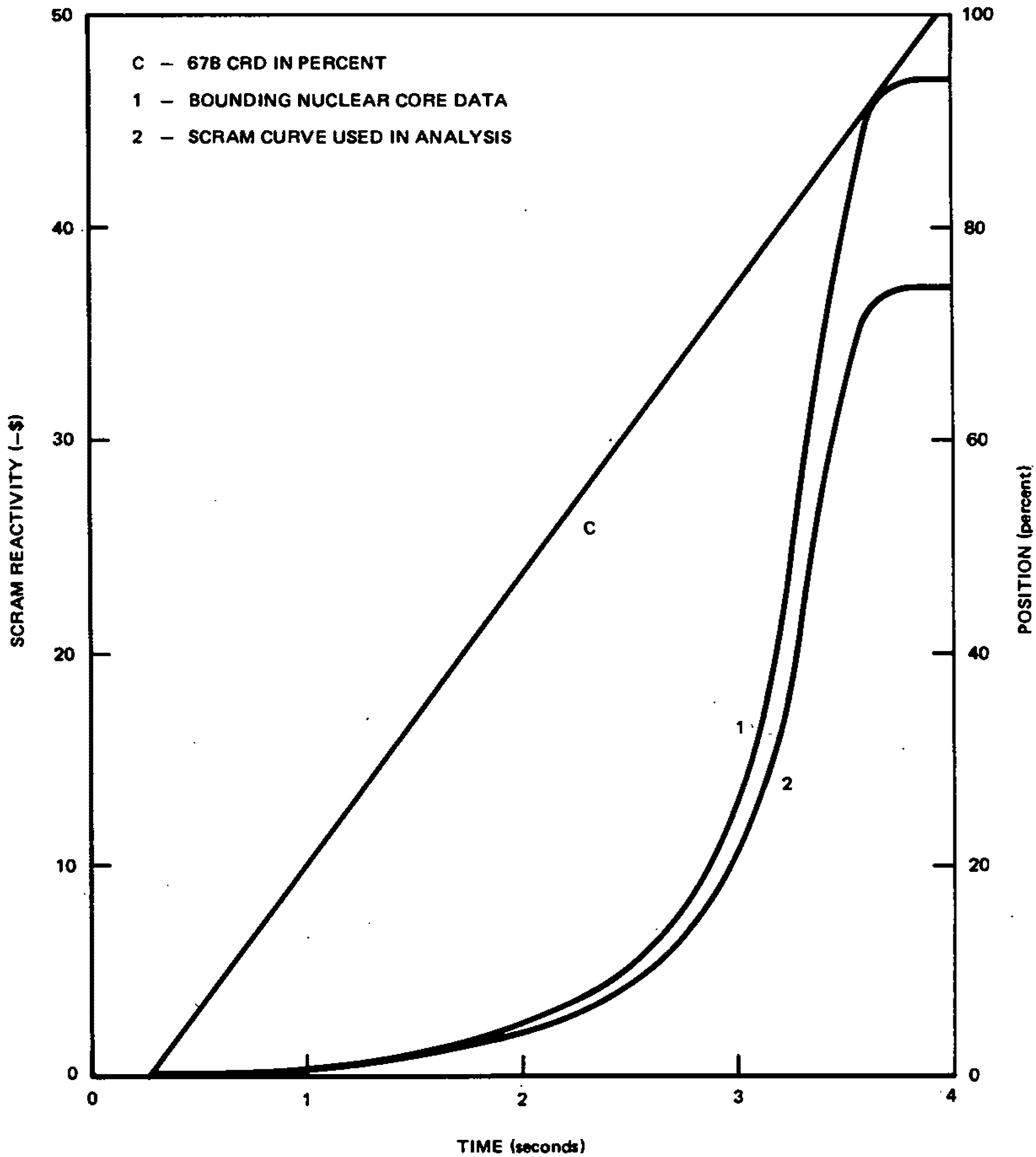
NONSAFETY-GRADE SYSTEMS/COMPONENTS ASSUMED
IN TRANSIENT ANALYSES

Section	Transient	Nonsafety-Grade System or Component
MODERATE FREQUENCY EVENTS		
15.1.2	Feedwater controller failure maximum demand	Level 8 turbine and with feedwater pump trip, turbine bypass, relief valves(1)
15.1.3	Pressure regulator failure open	Relief valves
15.2.2	Load rejection	Turbine bypass, relief valves
15.2.3	Turbine trip	Turbine bypass, relief valves
15.2.4	Closure of all MSIVs	Relief valves
15.2.5	Loss of condenser vacuum	Turbine bypass, relief valves
15.2.6	Loss of AC power	Turbine bypass, relief valves
15.2.7	Loss of all feedwater flow	Recirculation runback
15.3.1	Trip of one or both recirculation pumps	Level 8 turbine trip and feedwater pump trip, turbine bypass, relief valves
15.3.2	Recirculation flow control failure with decreasing flow	Level-8 turbine trip and feedwater pump trip, turbine bypass, relief valves
15.4.1	Rod withdrawal error at low power	Rod worth minimizer
15.4.2	Rod withdrawal error at power	Rod block monitor

TABLE 15.0-5 (Cont)

Section	Transient	Nonsafety-Grade System or Component
INFREQUENT EVENTS		
15.2.2	Load rejection without bypass	Relief valves
15.2.3	Turbine trip without bypass	Relief valves
LIMITING EVENTS		
15.3.3	Recirculation pump seizure	Level 8 turbine trip and feedwater pump trip, turbine bypass relief valves
15.3.4	Recirculation pump shaft break	Level 8 turbine trip and feedwater pump trip, turbine bypass, relief valves
15.6.5	Loss-of-Coolant Accident	Feedwater piping outside containment

(1) "Relief valves" refers to the nonsafety-related manual relief mode of the SRVs. Although this mode does not serve a safety function, its electrical components and power supply are safety grade.



*APPLICABLE TO EVENTS ANALYZED WITH REDY CODE.
ODYN CODE SCRAM REACTIVITY IS CALCULATED
INTERNALLY DURING THE TRANSIENT EVENT

REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

SCRAM POSITION AND REACTIVITY CHARACTERISTICS

UPDATED FSAR

FIGURE 15.0-1

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PSEG NUCLEAR L.L.C.
HOPE CREEK GENERATING STATION

HOPE CREEK UFSAR - REV 11	SHEET 1 OF 1
November 24, 2000	F15.0-2

15.1 DECREASE IN REACTOR COOLANT TEMPERATURE

15.1.1 Loss of Feedwater Heating

The loss of feedwater heating (LOFH) event is considered a potentially limiting event and is re-analyzed for each reload. The results of the re-analysis of the LOFH event are presented in Appendix 15D.

The re-analysis of the LOFH event is performed with the assumption of operation in the manual flow control mode (the original automatic flow control mode has been eliminated).

15.1.1.1 Identification of Causes and Frequency Classification

15.1.1.1.1 Identification of Causes

Feedwater heating can be lost in at least two ways:

1. The steam extraction line valve to a heater is closed

2. Feedwater flow is bypassed around portions of the heaters.

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. The maximum number of feedwater heaters that can be isolated or bypassed by a single event represents the most severe transient for analysis considerations. This transient is analyzed by assuming that a conservative decrease in feedwater temperature occurs. The decrease in feedwater temperature will cause an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient.

15.1.1.1.2 Frequency Classification

The probability of this transient is considered low enough to warrant it being categorized as an infrequent incident. However, because of the lack of a sufficient frequency data base, this transient is analyzed as an incident of moderate frequency.

This event is analyzed under worst-case conditions, which assumes a conservative decrease in feedwater temperature at rated power. The probability of occurrence of this event is regarded as small.

15.1.1.2 Sequence of Events and Systems Operation

15.1.1.2.1 Sequence of Events

Table 15.1-1 provides the sequence of events for this transient. The results presented in the aforementioned table are based on the initial CPPU analysis and are not representative of the currently loaded core. References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the CPPU licensed power level of 3840 MWt. The results are considered typical.

The response of the LOFH reload event is presented in Appendix 15D.

15.1.1.2.1.1 Identification of Operator Actions

As the LOFH event progresses, the reactor settles out at essentially the same recirculation flow with an increase in steam output. An average power range monitor (APRM) neutron flux or thermal power alarm alerts the operator that control rods should be inserted to return to the rated flow control line, or to reduce flow. If reactor scram occurs, the operator will monitor the reactor water level and pressure controls and turbine-generator auxiliaries during coastdown.

15.1.1.2.2 Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it was assumed that normal functioning occurred in the plant instrumentation and controls, plant protection, and Reactor Protection Systems (RPSs).

The high simulated thermal power scram is the primary protection system action in mitigating the consequences of this event.

Operator intervention requiring the activation of engineered safety features (ESF) is not expected for this transient.

15.1.1.2.3 The Effect of Single Failures and Operator Errors

This transient leads to an increase in reactor power level. Single failures are not expected to result in a more severe transient than analyzed. See Section 15.9 for a detailed discussion of this subject.

15.1.1.3 Core and System Performance

15.1.1.3.1 Mathematical Model

The LOFH event for CPPU conditions was analyzed using PANACEA (Reference 15.1-4). The LOFH is analyzed according to the description in Reference 15.1-3.

15.1.1.3.2 Input Parameters and Initial Conditions

The plant is assumed to be operating at 100 percent of nuclear boiler rated (NBR) power and at thermally limited conditions.

The transient is simulated by programming a change in feedwater enthalpy corresponding to a 110°F loss in feedwater heating.

15.1.1.3.3 Results

The results presented below are representative of CPPU conditions. These results are considered bounded by the reload licensing analysis.

As the LOFH event progresses, reactor power increases as a result of the increased core inlet subcooling. A trip may occur on high APRM neutron flux. If the core power does not reach the scram setpoint, a new steady state operating condition is achieved. Vessel steam flow increases and the system pressure increases. The response of the key plant variables for the reload re-analysis is shown in Appendix 15D.

This transient is less severe from lower initial power levels for two main reasons:

1. Lower initial power levels have initial MCPR values greater than the limiting initial value assumed.

2. The magnitude of the power rise decreases with lower initial power conditions.

Therefore, transients from lower power levels are less severe.

15.1.1.3.4 Considerations of Uncertainties

Important factors such as initial operating conditions, trip characteristics, and magnitude of the feedwater temperature change are assumed to be at their worst values, so that any deviations seen in the actual plant operation reduce the severity of the event.

15.1.1.4 Barrier Performance

As noted above, and in Appendix 15D, the consequences of this transient 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.

15.1.1.5 Radiological Consequences

Since this transient does not result in any additional fuel failures, or any release of primary coolant to either the reactor building or to the environment, there are no radiological consequences associated with this transient.

15.1.2 Feedwater Controller Failure - Maximum Demand

The feedwater controller failure - Maximum Demand event is considered a potentially limiting event and is re-analyzed for each reload. The results of the re-analysis of this event are presented in Appendix 15D.

15.1.2.1 Identification of Causes and Frequency Classification

15.1.2.1.1 Identification of Causes

This transient is postulated on the basis of a single failure of a control device, specifically one that can affect an increase in reactor coolant inventory by increasing the feedwater flow. The

most severe applicable transient is a feedwater controller failure during maximum flow demand. The feedwater controller is forced to its upper limit at the beginning of the transient.

15.1.2.1.2 Frequency Classification

This transient is considered to be an incident of moderate frequency.

15.1.2.2 Sequence of Events and Systems Operation

15.1.2.2.1 Sequence of Events

With excess feedwater flow, the reactor water level rises to the high level reference point, at which time the feedwater pumps and the main turbine are tripped and a scram is initiated. The sequence of events are presented in Appendix 15D.

15.1.2.2.1.1 Identification of Operator Actions

The operator will:

1. Observe that high level feedwater pump trip has terminated the transient

2. Switch the feedwater controller from auto to manual control in order to try to regain a correct output signal

3. Identify causes of the failure and report all key plant parameters during the transient.

15.1.2.2.2 Systems Operation

To properly simulate the expected sequence of events, the analysis of this transient assumes normal functioning of plant

instrumentation and controls, plant protection, and Reactor Protection Systems (RPSs). Important system operational actions for this transient are high level tripping of the main turbine, main stop valve reactor trip initiation, recirculation pump trip (RPT), and low water level initiation of the Reactor Core Isolation Cooling (RCIC) System and the High Pressure Coolant Injection (HPCI) System to maintain long term water level control following tripping of feedwater pumps.

15.1.2.2.3 The Effect of Single Failures and Operator Errors

The first sensed event to initiate automatic corrective action to the transient is the vessel high water level (L8) trip. Multiple level sensors are used to sense and detect when the water level reaches the level 8 (L8) setpoint. At this point in the logic, a single failure does not initiate or prevent a turbine trip signal. Turbine trip signal transmission, however, is not built to single failure criterion. The result of a failure at this point would have the effect of delaying the pressurization. High levels in the turbine moisture separators result in a trip of the unit before high moisture levels enter the low pressure turbine. However, if excessive moisture enters the turbine, it causes vibration to the point where the operator may manually trip the unit.

Reactor trip signals from the turbine are designed such that a single failure neither initiates nor impedes a scram initiation. See Section 15.9 for a detailed discussion of this subject.

15.1.2.3 <u>Core and System Performance</u>

15.1.2.3.1 Mathematical Model

The predicted dynamic behavior has been determined using a computer simulated, analytical model of a generic direct cycle BWR. This model is described in detail in Reference 15.1-2. This computer model has been improved and verified through extensive comparison of its predicted results with boiling water reactor (BWR) test data.

The non-linear computer simulated analytical model is designed to predict associated transient behavior of the reactor. Some of the significant features of the model are:

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15.1.2.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions as shown in Appendix 15D.

A description of the plant initial conditions and assumptions are presented in section 7.4.5.2 of reference 15.1-3.

15.1.2.3.3 Results

The results of the feedwater controller failure - maximum demand event are presented in Appendix 15D.

15.1.2.3.4 Consideration of Uncertainties

All systems used for protection in this transient were assumed to have the most conservative allowable response, e.g., relief valve setpoints, scram control rod travel time, and reactivity

characteristics. Expected plant behavior is therefore expected to lead to a less severe transient.

15.1.2.4 Barrier Performance

As noted above, the consequences of this transient 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.

15.1.2.5 Radiological Consequences

While this transient does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Because this activity is contained in the primary containment, there is no exposure to operating personnel. This transient does not result in an uncontrolled release to the environment, so the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release is in accordance with established technical specification limits.

15.1.3 Pressure Regulator Failure - Open

The pressure regulator failure in the open position event is considered a non-Limiting event. Therefore it is not required to be re-analyzed as a part of the reload licensing analysis for Hope Creek, unless the disposition for this event changes (Reference 15.1-3).

The results referenced within this section are representative of cycle 1. **References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed** power level of 3293 MW_{th}. These results are considered bounded by the reload licensing analysis.

15.1.3.1 Identification of Causes and Frequency Classification

15.1.3.1.1 Identification of Causes

The total steam flow rate to the main turbine resulting from a pressure regulator malfunction is limited by a maximum flow limiter imposed at the turbine controls. This limiter is set to limit maximum steam flow to approximately 130 percent nuclear boiler rated (NBR).

If the pressure regulator fails to the open position, which requires the failure of two of pressure channels, the turbine control valves can be fully opened, and the turbine bypass valves can be partially opened until the maximum steam flow is established.

15.1.3.1.2 Frequency Classification

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

15.1.3.2 Sequence of Events and Systems Operation

15.1.3.2.1 Sequence of Events

Table 15.1-4 lists the sequence of events for Figure 15.1-4.

The above results are representative of cycle 1. These results are considered bounded by the reload licensing analysis.

15.1.3.2.1.1 Identification of Operator Actions

Water level would reach high level (L8) to trip the turbine and feedwater pump and cause the reactor to scram. Once the turbine trip occurs, the pressure increases to the point where the main steam safety/relief valves (SRVs) open. The operator will:

1. Verify that all rods are in their fully inserted position.

2. Monitor reactor water level and pressure.

3. Monitor turbine coastdown and break vacuum before the loss of steam seals, and check turbine auxiliaries.

4. Observe that the reactor pressure relief valves open at their setpoint.

5. Observe that reactor core isolation cooling (RCIC) and high pressure coolant injection (HPCI) initiate on low water level.

6. Secure both HPCI and RCIC when reactor pressure and level are under control.

7. Monitor reactor water level and continue cooldown per the normal procedure.

8. Complete the scram report and initiate a maintenance survey of pressure regulator before reactor restart.

15.1.3.2.2 Systems Operation

To simulate the expected sequence of events properly, the analysis of this transient assumes normal functioning of plant instrumentation and controls, plant protection, and Reactor Protection Systems (RPSs), except as otherwise noted.

HPCI and RCIC system functions are initiated when the vessel water level reaches the L2 setpoint. Normal startup and actuation can take up to 30 seconds before full flow is realized. If these events occur, they follow sometime after the primary concerns of fuel thermal margin and overpressure effects have occurred, and are less severe than those already experienced by the system.

15.1.3.2.3 The Effect of Single Failures and Operator Errors

This transient leads to a loss of pressure control such that the increased steam flow demand causes a depressurization. Instrumentation for pressure sensing of the turbine inlet pressure is designed to accommodate the effects of single failure for initiation of main steam line isolation valve (MSIV) closure.

Reactor scram sensing, originating from limit switches on the MSIVs, is designed to accommodate the effects of single failure.

It is therefore concluded that the basic phenomenon of pressure decay is adequately terminated. See Section 15.9 for a detailed discussion of this subject.

15.1.3.3 <u>Core and System Performance</u>

15.1.3.3.1 Mathematical Model

The predicted dynamic behavior has been determined using a computer-simulated analytical model of a generic direct cycle boiling water reactor (BWR). This model is described in detail in Reference 15.1-1. It has been verified through extensive comparison of its predicted results with actual BWR test data.

The nonlinear, computer-simulated, analytical model is designed to predict associated transient behavior of this reactor. Some of the significant features of the model are:

1. A point kinetic model is assumed with reactivity feedbacks from control rods (absorption), voids (moderation), and Doppler (capture) effects.

2. The fuel is represented by three four-node cylindrical elements, each enclosed in a cladding node. One of the cylindrical elements is used to represent core average power and fuel temperature conditions, providing the source of Doppler feedback. The other two are used to represent "hot spots" in the core to simulate peak fuel center temperature and cladding temperature.

3. Four primary system pressure nodes are simulated. The nodes represent the core exit pressure, vessel dome pressure, steam line pressure at a point representative of the main steam safety/relief valve (SRV) location, and turbine inlet pressure.

4. The active core void fraction is calculated from a relationship between core exit quality, inlet subcooling, and pressure. This relationship is generated from multinode core steady state calculations. A second order void dynamic model with the void boiling sweep time calculated as a function of core flow and void conditions is also used.

5. Principle controller functions, such as feedwater flow, recirculation flow, reactor water level, pressure, and load demand are represented together with their dominant nonlinear characteristics.

6. The ability to simulate necessary RPS functions is provided.

15.1.3.3.2 Input Parameters and Initial Conditions

This transient is simulated by setting the controlling regulator output to a high value, which causes the turbine control valves to open fully and the turbine bypass valves to open partially. A regulator failure with 130 percent steam flow was simulated as a worst case since 130 percent is the normal maximum flow limit.

This analysis has been performed, unless otherwise noted, with the plant conditions listed in Table 15.0-3.

15.1.3.3.3 Results

The results presented below are representative of cycle 1. These results are considered bounded by the reload licensing analysis.

Figure 15.1-4 shows the response of important nuclear system variables for this transient. The water level rises to the L8 trip setpoint in 5.4 s and initiates trip of the main and feedwater turbines. Closure of the main stop valves (MSVs) initiates a scram and recirculation pump trip (RPT). After the pressurization resulting from the MSV closure, pressure again drops and continues to drop until turbine inlet pressure is below the low turbine pressure isolation setpoint, when the main steam line isolation finally terminates the depressurization.

A reactor L8 trip limits the duration and severity of the depressurization so that no significant thermal stresses are imposed on the reactor coolant pressure boundary (RCPB). After the rapid portion of the transient is complete, the nuclear system SRVs operate intermittently to relieve the pressure rise that results from decay heat generation. No significant reductions in fuel thermal margins occur. Because the rapid portion of the transient results in only momentary depressurization of the nuclear system, and because the SRVs operate only to relieve the pressure increase caused by decay heat, the RCPB is not threatened by high internal pressure for this pressure regulator malfunction.

15.1.3.3.4 Consideration of Uncertainties

If the maximum flow limiter were set higher or lower than normal, a faster or slower loss in nuclear steam pressure results. The rate of depressurization may be limited by the bypass capacity. For example, the turbine valves open to the wide open state, admitting slightly more than the rated steam flow. With the limiter in this analysis set to fail at 130 percent, we would expect something less than 25 percent to be bypassed. This is therefore not a limiting factor on this plant. If the rate of depressurization does change, it is terminated by the low turbine inlet pressure trip setpoint.

Depressurization rate has a proportional effect upon the voiding action of the core. If it is not large enough, the L8 trip setpoint may not be reached. Then a turbine and feedwater pump trip will not occur in the transient. In this case, the turbine inlet pressure will drop below the pressure isolation setpoint, and the expected transient will conclude with an isolation of the main steam lines. The reactor will be shut down by the scram initiated from the MSIV closure.

15.1.3.4 Barrier Performance

Barrier performance analyses were not required, since the consequences of this transient do not result in any temperature or pressure transient in excess of the criteria for which fuel, pressure vessel, or containment are designed. Peak pressure in the bottom of the vessel reaches 1142 psig, which is below the ASME B&PV Code limit of 1375 psig for the RCPB.

15.1.3.5 Radiological Consequences

While this transient does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Because this activity is contained in the primary containment, there is no exposure to operating personnel. This transient does not result in an uncontrolled release to the environment, so the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release is in accordance with established technical specification limits.

15.1.4 Inadvertent Main Steam Relief Valve Opening

The inadvertent main steam relief valve opening event is considered a non-Limiting event. Therefore it is not required to be re-analyzed as a part of the reload licensing analysis for Hope Creek, unless the disposition for this event changes (Reference 15.1-3). This event was not re-evaluated for CPPU conditions.

15.1.4.1 Identification of Causes and Frequency Classification

15.1.4.1.1 Identification of Causes

Cause of inadvertent main steam safety/relief valve (SRV) opening is attributed to malfunction of the valve or an operator initiated opening. Opening and closing circuitry at the individual valve level, as opposed to groups of valves, is subject to a single failure event. It is therefore postulated that a failure occurs, and the transient is analyzed accordingly. Detailed discussion of the valve design is provided in Section 5.

15.1.4.1.2 Frequency Classification

This transient is categorized as an infrequent incident, but due to a lack of a comprehensive data basis, it is analyzed as an incident of moderate frequency.

15.1.4.2 <u>Sequence of Events and Systems Operation</u>

15.1.4.2.1 Sequence of Events

Table 15.1-5 lists the sequence of events for this transient.

15.1.4.2.1.1 Identification of Operator Actions

The plant operator must "reclose" the valve and check that reactor and turbine-generator output return to normal. If the valve cannot be closed, plant shutdown must be initiated.

15.1.4.2.2 Systems Operation

This event assumes normal functioning of normal plant instrumentation and controls, specifically the operation of the pressure regulator and level control systems.

15.1.4.2.3 The Effect of Single Failures and Operator Errors

Failure of additional components, e.g., pressure regulator, feedwater flow controller, is discussed in Section 15.9.

15.1.4.3 <u>Core and System Performance</u>

15.1.4.3.1 Mathematical Model

The reactor model briefly described in Section 15.1.3.3.1 was previously used to simulate this event in earlier FSARs. This model is discussed in detail in Reference 15.1-1. It has been

determined that this transient is not limiting from a core performance standpoint. Therefore, a qualitative presentation of results is included below.

15.1.4.3.2 Input Parameters and Initial Conditions

It is assumed that the reactor is operating at an initial power level corresponding to 105 percent nuclear boiler rated (NBR) steam flow conditions when an SRV is inadvertently opened.

15.1.4.3.3 Qualitative Results

The opening of an SRV allows steam to be discharged into the suppression chamber. The sudden increase in the rate of steam flow leaving the reactor vessel causes a mild depressurization transient.

The pressure regulator senses the nuclear system pressure decrease and closes the turbine control valve far enough to stabilize reactor vessel pressure at a slightly lower value, and reactor power settles at nearly the initial power level. Thermal margins decrease only slightly through the transient, and no fuel damage results from the transient. Minimum critical power ratio (MCPR) is essentially unchanged, and therefore the safety limit margin is unaffected.

15.1.4.4 Barrier Performance

As discussed above, the transient resulting from an inadvertent SRV opening is a mild depressurization that is within the range of normal load following, and, therefore, has no significant effect on RCPB and containment design pressure limits.

15.1.4.5 Radiological Consequences

While this transient does not result in fuel failure, it does result in the discharge of normal coolant activity to the

suppression chamber via SRV operation. Because this activity is contained in the primary containment, there are no exposures to operating personnel. This transient does not result in an uncontrolled release to the environment. So, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release is in accordance with the established technical specification limits.

15.1.5 Spectrum of Steam System Piping Failures Inside and Outside of Containment in a PWR

This event is not applicable to boiling water reactor (BWR) plants.

15.1.6 Inadvertent RHR Shutdown Cooling Operation

The inadvertent RHR shutdown cooling operation event is considered a non-Limiting event. Therefore it is not required to be re-analyzed as a part of the reload licensing analysis for Hope Creek, unless the disposition for this event changes (Reference 15.1-3).

15.1.6.1 Identification of Causes and Frequency Classification

15.1.6.1.1 Identification of Causes

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

If the reactor were critical or near critical in startup or cooldown conditions, a very slow increase in reactor power can result. A shutdown cooling malfunction leading to a moderator temperature decrease can result from misoperation of the cooling water controls for the residual heat removal (RHR) heat exchangers. The resulting temperature decrease causes a slow insertion of positive reactivity into the core. If the operator does not control the power level, a high neutron flux reactor scram terminates the transient without violating fuel thermal limits and without any measurable increase in nuclear system pressure.

15.1.6.1.2 Frequency Classification

Although no single failure can cause this transient, it is conservatively categorized as a transient of moderate frequency.

15.1.6.2 Sequence of Events and Systems Operation

15.1.6.2.1 Sequence of Events

A shutdown cooling malfunction leading to a moderator temperature decrease could result from misoperation of the cooling water controls for RHR heat exchangers. The resulting temperature decrease causes a slow insertion of positive reactivity into the core. Scram occurs before any thermal limits are reached if the operator does not take action. The sequence of transients for this event is shown in Table 15.1-6.

The above results are representative of cycle 1. These results are considered bounded by the reload licensing analysis.

15.1.6.2.2 System Operation

A shutdown cooling malfunction causing a moderator temperature decrease must be considered in all operating states. However, this transient is not considered while at power operation, since the nuclear system pressure is too high to permit operation of the RHR shutdown cooling mode.

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

15.1.6.2.3 Effect of Single Failures and Operator Action

No single failures can cause this transient to be more severe. If the operator takes action, the slow power rise is controlled in the normal manner. If no operator action is taken, a scram terminates the power increase before thermal limits are reached. See Section 15.9 for details.

15.1.6.3 Core and System Performance

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

15.1.6.4 Barrier Performance

As noted above, this event does 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.

15.1.6.5 Radiological Consequences

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

15.1.7 References

15.1-1 R.B. Linford, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802-A, General Electric, December 1986.

15.1-2 General Electric, "Qualification of the One Dimensional Core Transient Model for BWR," NEDO 24154, October 1978.

15.1-3 "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (latest approved revision) and "General Electric Standard Application for Reactor Fuel (Supplement for United States)", NEDE-24011-P-A-US (latest approved revision)

15.1-4 Three Dimensional Boiling Water Reactor Core Simulator, NEDO-20953-A, January 1977

TABLE 15.1-1

SEQUENCE OF EVENTS FOR LOSS OF FEEDWATER HEATING

Time, s	Event
0	A 110°F temperature reduction is initiated in the feedwater system.
5	Initial effect of loss of feedwater heating starts to raise core power level.
Approx 120	Reactor variables settle into new steady state.

TABLE 15.1-2

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TABLE 15.1-3

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TABLE 15.1-4

SEQUENCE OF EVENTS FOR PRESSURE REGULATOR FAILURE TO 130 PERCENT
(FIGURE 15.1-4)

Time, s	Event
0	Simulate steam flow demand to 130 percent.
0.1	Main turbine bypass opens.
5.38	L8 setpoint trips main turbine and feedwater pumps.
5.39	Reactor scram initiated from MSV position switches.
5.39	RPT initiated from MSV position switches.
10.71	First group of SRVs open due to high pressure.

TABLE 15.1-5

SEQUENCE OF EVENTS FOR INADVERTENT SAFETY RELIEF VALVE OPENING

Time, s	Event
0	Opening of one SRV is initiated.
0.5	SRV flow reaches full flow.
15	System establishes new steady state operation.

TABLE 15.1-6

SEQUENCE OF EVENTS FOR INADVERTENT RHR SHUTDOWN COOLING OPERATION

<u>Time, s</u>	<u>Event</u>
0	Reactor at states B or D (of Section 15.9) when RHR shutdown cooling inadvertently activated.
0-10 min	Slow rise in reactor power.
>10 min	Operator may take action to limit power rise. Flux scram occurs if no action is taken.

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Revision 17, June 23, 2009

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station LOSS OF 100 DEGREE F FEEDWATER HEATING AUTO FLOW CONTROL
	Updated FSAR — Figure 15.1-1

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HOPE CREEK GENERATING STATION

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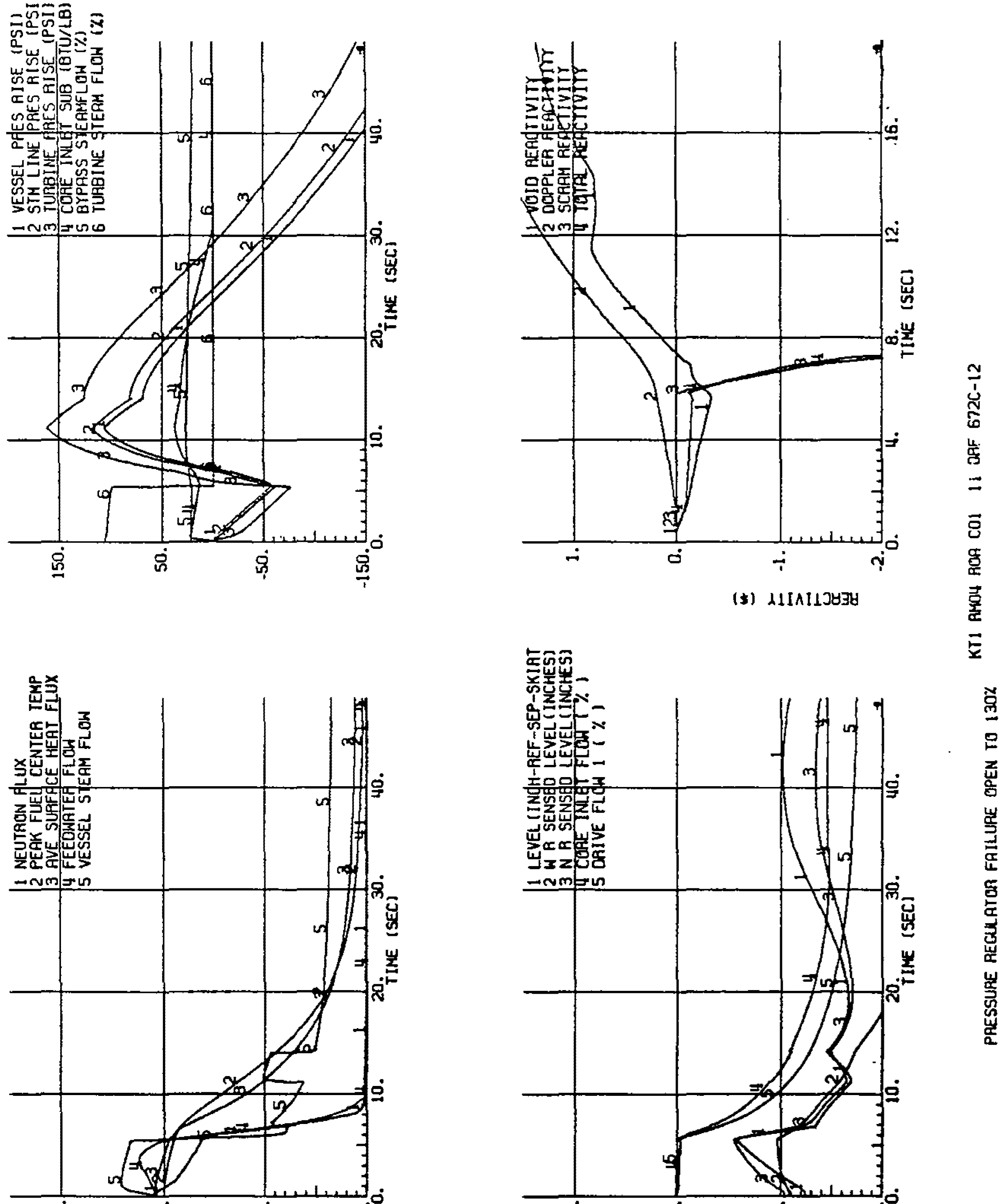
PSEG NUCLEAR L.L.C. HOPE CREEK GENERATING STATION



REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PRESSURE REGULATOR
FAILURE OPEN

UPDATED FSAR FIGURE 15.1-4

15.2 INCREASE IN REACTOR PRESSURE

15.2.1 Pressure Regulator Failure - Closed

The pressure regulator failure in the closed position event is considered a non-Limiting event. Therefore it is not required to be re-analyzed as a part of the reload licensing analysis for Hope Creek, unless the disposition for this event changes (Reference 15.2-2).

15.2.1.1 Identification of Causes and Frequency Classification

15.2.1.1.1 Identification of Causes

Three identical pressure regulators are provided to maintain primary system pressure control. They independently sense pressure just upstream of the main stop valves and compare it to pressure demand to create proportional error signals that produce each regulator output. The output of the regulators feeds into a median select logic, where the median value controls the turbine control valves (TCV).

It is assumed for purposes of this transient analysis that a single personnel error or momentary failure of two pressure regulator channels occurs that erroneously causes the regulator to momentarily close the turbine control valves and thereby increases reactor pressure.

15.2.1.1.2 Frequency Classification

This event is treated as a moderate frequency event.

15.2.1.2 Sequence of Events and System Operation

15.2.1.2.1 Sequence of Events

A postulated failure of the pressure regulator in the closed mode, as discussed in Section 15.2.1.1.1, causes the turbine control valves to close momentarily. The

pressure increases because the reactor is still generating the initial steam flow. The regulator reopens the valves and reestablishes steady state operation at the initial pressure equal to the setpoint.

15.2.1.2.1.1 Identification of Operator Actions

The operator will verify that the regulator assumes proper control.

15.2.1.2.2 Systems Operation

Plant instrumentation and controls are assumed to function normally. This event requires no protection system or safeguard systems operation.

15.2.1.2.3 The Effect of Single Failures and Operator Errors

A single failure is not credible in this scenario since the pressure controller is single failure proof. An Operator error is the assumed cause of a slight pressure increase.

The assumed failure produces a slight pressure increase in the reactor until the backup regulator gains control.

15.2.1.3 Core and System Performance

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 Sections 15.2.2 and 15.2.3.

15.2.1.3.1 Mathematical Model

Only qualitative evaluation is provided.

15.2.1.3.2 Input Parameters and Initial Conditions

Only qualitative evaluation is provided.

15.2.1.3.3 Results

Response of the reactor during this regulator failure is such that pressure at the turbine inlet increases quickly, less than 2 seconds, due to the sharp closing action of the turbine control valves that reopen when the regulator gains control. This pressure disturbance in the vessel is not expected to exceed flux or pressure scram setpoints.

15.2.1.3.4 Consideration of Uncertainties

All systems used for protection in this event are assumed to have the most conservative allowable response, e.g., relief setpoints, scram stroke time, and control rod worth characteristics. Expected plant behavior is, therefore, expected to reduce the actual severity of the transient.

15.2.1.4 <u>Barrier Performance</u>

Since the consequences of this event do not result in any temperature or pressure transient, as shown by Table 15.0-1, in excess of the criteria for which the fuel, pressure vessel, or containment are designed, these barriers maintain their integrity and function as designed.

15.2.1.5 <u>Radiological Consequences</u>

Since this event does not result in any fuel failures, or any release of primary coolant to either the Reactor Building or to

the environment, there are no radiological consequences associated with this event.

15.2.2 Generator Load Rejection

The generator load rejection event is considered a potentially limiting event and is re-analyzed for each reload. The results of the re-analysis of the generator load rejection event are presented in Appendix 15D.

The re-analysis of the generator load rejection event is performed with the failure of the main steam bypass system. This event without the operability of the main steam bypass system is more limiting, so the generator load rejection with main steam bypass system operable is not re-analyzed (Ref. 15.2-2).

15.2.2.1 Identification of Causes and Frequency Classification

15.2.2.1.1 Identification of Causes

Fast closure of the turbine control valves is initiated whenever there is an electrical grid disturbance resulting in significant loss of electrical load on the generator. The TCVs are required to close as rapidly as possible to prevent excessive overspeed of the turbine generator rotor. Closure of the turbine control valves causes a sudden reduction in steam flow, which results in an increase in system pressure and reactor shutdown.

15.2.2.1.2 Frequency Classification

15.2.2.1.2.1 Generator Load Rejection

This event is categorized as an incident of moderate frequency.

15.2.2.1.2.2 Generator Load Rejection with Bypass Failure

This event is categorized as an infrequent incident with the following characteristics:

1. Frequency of 0.0036/plant year
2. Mean time between events (MTBE) of 278 years.

Thorough searches of domestic plant operating records have revealed three instances of bypass failure during 628 bypass system operations. This gives a probability of bypass failure of 0.0048/event. Combining the actual frequency of a generator load

rejection with the failure rate of the bypass yields a frequency of a generator load rejection with bypass failure of 0.0036 event/plant year.

15.2.2.2 Sequence of Events and System Operation

15.2.2.2.1 Sequence of Events

15.2.2.2.1.1 Generator Load Rejection - Turbine Control Valve Fast Closure

A loss of generator electrical load from high power conditions produces the sequence of events listed in Table 15.2-1.

The sequence of events listed in Table 15.2-1 and the results in Figure 15.2-1, which are referred to in the Table 15.2-1, are representative of CPPU operation. References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the CPPU licensed power level of 3480 MW_{th}. Failure of a single SRV to open has also been included in the CPPU analysis. These results are considered bounded by the reload licensing analysis.

15.2.2.2.1.2 Generator Load Rejection with Failure of Bypass

A loss of generator electrical load at high power with bypass failure produces the sequence of events listed in Appendix 15D.

15.2.2.2.1.3 Identification of Operator Actions

In the event of generator load rejection, the operator will:

1. Verify proper bypass valve performance
2. Observe that the feedwater/level controls have maintained a satisfactory reactor water level
3. Observe that the pressure regulator is maintaining the desired reactor pressure
4. Record peak power and pressure
5. Verify main steam safety/relief valve (SRV) operation.

15.2.2.2.2 System Operation

15.2.2.2.2.1 Generator Load Rejection with Bypass

To properly simulate the expected sequence of events, this analysis assumes normal functioning of plant instrumentation and controls, plant protection, and Reactor Protection Systems (RPS), unless otherwise stated.

Generator load rejection causes turbine control valve fast closure which initiates a scram signal for power levels greater than 24 percent nuclear boiler rated (NBR). In addition, recirculation pump trip (RPT) is initiated. Both of these trip signals satisfy the single failure criterion, and credit is taken for these protection features.

The pressure relief system, which operates the SRVs independently when system pressure exceeds SRV instrumentation setpoints, is assumed to function normally during the time period analyzed.

15.2.2.2.2.2 Generator Load Rejection with Failure of Bypass

This operation is similar to that described in Section 15.2.2.2.2.1, except that failure of the main turbine bypass valves is assumed for the entire transient.

15.2.2.2.3 The Effect of Single Failures and Operator Errors

Mitigation of pressure increase, the basic nature of this transient, is accomplished by the RPT function. Turbine control valve trip scram and RPT are designed to satisfy the single failure criterion. An evaluation of the most limiting single failure, i.e., failure of the bypass system, was considered in this event. Details of single failure analysis can be found in Section 15.9.

15.2.2.3 Core and System Performance

15.2.2.3.1 Mathematical Model

The computer model described in Section 15.1.2.3.1 is used to simulate this event.

15.2.2.3.2 Input Parameters and Initial Conditions

The turbine Electrohydraulic Control System (EHC) 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 partial arc mode and utilize the full stroke closure time, from fully open to fully closed, of 0.15 seconds.

Auxiliary power is normally independent of any turbine-generator overspeed effects and is continuously supplied at rated frequency as automatic fast transfer to auxiliary power supplies normally occurs. For the purposes of worst-case analysis, the reactor recirculation pumps are assumed to be tripped by the RPT system.

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

Events caused by low water level (L2) trips, including initiation of the high pressure coolant injection (HPCI) and Reactor Core Isolation

Cooling (RCIC) Systems, are not included in the simulation. If these events occur, they will follow sometime after the primary concerns of fuel margin and overpressure effects have passed and are expected to result in effects less severe than those already experienced by the reactor system.

15.2.2.3.3 Results

15.2.2.3.3.1 Generator Load Rejection with Bypass

The results presented below are representative of CPPU conditions. References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the CPPU licensed power level of 3840 MW_{th}. These results are considered bounded by the reload licensing analysis.

Figure 15.2-1 shows the results of the generator trip from 100 percent rated power and 105 percent rated core flow. Peak neutron flux rises to 243.4 percent. The average surface heat flux peaks at 110.3 percent of its initial value, and minimum critical power ratio (MCPR) decreases by 0.19 below its initial value.

15.2.2.3.3.2 Generator Load Rejection with Failure of Bypass

The results of the generator load rejection with failure of bypass are presented in Appendix 15D.

15.2.2.3.4 Consideration of Uncertainties

The full-stroke closure time of the turbine control valve of 0.15 seconds is conservative. Typically, the actual closure time is more like 0.2 seconds. Clearly, the less time the valve takes to close, the more severe the pressurization effect.

Changing from full-arc to partial-arc turbine control results in the load rejection event being slightly less severe (0.01 to 0.02 delta CPR decrease). This results because all control valves are fractionally closed initially in the full-arc mode and thus, during the load rejection, the steam flow is shut off sooner than it would be with partial-arc control. For this reason, the load rejection, with full-arc control, bounds the partial-arc control condition.

All systems used for protection in this event are assumed to have the most conservative allowable response, e.g., relief setpoints, scram stroke time, and control rod worth characteristics.

Therefore, anticipated plant behavior is expected to reduce the actual severity of the transient.

15.2.2.4 Barrier Performance

15.2.2.4.1 Generator Load Rejection

Peak pressure remains within normal operating range and no threat to the barrier exists.

15.2.2.4.2 Generator Load Rejection with Failure of Bypass

The results presented in Appendix 15D show that the peak nuclear system pressure is well below the reactor coolant pressure boundary (RCPB) pressure limit of 1375 psig.

15.2.2.5 Radiological Consequences

While 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 is no exposure to operating personnel. Also, since this event does not result in an uncontrolled release to the environment, the plant operator can choose to hold the activity in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will be in accordance with established Hope Creek Technical Specification limits.

15.2.3 Turbine Trip

The turbine trip event is considered a potentially limiting event and is re-analyzed for each reload. The results of the re-analysis of the turbine trip event are presented in Appendix 15D.

The re-analysis of the turbine trip event is performed with the failure of the main steam bypass system. This event without the operability of the main steam bypass system is more limiting, so the turbine trip with main steam bypass system operable is not re-analyzed (Reference 15.2-2).

15.2.3.1 Identification of Causes and Frequency Classification

15.2.3.1.1 Identification of Causes

A variety of turbine or nuclear system malfunctions can initiate a turbine trip. Some examples are moisture separator drain tank high levels, operational lock-out, loss of control fluid pressure, low condenser vacuum, and reactor vessel high water level.

15.2.3.1.2 Frequency Classification

15.2.3.1.2.1 Turbine Trip

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

15.2.3.1.2.2 Turbine Trip with Failure of the Bypass

This transient disturbance is categorized as an infrequent incident. Frequency is expected to be as follows:

1. Frequency of 0.0064/plant year
2. Mean time between events of 156 years.

As discussed in Section 15.2.2.1.2.2, the probability of bypass failure is 0.0048 per event. Combining this with the turbine trip

frequency of 1.22 events per plant year yields the frequency of 0.0064 per plant year.

15.2.3.2 Sequence of Events and Systems Operation

15.2.3.2.1 Sequence of Events

15.2.3.2.1.1 Turbine Trip

Turbine trip at high power produces the sequence of events listed in Table 15.2-3.

The sequence of events listed in Table 15.2-3 and the results in Figure 15.2-3, which are referred to in the Table 15.2-3, are representative of CPPU conditions. References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the CPPU licensed power level of 3840 MW_{th}. The failure of a single SRV to open has also been included in the CPPU analysis. These results are considered bounded by the reload licensing analysis.

15.2.3.2.1.2 Turbine Trip with Failure of the Bypass

Turbine trip at high power with bypass failure produces the sequence of events listed in Appendix 15D.

15.2.3.2.1.3 Identification of Operator Actions

In the event of a turbine trip, the operator will:

1. Monitor and maintain reactor water level at the required level.

2. Check the turbine for proper operation of all auxiliaries during coastdown.

3. Depending on conditions, initiate normal operating procedures for cooldown of the core, or maintain pressure for restart purposes.

4. Put the Reactor Protection System (RPS) mode switch in the startup position before the reactor pressure decays to <850 psig.

5. Secure the Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) System operation if automatic initiation occurs due to low water level.

6. Monitor control rod drive (CRD) positions and insert both the intermediate range monitors (IRMs) and source range monitors (SRMs).

7. Investigate the cause of the trip, make repairs as necessary, and complete the scram report.

8. Cool down the reactor per standard procedure if a restart is not intended.

15.2.3.2.2 Systems Operation

15.2.3.2.2.1 Turbine Trip

All plant control systems maintain normal operation unless specifically designated to the contrary.

Main stop valve closure initiates a reactor scram by position signals to the RPS. Credit is taken for successful operation of the RPS.

Main stop valve closure initiates recirculation pump trip (RPT), thereby terminating the jet pump drive flow.

The pressure relief system, which operates the main steam safety relief valves (SRVs) independently when system pressure exceeds the SRVs' instrumentation setpoints, is assumed to function normally during the time period analyzed.

15.2.3.2.2.2 Turbine Trip with Failure of the Bypass

This event occurs as described in Section 15.2.3.2.2.1, except that failure of the main turbine bypass system is assumed for the entire transient time period analyzed.

15.2.3.2.2.3 Turbine Trip at Low Power with Failure of the Bypass

This event occurs as described in Section 15.2.3.2.2.1, except that failure of the main turbine bypass system is assumed.

It should be noted that below 24 percent nuclear boiler rated (NBR) power level, a MSV scram trip inhibit signal, derived from the first stage pressure of the turbine, is activated. This prevents the MSV trip scram signal from scramming the reactor, provided the bypass system functions properly. In other words, the bypass would be sufficient at this low power to accommodate a turbine trip without the necessity of shutting down the reactor. All other protection system functions remain operational, as before, and credit is taken for those protection system trips.

15.2.3.2.3 The Effect of Single Failures and Operator Errors

15.2.3.2.3.1 Turbine Trips at Power Levels Greater Than 24 percent

Mitigation of pressure increase, the basic nature of this transient, is accomplished by the RPS function. MSV closure trip scram and recirculation pump trip (RPT) are designed to satisfy the single failure criterion.

15.2.3.2.3.2 Turbine Trips at Power Levels Less Than 24 Percent NBR

This event occurs as described in Section 15.2.3.2.3.1, except RPT and MSV closure trip scram is normally inoperative. Since protection is still provided by high flux, high pressure, etc, these also continue to function and scram the reactor if a single failure occurs.

15.2.3.3 Core and System Performance

15.2.3.3.1 Mathematical Model

The computer model described in Section 15.1.2.3.1 was used to simulate these events.

15.2.3.3.2 Input Parameters and Initial Conditions

The full stroke closure time of the MSV is 0.1 second.

A reactor scram is initiated by position switches on the MSVs when the valves are less than 90 percent open. This MSV scram trip signal is automatically bypassed when the reactor is below 24 percent NBR power level.

Reduction in core recirculation flow is initiated by position switches on the MSVs, which actuate trip circuitry that trips the recirculation pumps.

15.2.3.3.3 Results

15.2.3.3.3.1 Turbine Trip

The results presented below are representative of CPPU conditions. References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the CPPU licensed power level of 3840 MW_{th}. These results are considered bounded by the reload licensing analysis.

A turbine trip with the bypass system operating normally is simulated at 100 percent NBR power and 105percent core flow conditions on Figure 15.2-3.

Neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the flux increase is limited to 273.5 percent NBR by the scram initiated by the MSV trip and the RPT system. Peak fuel surface heat flux does not exceed 112.2 percent of its initial value. Minimum critical power ratio (MCPR) remains above the safety limit.

15.2.3.3.3.2 Turbine Trip with Failure of Bypass

The results of the turbine trip with failure of bypass are presented in Appendix 15D.

15.2.3.3.3.3 Turbine Trip with Bypass Valve Failure, Low Power

This transient is less severe than a similar one at high power. Below 24 percent of rated power, the MSV closure, turbine control valve closure scrams and the RPT are automatically bypassed. At these lower power levels, turbine first stage pressure is used to initiate the scram logic bypass. The scram that terminates the transient is initiated by high neutron flux or high vessel pressure. The bypass valves are assumed to fail. Therefore, system pressure increases until the pressure relief setpoints are reached. At this time, because of the relatively low power of this transient event, relatively few SRVs open to limit reactor pressure.

15.2.3.3.4 Considerations of Uncertainties

Uncertainties in these analyses involve protection system settings, system capacities, and system response characteristics. In all cases, the most conservative values are used in the analyses. For example:

1. Slowest allowable control rod scram motion is assumed.

2. Scram worth shape for all-rods-out conditions is assumed.

3. Minimum specified SRVs capacities are used for overpressure protection.

4. Analytical setpoints of the SRVs are approximately 3 percent higher than the valves' specified setpoints.

15.2.3.4 Barrier Performance

15.2.3.4.1 Turbine Trip

Peak pressure in the bottom of the vessel reaches 1195 psig, which is below the ASME B&PV Code limit of 1375 psig for the RCPB. 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 steam flow is within the bypass capability.

15.2.3.4.2 Turbine Trip with Failure of the Bypass

The SRVs open and close sequentially as the stored energy is dissipated and the pressure falls below the setpoints of the valves. The results presented in Appendix 15D show that the peak nuclear system pressure remains below the RCPB pressure limit of 1375 psig.

15.2.3.4.2.1 Turbine Trip with Failure of Bypass at Low Power

Qualitative discussion is provided in Section 15.2.3.3.3.3.

15.2.3.5 Radiological Consequences

While this event does not result in fuel failure, it does result in the discharge of normal coolant to the suppression pool via SRVs. Since this activity is contained in the primary containment, there is no exposure to operating personnel. Since this event also does not result in an uncontrolled release to the environment, the plant operator can choose to hold the activity in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release is in accordance with established Hope Creek Technical Specification limits.

15.2.4 Main Steam Isolation Valve Closures

The main steam isolation valve closure events are considered a non-Limiting event. Therefore it is not required to be re-analyzed as a part of the reload licensing analysis for Hope Creek, unless the disposition for this event changes (Reference 15.2-2).

The results referenced within this section are representative of CPPU conditions. References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the CPPU licensed power level of 3840 MW_{th}. These results are considered bounded by the reload licensing analysis.

15.2.4.1 Identification of Causes and Frequency Classification

15.2.4.1.1 Identification of Causes

Various steam line and nuclear system malfunctions or operator actions can initiate main steam isolation valve (MSIV) closure. Examples are low steam line pressure, high steam line flow, low water level, or manual action.

15.2.4.1.2 Frequency Classification

15.2.4.1.2.1 Closure of All MSIVs

This event is categorized as an incident of moderate frequency. To define the frequency of this event as an initiating event and not as the byproduct of another transient, only the following contribute to the frequency:

1. Manual action (intended or inadvertent)

2. Spurious signals, such as low pressure, low reactor water level, and low condenser vacuum

3. Equipment malfunctions, such as faulty valves.

Depending on reactor conditions, closure of one MSIV may cause immediate closure of all the other MSIVs. If this occurs, it is also included in this event category. During the MSIV closure, position switches on the valves provide a reactor scram if the valves in three or more main steam lines are less than 90 percent open, (See Note 1, Table 15.2-5) except for interlocks that permit proper plant startup. However, protection system logic permits the test closure of one MSIV without initiating scram from the position switches.

15.2.4.1.2.2 Closure of One MSIV

This event is categorized as an incident of moderate frequency. One MSIV may be manually closed for testing purposes. (The MSIVs are tested weekly at five percent closure and quarterly at 100 percent closure.) Operator error or equipment malfunction may cause a single MSIV to be inadvertently closed. If reactor power is greater than about 80 percent when this occurs, a high flux scram or high steam line flow isolation may result. If all MSIVs close as a result of the single closure, the event is considered a closure of all MSIVs.

15.2.4.2 Sequence of Events and Systems Operation

15.2.4.2.1 Sequence of Events

Table 15.2-5 lists the sequence of events for Figure 15.2-5.

The above results are representative of CPPU conditions. References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the CPPU licensed power level of 3840 MW_{th}. These results are considered bounded by the reload licensing analysis.

15.2.4.2.1.1 Identification of Operator Actions

The following is the sequence of operator actions expected during the course of the event, assuming no restart of the reactor. In this case, the operator will:

1. Observe that all rods have been inserted.

2. Observe that the main steam safety/relief valves (SRVs) have opened for reactor pressure control.

3. Check that reactor core isolation cooling (RCIC)/high pressure coolant injection (HPCI) automatically starts at low-low (L2) reactor water level.

4. Switch the feedwater controller to the manual position.

5. Determine the cause of valve closure before resetting the MSIV isolation.

6. Observe turbine coastdown and break vacuum before the loss of sealing steam; check turbine generator auxiliaries for proper operation.

7. Reset and open MSIVs if conditions warrant and ensure that the pressure regulator setpoint is above vessel pressure.

8. Secure RCIC/HPCI after the reactor vessel level has recovered to a satisfactory level.

9. Initiate shutdown cooling after reactor pressure has decayed sufficiently for RHR operation.

10. Survey maintenance requirements and complete the scram report.

15.2.4.2.2 Systems Operation

15.2.4.2.2.1 Closure of All MSIVs

MSIV closures initiate a reactor scram via position signals to the Reactor Protection System (RPS). Credit is taken for successful operation of the RPS.

The Pressure Relief System, which initiates the opening of the SRVs when system pressure exceeds relief valve instrumentation setpoints, is assumed to function normally during the time period analyzed. All plant control systems are assumed to be functional.

15.2.4.2.2.2 Closure of One MSIV

A closure of a single MSIV at any given time does not initiate a reactor scram. This is because the valve position trip scram logic is designed to accommodate single valve operation and testability during normal reactor operation at limited power levels. Credit is taken for the operation of the pressure and flux signals to initiate a reactor scram. All plant control systems are assumed to be functional.

15.2.4.2.3 The Effect of Single Failures and Operator Errors

Mitigation of pressure increase is accomplished by initiation of the reactor scram via MSIV position switches and the RPS. SRVs also operate to limit system pressure. All of these aspects are designed to the single failure criterion and additional single failures would not alter the results of this analysis.

Failure of a single SRV to open is included in the CPPU analysis. The peak pressure still remains below 1375 psig. The design basis and performance of the pressure relief system is discussed in Section 5.2.2.

15.2.4.3 Core and System Performance

15.2.4.3.1 Mathematical Model

The predicted dynamic behavior has been determined using the computer model described in Section 15.1.2.3.1. This model is described in detail in Reference 15.1-2.

15.2.4.3.2 Input Parameters and Initial Conditions

The MSIVs close in 3 to 5 seconds. The worst case, which is the 3-second closure time, is assumed in this analysis. The analysis performed for CPPU conditions conservatively assumes a 2.4 second closure time to address uncertainty in plant measurements when actual closure time tests are performed.

Position switches on the MSIVs initiate a reactor scram when the valves are less than 90 percent open (See Note 1, Table 15.2-5). Closure of these valves inhibits steam flow to the feedwater turbines terminating feedwater flow.

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

15.2.4.3.3 Results

The results presented below are representative of CPPU conditions. References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the CPPU licensed power level of 3840 MW_{th}. These results are considered bounded by the reload licensing analysis.

15.2.4.3.3.1 Closure of All MSIVs

Figure 15.2-5 shows the changes in important nuclear system variables for the simultaneous isolation of all main steam lines while the reactor is operating at 100 percent of nuclear boiler rated (SBR) power and 105 percent core flow. Due to the nonlinear valve characteristics, the initial movement of the valves would not cause any significant pressurization. However, the conservative assumed scram reactivity is insufficient to prevent neutron flux from increasing as the reactor pressure increases when the valves approach full closure. Therefore, the neutron flux increases to a level of 233.1 percent. The average surface heat flux peaks at 103.5 percent of its initial value.

15.2.4.3.3.2 Closure of One MSIV

To prevent a scram, only one isolation valve at a time is permitted to be closed for testing purposes. Normal test procedure requires an initial power reduction to the range 80 to 90 percent of design conditions in order to avoid high flux scram, high pressure scram, or full isolation from high steam flow in the live lines. With a 3-second closure of one MSIV at CPPU conditions, the steam flow disturbance may raise vessel pressure and reactor power enough to initiate a high neutron flux scram. This transient is considerably milder than closure of all MSIVs at full power. No quantitative analysis is furnished for this event. No significant change in thermal margins is experienced and no fuel damage occurs. Peak pressure remains below SRV setpoints.

As described in Section 15.9, inadvertent closure of one or all of the MSIV while the reactor is shut down produces no significant transient. Closures during plant heatup, operating state D, will be less severe than those in the maximum power cases discussed in Section 15.2.4.3.3.1.

Uncertainties in these analyses involve protection system settings, system capacities, and system response characteristics. In all cases, the most conservative values are used in the analyses. For example:

1. Slowest allowable control rod scram motion is assumed.
2. Scram worth shape for all-rods-out conditions is assumed.
3. Minimum specified SRVs' capacities are used for overpressure protection.
4. Analytical setpoints of the SRVs are assumed to be about three percent higher than the specified setpoints.

15.2.4.4 Barrier Performance

15.2.4.4.1 Closure of All MSIVs

The SRVs begin to open approximately 3 seconds after the start of isolation. The valves close sequentially as the stored heat is dissipated but continue to intermittently discharge the steam resulting from decay heat. Peak pressure at the vessel bottom reaches 1229 psig, clearly below the pressure limits of the reactor coolant pressure boundary (RCPB).

15.2.4.4.2 Closure of One MSIV

No significant effect is imposed on the RCPB, since if closure of the valve occurs at an unacceptably high operating power level, a flux or pressure scram results. The main Turbine Bypass System continues to regulate system pressure via the other three "live" steam lines.

15.2.4.5 Radiological Consequences

While this event does not result in fuel failures, it does result in the discharge of normal coolant to the suppression pool via SRVs. Since this activity is contained in the primary containment, there is no exposure to operating personnel. Since this event also does not result in an uncontrolled release to the environment, the plant operator can choose to hold the activity in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will be in accordance with established Hope Creek Technical Specification limits.

15.2.5 Loss of Condenser Vacuum

The loss of condenser vacuum event is considered a non-Limiting event. Therefore it is not required to be re-analyzed as a part of the reload licensing analysis for Hope Creek, unless the disposition for this event changes (Reference 15.2-2).

The results referenced within this section are representative of cycle 1. **References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}.** These results are considered bounded by the reload licensing analysis.

15.2.5.1 Identification of Causes and Frequency Classification

15.2.5.1.1 Identification of Causes

Various system malfunctions that can cause a loss of condenser vacuum due to single equipment failure are designated in Table 15.2-6.

15.2.5.1.2 Frequency Classification

This event is categorized as an incident of moderate frequency.

15.2.5.2 Sequence of Events and Systems Operation

15.2.5.2.1 Sequence of Events

Table 15.2-7 lists the sequence of events for the loss of condenser vacuum transient, shown on Figure 15.2-6.

The above results are representative of cycle 1. **References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}.** These results are considered bounded by the reload licensing analysis.

15.2.5.2.1.1 Identification of Operator Actions

In the event of loss of condenser vacuum, the operator will:

1. Monitor and maintain reactor water level at the required level.
2. Check the turbine for proper operation of all auxiliaries during coastdown.
3. Depending on conditions, initiate normal operating procedures for cooldown, or maintain pressure for restart purposes.

4. Put the Reactor Protection System (RPS) mode switch in the startup position before the reactor pressure decays to <850 psig.

5. Secure the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) operation if automatic initiation occurred due to low reactor vessel water level.

6. Monitor control rod drive (CRD) positions and insert both the intermediate range monitors (IRMs) and source range monitors (SRMs).

7. Investigate the cause of the trip, make repairs as necessary, and complete the scram report.

8. Cooldown the reactor per standard procedure if a restart is not intended.

15.2.5.2.2 Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it was assumed that the plant instrumentation and controls, plant protection, and reactor protection systems were functioning normally.

Trip functions initiated by sensing main turbine condenser vacuum pressure are designated in Table 15.2-8.

15.2.5.2.3 The Effect of Single Failures and Operator Errors

This event does not lead to a general increase in reactor power level. Power increase is mitigated by the scram.

Failure of the integrity of the Off-gas Treatment System is considered an accident situation and is described in Section 15.7.1.

Single failures do not affect the vacuum monitoring and turbine trip devices that are redundant. Further discussion of the effects of a single failure is presented Section 15.9.

15.2.5.3 <u>Core and System Performance</u>

15.2.5.3.1 Mathematical Model

The computer model described in Section 15.1.2.3.1 was used to simulate this transient event.

15.2.5.3.2 Input Parameters and Initial Conditions

This analysis was performed with plant conditions as tabulated in Table 15.0-3, unless otherwise noted.

The main stop valve (MSV) full stroke closure time is 0.1 second.

A reactor scram is initiated by position switches on the MSVs when the valves are less than 90 percent open. This MSV trip scram signal is automatically bypassed when the reactor is below 30 percent nuclear boiler rated (NBR) power level.

The analysis presented here is a hypothetical case with a vacuum decay rate of 2 inches of mercury per second. Thus, the bypass system is available for several seconds, because the bypass is signaled to close at a vacuum level of about 10 inches of mercury less than the MSV closure.

15.2.5.3.3 Results

The results presented below are representative of cycle 1. **References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}.** These results are considered bounded by the reload licensing analysis.

Using this rate of vacuum decay conditions, the turbine bypass valve and main steam isolation valve (MSIV) closure would follow main turbine and feedwater turbine trips about 5 seconds after the trips initiate the transient. This transient, therefore, is similar to a normal turbine trip with bypass. The effect of MSIV closure tends to be minimal, since the closure of MSVs and the

subsequent closure of the bypass valves have already shut off the main steam line flow. Figure 15.2-6 shows the transient expected for this event. It is assumed that the plant is initially operating at 105 percent of Cycle 1 NBR steam flow conditions. Peak neutron flux reaches 132.4 percent of NBR power while average fuel surface heat flux reaches 104.7 percent of rated value main steam safety/relief valves (SRVs) open to limit the pressure rise, and then sequentially reclose as the stored energy is dissipated.

15.2.5.3.4 Considerations of Uncertainties

The reduction or loss of vacuum in the main turbine condenser sequentially trips the main and feedwater turbines and closes the MSIVs and bypass valves. While these are the major events, scram from MSV closure and bypass opening with the main turbine trip are also included. Because the protective actions are actuated at various levels of condenser vacuum, the severity of the resulting transient is directly dependent upon the rate at which the vacuum pressure is lost. Normal loss of vacuum due to loss of cooling water pumps, or a steam jet air ejector problem, produces a very slow rate of loss of vacuum, i.e., in minutes, not seconds (see Table 15.2-6 for comparison). If corrective actions by the reactor operators are not successful, then the main and feedwater turbines trip simultaneously, and ultimately, complete isolation occurs by closing the MSIVs and the bypass valves that have been opened by the main turbine trip.

A faster rate of loss of the condenser vacuum would reduce the anticipatory action of the scram and the overall effectiveness of the bypass valves, since they would be closed more quickly.

Other uncertainties in these analyses involve RPS settings, system capacities, and system response characteristics. In all cases, the most conservative values are used in the analyses. For example:

1. Slowest allowable control rod scram motion is assumed.

2. Scram worth shape for all rods out conditions is assumed.

3. Minimum specified SRVs' capacities are used for overpressure protection.

4. Analytical setpoints of the SRVs are assumed about 1 percent higher than the specified setpoints.

15.2.5.4 Barrier Performance

Peak pressure is 1178 psig at the vessel bottom, which is below the reactor coolant pressure boundary (RCPB) transient pressure limit of 1375 psig. Vessel dome pressure does not exceed 1151 psig. A comparison of these values to those for turbine trip with bypass failure at high power (see Section 15.2.3.4.2) shows the similarities between these two transients. The differences are the loss of feedwater and main steam line isolation, and the resulting low water level trips.

15.2.5.5 Radiological Consequences

While 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 is no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to hold the activity in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will be in accordance with established Hope Creek Technical Specification limits.

15.2.6 Loss of AC Power

The loss of AC power event is considered a non-Limiting event. Therefore it is not required to be re-analyzed as a part of the reload licensing analysis for Hope Creek, unless the disposition for this event changes (Reference 15.2-2).

The results referenced within this section are representative of cycle 1. References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}. These results are considered bounded by the reload licensing analysis.

15.2.6.1 Identification of Causes and Frequency Classification

15.2.6.1.1 Identification of Causes

15.2.6.1.1.1 Loss of Auxiliary Power

Loss of auxiliary power to the station auxiliary power buses can be caused by any one of the following:

1. Loss of the two interconnections between the 500-kV switchyard at Hope Creek and the 13.8-kV ring bus.

2. Failure of station power and station service transformers such that none of the buses is energized. Section 8.3 discusses the electrical distribution.

The loss of auxiliary-power event is the same as the loss of all grid-connections event, because of the HCGS switchyard design. Section 8.1 discusses the switchyard design. Hence, the loss of ac power event will only be discussed in terms of loss of all grid-connection event consequences.

15.2.6.1.1.2 Loss of All Grid Connections

The loss of all grid connections can result from major shifts in electrical loads, loss of loads, lightning, storms, wind, etc, which contribute to electrical grid instabilities. These instabilities cause equipment damage if unchecked. Protective relay schemes automatically disconnect electrical sources and loads to mitigate damage and regain electrical grid stability.

15.2.6.1.2 Frequency Classification

The loss of all grid connections is categorized as an incident of moderate frequency.

15.2.6.2 Sequence of Events and Systems Operation

15.2.6.2.1 Sequence of Events

Table 15.2-9 lists the sequence of events for loss of all grid connections. Figure 15.2-7 provides the results of the events analysis.

The above results are representative of cycle 1. **References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}.** These results are considered bounded by the reload licensing analysis.

15.2.6.2.1.1 Identification of Operator Actions

In the event of loss of ac power, the operator will:

1. Maintain the reactor water level by use of the Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) Systems.

2. Control reactor pressure by use of the main steam safety/relief valves (SRVs).

3. Verify that the turbine dc oil pump is operating satisfactorily to prevent turbine bearing damage.

4. Verify proper switching and loading of the standby diesel generators.

The following is the sequence of operator actions expected during the course of the events when no immediate restart is assumed. In this case, the operator will:

1. Verify all the rods are in, following the scram.

2. Check that the standby diesel generators (SDGs) start and carry the vital loads.

3. Check that both RCIC and HPCI systems start when the reactor vessel level drops to the initiation point after the SRV opens.

4. Break the vacuum before the loss of sealing steam occurs.

5. Check turbine generator auxiliaries during coastdown.

6. When both the reactor pressure and level are under control, secure both HPCI and RCIC systems as necessary.

7. Continue cooldown per the applicable procedures.

8. Complete the scram report and survey the maintenance requirements.

15.2.6.2.2 Systems Operation

The loss of all grid connections, unless otherwise stated, assumes and takes credit for normal functioning of plant instrumentation and controls, plant protection, and RPS.

The reactor is subjected to a complex sequence of events when the plant loses all grid connections. Estimates of the responses of the various reactor systems, assuming loss of all grid connections, provide the following simulation sequence:

1. A generator load rejection occurs at time t=0, which immediately forces the turbine control valves (TCVs) closed and causes a scram.

2. The reactor recirculation pumps are tripped at reference time t=0 with normal coastdown times.

3. Independent main steam isolation valve (MSIV) closure is initiated due to loss of power.

4. At approximately 2 seconds the feedwater pump trips are initiated.

Operation of the HPCI and RCIC system functions is not simulated in this analysis. Their operation occurs after fuel thermal margin and overpressure effects are of concern.

15.2.6.2.3 The Effect of Single Failures and Operator Errors

Loss of all grid connections leads to a reduction in power level due to rapid recirculation pump coastdown and pressurization due to MSIV closure after the reactor scram. Failures in protection systems have been considered, and satisfy the single failure criteria. No change in analyzed consequences is expected. See Section 15.9 for details on single failure analysis.

15.2.6.3 Core and System Performance

15.2.6.3.1 Mathematical Model

The computer model described in Section 15.1.2.3.1 was used to simulate this event.

Operation of the RCIC or HPCI system is not included in the simulation of this transient.

15.2.6.3.2 Input Parameters and Initial Conditions

The loss of all grid connections event has been performed, unless otherwise noted, with plant conditions as tabulated in Table 15.0-3 and under the assumed systems constraints described in Section 15.2.6.2.2.

15.2.6.3.3 Results

The results presented below are representative of cycle 1. **References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level** of 3293 MW_{th}. These results are considered bounded by the reload licensing analysis.

Figure 15.2-7 shows graphically the simulated loss of all grid connections transient. The results are similar to the load rejection discussed in Section 15.2.2. Peak neutron flux reaches 120.6 percent of nuclear boiler rated (NBR) power while fuel surface heat flux peaks at 100.1 percent of initial value.

15.2.6.3.4 Consideration of Uncertainties

The most conservative characteristics of protection features are assumed. Any actual deviations in plant performance are expected to make the results of this event less severe.

Operation of the RCIC or HPCI systems is not included in the simulation of the first 50 seconds of this transient. Startup of these pumps occurs in the latter part of this time period and has no significant effect on the results of this transient.

The trip of the feedwater turbines may occur earlier than simulated if the inertia of the condensate and booster pumps is not sufficient to maintain feedwater pump suction pressure above the low suction pressure trip setpoint. The simulation assumes sufficient inertia and thus the feedwater pumps are not tripped until 2 seconds after MSIV closure.

Following main steam line isolation, the reactor pressure is expected to increase until the SRV setpoints are reached. During this time, the SRVs operate in a cyclic manner to discharge the decay heat to the suppression pool.

15.2.6.4 <u>Barrier Performance</u>

For the loss of all grid connection event, the SRVs open in the pressure relief mode of operation as the pressure increases beyond their setpoints. The pressure at the bottom of the vessel is limited to a maximum value of 1198 psig, well below the vessel pressure limit of 1375 psig.

15.2.6.5 <u>Radiological Consequences</u>

While this event does not result in fuel failure, it does result in the discharge of normal coolant to the suppression pool via SRVs. Since this activity is contained in the primary containment, there is no exposure to operating personnel. Since

this event also does not result in an uncontrolled release to the environment, the plant operator can choose to hold the activity in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will be in accordance with established Hope Creek Technical Specification limits.

15.2.7 Loss of Feedwater Flow

The loss of feedwater flow event is considered a non-Limiting event. Therefore it is not required to be re-analyzed as a part of the reload licensing analysis for Hope Creek, unless the disposition for this event changes (Reference 15.2-2).

The results referenced within this section are representative of Cycle 1. References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}. In addition, results for reactor level response are provided for the CPPU condition. These results are considered bounded by the reload licensing analysis.

15.2.7.1 Identification of Causes and Frequency Classification

15.2.7.1.1 Identification of Causes

A loss of feedwater flow could occur from pump failures, feedwater controller failures, operator errors, the high vessel water level (L8) feedwater pump trip signal, or Reactor Protection System (RPS) trips.

15.2.7.1.2 Frequency Classification

This transient is categorized as an incident of moderate frequency.

15.2.7.2 Sequence of Events and Systems Operation

15.2.7.2.1 Sequence of Events

Table 15.2-10 lists the sequence of events for Figure 15.2-8.

The above results are representative of cycle 1. References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}. These results are considered bounded by the reload licensing analysis.

15.2.7.2.2 Identification of Operator Actions

In the event of loss of feedwater flow, the operator will:

1. Ensure reactor core isolation cooling (RCIC) and high pressure coolant injection (HPCI) actuation so that water inventory is maintained in the reactor vessel.

2. Monitor and control reactor water level and pressure.

3. Monitor turbine generator auxiliaries during shutdown.

The following is the sequence of operator actions expected during the course of the event when no immediate restart is assumed. The operator will:

1. Verify that all rods are in, following the scram.

2. Verify HPCI and RCIC initiation.

3. Verify that the main steam safety/relief valves (SRVs) open on reactor high pressure.

4. Verify that the reactor recirculation pumps trip on reactor low-low level.

5. Secure HPCI when reactor level and pressure are under control.

6. Continue operation of the RCIC system until decay heat diminishes to a point where the RHR system can be put into service.

7. Monitor the turbine coastdown and break the vacuum as necessary.

8. Complete the scram report and survey the maintenance requirements.

15.2.7.2.3 System Operation

Loss of feedwater flow results in a proportional reduction of vessel inventory causing the vessel water level to drop. The first corrective action is the low level, L3, scram actuation. RPS responds in about 1 second after this trip to scram the

reactor. The low level, L3, scram function meets the single failure criterion.

Containment isolation, when it occurs, would also initiate a main steam isolation valve (MSIV) position scram signal as part of the normal isolation event. The reactor, however, is already scrammed and shut down by this time.

15.2.7.2.4 The Effect of Single Failures and Operator Errors

This event results in a lowering of vessel water level. Key corrective efforts to shut down the reactor are automatic and are designed to satisfy the single failure criterion. Therefore, any additional failure in these shutdown methods would not aggravate or change the simulated transient. See Section 15.9 for details.

The potential exists for a single SRV failing to close once it is opened. This is discussed in Section 15.1.4. Either the HPCI or RCIC system is capable of maintaining adequate core coverage and provides long term inventory control.

15.2.7.3 Core and System Performance

15.2.7.3.1 Mathematical Model

The computer model described in Section 15.1.2.3.1 is used to simulate this event.

15.2.7.3.2 Input Parameters and Initial Conditions

These analyses have been performed with plant conditions as tabulated in Table 15.0-3.

15.2.7.3.3 Results

The results presented below are representative of cycle 1. **References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level** of 3293 MW_{th}. These results are considered bounded by the reload licensing analysis.

The results of this transient simulation are shown on Figure 15.2-8. Feedwater flow terminates at approximately

5 seconds after initiation of the accident. Subcooling decreases, causing a reduction in core power level and pressure. As power level is lowered, the turbine steam flow starts to drop off because the pressure regulator is initially attempting to maintain pressure for approximately 7 seconds. Water level continues to drop until the vessel level, L3, scram setpoint is reached, whereupon the reactor is shut down. When water level drops to L2, the recirculation system is also tripped, and HPCI and RCIC system operation is initiated. In the CPPU analysis, the HPCI system is assumed to fail. The minimum reactor water level inside the shroud reaches 446 inches above vessel zero (AVZ) which is 80 inches above top of active fuel (TAF). The minimum water level in the vessel downcommer reaches 389 inches AVZ. Minimum critical power ratio (MCPR) remains considerably above the safety limit since increases in heat flux are not experienced.

15.2.7.3.4 Considerations of Uncertainties

End of cycle scram characteristics are assumed.

This transient is more severe at high power conditions, because the rate of water level decrease is greater and the amount of stored and decay heat to be dissipated is higher.

Operation of the RCIC or HPCI systems is not included in the simulation of the first 50 seconds of this transient since startup of these pumps occurs in the latter part of this time period. The CPPU analysis demonstrates that the RCIC system alone is capable of maintaining reactor water level above the top of active fuel.

15.2.7.4 <u>Barrier Performance</u>

Peak pressure in the bottom of the vessel reaches 1059 psig, which is below the ASME B&PV Code limit of 1375 psig for the RCPB. Vessel dome pressure does not exceed 1020 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. These barriers maintain their integrity and function as designed.

15.2.7.5 <u>Radiological Consequences</u>

While this event does not result in fuel failure, it does result in the discharge of normal coolant to the suppression pool via SRVs. Since this activity is contained in the primary containment, there is no exposure to operating personnel. Since this event also does not result in an uncontrolled release to the environment, the plant operator can choose to hold the activity in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will be in accordance with established Hope Creek Technical Specification limits.

15.2.8 Feedwater Line Break

Feedwater line break is discussed in Section 15.6.6.

15.2.9 Failure of RHR Shutdown Cooling

The Failure of RHR Shutdown Cooling event is considered a non-Limiting event. Therefore it is not required to be re-analyzed as a part of the reload licensing analysis for Hope Creek, unless the disposition for this event changes (Reference 15.2-2). This event was not re-evaluated for CPPU conditions.

Normally, in evaluating component failure considerations associated with the RHR system shutdown cooling mode, active pumps or instrumentation (all of which are redundant for safety system portions of the RHR system) would be assumed to be the likely failed equipment. For purposes of worst case analysis, the single recirculation loop suction valve to the redundant RHR loops is assumed to fail or a loss of offsite power (LOP) isolates the RHR system shutdown cooling from the reactor vessel. This failure or the LOP would, of course, still leave four complete RHR loops for low pressure coolant injection (LPCI) and four Core Spray pumps, or two RHR loops for LPCI plus four Core Spray pumps and two RHR loops for pool and containment cooling minus the normal RHRS shutdown cooling loop connections.

15.2.9.1 <u>Identification of Causes and Frequency Classification</u>

15.2.9.1.1 Identification of Causes

The plant is operating at 105 percent nuclear boiler rated (NBR) steam flow when a loss of offsite power (LOP) occurs, causing

multiple main steam safety/relief valve (SRV) actuation, as discussed in Section 15.2.6, and subsequent heatup of the suppression pool. Reactor vessel depressurization is initiated to bring the reactor pressure to approximately 100 psig. A LOP or a single failure occurrence would prevent the operator from establishing the normal shutdown cooling path through the RHR shutdown cooling lines. The operator would then establish a shutdown cooling path for the vessel through the SRVs and vessel inventory makeup.

15.2.9.1.2 Frequency Classification

Recent analytical evaluations of this event have required additional worst case assumptions including:

1. Loss of all offsite ac power

2. Use of safe shutdown equipment only

3. Operator action after 10 minutes.

These accident assumptions change the initial incident, which was the malfunction of an RHR shutdown cooling suction supply valve, from a moderate frequency incident to a classification in the design basis accident (DBA) status. However, the event is evaluated as a moderate frequency event.

15.2.9.2 <u>Sequence of Events and System Operation</u>

15.2.9.2.1 Sequence of Events

The sequence of events for this event is shown in Table 15.2-11.

15.2.9.2.1.1 Identification of Operator Actions

The operator will:

1. At 10 minutes after the isolation/scram, initiate reactor pressure vessel (RPV) shutdown depressurization at approximately 100°F/h by manual actuation of the SRVs; maintain water level with the Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) Systems.

2. After 10 minutes into the transient, initiate suppression pool cooling (again, for purposes of this analysis, it is assumed that only one RHR heat exchanger is available).

3. After the RPV is depressurized to approximately 100 psig, attempt to restore offsite power and RHR shutdown cooling.

4. If Step 3 is unsuccessful, actuate SRVs as required to establish a closed cooling path, as described in the notes for Figure 15.2.9.

15.2.9.2.2 System Operation

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 use of engineered safety features (ESFs).

15.2.9.2.3 The Effect of Single Failures and Operator Errors

The worst case single failure, loss of one DC division, is considered in this event. Therefore, no single failure or operator error makes the consequences of this event any worse. See Section 15.9 for a more detailed discussion.

15.2.9.3 Core and System Performance

15.2.9.3.1 Methods, Assumptions, and Conditions

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. During this time, minimum critical power ratio (MCPR) remains high and nucleate boiling heat transfer is not exceeded. Therefore, the core thermal safety margin remains essentially unchanged. The 10-minute time period assumed for operator action is an estimate of how long it would take the operator to initiate the necessary actions; it is not a time by which action must be initiated.

15.2.9.3.2 Mathematical Model

Only a qualitative evaluation is provided.

15.2.9.3.3 Input Parameters and Initial Conditions

Only a qualitative evaluation is provided.

15.2.9.3.4 Qualitative 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 of a loss of offsite power or where both of the RHR shutdown cooling suction valves cannot be opened, alternate paths are available to accomplish the shutdown cooling function as shown on Figure 15.2-9.

The evaluation demonstrates the capability to safely transfer fission product decay heat and other residual heat from the

reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded. It ensures that the safety function can be accomplished, assuming a worst case single failure.

The alternate cooldown path chosen to accomplish the shutdown cooling function uses the RHR and SRV systems. For more detail, see Reference 15.2-1.

The alternate shutdown systems are capable of performing the function of transferring heat from the reactor to the environment using only safety grade systems. Even if it is additionally postulated that all of the SRV discharge piping breaks, the shutdown cooling function eventually is accomplished as the cooling water runs directly out of the SRVs, flooding into the drywell and then into the suppression pool.

These systems also have suitable redundancy in components such that their safety function can be accomplished assuming an additional single failure in either power mode for both onsite electrical power operation, assuming offsite power is not available, and offsite electrical power operation, assuming onsite power is also not available. The systems can be fully operated from the main control room, and sufficient SRVs can be operated from the Remote Shutdown Panel or lower relay room, such that functioning of the Alternate Shutdown Method is assured.

The design evaluation is divided into the following two phases:

1. Full power operation to approximately 100-psig vessel pressure

2. Approximately 100 psig vessel pressure to cold shutdown (14.7 psia and 200°F) conditions.

15.2.9.3.4.1 Full Power to Approximately 100 psig

Independent of the event that initiated plant shutdown, whether a normal plant shutdown or a forced plant shutdown, the reactor is

normally brought to approximately 100 psig using either the main condenser or, if the main condenser is unavailable, the RCIC and HPCI systems, together with the SRVs.

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

These systems are designed to routinely perform their functions for both normal and forced plant shutdown. Since the RCIC, HPCI and RHR systems are divisionally separated, no single failure together with the LOP is capable of preventing reaching the 100 psig level.

15.2.9.3.4.2 Approximately 100 psig to Cold Shutdown

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

1. The vessel is at 100 psig and saturated conditions

2. A worst case single failure is assumed to have occurred, i.e., loss of a division of emergency power

3. No offsite power is available.

Since RHR shutdown cooling mode is isolated due to the assumed loss of offsite power, the alternate shutdown cooling mode is used. In the event that offsite power is available and the RHR shutdown suction line is not available because of single failure, personnel must gain access and attempt

to effect repairs. For example, if a single electrical failure caused a suction valve to fail in the closed position, a hand wheel is provided on the valve to allow manual operation. If, for some reason, the normal shutdown cooling suction line cannot be repaired or for the assumed loss of offsite power, the capabilities described below satisfy the normal shutdown cooling requirements and fully comply with General Design Criterion (GDC) 34.

To satisfy containment isolation criteria, the RHR shutdown cooling line valves are divided as follows:

1. AC division 1 - the inboard valves

2. AC division 4 - the outboard valves

For evaluation purposes, the worst case failure is assumed to be the loss of a division of emergency power, since this prevents actuation of one alternate shutdown cooling function. ESF equipment available for accomplishing the alternate shutdown cooling function for the selected path includes:

1. ADS (dc divisions 2 and 4)

2. RHR loops A and C (dc divisions 1 and 3)

3. RHR loops B and D (dc divisions 2 and 4)

4. HPCI (dc divisions 1 and 3)

5. RCIC (dc divisions 2 and 4)

6. Core spray A and C (dc divisions 1 and 3)

7. Core spray B and D (dc divisions 2 and 4)

For failures of dc division 1 or 2, the following systems are assumed functional:

1. DC division 1 fails, dc divisions 2, 3, and 4 function:

Failed systems	Functional systems
RHR pumps A	RCIC
CS loop A	ADS
HPCI	RHR loop B
	CS loop B
	RHR pumps B, C, & D

2. DC division 2 fails, dc divisions 1, 3, and 4 function:

Failed systems	Functional systems
RHR pump B	CS loop A
CS loop B	HPCI
RCIC	RHR loop A
	RHR pumps A, C, & D
	ADS (one solenoid)

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 15.2-11. If the assumed single failure is division 1, the safety function is accomplished by establishing the cooling loop described as activity C1 of Figure 15.2-10.

15.2.9.4 Barrier Performance

This event does not result in any temperature or pressure transient in excess of the design criteria for the fuel, pressure vessel, or containment. Coolant is released to the containment by SRVs. Release of radiation to the environment is described below.

15.2.9.5 Radiological Consequences

While 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 are no exposures to operating personnel. Since this event also does not result in an uncontrolled release to the environment, the plant operator can choose to hold the activity in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will be in accordance with established Hope Creek Technical Specification limits.

15.2.10 References

15.2-1 Fukushima, T. Y, "HEX01 User Manual," NEDE-23014, July 1976.

15.2-2 "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (latest approved revision), and "General Electric Standard Application for Reactor Fuel (Supplement for United States)", NEDE-24011-P-A-US (latest approved revision).

15.2-3 Deleted

TABLE 15.2-1

SEQUENCE OF EVENTS FOR GENERATOR LOAD REJECTION WITH BYPASS
(FIGURE 15.2-1)

Time, s	Event
<0	Loss of electrical load detected by the turbine generator.
0	Turbine generator load rejection sensing devices trip to initiate TCV fast closure and main turbine bypass system operation.
0.03	Fast closure of TCV initiates scram and RPT.
0.1	Turbine bypass valves start to open.
0.15	TCVs are closed.
0.28	Start of control rod motion.
0.38	Recirculation Pump Trip (RTP) occurs.
2.27	Group 1 SRVs are actuated.

TABLE 15.2-2

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TABLE 15.2-3

SEQUENCE OF EVENTS FOR TURBINE TRIP
(FIGURE 15.2-3)

Time, s	Event
0	Turbine trip initiates closure of MSVs.
0	Turbine trip initiates bypass operation.
0.01	MSVs reach 90 percent open position.
0.02	Reactor scram trip and RPT initiated.
0.1	MSVs are closed.
0.1	Turbine bypass valves start to open to regulate pressure.
0.28	Start of control rod motion.
0.38	Recirculation Pump Trip (RPT) occurs.
2.18	Group 1 SRVs are actuated.

TABLE 15.2-4

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TABLE 15.2-5

SEQUENCE OF EVENTS FOR MAIN STEAM ISOLATION VALVE CLOSURE
(FIGURE 15.2-5)

Time, s	Event
0	Initiate closure of all MSIVs.
0.46	MSIV position trip scram is initiated.[1]
0.72	Start of control rod motion.
2.02	Recirculation Pump Trip initiated (high pressure).
2.40	MSIVs fully closed.
2.64	Group 1 SRVs are activated.

(1) The 90 percent open value was used for the Hope Creek analysis. The use of an 85 percent open value for the position scram would not have significant impact on the transient results.

TABLE 15.2-6

TYPICAL RATES OF DECAY FOR CONDENSER VACUUM

Cause	Estimated Vacuum 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

TABLE 15.2-7

SEQUENCE OF EVENTS FOR LOSS OF CONDENSER VACUUM

(FIGURE 15.2-6)

Time, s	Event
-3.0	Simulated loss of condenser vacuum at 2 inches Hg/s is initiated.
0.0	Low condenser vacuum main turbine trip is actuated.
0.0	Low condenser vacuum feedwater trip is actuated.
0.01	Main turbine trip initiates reactor scram.
0.01	Main turbine trip initiates RPT.
1.84	Group 1 SRVs setpoints are actuated.
1.98	Group 2 SRVs setpoints are actuated.
2.25	Group 3 SRVs setpoints are actuated.
5.0	Low condenser vacuum initiates MSIV closure.
5.0	Low condenser vacuum initiates bypass valve closure.
13.3	Group 1 SRVs close.

TABLE 15.2-8

TRIP SIGNALS ASSOCIATED WITH LOSS OF CONDENSER VACUUM

Vacuum inches of Hg	Protective Action Initiated
27 to 28	Normal vacuum range
20	Main turbine trip and feedwater turbine trip (stop valve closures)
10	MSIV closure and bypass valve closure

TABLE 15.2-9

SEQUENCE OF EVENTS FOR LOSS OF ALL GRID CONNECTIONS
(FIGURE 15.2-7)

Time, s	Event
<0	Loss of grid causes the turbine generator to detect a loss of electrical load.
0	TCV fast closure is initiated.
0	Turbine-generator power load unbalance trip initiates the main turbine bypass system operation.
0	Reactor recirculation system pump motors are tripped.
0	TCV closure initiates a reactor scram trip.
0	MSIV closure is initiated.
0.07	TCVs are closed.
0.1	Turbine bypass valves open.
1.49	Group 1 SRVs are actuated.
1.70	Group 2 SRVs are actuated.
1.75	Group 3 SRVs are actuated.
2.0	Feedwater turbines tripped.
13.0	Group 1 safety/relief valves close.

TABLE 15.2-10

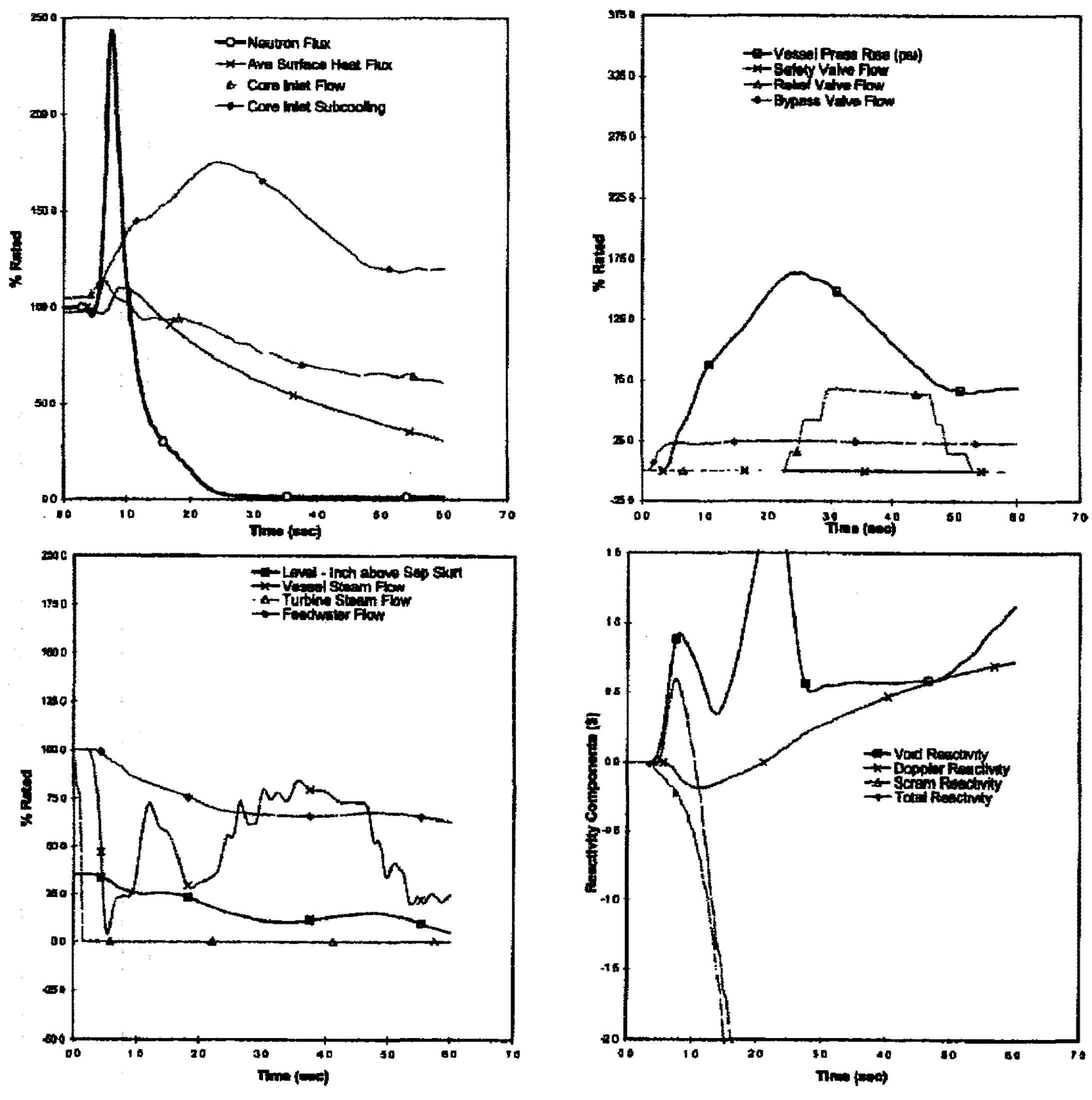
SEQUENCE OF EVENTS FOR LOSS OF FEEDWATER FLOW

Time, s	Event
0	Trip of all feedwater pumps initiated.
5.0	Feedwater flow decays to zero.
8.2	Vessel water level, L3, trip initiates scram trip.
21.6	Vessel water level, L2, trip initiates recirculation pump system trip.
21.6	Vessel water level, L2, trip initiates HPCI and RCIC system operations (not simulated).
51.6	RCIC flow enters the vessel.
56.6	HPCI flow enters the vessel.

TABLE 15.2-11

SEQUENCE OF EVENTS FOR FAILURE OF RHR SHUTDOWN COOLING RESULTING FROM A LOSS OF OFFSITE POWER

Approximate Elapsed Time	Event
0	Loss of offsite power occurs.
0	Loss of one division of power occurs.
10 min	Depressurization is initiated at 100□F/h.
>10 min	Suppression pool cooling is initiated.
2-3 h	Blowdown to approximately 100 psig is completed.
>2-3 h	Core spray is actuated and selected SRVs are opened to establish a flow path through the reactor and back into the suppression pool.
>4-5 h	Vessel temperature decreases and eventually reaches the cold shutdown condition.

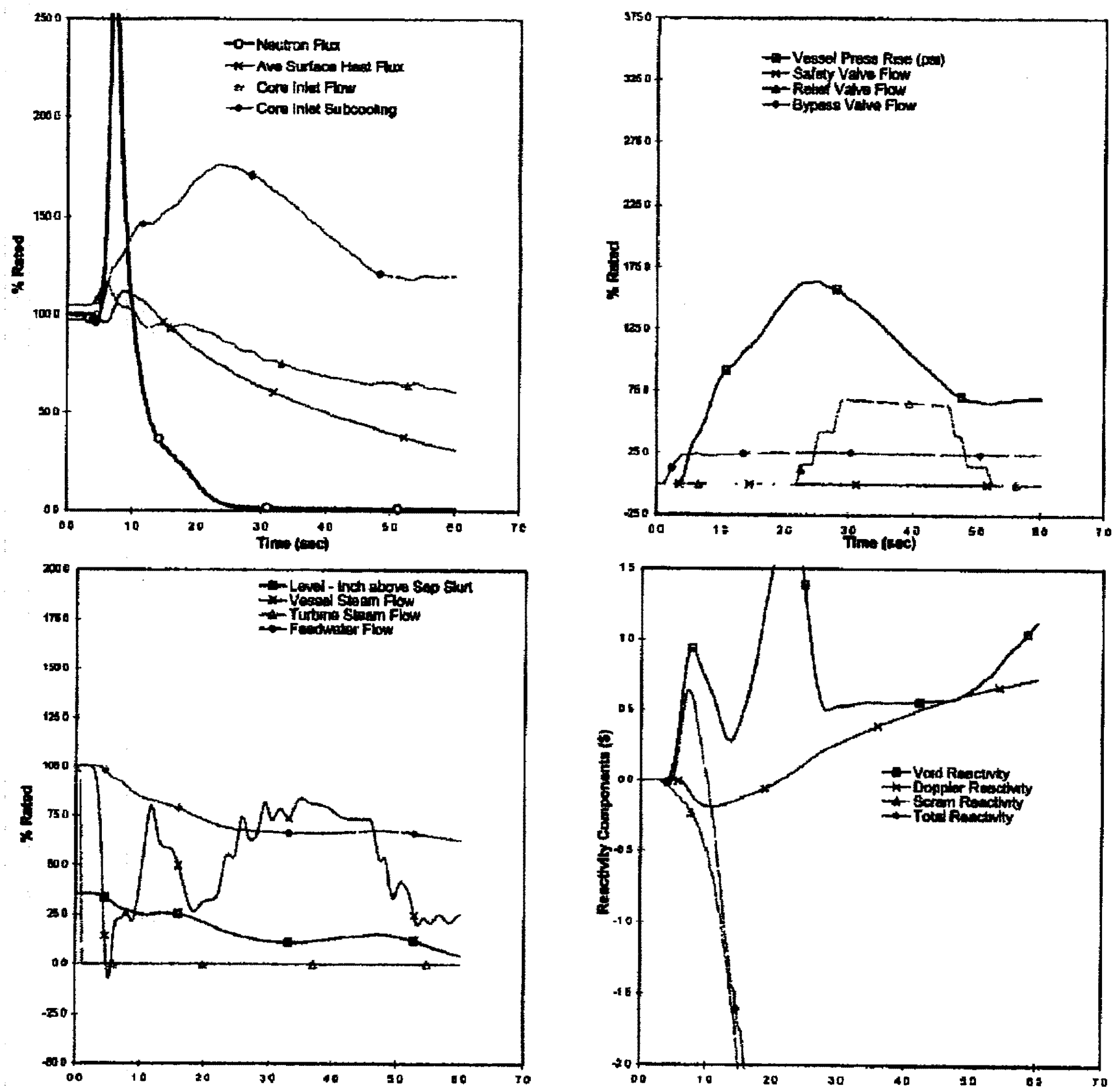


Revision 17, June 23, 2009

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station GENERATOR LOAD REJECTION TRIP, REACTOR SCRAM BYPASS - ON
	Updated FSAR — Figure 15.2-1

THIS FIGURE HAS BEEN DELETED

PSEG NUCLEAR L.L.C. HOPE CREEK GENERATING STATION
HOPE CREEK UFSAR - REV 11 SHEET 1 OF 1 November 24, 2000 F15.2-2

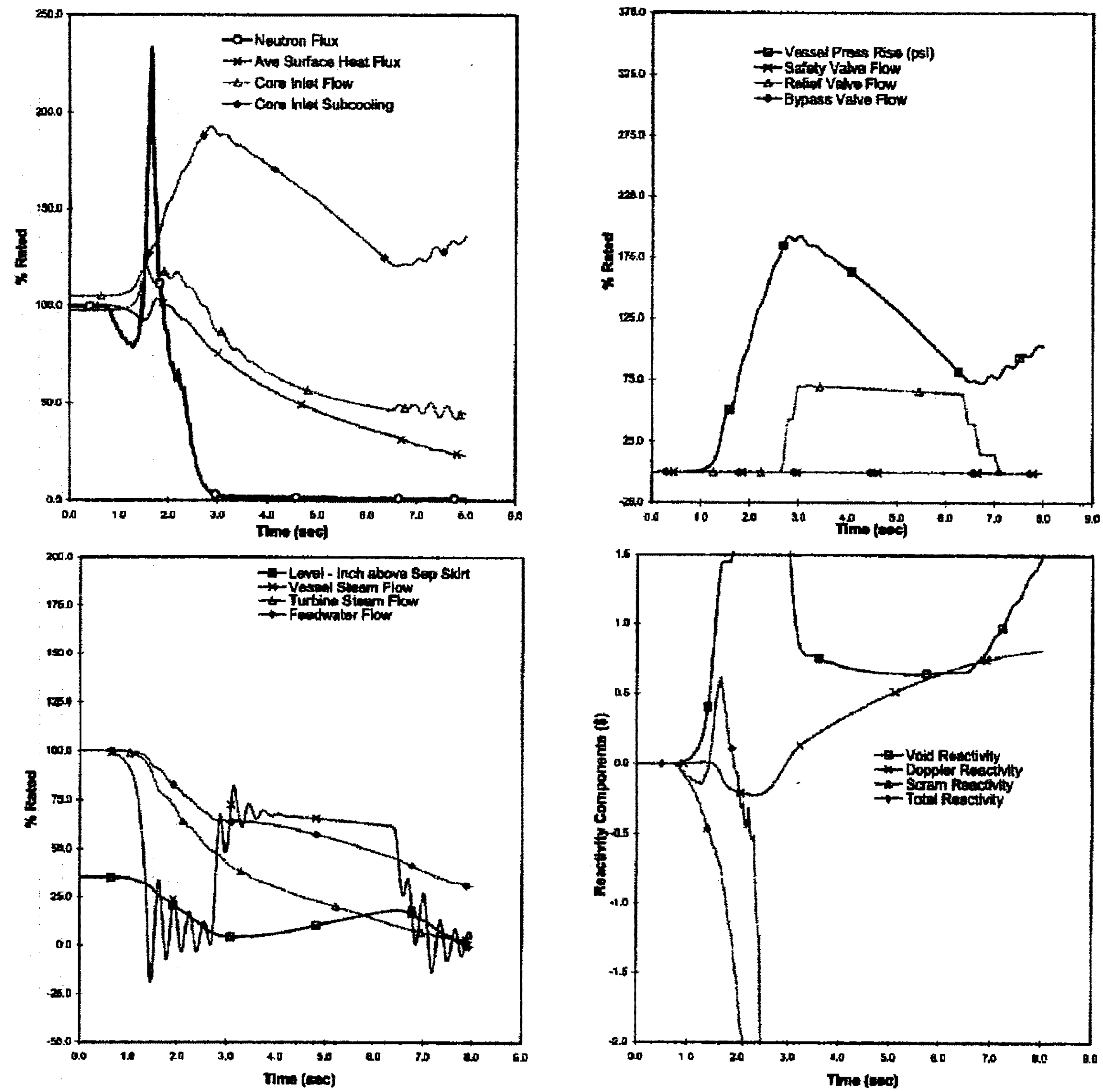


Revision 17, June 23, 2009

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station TRUBINE TRIP, REACTOR SCRAM, BYPASS AND RPT - ON
	Updated FSAR — Figure 15.2-3

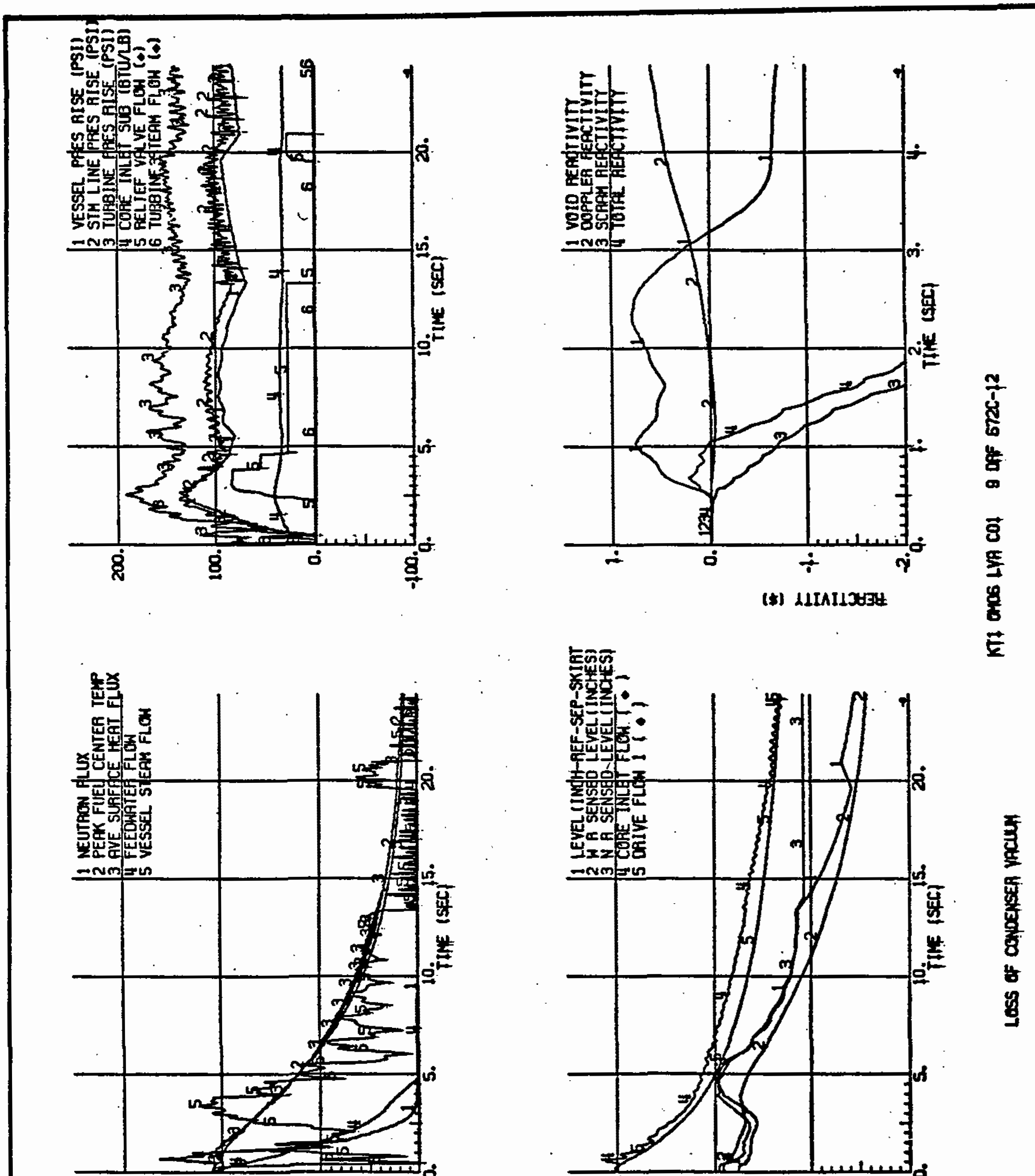
THIS FIGURE HAS BEEN DELETED

PSEG NUCLEAR L.L.C. HOPE CREEK GENERATING STATION



Revision 17, June 23, 2009

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station THREE-SECOND CLOSURE OF ALL MAIN STEAM ISOLATION VALVES WITH POSITION SWITCH REACTOR TRIP
	Updated FSAR — Figure 15.2-5

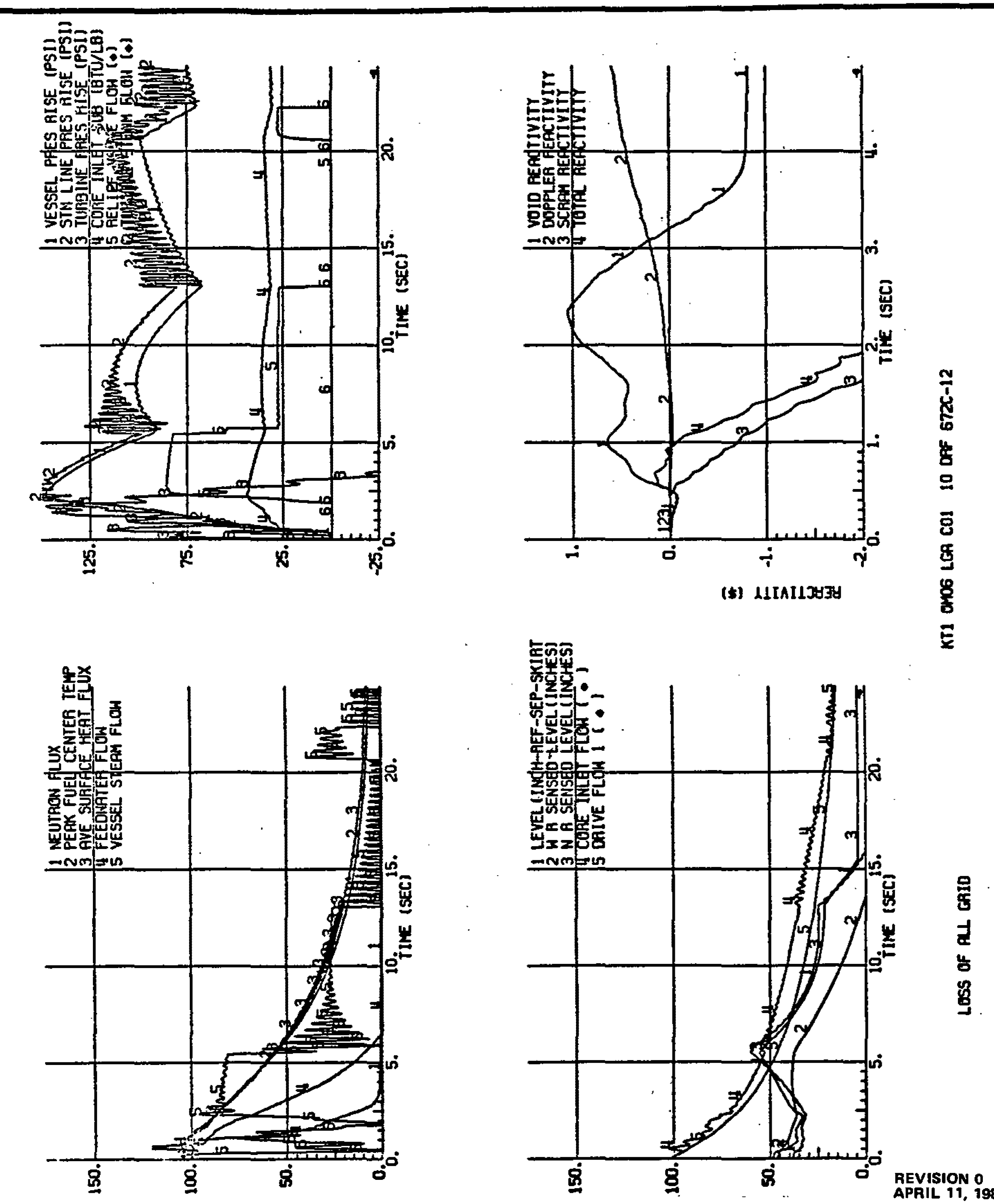


REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

LOSS OF CONDENSER VACUUM AT
2 INCHES PER SECOND

UPDATED FSAR | **FIGURE 15.2-6**



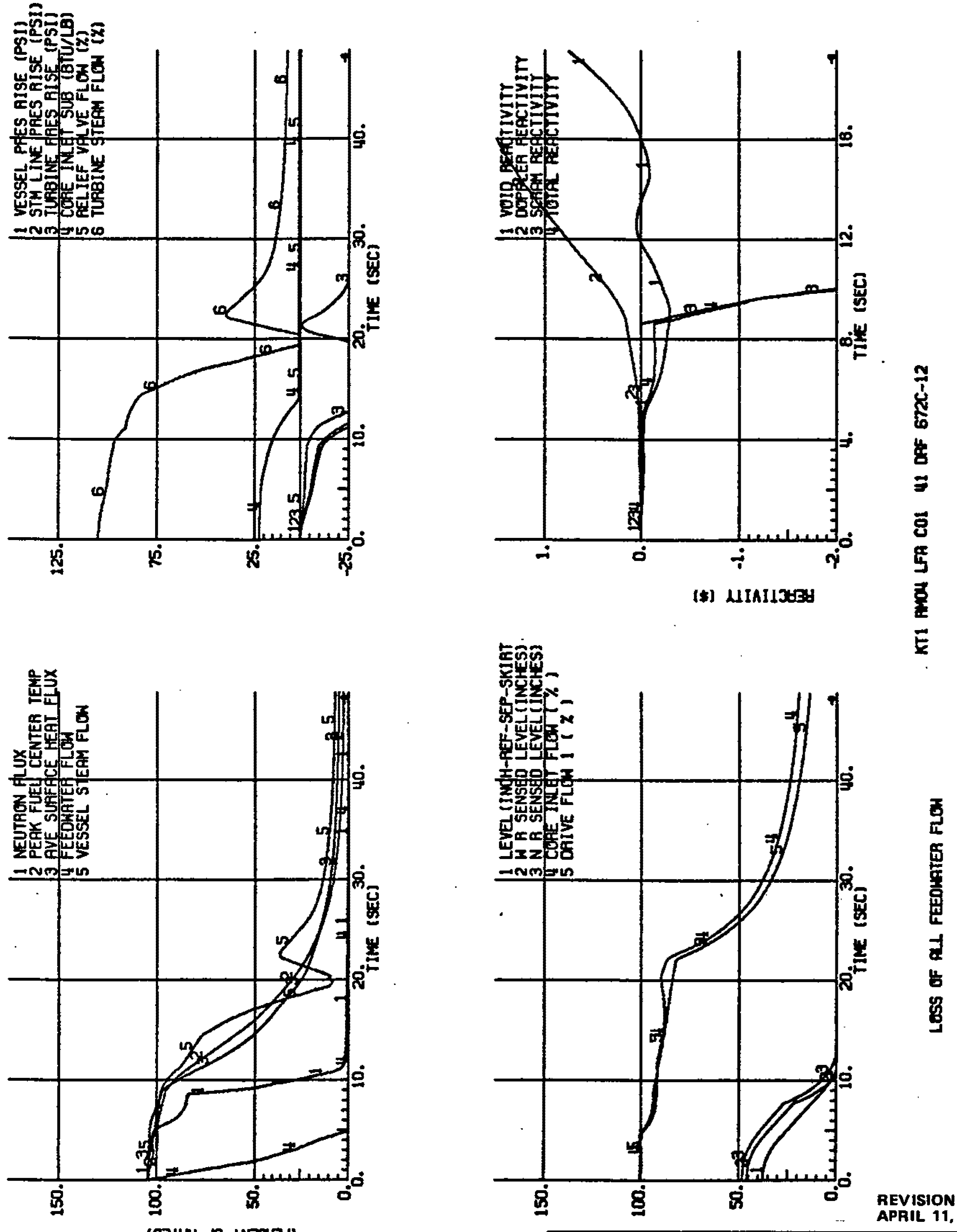
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

LOSS OF ALL GRID CONNECTIONS

UPDATED FSAR

FIGURE 15.2-7



REVISION 0
APRIL 11, 1988

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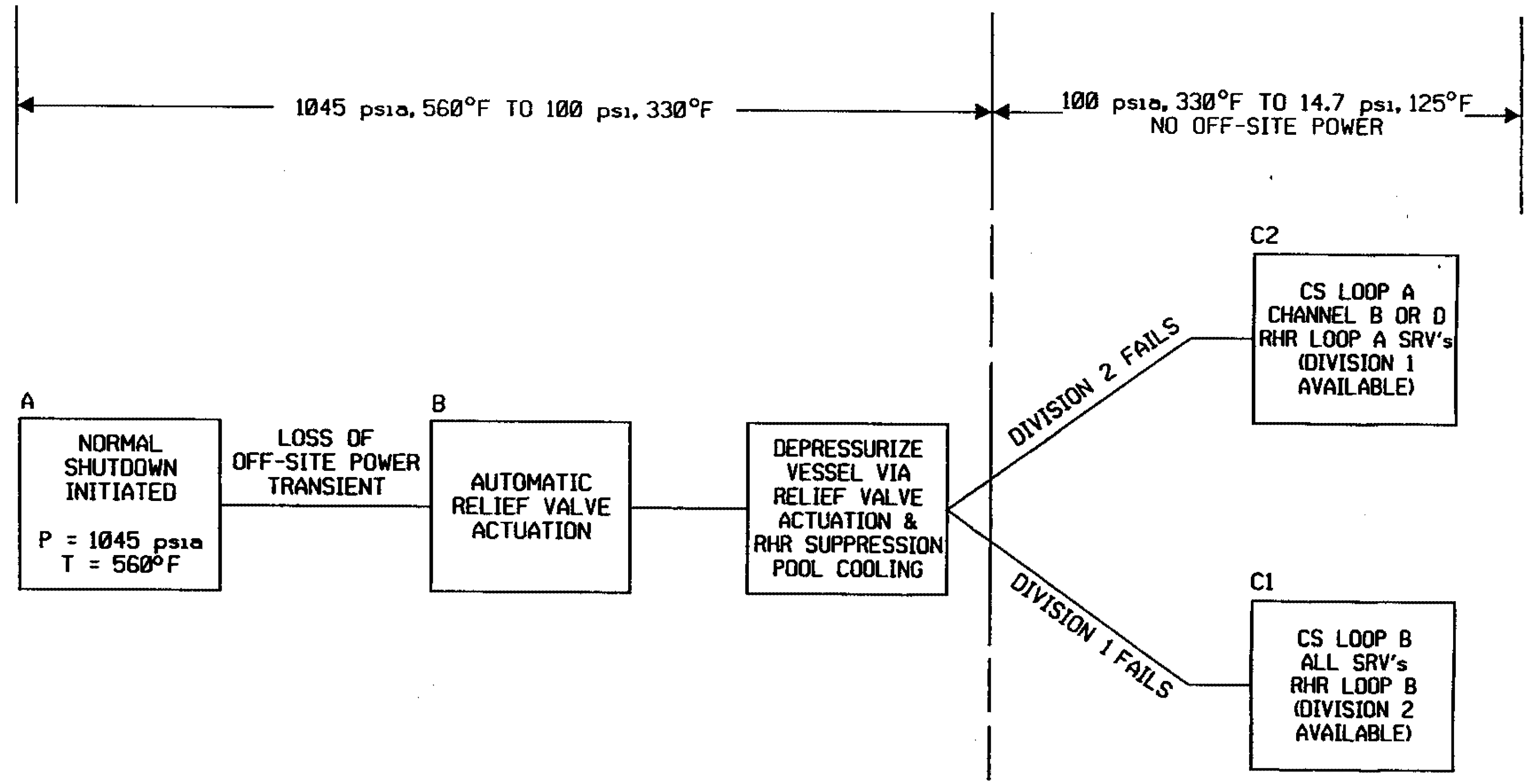
HOPE CREEK NUCLEAR GENERATING STATION

LOSS OF ALL FEEDWATER FLOW

UPDATED FSAR

FIGURE 15.2-8



NOTES:

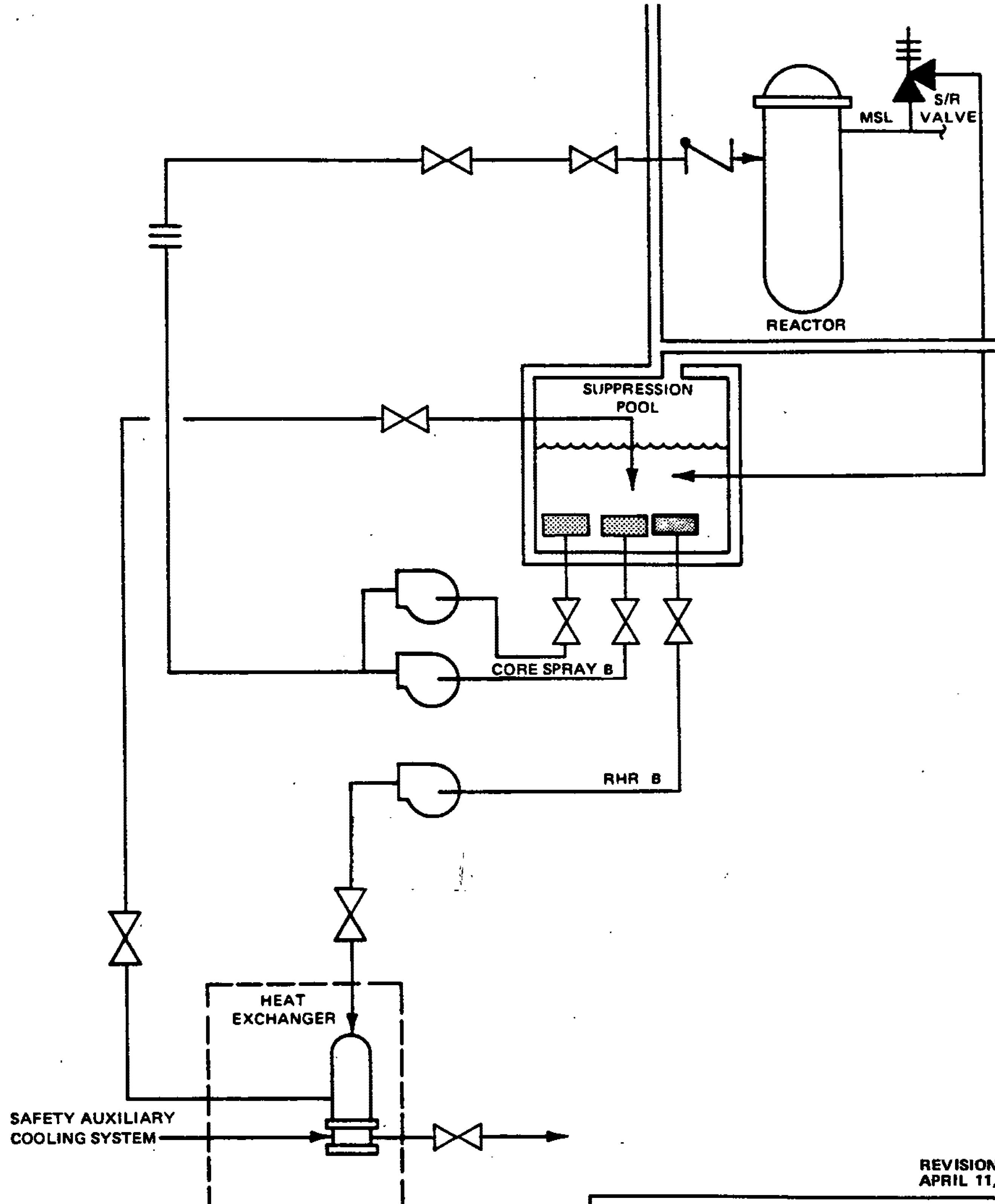
ACTIVITY A
THE VESSEL IS OPERATING AT THE INITIAL CONDITION. FOR MORE DETAILS SEE TABLE 15.0-3.

ACTIVITY B
AFTER THE LOSS OF OFF-SITE POWER, THE OPERATOR INITIATES VESSEL DEPRESSURIZATION.

ACTIVITY C (C1 AND C2)
POSSIBLE PATHS TO REACH REACTOR COLD SHUTDOWN. FOR MORE DETAILS, SEE FIGURES 15.2-10 AND 15.2-11

Revision 13, Nov 14, 2003

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station SRV /RHR COOLING LOOPS
	Updated FSAR — Figure 15.2-9



REVISION 0
APRIL 11, 1988

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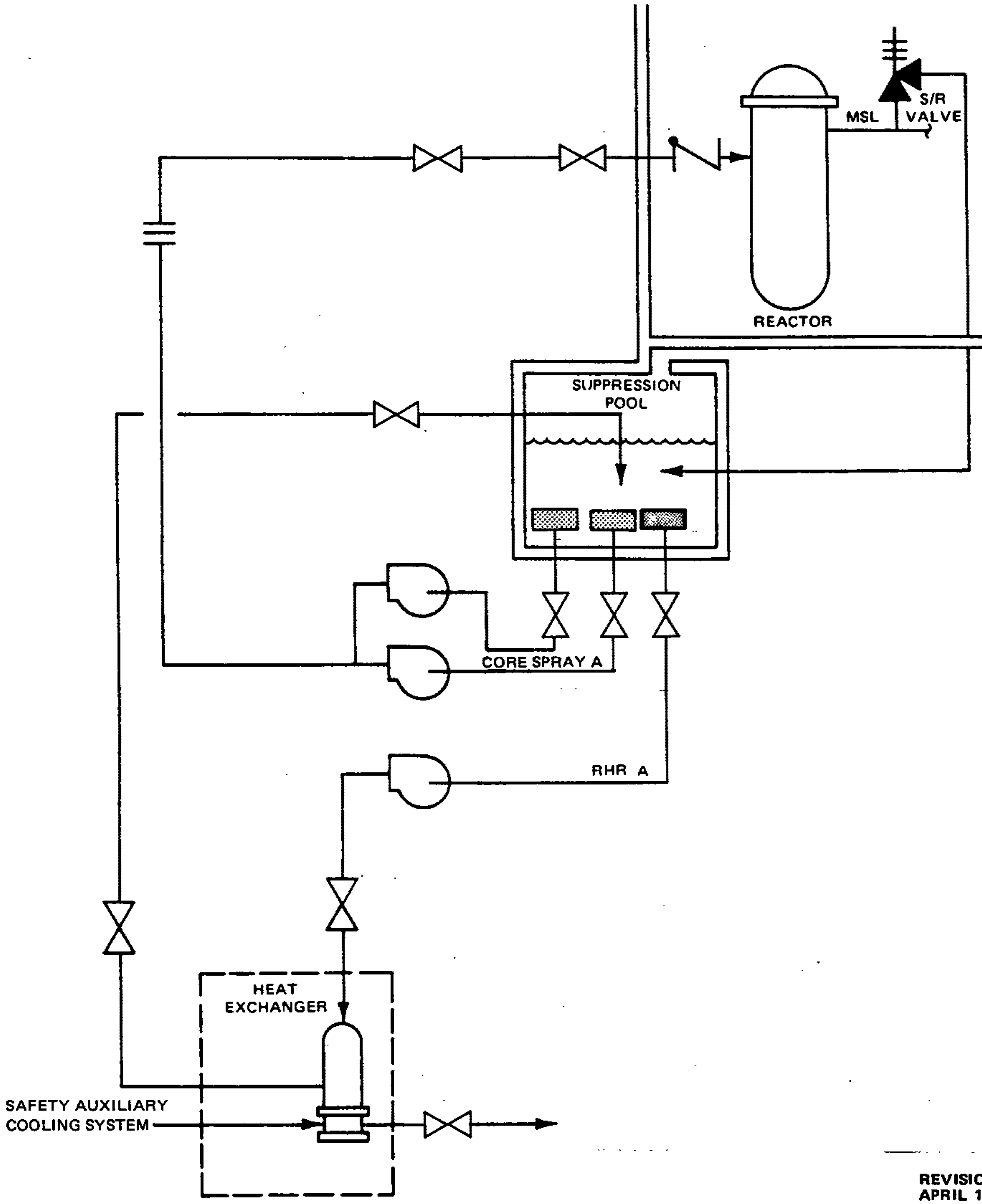
HOPE CREEK NUCLEAR GENERATING STATION

ACTIVITY C1 ALTERNATE SHUTDOWN
COOLING PATH UTILIZING
RHR LOOP B

UPDATED FSAR

FIGURE 15.2-10



REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

ACTIVITY C2 ALTERNATE SHUTDOWN COOLING PATH USING RHR LOOP A

UPDATED FSAR

FIGURE 15.2-11

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

15.3.1 Reactor Recirculation Pump Trip

The reactor recirculation pump trip events are considered a non-Limiting event. Therefore it is not required to be re-analyzed as a part of the reload licensing analysis for Hope Creek, unless the disposition for this event changes (Reference 15.3-1).

The results referenced within this section are representative of cycle 1. References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}. These results are considered bounded by the reload licensing analysis.

15.3.1.1 <u>Identification of Causes and Frequency Classification</u>

15.3.1.1.1 Identification of Causes

A reactor recirculation pump motor can be tripped by design, for reducing reactor coolant pressure boundary (RCPB) effects, and randomly by unpredictable operational failures. Intentional tripping will occur in response to:

1. Reactor vessel water level (L2) setpoint trip
2. Turbine control valve fast closure or main stop valve closure
3. Failure to scram at high pressure setpoint trip
4. Motor branch circuit overcurrent protection
5. Motor overload protection
6. Suction block valve not fully open.

Random tripping will occur in response to:

1. Operator error
2. Loss of electrical power source to the pumps
3. Equipment or sensor failures and malfunctions that initiate the above intended trips.

15.3.1.1.2 Frequency Classification

15.3.1.1.2.1 Trip of One Reactor Recirculation Pump

This transient event is categorized as one of moderate frequency.

15.3.1.1.2.2 Trip of Two Reactor Recirculation Pumps

This transient event is categorized as one of moderate frequency.

15.3.1.2 Sequence of Events and Systems Operation

15.3.1.2.1 Sequence of Events

The results below are representative of cycle 1. **References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}.** These results are considered bounded by the reload licensing analysis.

15.3.1.2.1.1 Trip of One Reactor Recirculation Pump

Table 15.3-1 lists the sequence of events for Figure 15.3-1.

15.3.1.2.1.2 Trip of Two Reactor Recirculation Pumps

Table 15.3-2 lists the sequence of events for Figure 15.3-2.

15.3.1.2.2 Identification of Operator Actions

15.3.1.2.2.1 Trip of One Reactor Recirculation Pump

Since no scram occurs, no immediate operator action is required. As soon as possible, the operator should verify that no operating limits are being exceeded and reduce flow of the operating pump to conform to the single pump flow criteria. Also, the operator will determine the cause of failure prior to returning the system to normal and follow the restart procedure.

15.3.1.2.2.2 Trip of Two Reactor Recirculation Pumps

Tripping of two reactor recirculation pumps will cause a reactor water level swell that will trip the main turbine. This in turn

will cause a reactor scram. The operator will ascertain that the reactor has been scrammed. The operator should regain control of reactor water level through reactor core isolation cooling (RCIC) operation, monitoring reactor water level and pressure control after shutdown. When both reactor pressure and level are under control, the operator may secure both high pressure coolant injection (HPCI) and RCIC, as necessary. The operator will also determine the cause of the trip before returning the system to normal.

15.3.1.2.3 Systems Operation

15.3.1.2.3.1 Trip of One Reactor Recirculation Pump

Tripping a single reactor recirculation pump does not require any protection system or safeguard system operation. This analysis assumes normal functioning of plant instrumentation and controls.

15.3.1.2.3.2 Trip of Two Reactor Recirculation Pumps

Analysis of this event assumes normal functioning of both the plant instrumentation and controls and the plant protection and reactor protection systems.

Specifically this transient takes credit for vessel level (L8) instrumentation to trip the turbine. Reactor shutdown relies on scram trips from the turbine. High system pressure is limited by the main steam safety/relief valve (SRV) operation.

15.3.1.2.4 The Effect of Single Failures and Operator Errors

15.3.1.2.4.1 Trip of One Reactor Recirculation Pump

Since no corrective action is required per Section 15.3.1.2.3.1, no additional effects of single failures need be discussed. If additional single active failures (SAFs) or single operator errors (SOEs) are assumed (for envelope purposes the other pump is assumed

tripped), then the following two pump trip analysis is provided. Refer to Section 15.9 for specific details.

15.3.1.2.4.2 Trip of Two Reactor Recirculation Pumps

Table 15.3-2 lists the reactor vessel level (L8) trip event as the first response to initiate corrective action in this transient. The high level (L8) is intended to prohibit moisture carryover to the main turbine. Multiple level sensors are used to sense and detect when the water level reaches the L8 setpoint. At this point, a single failure will neither initiate nor impede a turbine trip signal. Turbine trip signal logic circuitry, however, is not designed to meet single failure criterion. The result of a failure at this point would have the effect of delaying the pressurization. However, high moisture levels entering the turbine can cause vibration, which may lead to the operator manually tripping the unit.

Scram signals from the turbine trip are designed such that a single failure will neither initiate nor impede a reactor scram initiation. See Section 15.9 for specific details.

15.3.1.3 <u>Core and System Performance</u>

15.3.1.3.1 Mathematical Model

The nonlinear, dynamic model described briefly in Section 15.1.1.3.1 is used to simulate this event.

15.3.1.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-3.

Pump motors and pump rotors are simulated with minimum specified rotating inertias.

15.3.1.3.3 Results

The results presented below are representative of cycle 1. References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}. These results are considered bounded by the reload licensing analysis.

15.3.1.3.3.1 Trip of One Reactor Recirculation Pump

Figure 15.3-1 shows the results of losing one recirculation pump. The tripped loop diffuser flow reverses in approximately 4.3 seconds. However, the ratio of diffuser mass flow to pump mass flow in the active jet pumps increases considerably and produces approximately 14.8 percent of normal diffuser flow. Minimum critical power ratio (MCPR) remains above the operating limit, thus, the fuel thermal limits are not violated. During this transient, level swell is not sufficient to cause turbine trip and scram.

15.3.1.3.3.2 Trip of Two Reactor Recirculation Pumps

Figure 15.3-2 shows graphically this transient with minimum specified rotating inertia. MCPR remains unchanged at the operating limit. 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 main turbine and feed pump turbines and causing a reactor scram. Subsequent events, such as main steam line isolation and initiation of RCIC and HPCI systems occurring late in this event, have no significant effect on the results.

15.3.1.3.4 Consideration of Uncertainties

Initial conditions chosen for these analyses are conservative and tend to force analytical results to be more severe than expected under actual plant conditions.

Actual pump and pump-motor drive line rotating inertias are expected to be somewhat greater than the minimum design values assumed in this simulation. Actual plant deviations regarding inertia are expected to lessen the severity as analyzed. Minimum design

inertias were used as well as the least negative void coefficient, since the primary interest is in the flow reduction.

15.3.1.4 Barrier Performance

15.3.1.4.1 Trip of One Reactor Recirculation Pump

The results shown on Figure 15.3-1 indicate a basic reduction in system pressures from the initial conditions. Therefore, the RCPB barrier is not threatened.

15.3.1.4.2 Trip of Reactor Two Recirculation Pumps

The results shown on Figure 15.3-2 indicate peak pressures stay well below the 1375 psig limit allowed by the applicable code. Therefore, the RCPB barrier is not threatened.

15.3.1.5 Radiological Consequences

While the consequence of this transient does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression chamber via SRV operation. Since this activity is contained in the primary containment, there will be no exposures to operating personnel. Because this transient does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity in the primary containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will be in accordance with the Offsite Dose Calculation Manual.

15.3.2 Recirculation Flow Control Failure - Decreasing Flow

The recirculation flow control failure - decreasing flow events are considered a non-Limiting event. Therefore it is not required to be re-analyzed as a part of the reload licensing analysis for Hope Creek, unless the disposition for this event changes (Reference 15.3-1).

The results referenced within this section are representative of cycle 1. References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}. These results are considered bounded by the reload licensing analysis.

15.3.2.1 Identification of Causes and Frequency Classification

15.3.2.1.1 Identification of Causes

Two causes of recirculation flow control failure are:

1. Failure of an individual recirculation motor generator set controller (one per loop), or the positioning control of an individual scoop tube actuator, can result in a rapid flow decrease in only one recirculation loop.

2. A loss of power to both flow controllers, or an excessive manual speed demand setpoint change on both speed controllers, can generate a zero flow demand signal to both recirculation flow control loops.

15.3.2.1.2 Frequency Classification

This transient is categorized as an incident of moderate frequency.

15.3.2.2 <u>Sequence of Events and Systems Operation</u>

15.3.2.2.1 Sequence of Events

The results presented below are representative of cycle 1. References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}. These results are considered bounded by the reload licensing analysis.

15.3.2.2.1.1 Failure of One Controller - Closed

The sequence of events for this transient is similar to, and is less severe than, that listed in Table 15.3-1 for the trip of one reactor recirculation pump.

15.3.2.2.1.2 Failure of Two Controllers - Closed

The sequence of events for this transient is similar to, and can never be more severe than, that listed in Table 15.3-2 for the trip of both reactor recirculation pumps.

15.3.2.2.1.3 Identification of Operator Actions

The operator will verify that no operating limits are being exceeded. The operator will also determine the cause of the trip prior to returning the system to normal.

15.3.2.2.2 Systems Operation

Normal plant instrumentation and control is assumed to function. Credit is taken for scram in response to vessel high water level (L8) turbine trip if it occurs. This credit applies to both the single and master controller failure events.

15.3.2.2.3 The Effect of Single Failures and Operator Errors

The single failure and operator error considerations for these events are the same as discussed in Section 15.3.1.2.4 on reactor recirculation pump trips. Failure of one motor-generator set, and, thus, a recirculation pump trip (RPT), or the failure of both controllers, and, thus, two RPTs, would be the envelope cases for additional single active failures (SAFs) or single operator errors (SOEs). Refer to Section 15.9 for specific details.

15.3.2.3 <u>Core and System Performance</u>

15.3.2.3.1 Mathematical Model

The nonlinear dynamic model described briefly in Section 15.1.1.3.1 is used to simulate these transient events.

15.3.2.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions listed in Table 15.0-3. The lower negative void coefficient in Table 15.0-3 was used for these analyses.

15.3.2.3.3 Results

The results presented below are representative of cycle 1. References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}. These results are considered bounded by the reload licensing analysis.

In the case of zero demand to both controllers, each individual motor generator set speed controller has an error limiter that limits the maximum rate of change of speed in each loop. Thus, this transient can never be more severe than the simultaneous trip of both reactor recirculation pumps, evaluated in Section 15.3.1.3.3.2.

In the case of failure of one controller, the scoop tube positioners are designed so that the flow change rate limit is determined by the individual stroking rate, which is approximately 25 percent per second. This case is similar to the trip of one reactor recirculation pump, described in Section 15.3.1.3.3.1, and is less severe than the transient that results from the simultaneous trip of both reactor recirculation pumps.

15.3.2.3.4 Consideration of Uncertainties

Initial conditions chosen for these analyses are conservative and tend to force analytical results to be more severe than otherwise expected. These analyses, unlike the pump trip series, are unaffected by deviations in pump, pump motor, and driveline inertias, since it is the flow controllers that cause rapid recirculation decreases.

15.3.2.4 Barrier Performance

The reactor coolant pressure boundary (RCPB) barrier performance considerations for these transients are the same as discussed in Section 15.3.1.4 on recirculation pump trips.

15.3.2.5 Radiological Consequences

The radiological consequences for these transients are the same as discussed in Section 15.3.1.5.

15.3.3 Reactor Recirculation Pump Shaft Seizure

The Reactor Recirculation Pump Shaft Seizure accident is considered a non-Limiting event. Therefore it is not required to be re-analyzed as a part of the reload licensing analysis for Hope Creek, unless the disposition for this event changes (Reference 15.3-1).

The results referenced within this section are representative of cycle 1. **References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}.** These results are considered bounded by the reload licensing analysis.

15.3.3.1 Identification of Causes and Frequency Classification

The seizure of a reactor recirculation pump is considered a design basis accident (DBA). It has been evaluated as a very mild accident in relation to other DBAs, such as a loss-of-coolant accident (LOCA). The analysis has been conducted with consideration to a

single or double loop operation. A detailed discussion is given in Section S.2.3.4 of GESTAR II (Reference 15.3-1).

The seizure event postulated certainly would not be the mode of failure of the pump. Safe shutdown components, e.g., electrical breakers, protective circuits, would preclude an instantaneous seizure event.

15.3.3.1.1 Identification of Causes

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

15.3.3.1.2 Frequency Classification

This event is considered a limiting fault, but results in effects that can easily satisfy an event of greater probability, i.e., infrequent incident classification.

15.3.3.2 Sequence of Events and Systems Operations

15.3.3.2.1 Sequence of Events

Table 15.3-3 lists the sequence of events for Figure 15.3-3.

The above results are representative of cycle 1. **References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}.** These results are considered bounded by the reload licensing analysis.

15.3.3.2.1.1 Identification of Operator Actions

The operator will first make certain that the reactor scrams with the turbine trip resulting from reactor water level swell. The operator will then regain control of reactor water level through high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) operation, or by restart of a feedwater pump. The operator must also monitor reactor water level and pressure after shutdown.

15.3.3.2.2 Systems Operation

To properly simulate the expected sequence of events, the analysis of this accident assumes normal functioning of plant instrumentation and controls, plant protection, and Reactor Protection Systems (RPSs).

Operation of safe shutdown features, though not included in this simulation, is expected to be used in order to maintain adequate water level.

15.3.3.2.3 The Effect of Single Failures and Operator Errors

Single failures in the scram logic, originating via the high vessel level (L8) trip, are similar to the considerations in Section 15.3.1.2.4.2.

Refer to Section 15.9 for further details.

15.3.3.3 <u>Core and System Performance</u>

15.3.3.3.1 Mathematical Model

The Reactor Recirculation Pump Shaft Seizure event for the HCGS initial core was analyzed using the REDY computer code (Reference 15.3-2). The FSAR contains an analysis of the pump seizure from full power two loop conditions. This analysis has not changed.

For the introduction of the GE-14 and GNF2 fuel, Reactor Recirculation Pump Shaft Seizure is analyzed from Single Loop Operation and the AOO criteria is applied to provide a bounding analysis. This analysis was performed using the ODYN computer code (Reference 15.3-3)

15.3.3.3.2 Input Parameters and Initial Conditions

This analysis has been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-3.

For the purpose of evaluating consequences to the fuel thermal limits, 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 105 percent of nuclear boiler rated (NBR) steam flow. Also, the reactor is assumed to be operating at thermally limited conditions.

The void coefficient is adjusted to the most conservative value, that is, the least negative value in Table 15.0-3.

15.3.3.3.3 Results

The results presented below are representative of cycle 1. References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}. These results are considered bounded by the reload licensing analysis.

Figure 15.3-3 presents the results of the accident. Core coolant flow drops rapidly, reaching its minimum value of 53.6 percent in approximately 1.6 seconds. Minimum critical power ratio (MCPR) does not decrease significantly before fuel surface heat flux begins dropping enough to restore greater thermal margins. The level swell produces a trip of the main and feedwater turbines and main stop valve closure, reactor scram, and recirculation pump trip (RPT). Since, after the time at which MCPR occurs, heat flux decreases much more rapidly than the rate at which heat is removed by the coolant, the scram conditions impose no threat to thermal limits. Additionally, the bypass valves and momentary opening of some of the safety/ relief valves (SRVs) limit the pressure well within the range allowed by the ASME B&PV Code. Therefore, the reactor coolant pressure boundary (RCPB) is not threatened by overpressure.

15.3.3.3.4 Considerations of Uncertainties

Considerations of uncertainties are discussed in Section 15.0.3.3.

15.3.3.4 <u>Barrier Performance</u>

After the opening of bypass valves and after a turbine trip, the pressure well is limited within the range allowed by the ASME B&PV Code. Therefore, the RCPB barrier is not threatened by overpressure.

15.3.3.5 <u>Radiological Consequences</u>

The radiological consequences of this accident are the same as discussed in Section 15.3.1.5.

15.3.3.6 SRP Rule Review

Evaluation of SRP 15.3.3 is included in Section 15.3.4.6.

15.3.4 Reactor Recirculation Pump Shaft Break

The reactor recirculation pump shaft break accident is considered a non-Limiting event. Therefore it is not required to be re-analyzed as a part of the reload licensing analysis for Hope Creek, unless the disposition for this event changes (Reference 15.3-1).

15.3.4.1 Identification of Causes and Frequency Classification

The breaking of the shaft of a reactor recirculation pump is considered a design basis accident (DBA). It has been evaluated as a very mild accident in relation to other DBAs, such as a loss-of-coolant accident (LOCA). The analysis has been conducted with consideration to a single or double loop operation.

This postulated event is bounded by the more limiting case of recirculation pump seizure. Quantitative results for this more limiting case are presented in Section 15.3.3.

15.3.4.1.1 Identification of Causes

The case of reactor recirculation pump shaft breakage represents the extremely unlikely event of instantaneous stoppage of the pump motor operation of one recirculation pump. This event produces a very rapid decrease of core flow as a result of the break of the pump shaft.

15.3.4.1.2 Frequency Classification

This event is considered a limiting fault, but results in effects that can easily satisfy an event of greater probability, i.e., infrequent incident classification.

15.3.4.2 <u>Sequence of Events and Systems Operations</u>

15.3.4.2.1 Sequence of Events

A postulated instantaneous break of the pump motor shaft of one reactor recirculation pump, as discussed in Section 15.3.4.1.1, will cause the core flow to decrease rapidly, resulting in water level swell in the reactor vessel. When the vessel water level reaches the high water level setpoint, L8, main turbine trip and feedwater pump trip will be initiated. Subsequently, reactor scram and the remaining recirculation pump trip (RPT) will be initiated due to the turbine trip. Eventually, the vessel water level will be controlled by high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) flow.

15.3.4.2.1.1 Identification of Operator Actions

The operator will first make certain that the reactor scrams as a result of the turbine trip due to reactor water level swell. The operator will then regain control of reactor water level through HPCI and RCIC operation or by restart of a feedwater pump. The operator must also monitor reactor water level and pressure after shutdown.

15.3.4.2.2 Systems Operation

Normal operation of plant instrumentation and control is assumed. This event takes credit for vessel water level (L8) instrumentation to scram the reactor and trip the main turbine and feedwater pumps. High system pressure is limited by the safety/relief valve (SRV) system operation.

Operation of HPCI and RCIC systems is expected in order to maintain adequate water level control.

15.3.4.2.3 The Effect of Single Failures and Operator Errors

Effects of single failures in the high vessel level (L8) trip are similar to the considerations in Section 15.3.1.2.4.2.

Assumption of single active failure (SAF) or single operation error (SOE) in other equipment has been examined and has led to the conclusion that no other credible failure exists for this event. Therefore, the bounding case has been considered. (Refer to Section 15.9 for more details.)

15.3.4.3 <u>Core and System Performance</u>

Since this event is less limiting than the event described in Section 15.3.3, only qualitative evaluation is provided. Therefore, no discussion of the mathematical model, input parameters, nor consideration of uncertainties, etc, is necessary.

15.3.4.3.1 Qualitative Results

If this extremely unlikely event occurs, core coolant flow will drop rapidly. The level swell produces a trip of the main and feedwater turbines. Subsequently, scram is initiated due to turbine trip. Since heat flux decreases much more rapidly than the rate at which heat is removed by the coolant, there is no threat to thermal limits. Additionally, the bypass valves and momentary opening of some of the SRVs limit the pressure to well within the range allowed by the ASME B&PV Code. Therefore, the RCPB barrier is not threatened by overpressure.

The severity of this pump shaft break is bounded by the pump shaft seizure event (see Section 15.3.3). This can be demonstrated easily by consideration of these two events. In either of these two events, the recirculation drive flow of the affected loop decreases rapidly. In the case of the pump shaft seizure event, the loop flow decreases faster than the normal flow coastdown, as a result of the large hydraulic resistance introduced by the stopped rotor. For the

pump shaft break event, the hydraulic resistance caused by the broken pump shaft is less than that of the stopped rotor for the pump shaft seizure event. Therefore, the core flow decrease following a pump shaft break effect is slower than the pump shaft seizure event. Thus, it can be concluded that the potential effects of the hypothetical pump shaft break event are bounded by the effects of the pump shaft seizure event.

15.3.4.4 Barrier Performance

The bypass valves and momentary opening of some of the SRVs maintain the pressure well within the limits allowed by the ASME B&PV Code. Therefore, the RCPB is not threatened by overpressure.

15.3.4.5 Radiological Consequences

The radiological consequences of this event are the same as discussed in Section 15.3.1.5.

15.3.4.6 SRP Rule Review

SRP 15.3.3 - 15.3.4 acceptance criterion II.10 states that analysis for reactor coolant pump rotor seizure and reactor coolant pump shaft break events should include assumptions of turbine trip and coincidental loss of offsite power (LOP) and coastdown of undamaged pumps.

Coincidental LOP and turbine trip are not assumed in the HCGS analysis but would, if included, produce consequences less severe than those of Section 15.2-6.

The turbine trip or, indirectly, the loss of offsite power, will initiate reactor scram and rapid power reduction. The severity of pump shaft seizure or pump shaft break without assuming LOP is evidenced by the fast coastdown of core flow, which reduces the thermal margin significantly before a reactor scram is initiated by an L8 signal.

15.3.5 References

15.3-1 General Electric, "General Electric Standard Application for Reactor Fuel, including the United States Supplement," NEDE-24011-P-A- and NEDE-24011-P-A-US latest revision.

15.3-2 R.B. Linford, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802-A, General Electric, December 1986.

15.3-3 General Electric, "Qualification of the One Dimensional Core Transient Model for BWR," NEDO 24154-A, August 1986.

TABLE 15.3-1

SEQUENCE OF EVENTS FOR A TRIP OF ONE RECIRCULATION PUMP
(FIGURE 15.3-1)

Time, s	Event
0	Trip of one recirculation pump initiated
4.3	Diffuser flow decreases significantly in the tripped loop
20	Core flow stabilizes at new equilibrium conditions
40	Power level stabilizes at new equilibrium conditions

TABLE 15.3-2

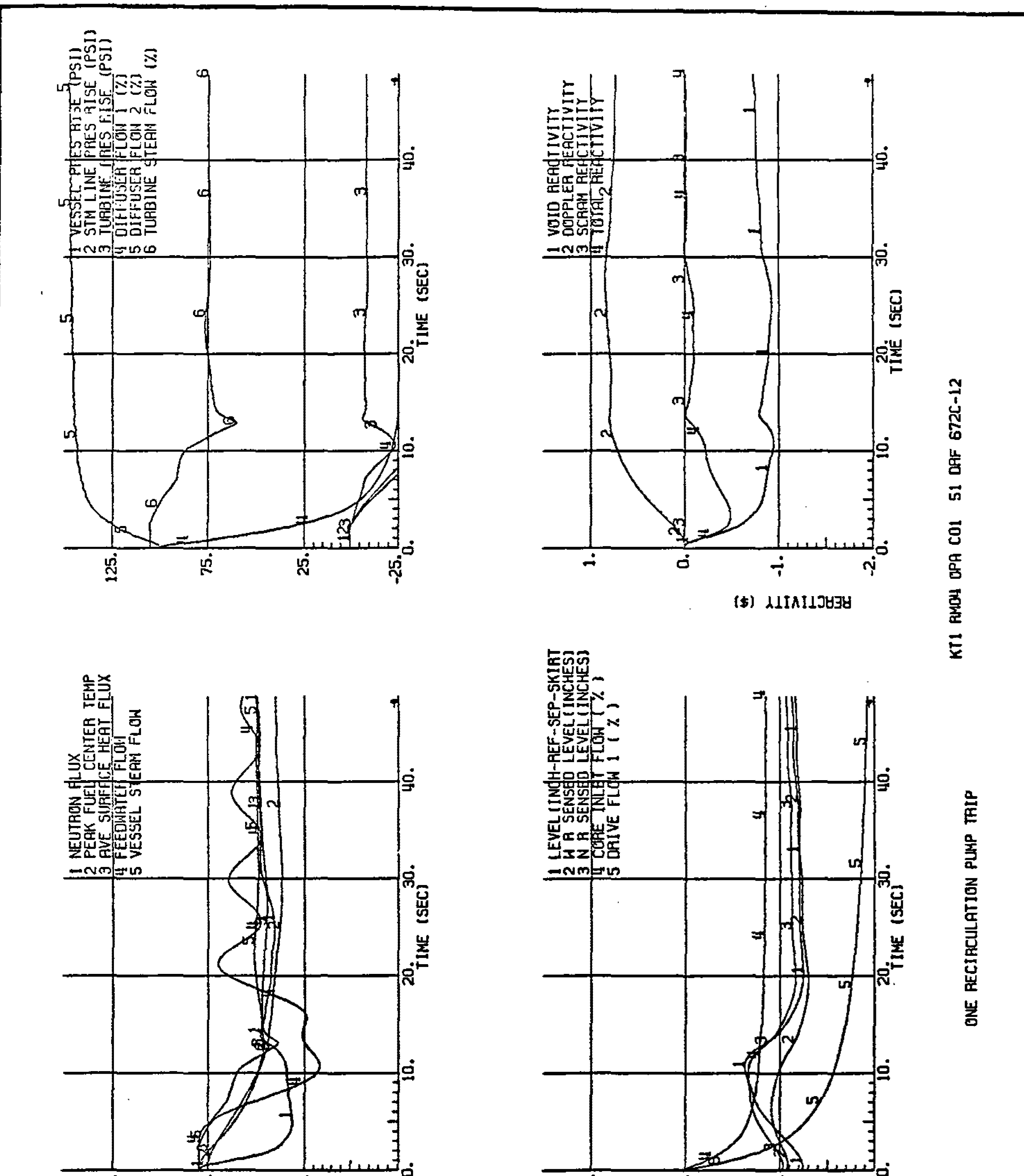
SEQUENCE OF EVENTS FOR A TRIP OF TWO RECIRCULATION PUMPS
(FIGURE 15.3-2)

Time, s	Event
0	Trip of both recirculation pumps initiated
5.5	Vessel water level (L8) trip initiates turbine trip and feedwater pumps trip
5.5	Turbine trip initiates bypass operation
5.51	Turbine trip initiates reactor scram

TABLE 15.3-3

SEQUENCE OF EVENTS FOR A RECIRCULATION PUMP SEIZURE
(FIGURE 15.3-3)

Time-s	Event
0	Single pump seizure is initiated
0.7	Jet pump diffuser flow reverses in seized loop
6.7	Vessel level (L8) trip initiates turbine trip
6.7	Feedwater pumps are tripped
6.7	Turbine trip initiates bypass operation
6.71	Turbine trip initiates reactor scram



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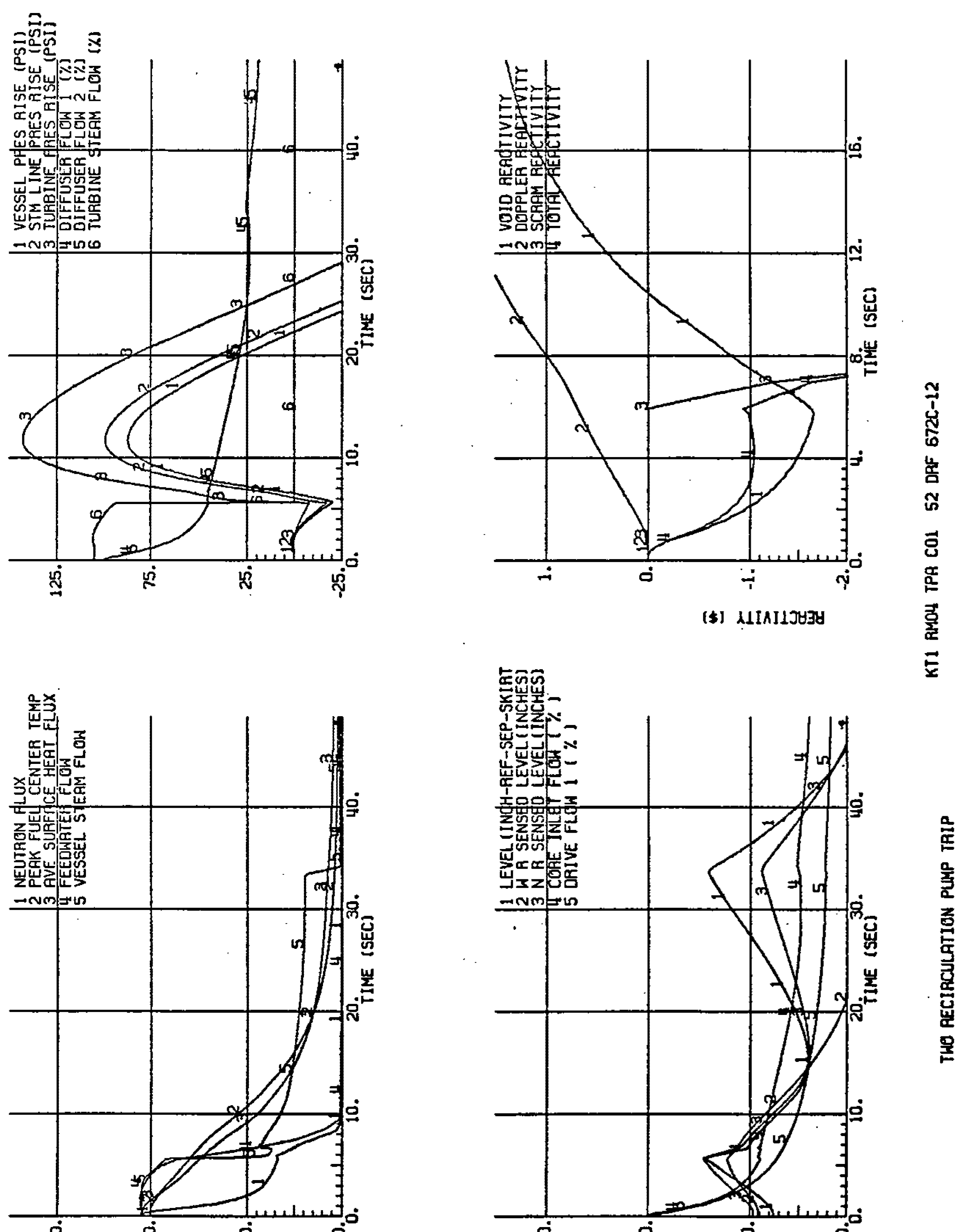
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

TRIP OF ONE
RECIRCULATION PUMP

UPDATED FSAR

FIGURE 15.3-1



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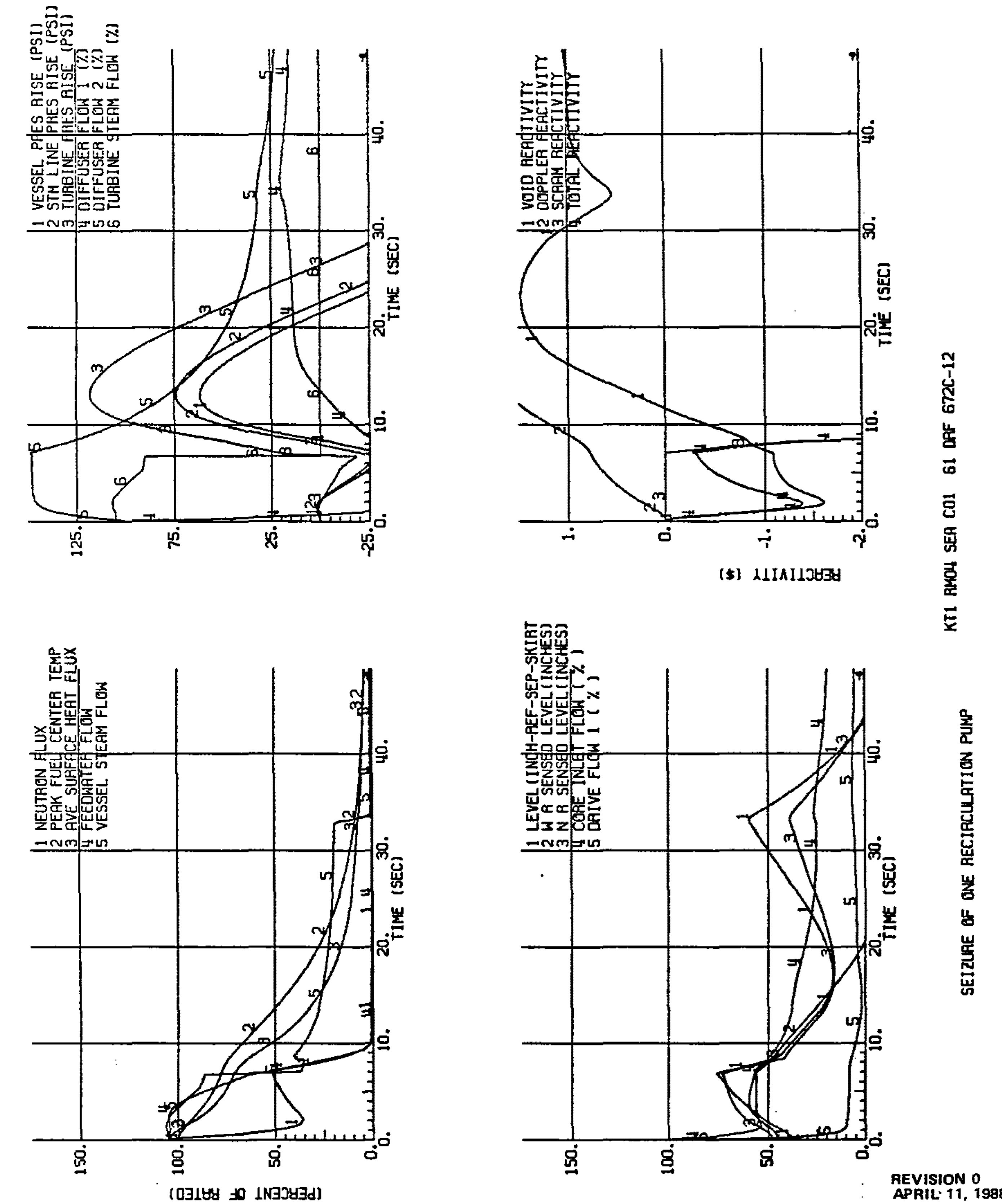
TRIP OF TWO RECIRCULATION PUMPS

UPDATED FSAR | **FIGURE 15.3-2**



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HOPE CREEK NUCLEAR GENERATING STATION

RECIRCULATION PUMP SEIZURE

UPDATED FSAR

FIGURE 15.3-3

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

15.4.1 Rod Withdrawal Error - Low Power

The rod withdrawal error - low power events are considered a non-Limiting event. Therefore it is not required to be re-analyzed as a part of the reload licensing analysis for Hope Creek, unless the disposition for this event changes (Reference 15.4-2).

15.4.1.1 Control Rod Removal Error During Refueling

15.4.1.1.1 Identification of Causes and Frequency Classification

The event considered here is inadvertent criticality due to the complete withdrawal or removal of the most reactive rod during refueling. The probability of the initial causes alone is considered low enough to warrant being categorized as an infrequent incident, since there is no postulated set of circumstances that results in an inadvertent rod withdrawal error (RWE) while in the refueling mode.

15.4.1.1.2 Sequence of Events and Systems Operation

15.4.1.1.2.1 Initial Control Rod Removal or Withdrawal

During refueling operations, safety system interlocks ensure that inadvertent criticality does not occur because a control rod has been removed or withdrawn in coincidence with another control rod.

15.4.1.1.2.2 Fuel Insertion With Control Rod Withdrawn

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods be fully inserted when fuel is being loaded into the core. This requirement is supplemented by refueling interlocks on rod withdrawal and 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.

15.4.1.1.2.3 Second Control Rod Removal or Withdrawal

When the platform is not over the core, or fuel is not on the hoist, and the mode switch is in the "refuel" position, only one control rod can be withdrawn. Any attempt to withdraw a second rod results in a rod block by refueling interlocks. Since the core is designed to meet shutdown requirements with the highest worth rod withdrawn, the core remains subcritical even with one rod withdrawn.

15.4.1.1.2.4 Control Rod Removal Without Fuel Removal

Finally, the design of the control rod 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.

15.4.1.1.2.5 Identification of Operator Actions

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

15.4.1.1.2.6 Effect of Single Failure and Operator Errors

If any one of the operations involved in initial failure or error is followed by any other single active failure (SAF) or single operator error (SOE), the necessary safety actions are taken, e.g., rod block or scram, automatically prior to limit violation. Refer to Section 15.9 for details.

15.4.1.1.3 Core and System Performances

Since the probability of inadvertent criticality during refueling is precluded, the core and system performances are not analyzed. The withdrawal of the highest worth control rod during refueling does not result in criticality. This is demonstrated by performing shutdown margin checks. See reference 15.4-2 for a

description of the methods and results of the shutdown margin analysis. Additional reactivity insertion is precluded by interlocks. See Section 7.7.1.1.

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

15.4.1.1.4 Barrier Performance

An evaluation of the barrier performance is 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.

15.4.1.2 Continuous Rod Withdrawal During Reactor Startup

15.4.1.2.1 Identification of Causes and Frequency Classification

The probability of initial causes or errors of this event alone is considered low enough to warrant 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 Rod Worth Minimizer (RWM) Systems (or the RWM bypassed with a second qualified verifier allowing out of sequence rod selection), concurrent with a high worth, out of sequence rod selection contrary to procedures, plus failure of the operator to acknowledge continuous alarm annunciations prior to safety system actuation.

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 or a second qualified verifier 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 percent to 75 percent control rod density range and limits rod movement to the banked position mode of rod withdrawal from the 75 percent rod density to the preset power level. Since only in--sequence control rods can be withdrawn in the 100 percent to 75 percent control rod density and control rods are withdrawn in the banked position mode from the 75 percent 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., 8.6 percent of CPPU rated core power. For RWE above low power setpoint, see Section 15.4.2. The banked position mode of the RWM is described in Reference 15.4-1.

A special analysis has been performed on the transient caused by continuous control rod withdrawal in the startup range to demonstrate that the licensing basis criterion for fuel failure will not be exceeded when an out of sequence control rod is withdrawn at the maximum allowable normal drive speed. See Appendix B for the details of this analysis.

15.4.1.2.2.1 Identification of Operator Actions

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

15.4.1.2.2.2 Effects of Single Failure and Operator Errors

If any one of the operations involves an initial failure or error and is followed by another SAF or SOE, the necessary safety actions are automatically taken, e.g., rod blocks, prior to any limit violation. Refer to Section 15.9 for details.

15.4.1.2.3 Core and System Performance

The performance of the RWM or a second qualified verifier prevents erroneous selection and withdrawal of an out of sequence control rod. The core and system performance is not affected by such an operator error.

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

15.4.1.2.4 Barrier Performance

An evaluation of the barrier performance is not made for this event since there is no postulated set of circumstances for which this error could occur.

15.4.1.2.5 Radiological Consequences

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

15.4.2 Rod Withdrawal Error - At Power

The rod withdraw error at power event is considered a potentially limiting event and is re-analyzed for each reload. The results of the re-analysis of the rod withdraw error at power event are presented in Appendix 15D.

This event and the analysis methodology are described in Reference 15.4-2. The limiting rod pattern and results from the reload rod withdraw error at power event are presented in Appendix 15D. The evaluation of a representative rod withdrawal error at CPPU conditions resulted in a minimum critical power ratio decrease of 0.17 below its initial value.

15.4.3 Control Rod Maloperation (System Malfunction or Operator Error)

This event is covered by the evaluation cited in Sections 15.4.1 and 15.4.2. As a result this event is less severe than the rod withdraw error analyzed for the reload, so the control rod maloperation event is not re-analyzed in the standard reload licensing analysis process.

15.4.4 Abnormal Startup of Idle Recirculation Pump

The abnormal startup of an idle recirculation pump event is considered a non-Limiting event. Therefore it is not required to be re-analyzed as a part of the reload licensing analysis for Hope Creek, unless the disposition for this event changes (Reference 15.4-2).

The results referenced within this section are representative of cycle 1. **References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}.** These results are considered bounded by the reload licensing analysis.

15.4.4.1 Identification of Causes and Frequency Classification

15.4.4.1.1 Identification of Causes

Abnormal startup of an idle recirculation pump results directly from the operator's manual initiation of pump operation. It assumes that the remaining loop is already operating.

15.4.4.1.2 Frequency Classification

15.4.4.1.2.1 Normal Restart of Recirculation Pump at Power

This transient is categorized as an incident of moderate frequency.

15.4.4.1.2.2 Abnormal Startup of Idle Recirculation Pump

This transient is categorized as an incident of moderate frequency.

15.4.4.2 Sequence of Events and Systems Operation

15.4.4.2.1 Sequence of Events

Table 15.4-1 lists the sequence of events for Figure 15.4-2.
The results referenced above are representative of cycle 1. **References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}.** These results are considered bounded by the reload licensing analysis.

15.4.4.2.1.1 Operator Actions

The normal sequence of operator actions expected in starting the idle loop is as follows. The operator will:

1. Adjust rod pattern as necessary for new power level following idle loop start.

2. Determine that the idle recirculation pump suction valve is open, the discharge block valve is closed, and the coupler in the idle loop is in the starting position; if not, place them in this configuration.

3. Readjust flow of the running loop downward to less than half of rated flow.

4. Determine that the temperature difference between the two loops is no more than 50°F.

5. Start the idle loop pump.

6. Open the discharge valve by using the manual jogging sequence or auto circuitry; monitor reactor power.

7. Adjust the loop flow to match the operating loop flow.

8. Readjust power, as necessary, to satisfy plant requirements per standard procedure.

15.4.4.2.2 Systems Operation

This event assumes normal functioning of plant instrumentation and controls, plant protection systems, and the Reactor Protection System (RPS). In particular, credit is taken for high neutron flux scram to terminate the transient. No engineered safety feature (ESF) action occurs as a result of the transient.

15.4.4.2.3 The Effect of Single Failures and Operator Errors

This transient leads to a quick rise in reactor power level. Corrective action first occurs in the high neutron flux trip and, as part of the RPS, it is designed to single failure criteria. Therefore, shutdown is ensured. Operator errors are of no concern here since the automatic shutdown events rapidly follow the postulated failure. See Section 15.9 for details.

15.4.4.3 <u>Core and System Performance</u>

15.4.4.3.1 Mathematical Model

The nonlinear dynamic model described briefly in Section 15.1.1.3.1 is used to simulate this event.

15.4.4.3.2 Input Parameters and Initial Conditions

Unless otherwise noted, this analysis has been performed with plant conditions tabulated in Table 15.0-3.

For the Cycle 1 analysis, one recirculation loop is idle and filled with cold water at 100°F. Normal procedure, when starting an idle loop with one pump already running, requires the coolant in the idle recirculation loop to within 50°F of core inlet temperature prior to idle loop startup and this is the input assumption used for the ARTS/MELLLA analysis (15.4-7) and CPPU operating conditions (15.0-2).

Reactor power is 55 percent of nuclear boiler rated (NBR). Normal procedures require startup of an idle loop at a lower power.

The idle recirculation pump suction valve is open, but the pump discharge valve is closed.

The idle pump fluid coupler is at a setting that approximates 50 percent generator speed demand.

15.4.4.3.3 Results

The results presented below are representative of cycle 1. References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}. These results are considered bounded by the reload licensing analysis.

The transient response to the incorrect startup of a cold, idle recirculation loop is shown on Figure 15.4-2. Shortly after the pump starts, a surge in flow from the started jet pump diffusers causes the core inlet flow to rise sharply.

A short duration neutron flux peak reaches the APRM neutron flux setpoint at 10 seconds, and reactor scram is initiated. The neutron flux peaks at 396.3 percent of NBR. Surface heat flux follows the slower response of the fuel and peaks at 80.5 percent of NBR. Nuclear system pressures do not increase significantly above initial. The water level does not reach either the high or low level setpoints.

For ARTS/MELLLA and CPPU operating conditions, the results are bounded by the power and flow dependent limits specified in the Core Operating Limits Report.

No evaluation of barrier performance is required for this event since no significant pressure increases are incurred during this transient. See Figure 15.4-2.

15.4.4.3.4 Deleted

15.4.4.4 Deleted

15.4.4.5 Radiological Consequences

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

15.4.5 Recirculation Flow Control Failure with Increasing Flow

The recirculation flow control failure with increasing flow event is considered a non-limiting event. Therefore, it is not required to be re-analyzed as a part of the reload licensing analysis for Hope Creek, unless the disposition for this event changes (Reference 15.4-2). The information described in this section for this event is based on pre-CPPU conditions. The results of an evaluation performed to validate that the event remains non-limiting and is bounded by the off-rated limits at CPPU conditions are included in Section 15.4.5.3.3.

15.4.5.1 Identification of Causes and Frequency Classification

15.4.5.1.1 Identification of Causes

Failure of a speed controller can cause a speed increase of a recirculation pump.

15.4.5.1.2 Frequency Classification

This transient is classified as an incident of moderate frequency.

15.4.5.2 Sequence of Events and Systems Operation

15.4.5.2.1 Sequence of Events

The sequence of events for the recirculation flow control failure with increasing flow event are listed in Table 15.4-2 and shown in Figure 15.4-3.

15.4.5.2.1.1 Identification of Operator Actions

Initial action by the operator will include:

1. Trip the Scoop Tube for the affected Recirculation Pump controller.
2. Reduce the non-affected Reactor Recirculation Pump speed to reduce power to pre-transient value.
3. Identification of the cause of failure.

Reactor pressure is controlled as required, depending on whether a restart or cooldown is planned. In general, the corrective action is to hold reactor pressure and condenser vacuum for restart after

the malfunctioning flow controller has been repaired. The following is the sequence of operator actions expected during the course of the event, assuming restart. The operator will:

1. Observe that all rods are fully inserted.

2. Check the reactor water level and maintain above low level (L2) trip to prevent main steam isolation valves (MSIVs) from isolating.

3. Switch the reactor mode switch to the "startup" position.

4. Continue to maintain condenser vacuum and turbine seals.

5. Reduce the Recirculation flow setpoint to minimum.

6. Survey maintenance requirements and complete the scram report.

7. Monitor the turbine coastdown and auxiliary systems.

8. Establish a restart of the reactor according to the normal procedure.

15.4.5.2.2 Systems Operation

The analysis of this transient assumes normal functioning of plant instrumentation and controls, and the Reactor Protection System (RPS). Operation of engineered safety features (ESFs) is not expected.

15.4.5.2.3 The Effect of Single Failures and Operator Errors

This transient leads to a quick rise in reactor power level, which results in a high neutron flux scram. See Section 15.9 for details. Operator errors are of no concern here in view of the fact that automatic shutdown events rapidly follow the postulated failure.

15.4.5.3 Core and System Performance

~~15.4.5.3.1 Mathematical Model~~

The nonlinear dynamic model described in section 15.1.1.3.1 is used to simulate this event.

15.4.5.3.2 Input Parameters and Initial Conditions

This analysis has been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-3.

15.4.5.3.3 Results

Figure 15.4-3 shows the results of the transient for pre-CPPU conditions. The changes in nuclear system pressure are not significant with regard to overpressure. Pressure decreases over most of the transient. The rapid increase in core coolant flow causes an increase in neutron flux, which initiates a reactor average power range monitor (APRM) high flux scram.

The peak neutron flux reaches 382.3 percent of NBR flux, and the accompanying transient fuel surface heat flux reaches 82.2 percent of rated. The MCPR remains above the safety limit and the fuel center temperature increases only 383°F. Reactor pressure is discussed in Section 15.4.5.4. Therefore, the design basis is satisfied.

The validation analysis for CPPU operation has been performed based on CPPU conditions at 30 percent CPPU power and 42 percent core flow. The reduced initial power and flow conditions are assumed since this condition results in the largest increase in core flow and power response. The changes in nuclear system pressure are not significant with regard to overpressure. Pressure increases slightly as the reactor power and steam flow increase to a new steady state value. The rapid increase in core coolant flow causes an increase in neutron flux, but does not initiate a reactor average power range monitor (APRM) high flux scram. The peak neutron flux reaches 74.4 percent of CPPU NBR flux, and the accompanying transient fuel surface heat flux reaches 45.9 percent of CPPU rated. The MCPR remains above the Safety Limit. Reactor pressure is discussed in Section 15.4.5.4. Based on this evaluation, the event remains non-limiting and does not require re-analysis on a reload basis. In addition, the event results are bounded by the off-rated thermal limits.

15.4.5.3.4 Considerations of Uncertainties

Some uncertainties in void reactivity characteristics, scram time, and worth are expected to be more optimistic and will therefore produce less severe consequences than those simulated.

15.4.5.4 Barrier Performance

This transient results in a very slight increase in reactor vessel pressure for both the pre-CPPU and CPPU evaluations and therefore represents no threat to the reactor coolant pressure boundary (RCPB).

15.4.5.5 Radiological Consequences

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

15.4.6 Chemical and Volume Control System Malfunctions

Not applicable to boiling water reactors (BWRs). This is a pressurized water reactor (PWR) event.

15.4.7 Misplaced Bundle Accident

The misplaced bundle accident is considered potentially limiting; therefore, it is considered for analysis for each reload. The results of the re-analysis of the misplaced bundle accident are presented in Appendix 15D.

15.4.7.1 Identification of Causes and Frequency Classification

15.4.7.1.1 Identification of Causes

The event discussed in this section is the improper loading of a fuel bundle and subsequent operation of the core. Three errors must occur for this event to take place in the initial core loading. First, a bundle must be loaded into a wrong location in the core. Second, the bundle that was supposed to be loaded where

the mislocation occurred would have to be overlooked and also loaded in an incorrect location. Third, the misplaced bundles would have to be overlooked during the core verification performed following initial core loading.

15.4.7.1.2 Frequency of Occurrence

This event occurs when a fuel bundle is loaded into the wrong location in the core. It is assumed the bundle is misplaced to the worst possible location, and the plant is operated with the mislocated bundle. This event is categorized as an infrequent incident based on an expected frequency of 0.004 events per operating cycle, which is based upon past experience. The only misloading events that have occurred in the past were in reload cores where only two errors are necessary. Therefore, the frequency of occurrence for initial cores is even lower since three errors must occur concurrently.

15.4.7.2 Sequence of Events and Failure Analysis

15.4.7.2.1 Sequence of Events

The postulated sequence of events for the misplaced bundle accident is presented in Appendix 15D.

Fuel loading errors, undetected by in-core instrumentation following fueling operations, may result in undetected reductions in thermal margins during power operations. No detection is assumed, and therefore, no corrective operator action or automatic protection system function occurs.

15.4.7.2.2 Effect of Single Failure and Operator Errors

This analysis represents the worst case, i.e., operation of a misplaced fuel bundle with three SAFs or SOEs, and there are no additional operator errors that could make this accident more severe. Refer to Section 15.9 for further details.

15.4.7.3 Core and System Performance

Analysis methods for this event are discussed in Section S.2.2.3.6 of GESTAR II (Reference 15.4-2).

Core performance are described in Appendix 15D.

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15.4.7.4 Barrier Performance

An evaluation of the barrier performance is not made for this event since it is a very mild and highly localized event. No change in the core pressure will be observed.

15.4.7.5 Radiological Consequences

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

15.4.8 Spectrum of Rod Ejection Accidents

This is not applicable to BWRs.

15.4.9 Control Rod Drop Accident

Hope Creek is a Banked Position Withdrawal Sequence (BPWS) plant, and therefore, in accordance with GESTAR II (Reference 15.4-2), does not need to analyze the control rod drop accident (CRDA) each reload. The results of the CRDA are presented in GESTAR II.

15.4.9.1 Identification of Causes and Frequency Classification

The causes and frequency of the control drop accident (CRDA) are described in Section S.2.2.3.1 of GESTAR II.

15.4.9.2 Sequence of Events and System Operations

A description of the sequence of events and operation of the system during a CRDA is provided in Section S.2.2.3.1 of GESTAR II.

15.4.9.3 Core and System Performance

15.4.9.3.1 Mathematical Model

The analytical methods, assumptions, and conditions for evaluating the excursion aspects of the CRDA are described in Section S.2.2.3.1 of GESTAR II.

15.4.9.3.2 Input Parameters and Initial Conditions

The input parameters and conditions for the CRDA are described in Section S.2.2.3.1 of GESTAR II.

15.4.9.3.3 Results

The radiological evaluations are based on the assumed failure of 850 fuel rods from an 8x8 fuel bundle design. The number of rods that exceed the damage threshold for the 8x8 fuel bundle design is less than 850 for all plant operating conditions or core exposures provided the peak enthalpy is less than the 280 cal/g design limit.

For the 10x10 fuel bundle design, the 280 cal/g design limit is not exceeded, and while the number of rods failing is greater than for the 8x8 rod array, the radiological consequences for these designs are essentially the same as for the 8x8 fuel designs due to lower plenum activity.

15.4.9.4 Barrier Performance

An evaluation of the 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 described in Reference 15.4-3 and 15.4-4, Appendix C, which are considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet 10CFR50.67 guidelines. This analysis is referred to as the Design Basis Analysis.

15.4.9.5.1 Design Basis Analysis

The specific models and assumptions used for this evaluation are described in Section S.2.2.3.1 of GESTAR II. Specific parametric values used in the valuation are presented in Table 15.4-6.

15.4.9.5.1.1 Fission Product Release from Fuel

The failure of 850 fuel rods is used for this analysis. The mass fraction of the fuel in the damaged rods that reaches or exceeds the initiation temperature of fuel melting, taken as 2842°C, is estimated to be 0.0077.

The failed fuel is assumed to be operating at 0.12 MWt/rod. In order to calculate the correct activity release from the fuel, a peaking factor of 1.75 is applied to the fuel pin powers to determine the fission product inventories of the damaged rods.

For the CRD accident, the release from the breached fuel is based on an NRC approved fuel vendor methodology for the number of fuel rods breached and the assumption that 10% of the core inventory of noble gases and iodine, and 12% of the core inventory of alkali metals (particulates) are in the fuel gaps. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and on the assumption that 100% of the noble gases and 50% of the iodines contained in that fraction are released to the reactor coolant. The activities released from the fuel gaps and melted fuel are assumed to be instantaneously mixed in the reactor coolant within the pressure vessel.

A maximum equilibrium inventory of fission products in the core is based on 1000 days of continuous operation at 4031 MWt. No delay time is considered between departure from the above power condition and initiation of the accident.

15.4.9.5.1.2 Fission Product Transport to the Environment

The transport pathway for this analysis, as shown in Figure 15.4-4, consists of carryover with steam to the turbine condenser, and subsequent release of all of the fission product activity to the environment via the condenser. The Main Steam Isolation Valves (MSIVs) are not assumed to close due to main steam line high radiation following the CRDA.

Consistent with the guidelines in Regulatory Guide 1.183, Appendix C (Ref. 15.4-4) and Reference 15.4-3, the condenser is isolated, and the activity airborne in the condenser leaks from the Turbine Building at ground level directly to the environment at a rate of 1.0 percent a day. Assuming that the mechanical vacuum pumps are tripped is consistent with these guidelines. No credit is taken for holdup and decay in the Turbine Building. Radioactive decay is accounted for during residence in the condenser, and is neglected after release to the environment. The release continues for 24 hours and then terminates.

15.4.9.5.1.3 Radiological Results

Site boundary doses based on a Hope Creek specific atmospheric dispersion factor were calculated using the results presented in Reference 15.4-3.

The calculated doses from the design basis analysis are presented in Table 15.4-10. The licensing basis CRDA radiological consequences are not impacted by the introduction of 12 GE14i assemblies at HCGS (Reference 15.4-5).

15.4.9.5.1.4 Main Control Room

Main control room habitability for the CRDA is bounded by the analysis for the design basis loss-of-coolant accident (LOCA) and is addressed in Section 6.4.

15.4.10 References

15.4-1 C. J. Paone "Bank Position Withdrawal Sequence," NEDO-21231, September 1976.

15.4-2 General Electric, "General Electric Standard Application For Reactor Fuel," including the "United States Supplement," NEDE-240111-P-A, and NEDE-24011-P-A-US latest revision.

15.4-3 General Electric, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Isolation Valve Closure Function and SCRAM Function of the Main Steam Line Radiation Monitor," NEDO-31400A, October 1992.

15.4-4 U.S. NRC Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000.

15.4-5 NRC letter to PSEG Nuclear dated October 7, 2010, "Hope Creek Generating Station - Issuance of Amendment 184 Re: Use of Isotopic Test Assemblies for Cobalt-60 Production (TAC No. ME2949)" (Adams Accession No. ML102700263).

15.4-6 Deleted

15.4-7 NEDC-33066P, Revision 2, "Hope Creek Generating Station, APRM/RBM/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA)," [Feb. 2005].

TABLE 15.4-1

SEQUENCE OF EVENTS FOR ABNORMAL STARTUP OF
IDLE RECIRCULATION LOOP PUMP
(FIGURE 15.4-6)

Time, s	Event
0	Start pump motor
9.6	Startup loop flow starts to increase significantly
10.6	High neutron flux scram initiated
>50	Vessel level returns to normal and stabilizes

TABLE 15.4-2

SEQUENCE OF EVENTS FOR RECIRCULATION FLOW CONTROL FAILURE
WITH INCREASING FLOW
(FIGURE 15.4-3)

Time, s	Event
0	Simulate failure of single loop control
1.95	APRM high neutron flux scram trip initiated
5.7	Turbine valves start to close upon falling turbine pressure
>50	Reactor variables settle into new steady state

TABLE 15.4-3

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TABLE 15.4-4

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TABLE 15.4-5

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TABLE 15.4-6

CONTROL ROD DROP ACCIDENT EVALUATION PARAMETERS

1.	Data and Assumptions Used to Estimate Radioactive Source from Postulated Accident		Assumptions
	a.	105% Core Power level, MWt	4031
	b.	Number of fuel rods damaged	850
	c.	Total number of fuel bundles in core	764
	d.	Number of rods per bundle	62
	e.	Peaking factor	1.75
	f.	Fission product released from failed fuel rods	
		melted	100% NG/50% I/25% Alkali Metals
		non-melted	10% NG/10% I/12% Alkali Metals
	g.	Mass fraction of fuel that reaches or exceeds the initiation temperatures for melting (2842°C)	0.0077
2.	Data and Assumptions Used to Estimate Activity Released		
	a.	Fraction of fission products transported to main condenser	100% NG/10% I/1% Alkali Metals
	b.	Fraction of fission products airborne in main condenser	100% NG/10% I/1% Alkali Metals
	c.	Condenser leak rate, percent/day	1.0
	d.	Duration of Release	24.0 hrs
3.	Dispersion Data (x/Q calculated by methodology in Section 2.3.4.2.1)		
	Exclusion Area Boundary (EAB):		
	a.	x/Q (s/m^3) for time interval 0-2 h	1.9E-4
	Low Population Zone (LPZ):		
	b.	x/Q (s/m^3) for time interval	
		0-2 h	1.9E-5
		2-4 h	1.2E-5
		4-8 h	8.0E-6
		8-24 h	4.0E-6
		24-96 h	1.7E-6
		96-720 h	7.7E-7

TABLE 15.4-7

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TABLE 15.4-9

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TABLE 15.4-10

CONTROL ROD DROP ACCIDENT (DESIGN BASIS ANALYSIS)

RADIOLOGICAL EFFECTS[1]

Description	Exclusion Area Boundary Maximum 2-hr dose Rem (TEDE)	Low Population Zone 30-day dose Rem (TEDE)
Release via isolated condenser	3.98E-2	7.76E-3

1. The above results of the radiological consequence evaluation are not impacted by the introduction of 12 GE14i assemblies at HCGS.

TABLE 15.4-11

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TABLE 15.4-12

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TABLE 15.4-17

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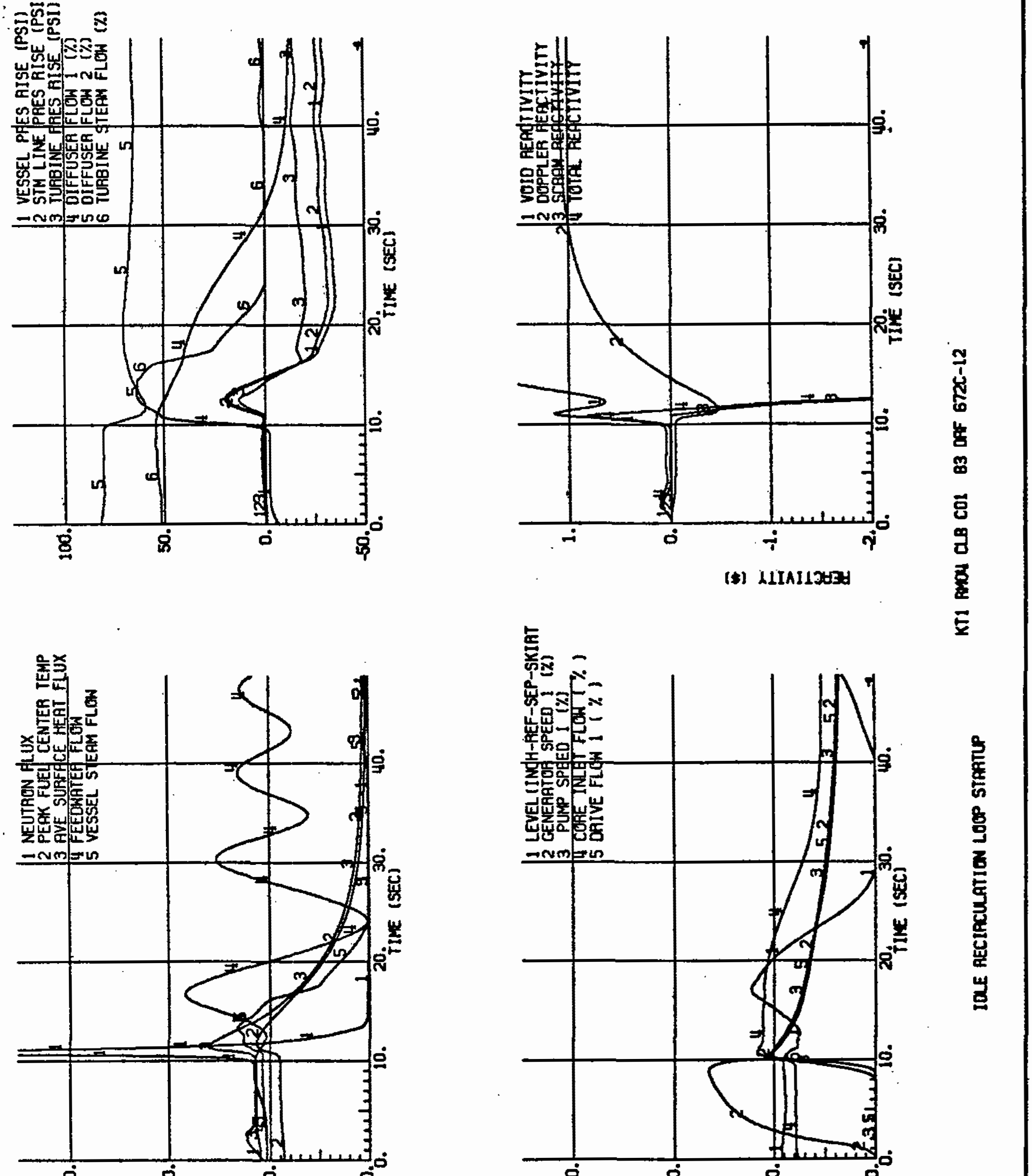
PSEG NUCLEAR L.L.C.
HOPE CREEK GENERATING STATION

HOPE CREEK UFSAR - REV 11 SHEET 1 OF 1
November 24, 2000 F15.4-1



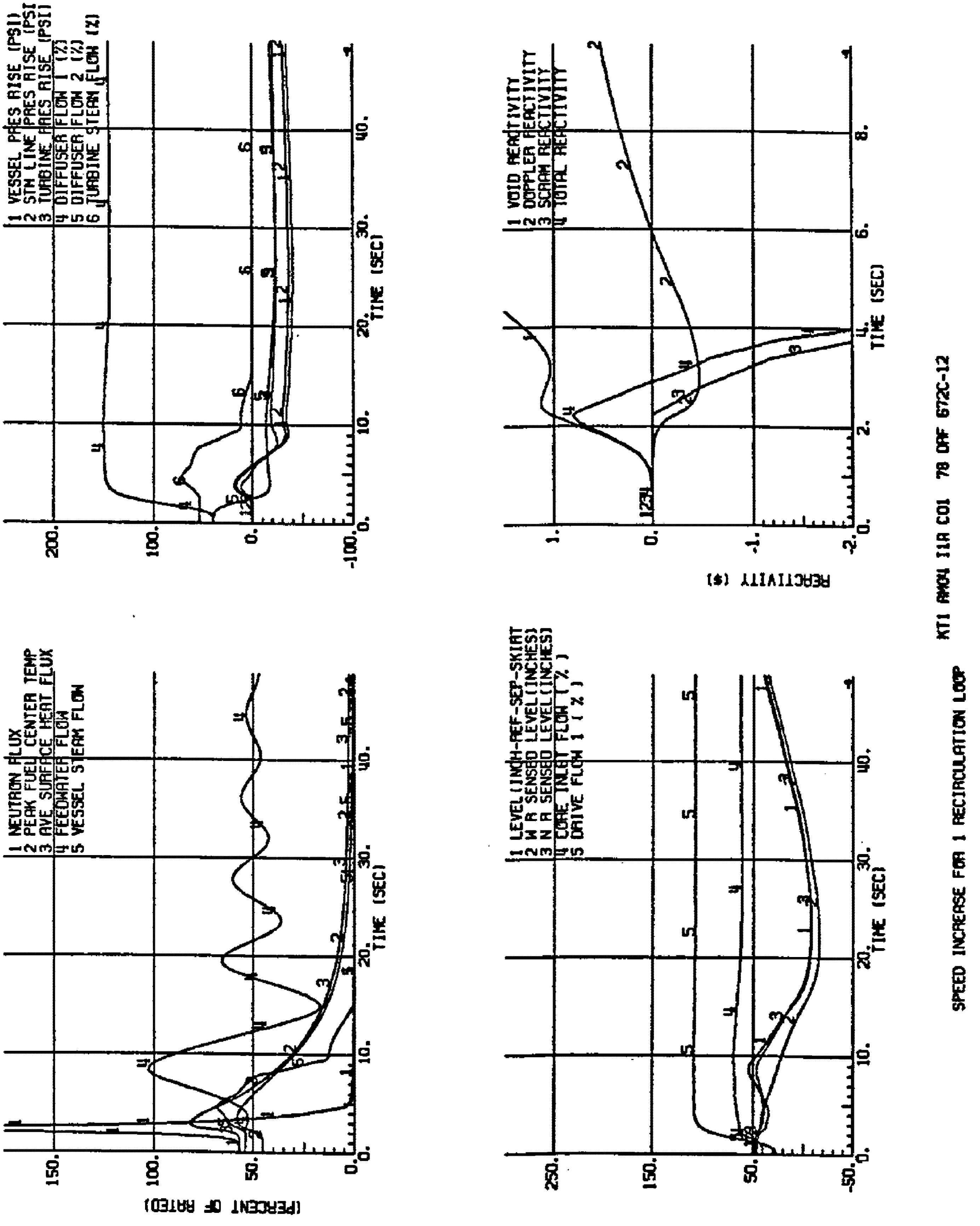
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

ABNORMAL STARTUP OF IDLE RECIRCULATION LOOP PUMP

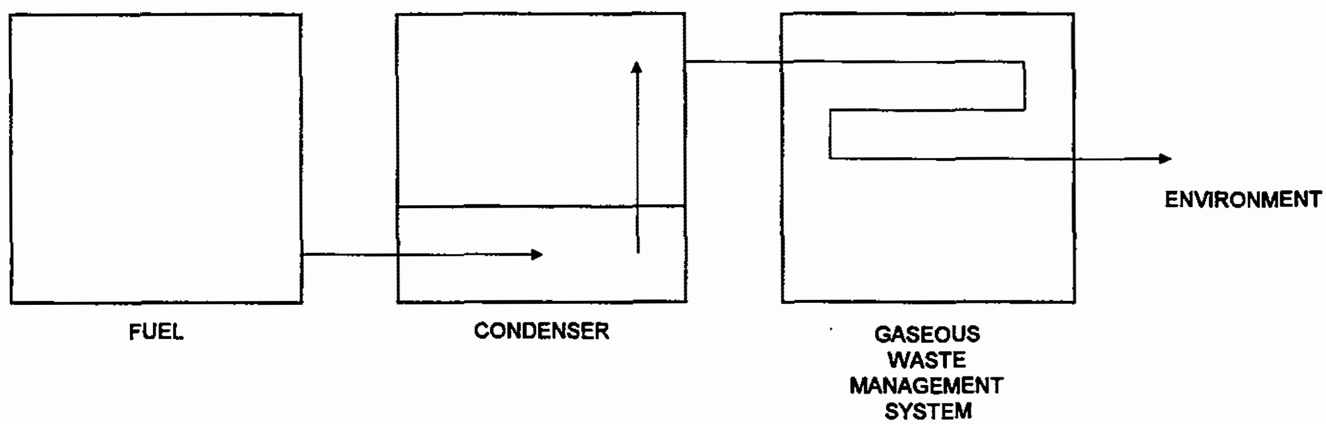
UPDATED FSAR

FIGURE 15.4-2



Revision 14, July 26, 2005

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station RECIRCULATION FLOW CONTROL FAILURE WITH INCREASING FLOW
	Updated FSAR Figure 15.4-3



PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

LEAKAGE PATH MODEL
FOR ROD DROP ACCIDENT

Updated FSAR
Revision 7, December 29, 1995

FIGURE 15.4-4

15.5 INCREASE IN REACTOR COOLANT INVENTORY

15.5.1 Inadvertent High Pressure Coolant Injection Startup

The inadvertent high pressure coolant injection startup event is considered a non-Limiting event. Sensitivity studies have shown that the inadvertent high pressure coolant injection startup event is similar to the loss of feedwater heating event. The loss of feedwater heating event bounds the inadvertent high pressure coolant injection startup event. Therefore it is not required to be re-analyzed as a part of the reload licensing analysis for Hope Creek, unless the disposition for this event changes (Reference 15.5-1)

The results referenced within this section are representative of cycle 1. References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}. These results are considered bounded by the reload licensing analysis.

15.5.1.1 Identification of Causes and Frequency Classification

15.5.1.1.1 Identification of Causes

Manual startup, i.e., operator error, of the High Pressure Core Injection (HPCI) System is postulated for this analysis.

15.5.1.1.2 Frequency Classification

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

15.5.1.2 Sequence of Events and Systems Operation

15.5.1.2.1 Sequence of Events

Table 15.5-1 lists the sequence of events for Figure 15.5-1.

15.5.1.2.1.1 Identification of Operator Actions

The operator will, upon indication that the HPCI has commenced operation, check reactor water level and drywell pressure. If conditions are normal, the operator will shut down the system.

15.5.1.2.2 System Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls; specifically, the reactor pressure regulator and reactor vessel level control that respond automatically to this event.

Required operation of engineered safety features (ESFs), other than what is described, is not expected for this transient event.

The system is assumed to be in the manual flow control mode of operation.

15.5.1.2.3 The Effect of Single Failures and Operator Errors

Inadvertent operation of the HPCI results in a mild pressurization. Corrective action by the pressure regulator and/or level control is expected to establish a new stable operating state. The effect of a failure in the pressure regulator is to aggravate the transient, depending upon the nature of the failure. Pressure regulator failures are discussed in Sections 15.1.3 and 15.2.1.

A single failure in the level control system will cause the RPV level to rise or fall by improper control of the feedwater system. Increasing water level will trip the turbine and automatically trip the HPCI system. This sequence is already described in the failure of feedwater controller with increasing flow. Decreasing RPV level will automatically initiate a scram at L3 level and will have results similar to loss of feedwater control with decreasing flow.

15.5.1.3 <u>Core and System Performance</u>

15.5.1.3.1 Mathematical Model

The detailed nonlinear dynamic model described briefly in Section 15.1.1.3.1 is used to simulate this transient.

15.5.1.3.2 Input Parameter and Initial Conditions

This analysis has been performed, unless otherwise noted, with plant conditions as tabulated in Table 15.0-3.

The water temperature of the HPCI system was assumed to be 40°F with an enthalpy of 11 Btu/lb.

Inadvertent startup of the HPCI system was chosen to be analyzed because it provides the greatest auxiliary source of cold water into the vessel.

15.5.1.3.3 Results

The results presented below are representative of cycle 1. References to percent power, percent of rated, etc., contained in the text, figures, and tables describing this event are relative to the Cycle 1 licensed power level of 3293 MW_{th}. These results are considered bounded by the reload licensing analysis.

Figure 15.5-1 shows the simulated transient event for the manual flow control mode. It begins with the introduction of cold water into the feedwater sparger. Within 1 second, the full HPCI flow is established at approximately 21.8 percent of the rated feedwater flow rate. No delays were considered because they are not relevant to the analysis.

Addition of cooler water to the core causes the neutron flux to increase to 118.7 percent of rated at approximately 16 seconds, and the plant parameters begin to stabilize.

15.5.1.3.4 Consideration of Uncertainties

Important analytical factors, including the reactivity coefficient and HPCI flow temperature, are assumed to be at the worst conditions so that any deviations in the actual plant parameters will produce a less severe transient.

15.5.1.4 Barrier Performance

Figure 15.5-1 indicates a slight pressure reduction from initial conditions; therefore, no further evaluation is required, since reactor coolant pressure boundary pressure margins are maintained.

15.5.1.5 Radiological Consequences

Since no activity is released during this event, a detailed evaluation is not required.

15.5.2 Chemical Volume Control System Malfunction (or Operator Error)

This section is not applicable to boiling water reactors (BWRs).

15.5.3 Increase in Reactor Coolant Inventory BWR Transients

These events are discussed and considered in Sections 15.1 and 15.2.

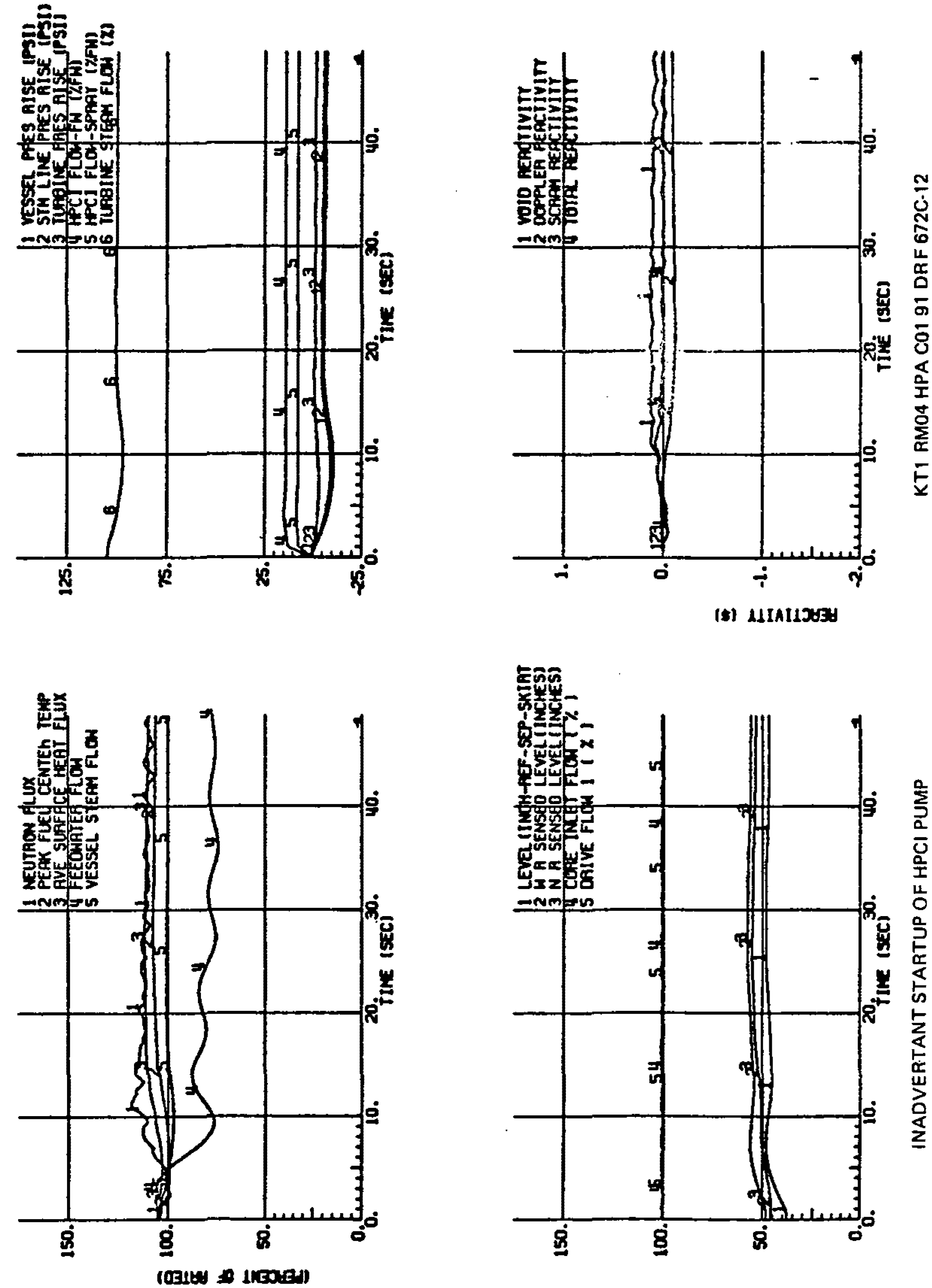
15.5.4 References

15.5-1 "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (latest approved revision), and "General Electric Standard Application for Reactor Fuel (Supplement for United States)", NEDE-24011-P-A-US (latest approved revision)

TABLE 15.5-1

SEQUENCE OF EVENTS FOR INADVERTENT HPCI PUMP STARTUP
(FIGURE 15.5-1)

Time, s	Event
0	Simulate HPCI cold water injection
1.0	Full flow established for HPCI
>16	Reactor variables settle into a new steady state



PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

INADVERTENT HPCI PUMP STARTUP

UPDATED FSAR

FIGURE 15.5-1

15.6 DECREASE IN REACTOR COOLANT INVENTORY

15.6.1 Inadvertent Safety/Relief Valve Opening

This event is discussed and analyzed in Section 15.1.4.

15.6.2 Instrument Line Pipe Break

The Instrument Line Pipe Break accident is considered a non-limiting event. Therefore it is not required to be re-analyzed as a part of the reload licensing analysis for Hope Creek, unless the disposition for this event changes (Reference 15.6-5).

This event involves the postulation of a small break in a steam or liquid line inside or outside the primary containment but within a controlled release structure. To bound the event, it is assumed that a small instrument line, instantaneously and circumferentially, breaks at a location where the break may not be able to be isolated and where immediate detection is not automatic or apparent.

This event is far less limiting than the postulated events in Sections 15.6.4 and 15.6.5.

This postulated event represents the envelope evaluation for small line failure inside and outside primary containment, relative to sensitivity to detection. It is summarized in Tables 15.6-2 and 15.6-5 and shown on Figure 15.6-1.

15.6.2.1 <u>Identification of Causes and Frequency Classification</u>

15.6.2.1.1 Identification of Causes

There is no specific event or circumstance identified that results in the failure of an instrument line. These lines are designed to applicable engineering codes and standards, and to appropriate seismic and environmental requirements. Flow control check valves and flow limiting orifices are also provided for each instrument line that penetrates the primary containment. However, for the purpose of evaluating the consequences of a small line rupture, the failure of an instrument line is assumed to occur.

15.6.2.1.1.1 Event Description

A circumferential rupture of an instrument line that is connected to the Reactor Coolant System is postulated to occur outside the primary containment but inside the Reactor Building. This failure results in the release of reactor coolant to the Reactor Building until the reactor pressure vessel (RPV) is depressurized. This event could also occur in the drywell. However, the associated effects from this would not be as significant as those from a failure in the Reactor Building.

15.6.2.1.2 Frequency Classification

This event is categorized as a limiting fault, as defined in Section 15.0.3.

15.6.2.2 Sequence of Events and Systems Operation

15.6.2.2.1 Sequence of Events

The sequence of events for this accident is shown in Table 15.6-1.

15.6.2.2.1.1 Identification of Operator Actions

The operator will isolate the affected instrument line. Depending on which line is broken, the operator will determine whether to continue plant operation until a scheduled shutdown can be made or to proceed with an immediate, orderly plant shutdown and initiate the Filtration, Recirculation, and Ventilating System (FRVS).

As a result of increased radiation, temperature, humidity, and fluid within the Reactor Building, operator action can be initiated by any one or any combination of the following:

1. Operator comparing readings with several instruments monitoring the same process variable such as reactor level, jet pump flow, steam flow, and steam pressure

2. Either a high or low indication in the main control room from the instrument served by the failed line

3. A half-channel scram if rupture occurred on a reactor protection system instrument line

4. A general increase in the area radiation monitor readings

5. An increase in the ventilation process radiation monitor readings

6. Increases in area temperature monitor readings in the Reactor Building

7. Leak detection system indications.

Upon receiving one or more of the above signals, the operator will proceed to shut down the system.

15.6.2.2.2 System Operation

Plant instrumentation and controls are assumed to be functional during the entire transient to ensure positive identification of the break and safe shutdown. Minimum reactor and plant protection system operations are assumed for the analysis, e.g., minimum Emergency Core Cooling System (ECCS) flow and suppression pool cooling capability. As a consequence of the accident, the reactor vessel is cooled and depressurized over a 5-hour period.

15.6.2.2.3 The Effect of Single Failures and Operator Errors

A discussion of the effects of single failures and operator errors is presented in Section 15.9.

15.6.2.3 Core and System Performance

15.6.2.3.1 Qualitative Results Summary

Instrument line breaks, because of their small size, are substantially less limiting from a core and systems performance standpoint than the events examined in Sections 15.6.4 and 15.6.5. Consequently, instrument line breaks are considered to be bounded specifically by the steam line break, Section 15.6.4. Details of the steam line break calculation, including those pertinent to core and system performance, are discussed in detail in Sections 6.3.3 and 15.6.4.3.

Since instrument line breaks result in a lower rate of coolant loss and are bounded by the calculations referenced above, the results presented here are qualitative rather than quantitative. Since the coolant loss is slow, RPV depressurization follows the reactor scram; and the RPV is cooled down and maintained without ECCS actuation. No fuel damage or exposure of the core occurs as a result of this accident.

15.6.2.3.2 Considerations of Uncertainties

The bounding analysis of the pipe break event is presented in Section 6.3.

15.6.2.4 Barrier Performance

The release of primary coolant through the orificed instrument line could result in an increase in reactor building pressure and the potential for isolation of the normal ventilation system. The Reactor Building Ventilation System (RBVS) trips, and the FRVS starts on an occurrence of high radioactivity concentration in the Reactor Building exhaust air. The FRVS can be also initiated manually after identification of the accident.

The following assumptions and conditions are the bases for the mass loss during the 5-hour reactor shutdown period of this event:

1. Shutdown and depressurization is initiated at 10 minutes after the break occurs.

2. Normal depressurization and cooldown of the RPV occurs.

3. The broken line contains a 1/4-inch diameter flow restricting orifice inside the drywell.

4. The Moody critical blowdown flow model, Reference 15.6-1, is applicable.

The total integrated mass of fluid released into the Reactor Building via the break during the blowdown is 25,000 pounds. Of this total, 11,500 pounds flash to steam. Release of this mass of coolant results in a reactor building pressure that is well below the design pressure.

15.6.2.5 Radiological Consequences

15.6.2.5.1 Design Basis Analysis

The guidance in the NRC Standard Review Plan (SRP) 15.6.2 and Safety Guide 11 are used to calculate design basis radiological consequences.

15.6.2.5.2 Analysis Approach

The specific models and assumptions used for evaluation are described in UFSAR Section 15A.4. The computer code is described in Appendix 15A. Specific values of parameters used in the evaluation are presented in Table 15.6-2. The leakage path used in these calculations is shown in Figure 15.6-1.

15.6.2.5.2.1 Fission Product Release from Fuel

No fuel damage is associated with this accident. As a result of depressurizing the Reactor Coolant System, an iodine spike occurs. It is assumed that that spike raises the iodine concentration in the reactor coolant to 4μCi/g of dose equivalent I-131. This is the maximum short term concentration allowed by the Technical Specifications.

15.6.2.5.2.2 Fission Product Release to the Environment

Of the 25,000 pounds of coolant released from the instrument line break, 11,500 pounds flash to steam. It is assumed that all the iodine in the coolant which flashes to steam enters the steam phase with the coolant and that 10 percent of the iodine in the unflashed coolant becomes airborne. No credit for plateout is taken. It is assumed that all activity is released instantaneously even though it would actually be released over a period of time.

The noble gas isotopic concentrations are calculated based on sufficient fuel cladding defects to result in a total release rate of 100,000 μCi/sec at time t=0 sec., uprated steam mass flow rate, and 100/E-bar.

The activity is assumed to mix with 50 percent Reactor Building volume consistent with the location of the break. It is assumed to be released unfiltered through the normal RBVS even though the FRVS would be activated within 20 minutes after the accident.

15.6.2.5.2.3 Radiological Results

Dose conversion factors for iodine are taken from Federal Guidance Report (FGR) 11 (Ref. 15.6-6), and breathing rates during the accident are taken from Regulatory Guide 1.183 (Ref. 15.6-7), as discussed in Appendix 15A. The calculated doses for the realistic analysis are presented in Table 15.6-5. The licensing basis Instrument Line Pipe Break accident radiological consequences are not impacted by the introduction of 12 GE14i assemblies at HCGS (Reference 15.6-13).

15.6.3 Steam Generator Tube Failure

This section is not applicable to the direct cycle boiling water reactor (BWR).

15.6.4 Steam System Piping Break Outside containment

The steam system piping break outside containment accident is considered a non-Limiting event. Therefore it is not required to be re-analyzed as a part of the reload licensing analysis for Hope Creek, unless the disposition for this event changes (Reference 15.6-5).

It is assumed that the main steam line instantaneously and circumferentially breaks at a location downstream of the outboard isolation valve as discussed in Section 3.6.1. The plant is designed to immediately detect such an occurrence, initiate isolation of the broken line, and actuate the necessary protective features. This postulated event represents the envelope evaluation of steam line failures outside primary containment. This accident is summarized in Tables 15.6-6 through 15.6-11 and on Figure 15.6-2.

15.6.4.1 Identification of Causes and Frequency Classification

15.6.4.1.1 Identification of Causes

These lines are designed to applicable engineering codes and standards and to appropriate seismic and environmental requirements. However, for the purpose of evaluating the consequences of the event, main steam line break is postulated without the cause being identified.

15.6.4.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.6.4.2 Sequence of Events and Systems Operation

15.6.4.2.1 Sequence of Events

Accidents that result in the release of radioactive materials directly outside primary containment are the results of postulated breaches in the reactor coolant pressure boundary (RCPB) 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 primary containment is a complete severance of one of the four main steam lines. The sequence of events and approximate times required to reach the events are given in Table 15.6-6.

Normally the operator maintains the vessel inventory and core cooling with the reactor core isolation cooling (RCIC) system. Following main steam isolation valve (MSIV) closure, the RCIC system initiates automatically on a signal of low water level. The core is covered throughout the accident, and there is no fuel damage. Without taking credit for the RCIC water makeup capability, and assuming high pressure coolant injection (HPCI) system failure, the Automatic Depressurization System (ADS) will automatically actuate at low water level, L1, to reduce reactor pressure. The subsequent actuations of the low pressure ECCS systems will reestablish water level above the core and terminate the accident without fuel damage.

15.6.4.2.2 Systems Operation

The postulated break of one of the four main steam lines 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 unbroken lines. Subsequent

closure of the main steam isolation valves (MSIVs) further limits the flow when the valve area becomes less than the limiter area and finally terminates the mass loss when full closure is reached.

A discussion of the responses of the plant, the Reactor Protection System (RPS), and engineered safety features (ESF) is given in Sections 6.3, 7.3, and 7.6.

15.6.4.2.3 The Effect of Single Failures and Operator Errors

The effect of single failures has been considered in analyzing this event. The Emergency Core Cooling System (ECCS) aspects are covered in Section 6.3. The break detection and isolation considerations are defined in Sections 7.3 and 7.6. Refer to Section 15.9 for further details.

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.

15.6.4.3.1 Input Parameters and Initial Conditions

Refer to Section 6.3 for initial conditions.

15.6.4.3.2 Results

There is no fuel damage as a consequence of this accident.

Refer to Section 6.3 for ECCS analysis.

15.6.4.3.3 Considerations of Uncertainties

Sections 6.3 and 7.3 contain discussions of the uncertainties associated with the performance of the ECCS and the containment isolation system, respectively.

15.6.4.4 Barrier Performance

Since this break occurs outside the primary containment, barrier performance within the containment envelope is not applicable. Details of the results of this event can be found in Section 6.2.3.

The following assumptions and conditions are used in determining the mass loss from the nuclear system from the inception of the break to full closure of the MSIVs:

1. The reactor is operating at the full power level given in Table 15.0-3.

2. Nuclear system pressure is 1060 psia and remains constant during MSIV closure.

3. An instantaneous circumferential break of the main steam line occurs.

4. Isolation valves start to close at 0.5 second on a high steam flow signal and are fully closed at 5.5 seconds.

5. The Moody critical flow model, Reference 15.6-1, is applicable.

6. Level rise time is conservatively assumed to be 1 second. Mixture quality is conservatively taken to be a constant 7 percent (steam weight percentage) during mixture flow.

Initially, only steam issues from the broken end of the steam line. The flow in each line is limited by critical flow at the limiter. Rapid depressurization of the RPV causes the water level to rise, resulting in a steam water mixture flowing from the break until the valves are closed. The total integrated reactor coolant mass leaving the RPV through the steam line break is 140,000 pounds. Although this release consists of two-phase flow of water and steam mixture with different iodine and noble gas concentrations in each phase, it is conservatively assumed that the reactor coolant iodine and noble gas concentrations are conservative for both phases. All iodine and noble gas activities in the reactor coolant are assumed to release to the environment.

15.6.4.5 Radiological Consequences

A design basis radiological analysis is provided for this accident that is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet Regulatory Guide 1.183, Table 6 and 10CFR50.67 guidelines.

A schematic of the release path is shown in Figure 15.6-2.

The design basis analysis is based on NRC Standard Review Plan (SRP) 15.6.4 and NRC Regulatory Guide 1.183, Appendix D. The specific models and assumptions used are described in Reference 15.6-4. Specific values of parameters used in the evaluation are presented in Table 15.6-7.

15.6.4.5.1 Fission Product Release from Break

There is no fuel damage as a result of this accident. The only activities available for release from the break are the iodine and noble gas activities, which are

present in the reactor coolant. The reactor coolant iodine and noble gas isotopic concentrations corresponding to the extended power uprate are obtained from Reference 15.6-12.

Because of the short half-life of N-16, the offsite radiological effects from this isotope are of no major concern and are not considered in the analysis.

Two separate iodine concentrations are used; one with an assumed pre-accident spike and one with a maximum equilibrium iodine. The iodine spike used is assumed to result in a reactor coolant iodine concentration of 4 μCi/g of dose equivalent I-131. This concentration is the maximum short term iodine concentration allowed in the Technical Specifications.

The iodine concentration used for the analysis with a maximum equilibrium iodine is assumed to be 0.2 μCi/g of dose equivalent I-131. This concentration is the Technical Specification limit for continuous operation.

15.6.4.5.2 Fission Product Transport to the Environment

The transport pathway is a direct, unfiltered release to the environment.

It is assumed that 140,000 pounds of blowdown is released from the break even though analysis has shown that only 99,480 pounds is released in the worst case break. Although this release would be partially water and partially steam with different iodine concentrations in each, it is assumed that reactor coolant iodine concentrations are appropriate for both. Similarly, the noble gas concentrations are assumed equal for both phases.

It is assumed that all activity released becomes airborne.

15.6.4.5.3 Radiological Results

Dose conversion factors for iodine are taken from Federal Guidance Report 11 (Reference 15.6-6) and breathing rates during the accident are taken from Regulatory Guide 1.183, Sections 4.1.3 & 4.2.6. The whole body dose is calculated using the dose conversion factors taken from Federal Guidance Report 12 (Reference 15.6-11. The calculated doses for the design basis analysis are presented in Table 15.6-9. The licensing basis Main Steam Line Break accident radiological consequences are not impacted by the introduction of 12 GE14i assemblies at HCGS (Reference 15.6-13).

15.6.5 Loss-of-Coolant Accident Resulting from the Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary Inside Primary Containment

This event involves the postulation of a spectrum of piping breaks inside primary containment varying in size, type, and location. The break type includes steam and/or liquid process system lines. This event is also coincident with a safe shutdown earthquake (SSE).

The Loss-of-Coolant Accident resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary inside primary containment is critical in determining the ECCS performance's compliance with the 10CFR50.46. The 10CFR50.46 compliance is re-evaluated for each reload application. The results of the re-evaluation are discussed in Appendix 15D.

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 other subject sections. All other information is covered by cross-references.

The postulated event represents the envelope evaluation for liquid or steam line failures inside primary containment. It is summarized in Table 15.6-12 and on Figure 15.6-3.

15.6.5.1 Identification of Causes and Frequency Classification

15.6.5.1.1 Identification of Causes

There are no realistic, identifiable events that would result in a pipe break inside the primary containment of the magnitude required to cause a LOCA coincident with an SSE plus single active component failure. The subject piping is designed to applicable engineering codes and standards, and for appropriate 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 causes being identified.

15.6.5.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.6.5.2 Sequence of Events and Systems Operation

15.6.5.2.1 Sequence of Events

The sequence of events associated with this accident is shown in Table 6.3-1 for core system performance and Table 6.2-8 for barrier (containment) performance.

Following the pipe break and scram, the low-low coolant level (L2) trip signal or high drywell pressure signal initiates the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems at time zero plus approximately 30 seconds. The low-low-low coolant level (L1) trip or high drywell pressure signal initiates main steam isolation valve (MSIV) closure and both core spray and low pressure coolant injection (LPCI) systems at time zero plus approximately 40 seconds.

15.6.5.2.1.1 Identification of Operator Actions

Since automatic actuation and operation of the Emergency Core Cooling System (ECCS) is a system design basis, no operator actions are required to mitigate the accident. However, the operator will perform the following:

1. Ensure that all rods have been inserted and determine plant condition by observing the control panels.

2. Observe that the ECCS flows are initiated.

3. Check that the diesel generators have started and are on standby condition.

4. Determine that the Safety Auxiliaries Cooling System (SACS) and Station Service Water System (SSWS) are available.

5. Initiate operation of the RHR system heat exchangers in the suppression pool cooling mode.

6. Monitor the hydrogen concentration in the drywell for proper activation of the post-LOCA recombiners.

15.6.5.2.2 Systems Operations

Accidents that could result in the release of radioactive fission products directly into the primary containment are the result of postulated nuclear system pipe breaks that violate the reactor coolant pressure boundary (RCPB). Pipe break sizes and locations are examined in Sections 6.2 and 6.3, including severance of small process system lines, the main steam lines upstream of the flow restrictors, and the recirculation loop pipelines. The most severe nuclear system effects, and the greatest releases of radioactive material to the containment, result from an instantaneous circumferential break of one of the two

recirculation loop pipelines. The minimum required functions of any reactor and plant protection system are discussed in Sections 6.2, 6.3, 7.3, 7.6, 8.3, and 15.9.

15.6.5.2.3 The Effect of Single Failures and Operator Errors

Single failures and operator errors are considered in the analysis of the entire spectrum of primary system breaks. The consequences of a LOCA with considerations for single failures are shown to be fully accommodated without the loss of any required safety function. See Section 15.9 for further details.

15.6.5.3 <u>Core and System Performance</u>

15.6.5.3.1 Mathematical Model

The analytical methods and assumptions that are used in evaluating the consequences of this accident provide a conservative assessment of the expected consequences of this event.

The details of these calculations, their justification, and bases for the models are discussed in Sections 6.3, 7.3, 7.6, 8.3, and 15.9.

15.6.5.3.2 Input Parameters and Initial Conditions

Input parameters and initial conditions used for the analysis of this event are given in Table 6.3-2.

15.6.5.3.3 Results

Results of this event are given in Section 6.3. 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. Post-accident tracking instrumentation and controls remain functional. Continued long term core cooling is demonstrated.

15.6.5.3.4 Consideration of Uncertainties

See Sections 6.3, 7.5, 7.6, 8.3, and 15.9 for details.

15.6.5.4 <u>Barrier Performance</u>

The design basis for the primary containment is to maintain its integrity after the instantaneous rupture of the largest coolant pipe within the structure. The design also accommodates the dynamic effects of the pipe break at the same time an SSE is occurring. Therefore, any postulated LOCA 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 <u>Radiological Consequences</u>

A radiological analysis is provided for this accident.

A schematic of the transport pathway is shown in Figure 15.6-3.

The methods, assumptions, and conditions used to evaluate this accident are in accordance with those guidelines set forth in the NRC Standard Review Plan (SRP) 15.0.1 and Regulatory Guide 1.183, Appendix A, Rev. 0 (Ref. 15.6-7). The specific models used to evaluate this event based on the above criteria are presented in Reference 15.6-8. The computer code used is described in Appendix 15A. Specific values of parameters used in this evaluation are presented in Table 15.6-12.

An evaluation of the main control room habitability is addressed in Section 6.4

15.6.5.5.1 Fission Product Inventory

The inventory of fission products in the reactor core available for release to the containment are based on the maximum full power operation of the core that considers fuel enrichment and fuel burnup with a thermal power level of 3,917 MWt. A core power level that considers rated thermal power and uncertainty is assumed. All fuel assemblies in the core are assumed to be affected and the core average inventory is used.

15.6.5.5.2 Fission Product Transport to the Environment

The fractions of radionuclides released from the fuel are provided in Table 1 of Regulatory Guide 1.183. Of the radioiodine released from the reactor core to the containment, 95 percent of the iodine released is assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products are assumed to be in particulate form and available for release to the environment from the containment.

There are three transport pathways:

(a) Primary containment leakage

(b) Leakage from engineered safety feature (ESF) components outside the primary containment

(c) Leakage from the Main Steam Isolation Valves (MSIVs)

The transport pathway for the initial 375 seconds of the accident consists of unfiltered leakage to the environment. For the remainder of the accident for primary containment leakage and leakage from ESF components outside the primary containment, the transport pathway consists of leakage from the primary containment to the Reactor Building and discharge to the environment through the Filtration, Recirculation, and Ventilation System (FRVS) exhaust vent.

The leakage mechanisms are discussed below.

1. Primary containment leakage

   The design basis leak rate of the primary containment (excluding the main steam lines) is 0.5 percent per day. As discussed above, after the initial 375 seconds, all of this leakage is to the Reactor Building and from there to the environment via the FRVS. Credit is taken for 50 percent mixing within the reactor building.

2. Leakage from engineered safety feature (ESF) components outside the primary containment.

   With the exception of noble gases, all the fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the suppression pool at the time of release from the core. The assumed total leakage rate is 5.7 gpm (= 2.85 x 2). It begins immediately and continues throughout the accident. It is assumed that 10% of the iodine in the leakage becomes airborne (as discussed in Section 6.2.1.1.3.2, the suppression pool temperature never exceeds 212°F).

As discussed above, after the initial 375 seconds, all the iodine from the leakage is assumed to be released to the reactor building since no ESF piping penetrates the reactor building. The radioiodine that is postulated to be available for release to the environment is assumed to be 97 percent elemental and 3 percent organic. Credit is taken for 50 percent mixing within the reactor building.

3. Leakage from the Main Steam Isolation Valves (MSIVs).

It is assumed that the MSIVs will leak at a combined rate of 250 scfh for all four main steam lines (150 scfh through one MSIV failed steam line and 100 scfh through one intact steam line).

Aerosol removal efficiencies in the main steam lines were calculated using the method described in Reference 15.6-9 with a 40^{th} percentile settling velocity. Elemental iodine removal efficiency is calculated using the method described in Reference 15.6-10 which included elemental iodine deposition and resuspension rates. Credit is also taken for elemental iodine plateout on wetted containment surfaces.

The post-LOCA containment, ESF, and MSIV leakage path releases are analyzed to include progeny from the decay of parent radionuclides.

The FRVS will maintain the reactor building differential pressure equal to or greater than 0.25 inches water gauge by exhausting air according to the following equation:

$$E(t) = 3324 + 5676 \exp. (-1.18t)$$

where: $E(t)$ = exhaust rate, cfm

t = time after the building reaches -0.25 inch w.g., h (assumed to be 375 s after LOCA)

The FRVS provides for filtered recirculation and filtered exhaust. A description of the FRVS design is provided in Section 6.8. A discussion of the mathematical modeling of the FRVS is in Section 15A.6.2.

15.6.5.5.1.3 Radiological Results

Dose calculations determine the total effective dose equivalent (TEDE), which is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure.

Exposure-to-CEDE factors for inhalation of radioactive material are derived consistent with the guidance provided in Regulatory Guide 1.183. Similarly, external exposure factors are also derived consistent with the guidance provided in Regulatory Guide 1.183.

Breathing rates during the accident are taken from Regulatory Guide 1.183 as presented in Appendix 15A.

The calculated doses for the design basis analysis are presented in Table 15.6-16. The licensing basis LOCA radiological consequences are revised by the introduction of 12 GE14i assemblies at HCGS (Reference 15.6-13).

(Historical Information)

15.6.5.5.2 Realistic Analysis

The specific models and assumptions used for evaluation are described in Reference 15.6-2. The computer codes used are described in Appendix 15A. Specific values of parameters used in the evaluation are presented in Table 15.6-12.

15.6.5.5.2.1 Fission Product Release from Fuel

Since this accident does not result in any fuel damage, the only activity released to the drywell is that activity contained in the

(Historical Information)

reactor coolant plus any additional activity which may be released as a consequence of reactor scram and vessel depressurization.

While there are various activation and corrosion products contained in the reactor coolant, the products of primary importance are the iodine isotopes I-131 to I-135. The design basis coolant concentration for these isotopes is:

I-131	1.3E-2 μCi/gm
I-132	7.6E-2 μCi/gm
I-133	5.6E-2 μCi/gm
I-134	9.7E-2 μCi/gm
I-135	5.6E-2 μCi/gm

Considering that approximately 40 percent of the reactor coolant flashes to steam, it is conservatively assumed that 40 percent of the total iodine activity in the coolant is airborne initially. It is also assumed that 10 percent of the iodine in the unflashed coolant becomes airborne.

However, as a result of plateout and condensation effects, only 50 percent of the activity initially airborne remains available for release to the environment.

As a consequence of reactor scram and depressurization, additional iodine activity is released from those rods which experienced cladding perforation during normal operation. Measurements performed (Reference 15.6-3) at operating BWRs during reactor shutdown have been used to develop an analytical model for the prediction of iodine and noble gas spiking as a consequence of reactor scram and vessel depressurization. Based on the 95th percentile probability, i.e., only 5 percent of the time will the release be greater, the I-131 release is calculated to be 2.1 Ci/bundle and the Xe-133 released to be 12 Ci/bundle. Other iodine and noble gas isotopes are determined in accordance with the method presented in Section 11.1 and are tabulated in Table 11.1-2.

(Historical Information)

Since no measurements have been obtained during a pressure transient as rapid as the LOCA, it is difficult to predict the actual release rate from the fuel as a consequence of iodine spiking. It is, therefore, arbitrarily assumed that 100 percent of the spiking source term is released during the time period that 40 percent of the discharged coolant is flashing to steam.

It is also assumed that plateout and condensation remove 50 percent of the airborne iodine spiking activity. The total activity airborne in the primary containment is presented in Table 15.6-17.

It is assumed that all airborne primary containment iodine consists of 99 percent elemental iodine and 1 percent organic iodine.

15.6.5.5.2.2 Fission Product Transport to the Environment

The leak rate from the primary containment to the reactor building is 0.5 percent/day where 100 percent mixing is assumed to occur. The leakage from the ESF components is assumed to be 5 gpm. This activity is assumed available for leakage and 10 percent becomes airborne. The leakage is assumed to have the concentration of design basis coolant. The initial airborne activity released by ESF component leakage is:

Isotope	Ci/s
I-131	4.10E-7
I-132	2.40E-6
I-133	1.77E-6
I-134	3.06E-6
I-135	1.77E-6

The MSIVs leakage is assumed to be 0.767 cfm (11.5 CFH for each of the four valves). It continues for 20 minutes and the operator action prevents further leakage.

The method of analysis is identical to that used in the design basis analysis. The isotopic activities in the Reactor Building and released to the environment are presented in Tables 15.6-18 and 15.6-19, respectively.

15.6.5.5.2.3 Radiological Results

The method of dose calculation is identical to that used in the design basis analysis.

The calculated radiological doses for this event are presented in Table 15.6-20.

15.6.5.5.3 Parametric Analysis

The HCGS Reactor Building design basis inleakage rate is 100 percent per day (2778 cfm). After thermal expansion of that inleakage and after adding 4 cfm for primary containment leakage (0.5 percent per day corrected to Reactor Building temperature and pressure), 3324 cfm must be exhausted to maintain the reactor building differential pressure equal to or greater than 0.25 inches water gauge. This is the steady state term of the FRVS exhaust equation presented in Section 15.6.5.5.1.2.

Different inleakage rates would modify the steady state term as follows:

Inleakage (percent/day)	Steady State Term (cfm)
10	336
50	1664
100	3324

The primary result of a change in the inleakage rate is to change the estimated time needed to bring the Reactor Building differential pressure to at least 0.25 in. wg. The calculated times are 168, 203, and 375 seconds for inleakage rates of 10, 50, and 100 percent per day

(Historical Information)

respectively. (For analysis purposes 175, 225, and 400 seconds are used respectively.) Figure 15.6-4 through 15.6-7 present calculated doses at the site boundary (SB) and low population zone (LPZ) versus drawdown time. Figure 15.6-8 presents a graph of drawdown time versus inleakage rate. Finally, Figures 15.6-9 through 15.6-12 present calculated doses at the SB and LPZ versus inleakage rates.

15.6.6 Feedwater Line Break - Outside Primary Containment

The feedwater line break outside primary containment accident is considered a non-Limiting event. Therefore it is not required to be re-analyzed as a part of the reload licensing analysis for Hope Creek, unless the disposition for this event changes (Reference 15.6-5).

To evaluate pipe breaks in a large liquid process line outside primary containment, the failure of a feedwater line is assumed. The postulated break of the feedwater line, representing the largest liquid line outside the primary containment, provides the design basis for this event. The break is assumed to be instantaneous, circumferential, and external to the outermost isolation valve.

A more limiting event from a core performance evaluation standpoint (feedwater line break inside containment) has been quantitatively analyzed in Section 6.3. Therefore, the following discussion provides new information not presented in Section 6.3.

15.6.6.1 Identification of Causes and Frequency Classification

15.6.6.1.1 Identification of Causes

A feedwater line break is assumed without the cause being identified. The subject piping is designed to applicable engineering codes and standards, and to appropriate seismic and environmental requirements.

15.6.6.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.6.6.2 Sequence of Events and Systems Operation

15.6.6.2.1 Sequence of Events

The sequence of events is shown in Table 15.6-21.

15.6.6.2.1.1 Identification of Operator Actions

Since automatic actuation and operation of the Emergency Core Cooling System (ECCS) is a system design basis, no operator actions are required to mitigate this accident. However, in accordance with procedural requirements, the operator will perform the actions listed below:

1. The operator will determine that a line break has occurred and will implement emergency instructions.

2. The operator will ensure that the reactor is shut down and that the Reactor Core Isolation Cooling (RCIC) System and/or the High Pressure Core Injection (HPCI) System are operating normally.

3. The operator will shut down the feedwater system and deenergize any electrical equipment damaged by water from the feedwater system in the Turbine Building.

4. The operation will begin normal reactor cooldown measures.

5. When the reactor pressure has decreased below 100 psig, the operator will initiate the Residual Heat Removal (RHR) System in the shutdown cooling mode to continue cooling down the reactor.

15.6.6.2.2 Systems Operations

It is assumed that the plant instruments and controls are functional. Credit is taken for vessel isolation and actuation of the ECCS.

15.6.6.2.3 The Effect of Single Failures and Operator Errors

The feedwater line outside the containment is a special case of the LOCA break spectrum considered in detail in Section 6.3. The general single failure analysis is discussed in detail in Section 6.3.3.3. Since the feedwater line break outside the primary containment can be isolated, either the RCIC system or the HPCI system can provide adequate flow to the vessel to maintain core cooling and prevent fuel rod cladding failure. A single failure in either the HPCI system or the RCIC system would not prevent sufficient flow to keep the core covered with water. See Section 6.3 and Section 15.9 for detailed description of the analysis.

15.6.6.3 <u>Core and System Performance</u>

15.6.6.3.1 Results

The feedwater line break outside the containment is less limiting than either the steam line break outside the containment, the analysis of which is presented in Sections 6.3.3 and 15.6.4, or the feedwater line break inside the containment, the analysis of which is presented in Sections 6.3.3 and 15.6.5. It is less limiting than the analyses presented in Sections 6.3.3 and 15.6.5.

For reload applications, sensitivity studies have demonstrated that there are no significant changes to the core thermal hydraulic conditions due to the introduction of new reload core condition or fuel design. Therefore, this event is not evaluated as a part of the standard reload licensing analysis process.

The reactor vessel is isolated on low-low-low (L1) water level. The RCIC system and the HPCI system 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.

15.6.6.3.2 Consideration of Uncertainties

See Section 6.3 for details.

15.6.6.4 Barrier Performance

A break spectrum analysis for the complete range of reactor conditions indicates that the limiting fault event for breaks outside the primary containment is a complete severance of one of the main steam lines as described in Section 15.6.4. The feedwater system piping break is less severe than the main steam line break. Results of the analysis of this event can be found in Sections 6.2.3 and 6.2.4.

15.6.6.5 Radiological Consequences

15.6.6.5.1 Design Basis Analysis

The specific models and assumptions used for evaluation are described in Reference 15.6-2. Specific values of parameters used in the evaluation are presented in Table 15.6-22.

15.6.6.5.1.1 Fission Product Release from Fuel

There is no fuel damage as a consequence of this accident. The activity in the main condenser hotwell prior to occurrence of the break is released.

The iodine concentration in the main condenser hotwell is 0.02 times the concentration in the reactor coolant based on the maximum equilibrium iodine concentration of 0.2 μCi/g Dose Equivalent I-131 allowed by the technical specification for normal operation of the plant. Noble gas activity in the condensate is negligible.

15.6.6.5.1.2 Fission Product Transport to the Environment

None of the 2,240,000 pounds of condensate released from the break flashes to steam since it is below 212°F. Ten percent of the iodine in the water is assumed to become airborne. It is assumed that none of the water passes through the condensate demineralizers before release.

It is assumed that the activity released from the feedwater line break is immediately released to the environment, taking no credit for holdup or plateout.

15.6.6.5.1.3 Radiological Results

Dose conversion factors for iodine are taken from Federal Guidance Report (FGR) 11 (Ref. 15.6-6) and breathing rates during the accident are taken from Regulatory Guide 1.183 (Ref. 15.6-7) as discussed in Appendix 15A.

The calculated doses for the design basis analysis are presented in Table 15.6-24. The licensing basis Feedwater Line Break accident radiological consequences are not impacted by the introduction of 12 GE14i assemblies at HCGS (Reference 15.6-13).

15.6.7 References

15.6-1 F.J. Moody, "Maximum Two-Phase Vessel Blowdown From Pipes," ASME Paper Number 65-WA/HT-1, March 15, 1965.

15.6-2 D. Nguyen, "Realistic Accident Analysis - The RELAC Code," NEDO-21142, October 1977.

15.6-3 F.J. Brutschy, et al, "Behavior of Iodine in Reactor Water During Plant Shutdown and Startup," NEDO-10585, August 1972.

15.6-4 P.P. Stancavage and E.J. Morgan, "Conservative Radiological Accident Evaluation - The CONACO1 Code," NEDO-21134, March 1976.

15.6-5 "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (latest approved revision), and "General Electric Standard Application for Reactor Fuel (Supplement for United States)", NEDE-24011-P-A-US (latest approved revision).

15.6-6 Federal Guidance Report 11, EPA-520/1-88-020, September 1988, Limiting Values Of Radionuclide Intake And Air Concentration And Dose Conversion Factors For Inhaled, Submersion, And Ingestion.

15.6-7 U.S. NRC Regulatory Guide 1.183, Alternate Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors, July 2000.

15.6-8 S. L. Humphreys et al., "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," NUREG/CR-6604, USNRC, April 1998.

15.6-9 "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term," AEB-98-03, USNRC, December 9, 1998.

15.6-10 J. E. Cline, "MSIV Leakage Iodine Transport Analysis," Letter Report dated March 26, 1991.

15.6-11 Federal Guidance Report 12, EPA-402-R-93-081, September 1993, External Exposure To Radionuclides In Air, Water, And Soil.

15.6-12 Vendor Technical Document (VTD) No. 430059, Volume 002, EPU TR T0807 - Coolant Radiation Sources.

15.6-13 NRC letter to PSEG Nuclear dated October 7, 2010, "Hope Creek Generating Station - Issuance of Amendment 184 Re: Use of Isotopic Test Assemblies For Cobalt-60 Production (TAC No. ME2949)" Adams Accession No. ML102700263).

TABLE 15.6-1

SEQUENCE OF EVENTS FOR INSTRUMENT LINE BREAK

Time, min	Event
0	Instrument line fails
0 to 10	Break identified
10	Activate RHR and initiate orderly shutdown
300	RPV depressurized.

TABLE 15.6-2

INSTRUMENT LINE FAILURE ACCIDENT - PARAMETERS TABULATED
FOR POSTULATED ACCIDENT ANALYSIS

	Parameter	Design Basis Assumptions
1.	Data and assumptions used to estimate radioactive source from postulated accidents	
	a. Power level	4,031 MWt
	b. Burnup	NA
	c. Fuel damaged	None
	d. Deleted	
	e. Iodine fractions	
	(1) Organic	3.0%
	(2) Elemental	97.0%
	(3) Particulate	0
	f. Mass of total coolant released (lb)	25,000
	g. Mass of flashed coolant (lb)	11,500
	h. Reactor coolant iodine activity	4.0 μCi/g
2.	Data and assumptions used to estimate activity released	
	a. Primary containment leak rate, percent/day	NA

	Parameter	Design Basis Assumptions
b.	Secondary containment leak rate, percent/day	NA
c.	Valve movement times	NA
d.	Normal ventilation system	
	(1) Recirculation flow	0
	(2) Recirculation filter efficiency	0
	(3) Exhaust flow, cfm	218,020 (to simulate 50% mixing in RB)
	(4) Exhaust iodine filter efficiency	0
e.	FRVS	
	(1) Recirculation flow	NA
	(2) Recirculation filter efficiency	NA
	(3) Exhaust flow	NA
	(4) Exhaust filter efficiency	NA
f.	Containment spray parameters (flow rate, drop size, etc)	NA
g.	Containment volumes, ft^3	
	(1) Primary	NA
	(2) Reactor Building	4.0E6
h.	All other pertinent data and assumptions	NA

3. Dispersion data (X/Qs calculated using methodology in Section 2.3.4.2.1)

		Parameter	Design Basis Assumptions
	a.	Exclusion area boundary (EAB)/ low population zone (LPZ) distance, m	901/8047
	b.	X/Q, s/m^3, for time intervals of	
		(1) 0-2 h - EAB/LPZ	1.9E-4/1.9E-5
		(2) 2-4 h - LPZ	1.2E-5
		(3) 4-8 h - LPZ	8.0E-6
		(4) 8-24 h - LPZ	4.0E-6
		(5) 1-4 days - LPZ	1.7E-6
		(6) 4-30 days - LPZ	4.7E-7
4.	Dose data		
	a.	EAB Breathing Rate (m^3/sec)	3.50E-4
	b.	LPZ Breathing Rates (m^3/sec)	
		0-8 hrs	3.5E-4
		8-24 hrs	1.8E-4
		24-720 hrs	2.3E-4
	c.	Deleted	
	d.	Doses	Table 15.6-5

TABLE 15.6-3

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TABLE 15.6-4

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TABLE 15.6-5

INSTRUMENT LINE FAILURE RADIOLOGICAL EFFECTS

(DESIGN BASIS ANALYSIS)[1]

	TEDE, rem
Exclusion Area boundary (Maximum 2-hour dose)	7.40E-2
Low population zone (30-day dose)	7.41E-3

1. The above results of the radiological consequence evaluation are not impacted by the introduction of 12 GE14i assemblies at HCGS.

TABLE 15.6-6

SEQUENCE OF EVENTS FOR A STEAM LINE BREAK OUTSIDE PRIMARY CONTAINMENT

Approximate Time, s	Event
0	Break of one main steam line outside primary containment.
Approx. 0.5	High steam line flow signal initiates closure of MSIVs.
<1	Reactor begins to scram.
≤5.5	MSIVs fully closed.
Approx. 27	RCIC and HPCI receive an initiation signal on low water level, L2 (RCIC considered unavailable, and HPCI assumed disabled by channel A dc power source failure).
Approx. 30	SRVs open upon high vessel pressure. The valves then open and close to maintain vessel pressure at approximately 1000 psi.
Approx. 90	Reactor water level above core begins to drop slowly due to the loss of steam through the SRVs. Reactor pressure remains at approximately 1000 psi.
Approx. 490	ADS receives a signal to initiate on low water level, L1; ADS high drywell pressure bypass timer started.
Approx. 970	All ADS timer's time delays are completed; ADS valves are actuated; rapid depressurization of vessel initiated.

Approximate Time, s	Event
Approx. 1215	Low-pressure ECCS systems begin injection with reactor fuel partially uncovered.
Approx. 1290	Core reflooded and clad temperature heatup terminated; no fuel rod failure.

TABLE 15.6-7

STEAM LINE BREAK ACCIDENT - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSES

		Design Basis Assumptions
1.	Data and assumptions used to estimate radioactive source from postulated accidents	
	a. Power level	4,031 MWt
	b. Burn-up	NA
	c. Fuel damaged	None
	d. Deleted	
	e. Iodine fractions	
	(1) Organic	0.0015
	(2) Elemental	0.0485
	(3) Particulate (CsI)	0.95
	f. Reactor coolant activity before the accident	Section 15.6.4.5.1
	g. Spiking term	Section 15.6.4.5.1
2.	Dispersion data (X/Qs calculated using methodology in Section 2.3.4.2.1)	
	a. Site boundary (SB)/low population zone (LPZ) distance, m	901/8047
	b. X/Q, s/m^3, for	
	Site boundary	1.9E-4
	Low population zone	1.9E-5

TABLE 15.6-7 (Cont)

			Design Basis Assumptions
3.	Dose data		
	a.	Method of dose calculation	Appendix 15A
	b.	Dose conversion assumptions	Section 15.6.4.5.3
	c.	Peak activity concentrations in containment	NA
	d.	Doses	Table 15.6-9

TABLE 15.6-8

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TABLE 15.6-9

STEAM LINE BREAK ACCIDENT (DESIGN BASIS ANALYSIS)
RADIOLOGICAL DOSE CONSEQUENCES

Accident Case Analyzed	TEDE Dose (rem)	
	EAB 2-Hr Maximum	LPZ
MSLB With Pre-accident Iodine Spike case	9.15E-01	9.44E-02
MSLB With Maximum Equilibrium Iodine Case	5.45E-02	5.62E-03

TABLE 15.6-10

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TABLE 15.6-11

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TABLE 15.6-12

PARAMETERS AND ASSUMPTIONS USED IN RADIOLOGICAL CONSEQUENCE CALCULATIONS FOR A LOSS-OF-COOLANT ACCIDENT

Parameter	Value[1]
Reactor power	3,917 MWt
Drywell air volume	1.69E+5 ft^3
Containment air volume	3.06E+5 ft^3
Reactor building air volume	4.0E+6 ft^3
Containment leak rate to environment:	
0 - 24 hours	0.5% per day
1 - 30 days	0.25% per day
Reactor building pressure drawdown time	375 seconds
Reactor building mixing efficiency	50%
FRVS exhaust filter efficiency:	
Aerosol (particulate)	99%
FRVS vent filter efficiency:	
Elemental iodine	90%
Organic iodine	90%
FRVS recirc filter efficiency:	
Elemental iodine	Not credited
Organic iodine	Not credited
FRVS recirculation flow rate	1.08E+5 cfm
ECCS leak rate	2.85 gpm
ECCS iodine partition factor	10%
ECCS leak initiation time	0 minutes
Suppression pool volume	1.18E+5 ft^3

1. Additional parameters and assumptions used in radiological consequence evaluations for use of GE14i ITAs are documented in H-1-ZZ-MDC-1880, Revision 5.

TABLE 15.6-12 (Cont)

Parameter	Value[1]
MSIV leak rate:	
0 - 30 days:	
Total all four lines	250 scfh
MSIV failed line	150 scfh
Intact line	100 scfh
Aerosol settling velocity on main steam lines	
Failed Line between inboard MSIV and TSV	8.1E-4 meters/sec
Intact Line between RPV and TSV	8.1E-4 meters/sec
Control room volume	8.5E+4 ft^3
CREF system outside air intake flow	1100 cfm
CREF recirculation flow	2600 cfm
Control room isolation time	30 minutes
Unfiltered air inleakage rate into control room:	
0 to 30 minutes	500 cfm
30 minutes to 30 days	250 cfm
CREF system filter efficiencies:	
Elemental iodine	99%
Organic iodine	99%
Aerosol (particulate)	99%

1. Additional parameters and assumptions used in radiological consequence evaluations for use of GE14i ITAs are documented in H-1-ZZ-MDC-1880, Revision 5 .

TABLE 15.6-12 (Cont)

Parameter	Value
Meteorological Data (Atmospheric Dispersion Factors)	
Exclusive Area Boundary:	
0 - 2 hours	1.9E-4 sec/m^3
Low Population Zone:	
0 - 2 hours	1.9E-5 sec/m^3
2 - 4 hours	1.2E-5 sec/m^3
4 - 8 hours	8.0E-6 sec/m^3
8 - 24 hours	4.0E-6 sec/m^3
24 - 96 hours	1.7E-6 sec/m^3
96 - 720 hours	4.7E-7 sec/m^3

TABLE 15.6-13

LOSS-OF-COOLANT ACCIDENT (DESIGN BASE ANALYSIS)
ACTIVITY AIRORNE IN PRIMARY CONTAINMENT, Ci

(Historical Information)

Isotope	Initial	1 h	2 h	4 h	8 h	1 day	4 days	30 days
I-131	2.17E7	2.16E7	2.15E7	2.14E7	2.11E7	1.98E7	1.51E7	1.42E6
I-132	3.29E7	2.43E7	1.79E7	9.76E6	2.90E6	2.25E4	7.17E-6	0
I-133	4.86E7	4.70E7	4.54E7	4.25E7	3.72E7	2.17E7	1.94E6	1.58E-3
I-134	5.69E7	2.57E7	1.16E7	2.36E6	9.78E4	2.89E-1	0	0
I-135	4.41E7	3.98E7	3.59E7	2.91E7	1.93E7	3.68E6	2.13E3	0
Kr-83m	1.44E7	9.90E6	6.81E6	3.22E6	7.19E5	1.79E3	3.47E-9	0
Kr-85m	4.49E7	3.83E7	3.27E7	2.39E7	1.27E7	1.02E6	1.17E1	0
Kr-85	1.42E6	1.42E6	1.42E6	1.42E6	1.42E6	1.41E6	1.39E6	1.22E6
Kr-87	8.07E7	4.67E7	2.70E7	9.03E6	1.01E6	1.59E2	1.21E-15	0
Kr-88	1.11E8	8.66E7	6.75E7	4.11E7	1.52E7	2.84E5	4.79E-3	0
Kr-89	1.38E8	2.91E2	6.15E-4	2.74E-15	0	0	0	0
Xe-131m	8.97E5	8.95E5	8.92E5	8.88E5	8.78E5	8.42E5	6.97E5	1.36E5
Xe-133m	4.79E6	4.73E6	4.67E6	4.55E6	4.32E6	3.51E6	1.38E6	4.16E2
Xe-133	1.94E8	1.93E8	1.92E8	1.90E8	1.85E8	1.69E8	1.12E8	3.25E6
Xe-135m	5.38E7	3.80E6	2.69E5	1.34E3	3.34E-2	1.29E-20	0	0
Xe-135	1.86E8	1.72E8	1.60E8	1.37E8	1.01E8	2.99E7	1.24E5	2.83E-16
Xe-137	1.77E8	3.36E3	6.37E-2	2.30E-11	0	0	0	0
Xe-138	1.65E8	8.80E6	4.70E5	1.34E3	1.09E-2	0	0	0

TABLE 15.6-14

LOSS-OF-COOLANT ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY IN REACTOR BUILDING, Ci

(Historical Information)

Isotope	1 h	2 h	4 h	8 h	1 day	4 days	30 days
I-131	7.39E3	1.42E4	2.76E4	5.39E4	5.49E4	4.58E4	6.25E3
I-132	8.34E3	1.19E4	1.27E4	7.45E3	2.49E2	1.47E0	4.75E-10
I-133	1.61E4	3.00E4	5.50E4	9.53E4	6.13E4	7.44E3	1.49E2
I-134	8.90E3	7.73E3	3.09E3	2.54E2	5.13E0	1.52E-5	0
I-135	1.36E4	2.36E4	3.77E4	4.94E4	1.13E4	2.73E2	1.57E-1
Kr-83m	2.39E3	3.06E3	2.79E3	1.22E3	8.81E0	6.12E-11	0
Kr-85m	9.25E3	1.47E4	2.07E4	2.16E4	4.99E3	2.07E-1	0
Kr-85	3.42E2	6.38E2	1.23E3	2.41E3	6.94E3	2.46E4	7.43E4
Kr-87	1.13E4	1.21E4	7.83E3	1.72E3	7.81E-1	1.93E-17	0
Kr-88	2.09E4	3.04E4	3.56E4	2.58E4	1.40E3	8.46E-5	0
Kr-89	7.35E-2	2.83E-7	2.00E-18	0	0	0	0
Xe-131m	2.16E2	4.01E2	7.69E2	1.49E3	4.14E3	1.23E4	8.29E3
Xe-133m	1.14E3	2.10E3	3.94E3	7.34E3	1.72E4	2.43E4	2.54E1
Xe-133	4.65E4	8.62E4	1.64E5	3.15E5	8.32E5	1.99E6	1.98E5
Xe-135m	9.23E2	1.21E2	1.16E0	5.69E-5	0	0	0
Xe-135	4.16E4	7.18E4	1.19E5	1.72E5	1.47E5	2.19E3	1.16E-17
Xe-137	8.39E-1	2.92E-5	2.01E-14	0	0	0	0
Xe-138	2.14E3	2.12E2	1.16E0	1.85E-5	0	0	0

TABLE 15.6-15

LOSS-OF-COOLANT ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY RELEASE TO ENVIRONMENT, Ci

(Historical Information)

Isotope	1 h	2 h	4 h	8 h	1 day	4 days	30 days
I-131	5.20E2	5.20E2	5.20E2	5.20E2	5.23E2	5.36E2	5.95E2
I-132	7.84E2	7.84E2	7.84E2	7.84E2	7.84E2	7.84E2	7.84E2
I-133	1.16E3	1.16E3	1.16E3	1.16E3	1.17E3	1.18E3	1.18E3
I-134	1.34E3	1.34E3	1.34E3	1.34E3	1.34E3	1.34E3	1.34E3
I-135	1.05E3	1.05E3	1.05E2	1.05E2	1.06E3	1.06E3	1.06E3
Total equivalent							
I-131	7.83E2	7.83E2	7.83E2	7.83E2	7.87E2	8.02E2	8.62E2
Kr-83m	2.58E2	2.59E2	2.60E2	2.60E2	2.65E2	2.65E2	2.65E2
Kr-85m	8.09E2	8.16E2	8.20E2	8.23E2	1.20E3	1.24E3	1.24E3
Kr-85	2.57E1	2.60E1	2.62E1	2.65E1	2.12E2	3.61E3	1.23E5
Kr-87	1.44E3	1.44E3	1.44E3	1.45E3	1.45E3	1.45E3	1.45E3
Kr-88	1.99E3	2.01E3	2.02E3	2.02E3	2.25E3	2.26E3	2.26E3
Kr-89	1.80E3	1.80E3	1.80E3	1.80E3	1.80E3	1.80E3	1.80E3
Xe-131m	1.63E1	1.64E1	1.66E1	1.67E1	1.29E2	1.94E3	2.83E4
Xe-133m	8.68E1	8.77E1	8.83E1	8.91E1	5.92E2	5.31E3	1.16E4
Xe-133	3.52E3	3.55E3	3.58E3	3.61E3	2.65E4	3.43E5	2.20E6
Xe-135m	8.99E2	8.99E2	8.99E2	8.99E2	8.99E2	8.99E2	8.99E2
Xe-135	3.36E3	3.39E3	3.41E3	3.44E3	9.59E3	1.46E4	1.46E4
Xe-137	2.42E3	2.42E3	2.42E3	2.42E3	2.42E3	2.42E3	2.42E3
Xe-138	2.74E3	2.74E3	2.74E3	2.74E3	2.74E3	2.74E3	2.74E3

TABLE 15.6-16

LOSS-OF-COOLANT ACCIDENT
OFFSITE RADIOLOGICAL EFFECTS[1]

Exclusion Area boundary (Maximum 2-hour dose)	3.02 rem TEDE
Low population zone (30-day dose)	0.879 rem TEDE

1. Additional results of the radiological consequence evaluations for use of GE14i ITAs are documented in H-1-ZZ-MDC-1880, Revision 6.

TABLE 15.6-17

LOSS-OF-COOLANT ACCIDENT (REALISTIC ANALYSIS)
ACTIVITY AIRBORNE IN PRIMARY CONTAINMENT, Ci

(Historical Information)

Isotope	Initial	1 h	2 h	4 h	8 h	1 day	4 days	30 days
I-131	8.03E2	8.00E2	7.97E2	7.91E2	7.79E2	7.33E2	5.58E2	5.24E1
I-132	1.23E3	9.08E2	6.70E2	3.65E2	1.08E2	8.40E-1	2.68E-10	0
I-133	1.91E3	1.85E3	1.79E3	1.67E3	1.46E3	8.54E2	7.63E1	6.19E-8
I-134	2.07E3	9.34E2	4.21E2	8.58E1	3.56E0	1.05E-5	0	0
I-135	1.84E3	1.66E3	1.50E3	1.22E3	8.04E2	1.53E2	8.88E-2	0
Kr-83m	6.88E2	4.73E2	3.25E2	1.54E2	3.44E1	8.57E-2	1.66E-13	0
Kr-85m	1.68E3	1.43E3	1.23E3	8.93E2	4.75E2	3.80E1	4.39E-4	0
Kr-85	3.82E2	3.82E2	3.82E2	3.82E2	3.81E2	3.80E2	3.74E2	3.27E2
Kr-87	3.29E3	1.90E3	1.10E3	3.68E2	4.12E1	6.48E-3	4.95E-20	0
Kr-88	4.66E3	3.63E3	2.83E3	1.72E3	6.38E2	1.19E1	2.01E-7	0
Kr-89	6.11E3	1.29E-2	2.72E-8	1.21E-19	0	0	0	0
Xe-131m	7.64E1	7.62E1	7.60E1	7.56E1	7.48E1	7.17E1	5.94E1	1.16E1
Xe-133m	2.29E2	2.26E2	2.23E2	2.17E2	2.06E2	1.68E2	6.58E1	1.99E-2
Xe-133	9.17E3	9.12E3	9.07E3	8.96E3	8.76E3	8.00E3	5.32E3	1.54E2
Xe-135m	1.38E3	9.75E1	6.89E0	3.44E-2	8.58E-7	0	0	0
Xe-135	8.40E3	7.78E3	7.21E3	6.19E3	4.57E3	1.35E3	5.61E0	1.28E-20
Xe-137	8.40E3	1.59E-1	3.03E-6	1.09E-15	0	0	0	0
Xe-138	8.40E3	4.48E2	2.39E1	6.81E-2	5.52E-7	0	0	0

TABLE 15.6-18

LOSS-OF-COOLANT ACCIDENT (REALISTIC ANALYSIS)
ACTIVITY IN REACTOR BUILDING, Ci

(Historical Information)

Isotope	1 h	2 h	4 h	8 h	1 day	4 days	30 days
I-131	1.94E-1	3.50E-1	6.57E-1	1.26E0	1.28E0	1.06E0	1.20E-1
I-132	2.25E-1	3.00E-1	3.10E-1	1.79E-1	1.87E-3	2.94E-6	9.52E-16
I-133	4.51E-1	7.88E-1	1.40E0	2.37E0	1.50E0	1.47E-1	1.49E-4
I-134	2.30E-1	1.88E-1	7.24E-2	5.84E-3	7.57E-6	2.24E-11	0
I-135	4.05E-1	6.61E-1	1.02E0	1.31E0	2.71E-1	4.63E-4	1.73E-7
Kr-83m	1.22E-1	1.52E-1	1.36E-1	5.90E-2	4.23E-4	2.93E-15	0
Kr-85m	3.71E-1	5.72E-1	7.89E-1	8.16E-1	1.87E-1	7.76E-6	0
Kr-85	9.88E-2	1.78E-1	3.37E-1	6.55E-1	1.87E0	6.61E0	2.00E1
Kr-87	4.92E-1	5.14E-1	3.25E-1	7.08E-2	3.20E-5	0	0
Kr-88	9.40E-1	1.32E0	1.52E0	1.10E0	5.89E-2	3.55E-9	0
Kr-89	3.34E-6	1.27E-11	0	0	0	0	0
Xe-131m	1.97E-2	3.55E-2	6.68E-2	1.29E-1	3.54E-1	1.05E0	7.06E-1
Xe-133m	5.85E-2	1.04E-1	1.92E-1	3.55E-1	8.27E-1	1.16E0	1.22E-3
Xe-133	2.36E0	4.23E0	7.92E0	1.51E1	3.95E1	9.39E1	9.38E0
Xe-135m	2.52E-2	3.22E-3	3.04E-5	1.47E-9	0	0	0
Xe-135	2.01E0	3.37E0	5.47E0	7.85E0	6.66E0	9.91E-2	0
Xe-137	4.12E-5	1.41E-9	9.63E-19	0	0	0	0
Xe-138	1.16E-1	1.12E-2	6.02E-5	9.49E-10	0	0	0

TABLE 15.6-19

LOSS-OF-COOLANT ACCIDENT (REALISTIC ANALYSIS)
ACTIVITY RELEASE TO ENVIRONMENT, Ci

(Historical Information)

Isotope	1 h	2 h	4 h	8 h	1 day	4 days	30 days
I-131	5.10E-6	6.50E-6	7.59E-6	8.96E-6	8.09E-5	3.64E-4	1.20E-3
I-132	7.15E-6	8.59E-6	9.33E-6	9.71E-6	1.18E-5	1.19E-5	1.19E-5
I-133	1.21E-5	1.53E-5	1.77E-5	2.04E-5	1.28E-4	2.73E-4	2.91E-4
I-134	1.02E-5	1.14E-5	1.17E-5	1.18E-5	1.18E-5	1.18E-5	1.18E-5
I-135	1.13E-5	1.41E-5	1.60E-5	1.78E-5	5.47E-5	6.44E-5	6.44E-5
Total equivalent							
I-131	7.81E-6	9.92E-6	1.15E-5	1.35E-5	1.06E-4	4.16E-4	1.25E-3
Kr-83m	3.99E-4	4.73E-4	5.07E-4	5.22E-4	7.71E-4	7.72E-4	7.72E-4
Kr-85m	1.05E-3	1.30E-3	1.45E-3	1.57E-3	1.57E-2	1.72E-2	1.72E-2
Kr-85	2.53E-4	3.22E-4	3.77E-4	4.46E-4	5.03E-2	9.65E-1	3.30E1
Kr-87	1.80E-3	2.08E-3	2.18E-3	2.21E-3	2.35E-3	2.35E-3	2.35E-3
Kr-88	2.82E-3	3.42E-3	3.75E-3	3.95E-3	1.36E-2	1.38E-2	1.38E-2
Kr-89	4.93E-4	4.93E-4	4.93E-4	4.93E-4	4.93E-4	4.93E-4	4.93E-4
Xe-131m	5.05E-5	6.44E-5	7.53E-5	8.89E-5	9.62E-3	1.64E-1	2.41E0
Xe-133m	1.51E-4	1.92E-4	2.23E-4	2.62E-4	2.38E-2	2.50E-1	5.48E-1
Xe-133	6.06E-3	7.72E-3	9.01E-3	1.06E-2	1.09E0	1.60E1	1.04E2
Xe-135m	4.32E-4	4.39E-4	4.39E-4	4.39E-4	4.39E-4	4.39E-4	4.39E-4
Xe-135	5.41E-3	6.78E-3	7.75E-3	8.72E-3	2.87E-1	5.11E-1	5.12E-1
Xe-137	8.29E-4	8.29E-4	8.29E-4	8.29E-4	8.29E-4	8.29E-4	8.29E-4
Xe-138	2.47E-3	2.50E-3	2.50E-3	2.50E-3	2.50E-3	2.50E-3	2.50E-3

TABLE 15.6-20

LOSS-OF-COOLANT ACCIDENT (REALISTIC ANALYSIS)
RADIOLOGICAL EFFECTS

(Historical Information)

	Whole Body, rem	Thyroid, rem
Site boundary (2-hour dose)	2.23E-7	3.22E-7
Low population zone (30-day dose)	4.51E-7	1.99E-7

TABLE 15.6-21

SEQUENCE OF EVENTS FOR FEEDWATER LINE BREAK OUTSIDE CONTAINMENT

Time, s	Event
0	One feedwater line breaks.
>0	Feedwater line check valves isolate the reactor from the break.
~5	Reactor scram on low water level.
≤30	At low-low reactor water level, RCIC and HPCI receive an initiation signal, and the recirculation pumps trip. At low-low-low reactor water level, MSIV closure initiates.
120	The SRVs open and close to maintain the reactor vessel pressure at approximately 1100 psig.
>3600	Normal reactor cooldown procedure established.

TABLE 15.6-22

FEEDWATER LINE BREAK ACCIDENT - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSES

	Parameter	Realistic Basis Assumptions
1.	Data and assumptions used to estimate radioactive source from postulated accidents	
	a. Power level	NA
	b. Burnup	NA
	c. Fuel damaged	None
	d. DELETED	
	e. Iodine fractions	
	(1) Organic	0.03
	(2) Elemental	0.97
	(3) Particulate	0
	f. Reactor coolant activity before the accident	Section 15.6.4.5.1
2.	Data and assumptions used to estimate activity released	
	a. Primary containment leak rate, percent/day	NA

Parameter		Realistic Basis Assumptions
b.	Reactor Building leak rate, percent/day	NA
c.	Isolation valve closure time, s	NA
d.	Adsorption and filtration efficiencies of Turbine Building Ventilation System	
	(1) Organic iodine	NA
	(2) Elemental iodine	NA
	(3) Particulate iodine	NA
	(4) Particulate fission products	NA
e.	Recirculation system parameters of Turbine Building	
	(1) Flow rate	NA
	(2) Mixing efficiency	NA
	(3) Filter efficiency	NA
f.	Containment spray parameters (flow rate, drop size, etc)	NA
g.	Containment volumes	NA
h.	All other pertinent data and assumptions	None

3. Dispersion data (X/Qs calculated using methodology of Section 2.3.4.2.1)

TABLE 15.6-22 (Cont)

		Parameter	Realistic Basis Assumptions
	a.	Exclusion Area boundary (EAB)/ low population zone (LPZ), distance, m	901/8047
	b.	X/Q, s/m^3, EAB/LPZ	1.9E-4/1.9E-5
4.	Dose data		
	a.	Method of dose calculation	Appendix 15A
	b.	Dose conversion assumptions	Appendix 15A
	c.	Peak activity concentrations in containment	NA
	d.	Doses	Table 15.6-24

TABLE 15.6-23

THIS TABLE INTENTIONALLY DELETED

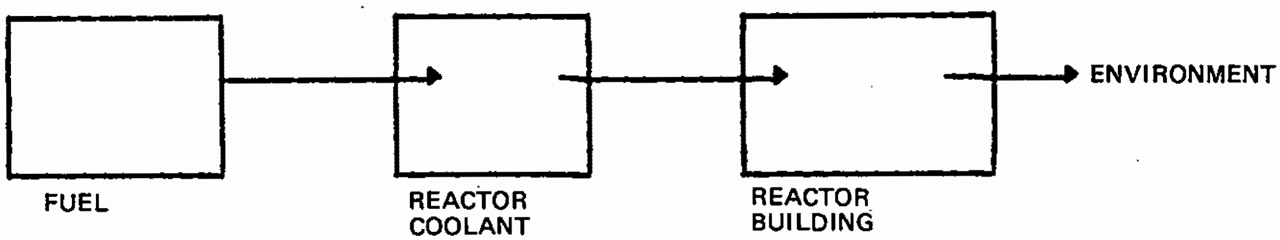
TABLE 15.6-24

FEEDWATER LINE BREAK RADIOLOGICAL EFFECTS[1]

Feedwater Line Break Accident TEDE Dose (rem)

	EAB (Maximum 2-hr Dose)	LPZ
Calculated Dose	1.50E-03	1.50E-04
Allowable TEDE Limit	2.50E+00	2.50E+00

1. The above results of the radiological consequence evaluations are not impacted by the introduction of 12 GE14i assemblies at HCGS.



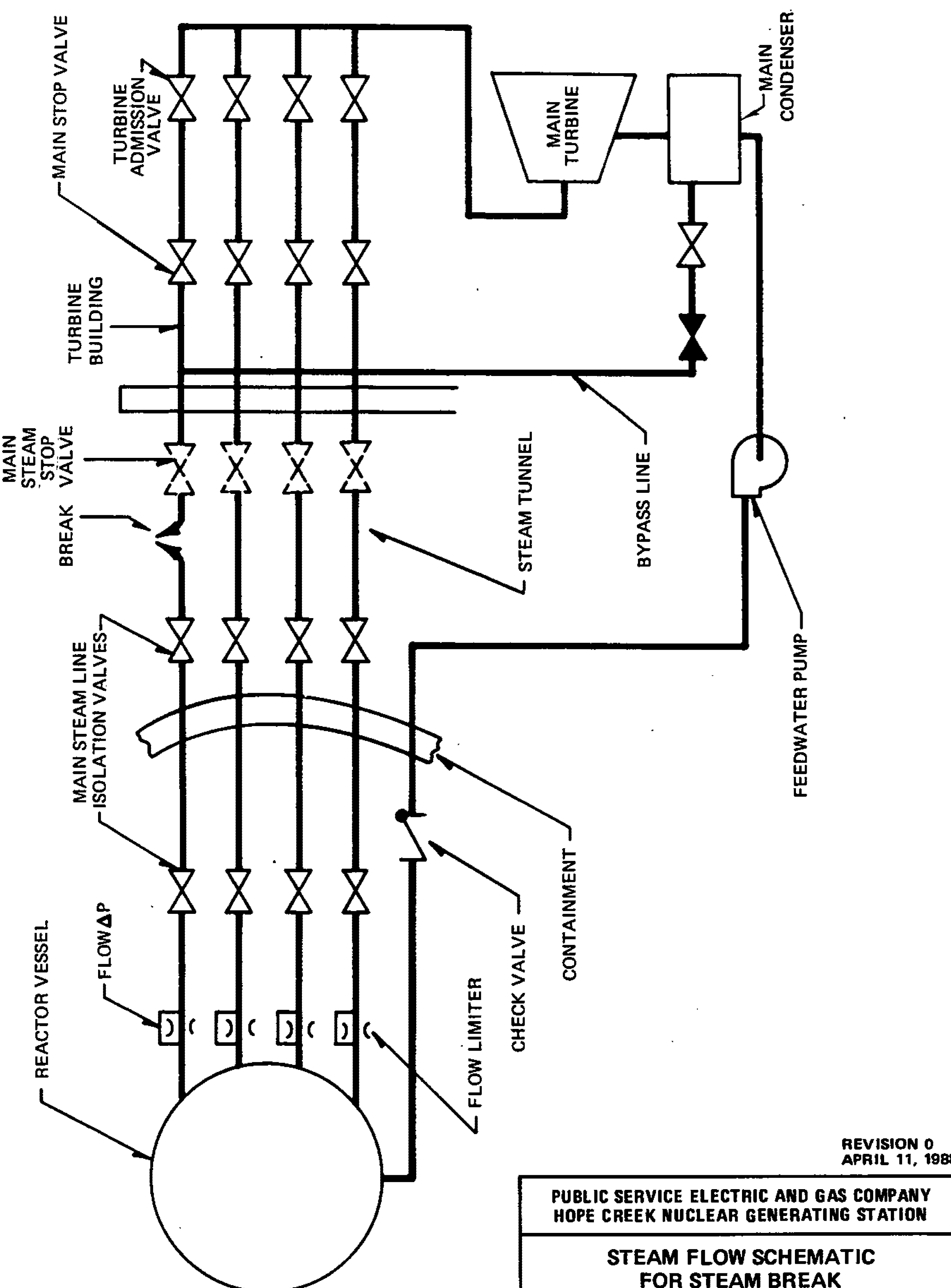
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

LEAKAGE PATH FOR
INSTRUMENT LINE BREAK

UPDATED FSAR

FIGURE 15.6-1



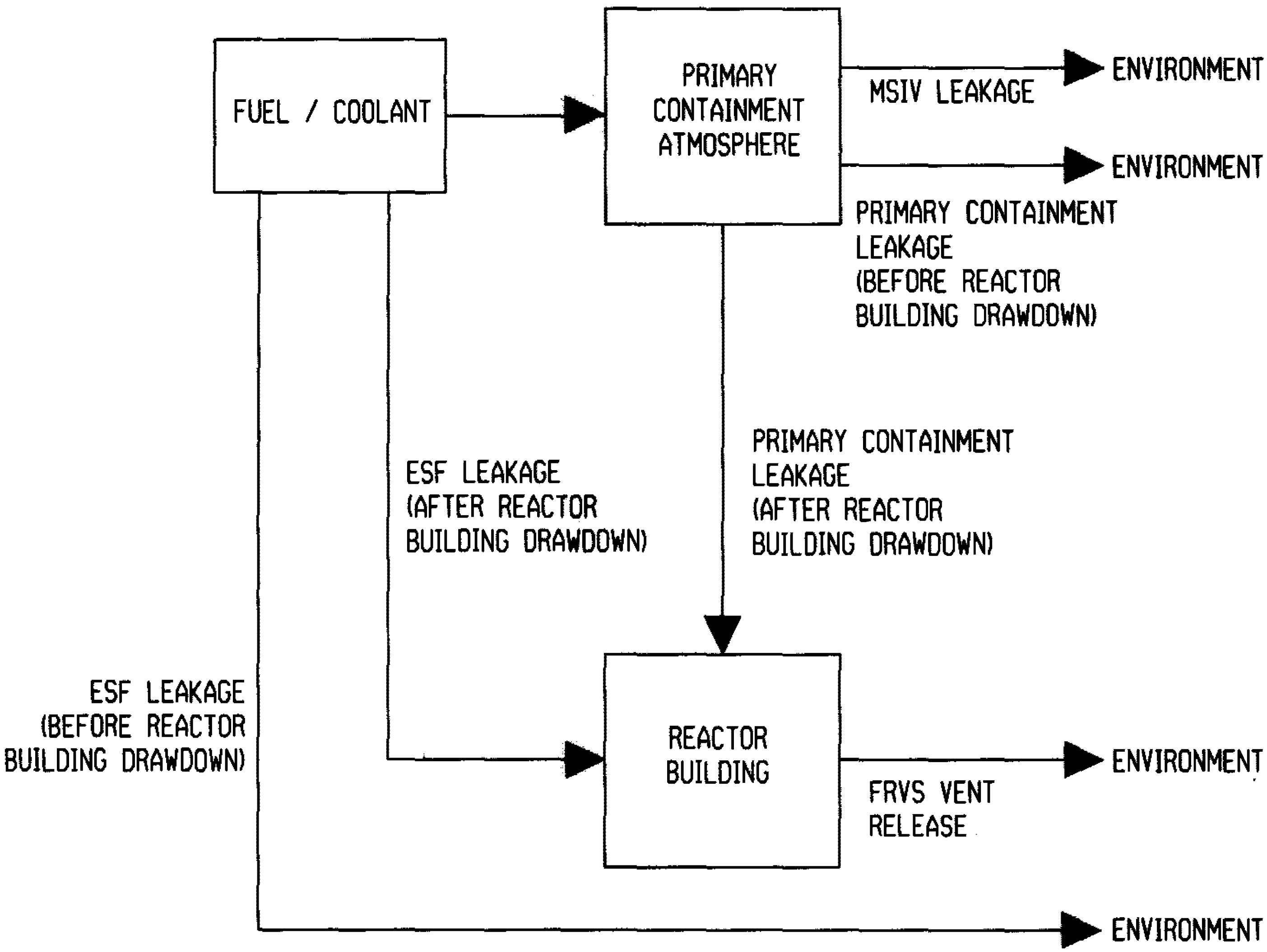
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

STEAM FLOW SCHEMATIC
FOR STEAM BREAK
OUTSIDE CONTAINMENT

UPDATED FSAR

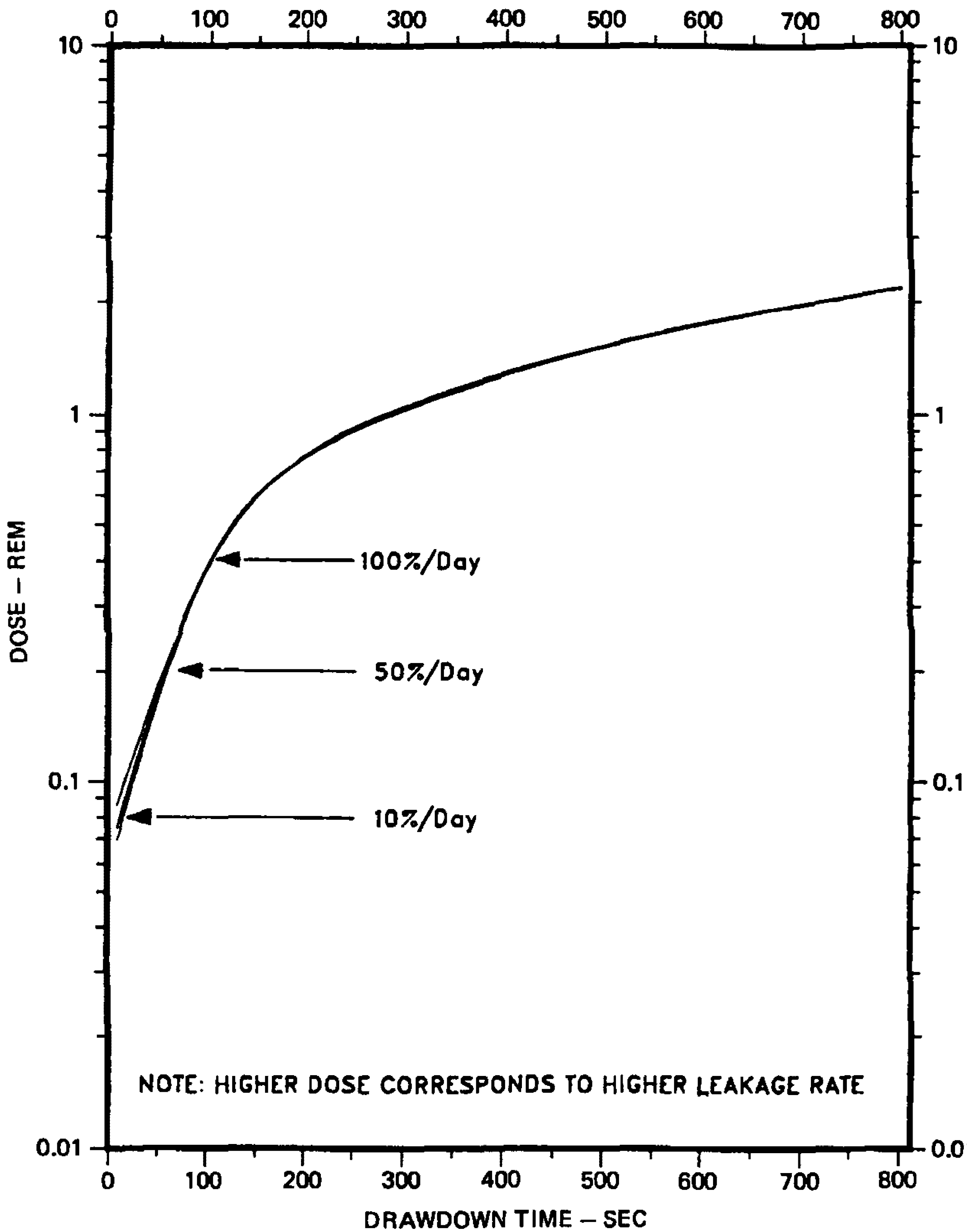
FIGURE 15.6-2



Revision 12, May 3, 2002

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station LEAKAGE FLOW FOR LOCA Updated FSAR — Figure 15.6-3

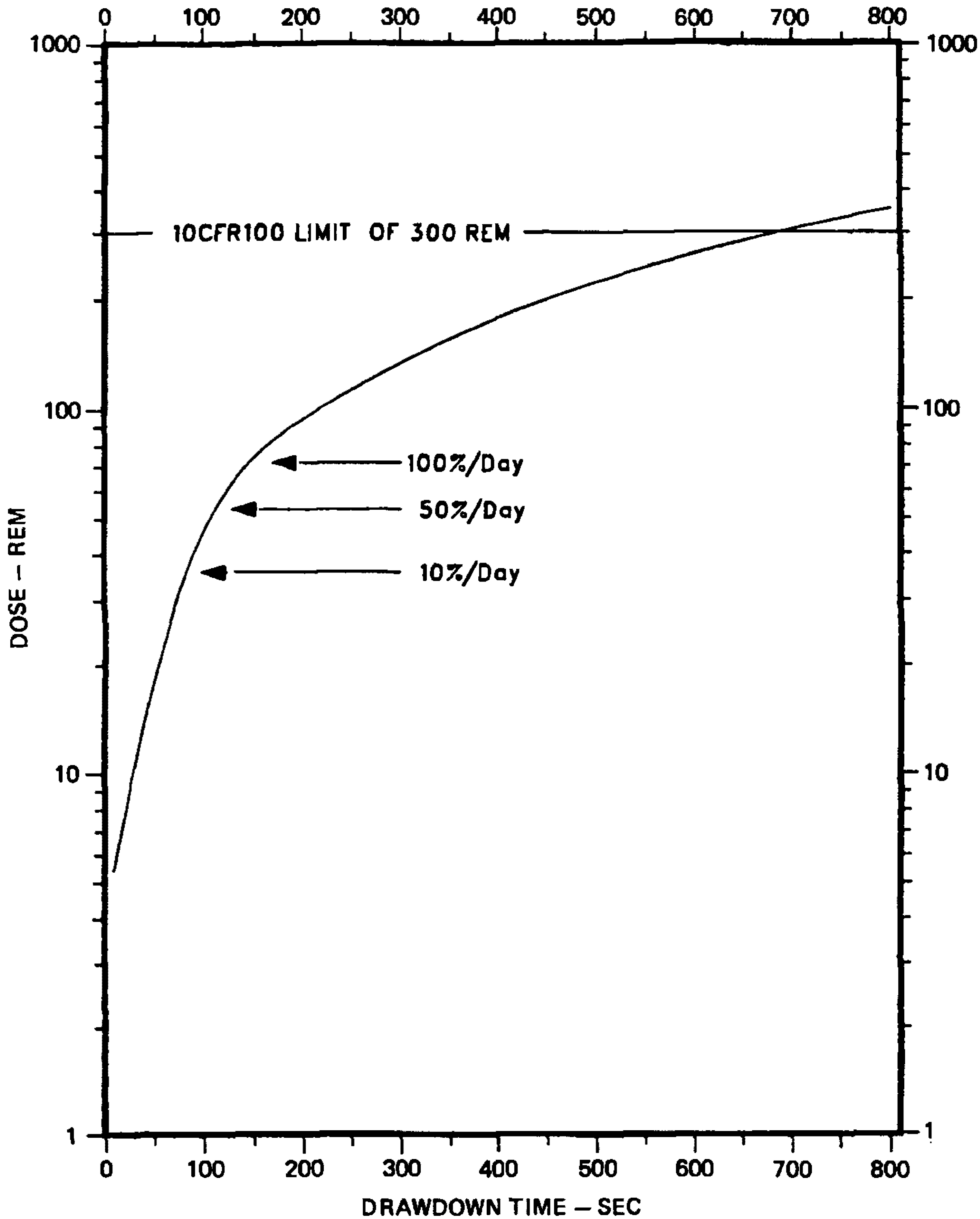
THIS FIGURE DISPLAYS HISTORICAL INFORMATION ONLY



Revision 12, May 3, 2002

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station SITE BOUNDARY WHOLE BODY DOSE (2-HOUR) VERSUS DRAWDOWN TIME FOR VARIOUS INLEAKAGE RATES
	Updated FSAR Figure 15.6-4

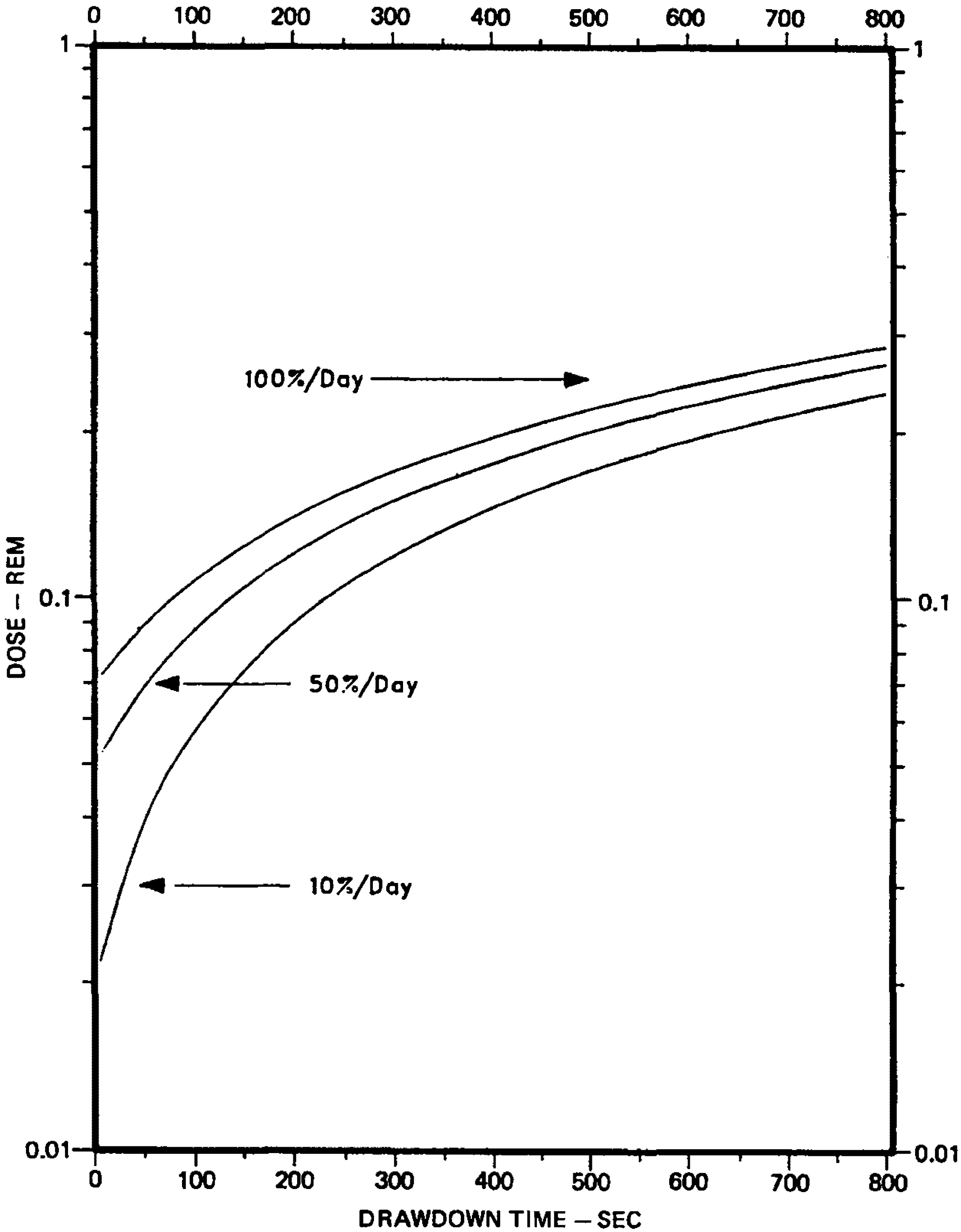
THIS FIGURE DISPLAYS HISTORICAL INFORMATION ONLY



Revision 12, May 3, 2002

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station SITE BOUNDARY THYROID DOSE (2-HOUR) VERSUS DRAWDOWN TIME FOR VARIOUS INLEAKAGE RATES
	Updated FSAR — Figure 15.6-5

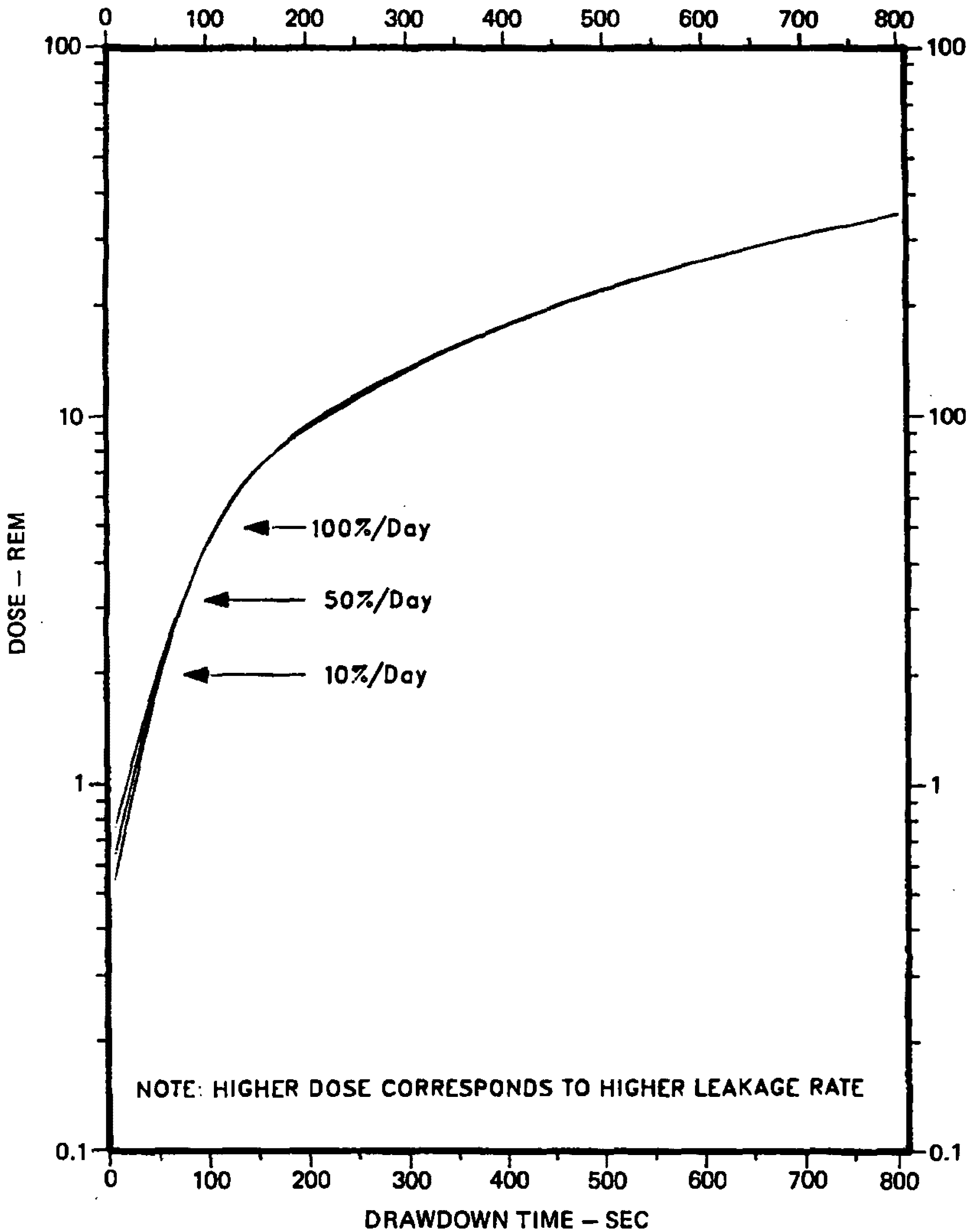
THIS FIGURE DISPLAYS HISTORICAL INFORMATION ONLY



Revision 12, May 3, 2002

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station LOW POPULATION ZONE WHOLE BODY DOSE (30-DAY) VERSUS DRAWDOWN TIME FOR VARIOUS INLEAKAGE RATES
	Updated FSAR — Figure 15.6-6

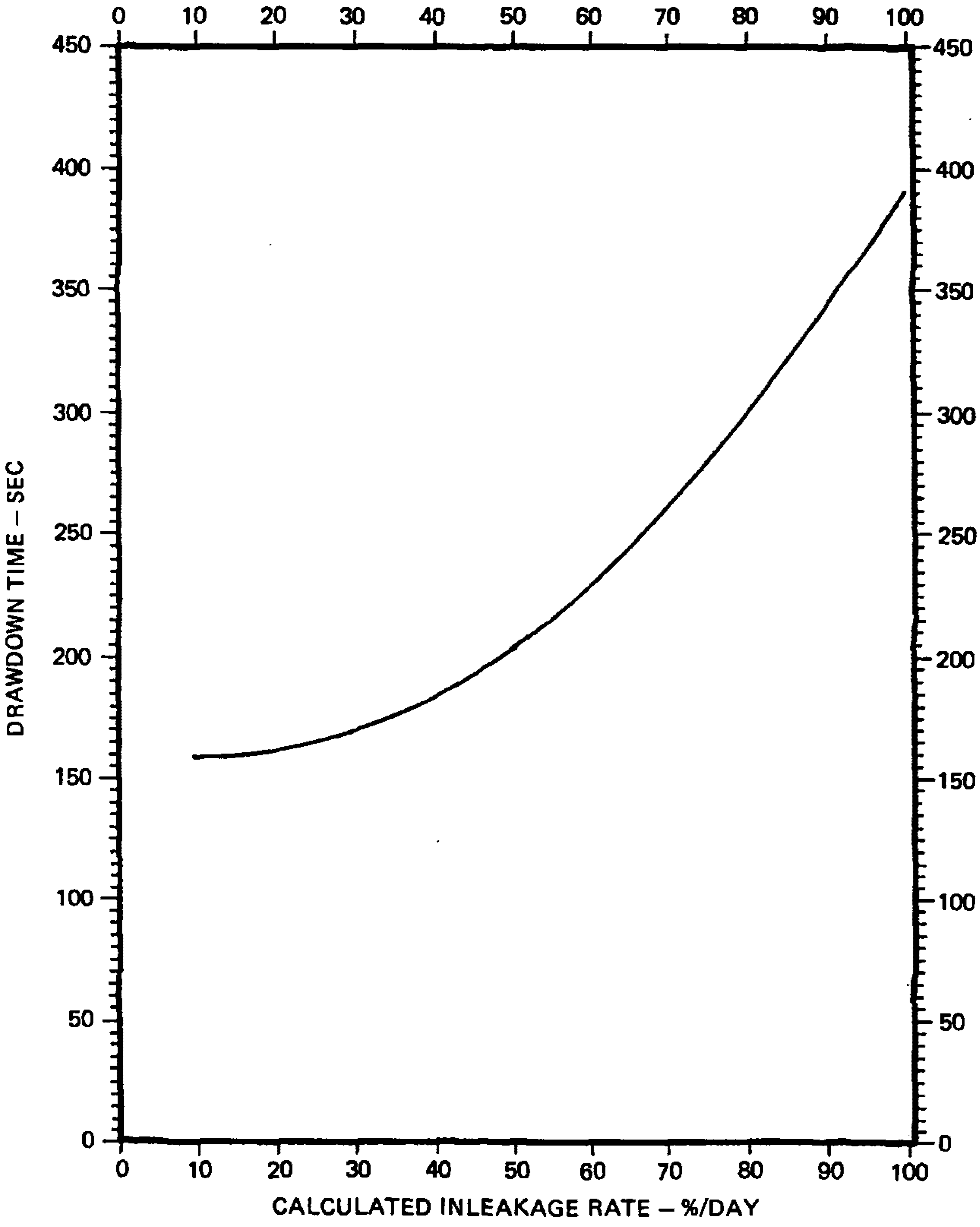
THIS FIGURE DISPLAYS HISTORICAL INFORMATION ONLY



Revision 12, May 3, 2002

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station LOW POPULATION ZONE THYROID DOSE (30-DAY) VERSUS DRAWDOWN TIME FOR VARIOUS INLEAKAGE RATES
	Updated FSAR Figure 15.6-7

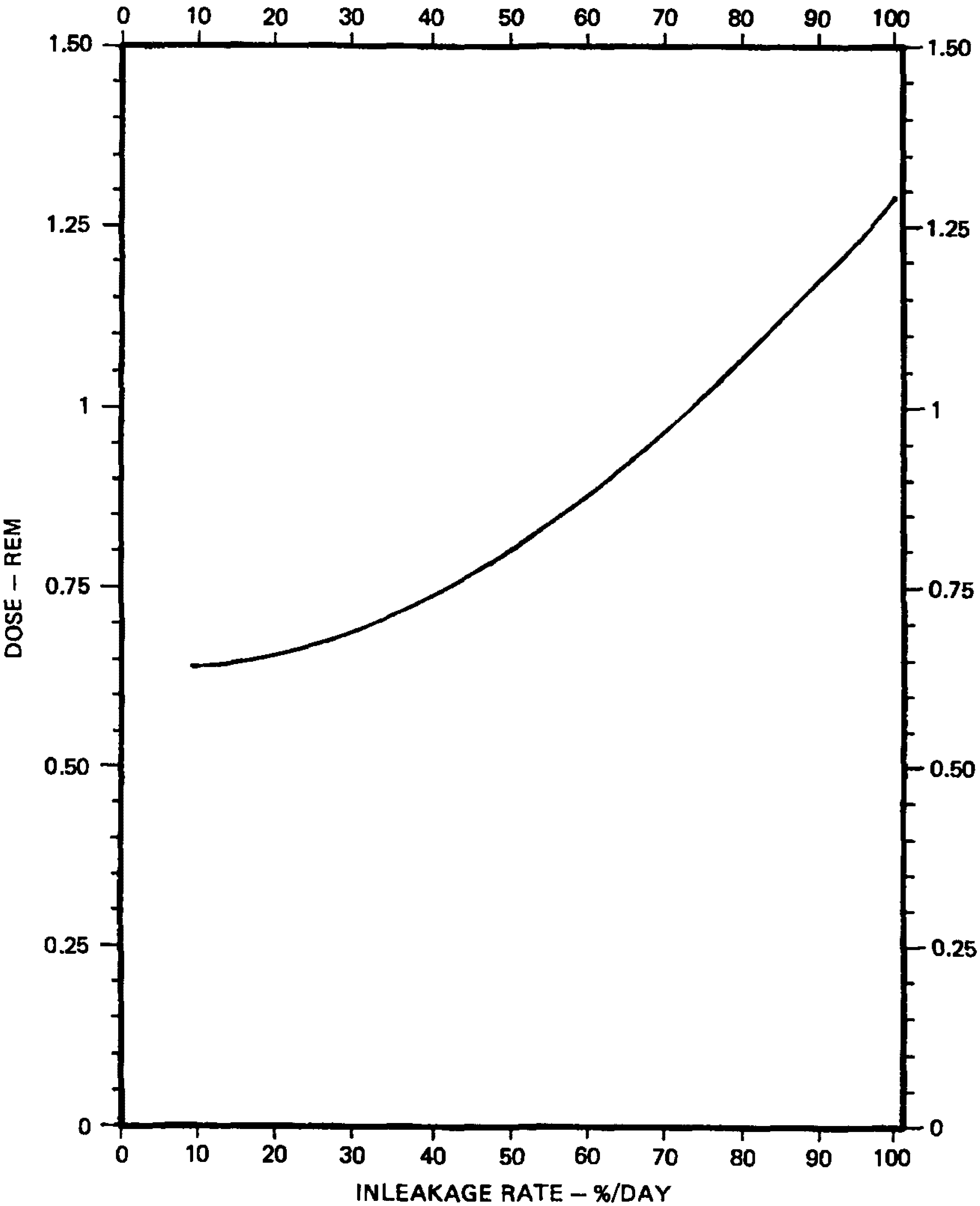
THIS FIGURE DISPLAYS HISTORICAL INFORMATION ONLY



Revision 12, May 3, 2002

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station DRAWDOWN TIME VERSUS CALCULATED INLEAKAGE RATE
	Updated FSAR Figure 15.6-8

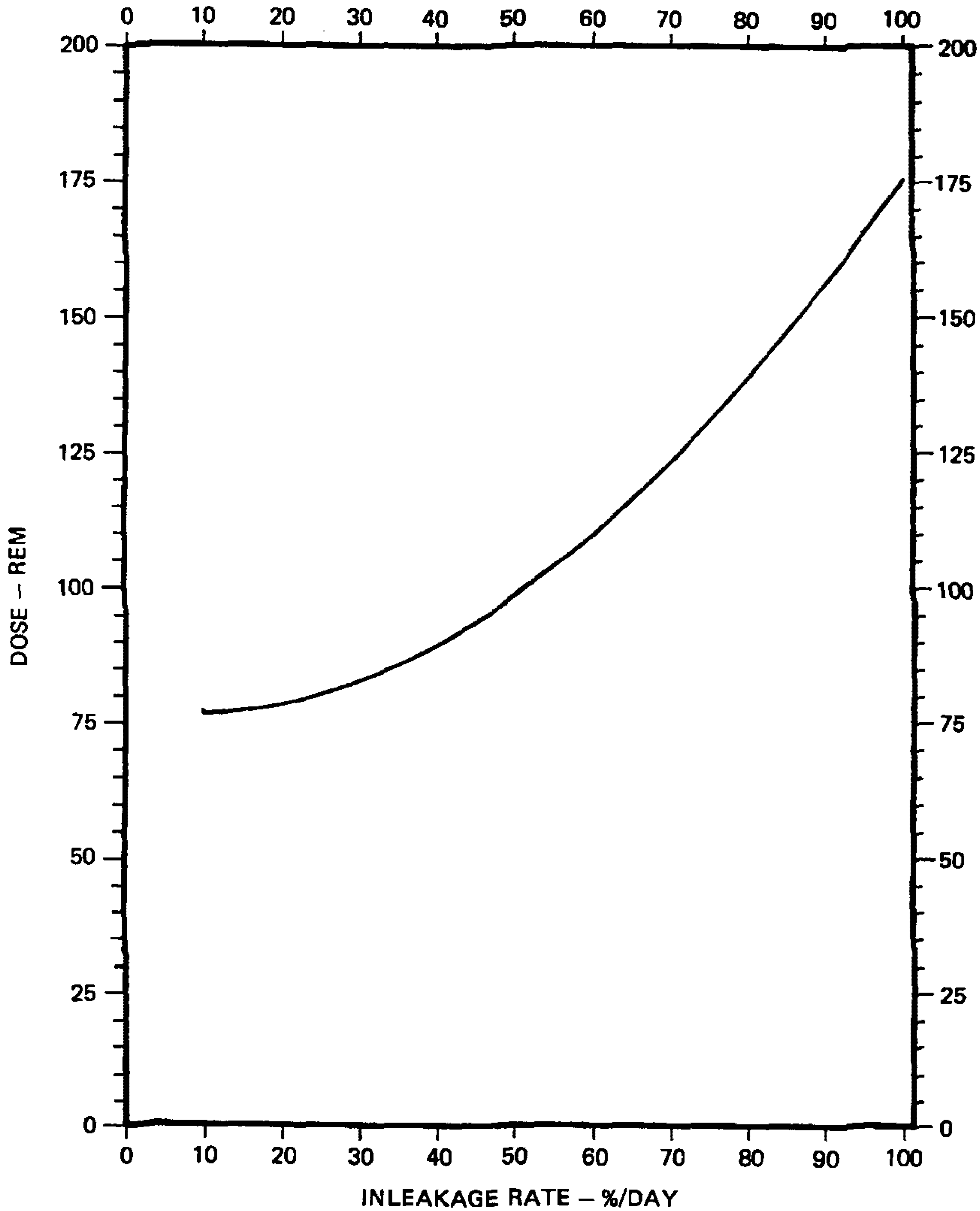
THIS FIGURE DISPLAYS HISTORICAL INFORMATION ONLY



Revision 12, May 3, 2002

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station SITE BOUNDARY WHOLE BODY DOSE (2 HOUR) VERSUS INLEAKAGE RATE
	Updated FSAR Figure 15.6-9

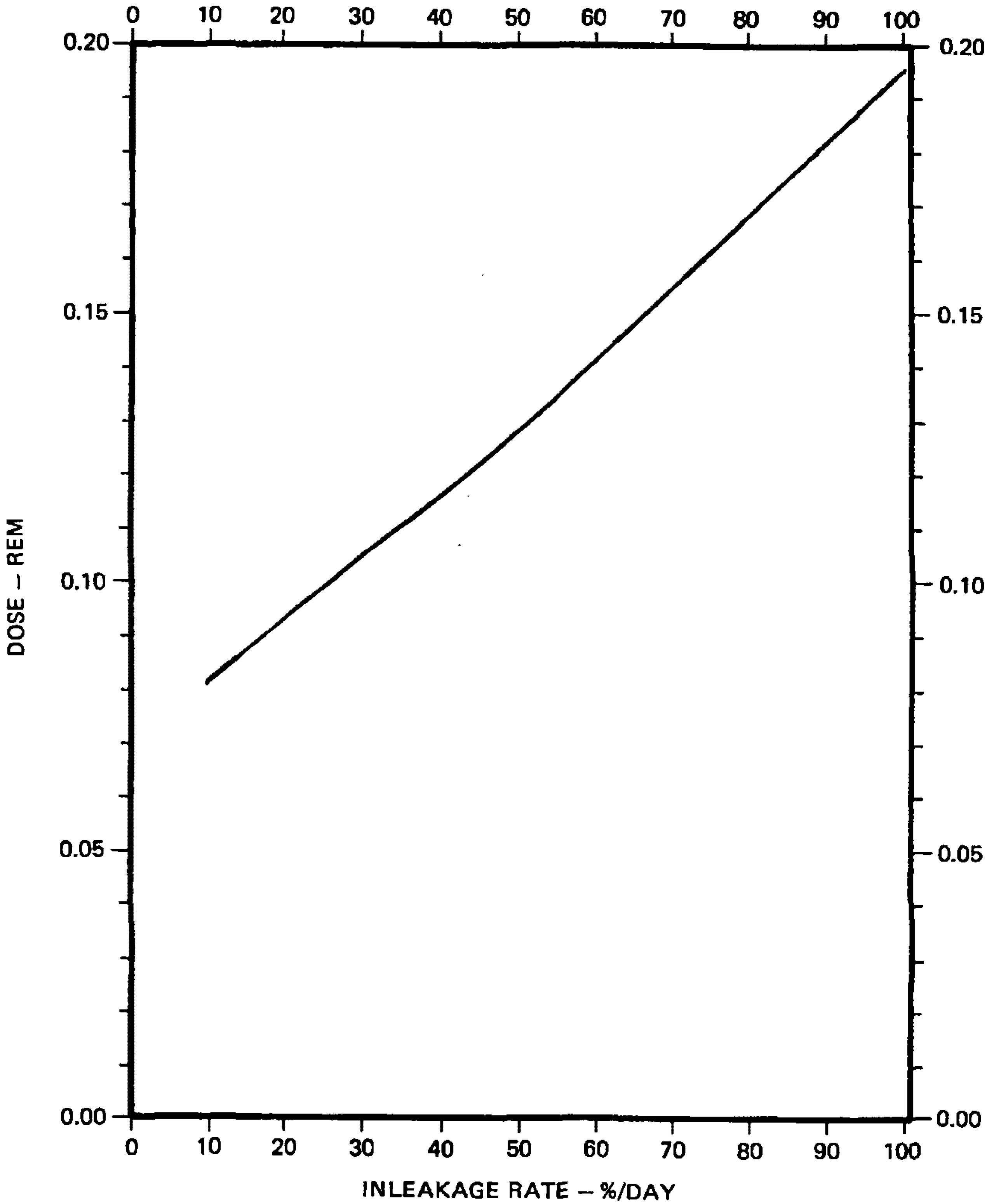
THIS FIGURE DISPLAYS HISTORICAL INFORMATION ONLY



Revision 12, May 3, 2002

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station SITE BOUNDARY THYROID DOSE (2 HOUR) VERSUS INLEAKAGE RATE
	Updated FSAR — Figure 15.6-10

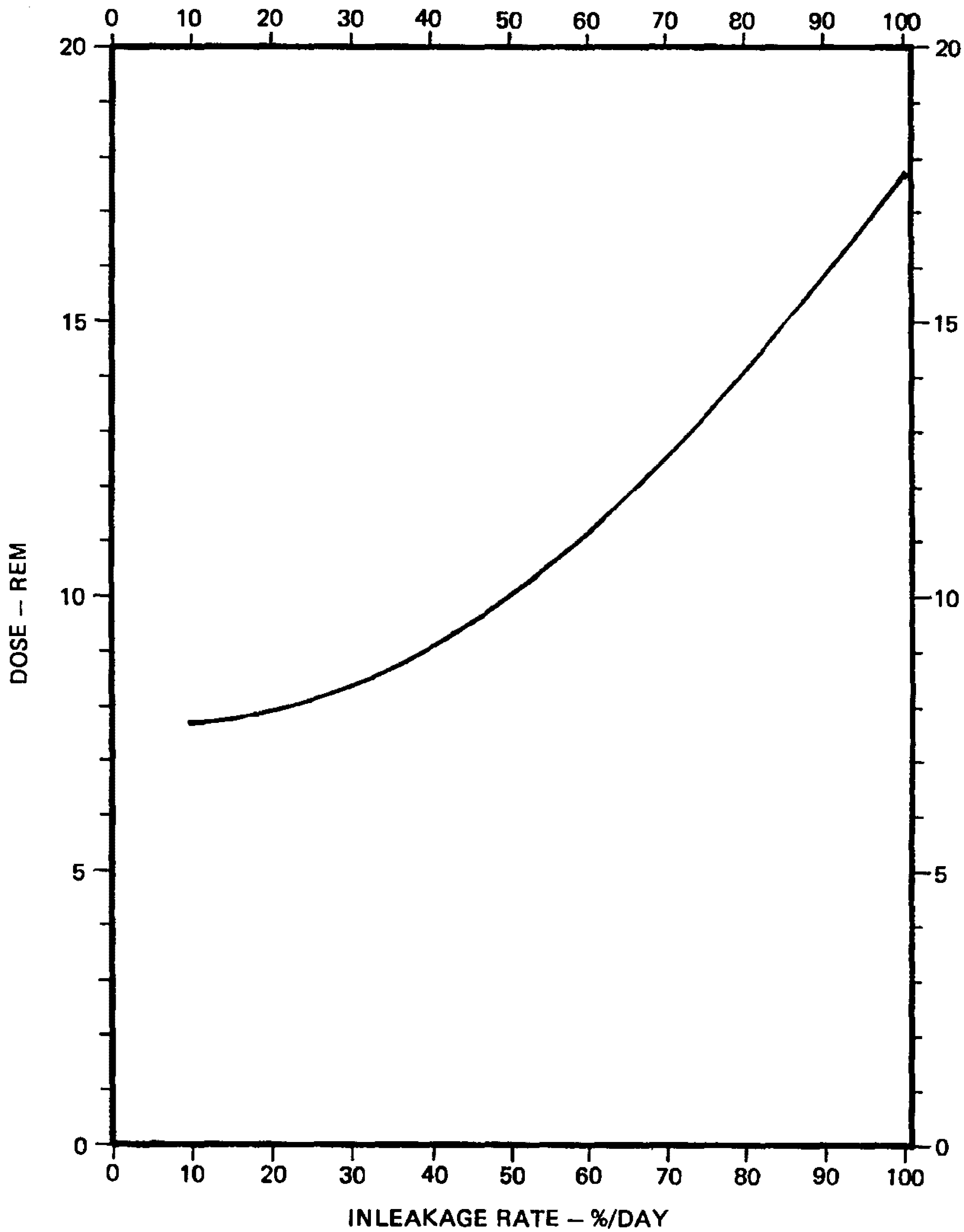
THIS FIGURE DISPLAYS HISTORICAL INFORMATION ONLY



Revision 12, May 3, 2002

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station LOW POPULATION ZONE WHOLE BODY DOSE (30 DAY) VERSUS INLEAKAGE RATE Updated FSAR — Figure 15.6-11

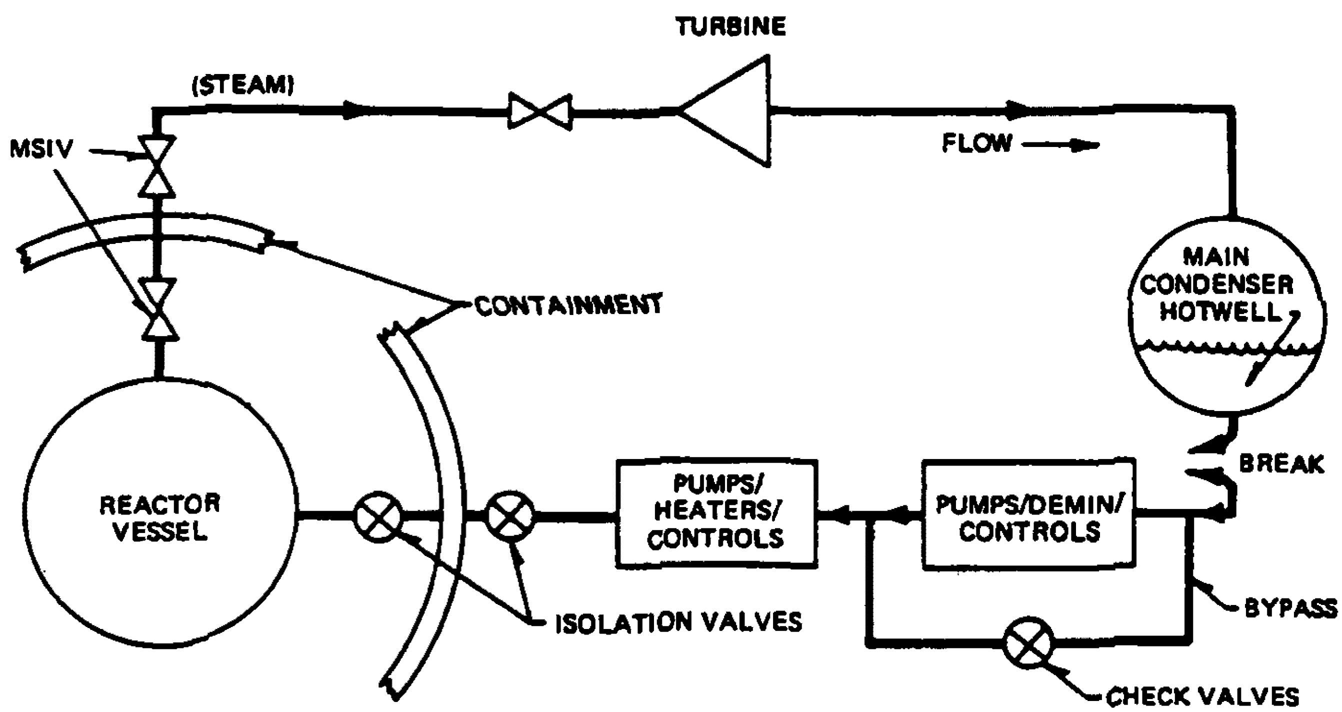
THIS FIGURE DISPLAYS HISTORICAL INFORMATION ONLY



Revision 12, May 3, 2002

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station LOW POPULATION ZONE THYROID DOSE (30 DAY) VERSUS INLEAKAGE RATE
	Updated FSAR — Figure 15.6-12

THIS FIGURE DISPLAYS HISTORICAL INFORMATION ONLY



Revision 12, May 3, 2002

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station LEAKAGE PATH FOR FEEDWATER LINE BREAK OUTSIDE CONTAINMENT
	Updated FSAR — Figure 15.6-13

15.7 RADIOACTIVE RELEASE FROM SUBSYSTEMS AND COMPONENTS

15.7.1 Gaseous Radwaste Subsystem Leak or Failure

SRP 15.7.1, Waste Gas System Failure, has been deleted. However, Branch Technical Position (BTP) ETSB 11-5, which is an attachment to SRP Section 11.3 (Gaseous Waste Management Systems), addresses the requirements for a postulated waste gas system failure.

For this class of accidents, the release of radioactive gases is limited by the design requirements for the systems, as specified in Section 11.3 and Regulatory Guide 1.143, and by the radiological effluent technical specifications (RETS) limits.

The systems and components addressed in this postulated accident are not impacted by cycle-to-cycle changes in the reactor core. Therefore, this event is not re-evaluated as a part of the standard reload licensing analysis process. (Reference 15.7-6)

15.7.1.1 Identification of Causes

An evaluation of events that could cause a gaseous radwaste system leak or failure indicates that a hydrogen explosion within the process boundary or a seismic event more serious than the system is designed to withstand could cause a gaseous radwaste system leak. These are events that could cause a gross system failure, such as a rupture of a tank or rupture of a line. Operator errors or system malfunctions resulting in a slower uncontrolled release of activity within the system are also feasible but are bounded by the consequences of the assumed gross system failure.

15.7.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.7.1.3 Sequence of Events

The sequence of events following this failure is shown in Table 15.7-1.

15.7.1.4 Identification of Operator Actions

A failure of an active component of the gaseous radwaste treatment system is assumed to occur. This event results in the activity normally processed by the waste gas system being released to the Turbine or Auxiliary Building, and subsequently, released through the ventilation system to the environment. For this event, the release is assumed to be to the Turbine Building.

The operator initiates a normal shutdown of the reactor to reduce gaseous activity being discharged and to isolate the waste gas system component. The operator initiates evacuation of the area, as needed. Radiation protection personnel will survey the evacuated area prior to reentry.

15.7.1.5 System Operation

In the gaseous radwaste leak or failure analysis, no credit is taken for the operation of plant and Reactor Protection Systems (RPSs), or of the engineered safety features (ESFs). Credit is, however, taken for the functioning of normally operating plant instruments and controls, i.e., gaseous release points to the atmosphere are monitored.

15.7.1.6 Effect of Single Failures and Operator Errors

After the initial system gross failure, the inability of the operator to isolate a system can affect the analysis. However, the seismic event that is assumed to occur beyond the present plant design basis for non-safety equipment causes a turbine trip that leads to a load rejection. This initiates a scram and negates a need for the operator to initiate a reactor shutdown via system isolation.

15.7.1.7 Core and System Performance

The system failure does not directly affect the reactor core or the Nuclear Steam Supply System (NSSS) safety performance.

15.7.1.8 Barrier Performance

The release of radioactive gases occurs outside the primary containment. Therefore, it does not involve any barrier integrity aspects.

15.7.1.9 Radiological Consequences

The instructions provided in BTP ETSB 11-5 were followed in this section. Specifically, the BWR-GALE code uses inputs specified in Table 11.2-1, except that the Krypton and Xenon dynamic adsorption coefficients used are 0.1193 cm^3 /g rather than their normal valves of 18.5 and 330.0, respectively. This results in Krypton and Xenon holdup times of 0.02 days, as specified in ETSB 11-5. The results from the GALE code are then used in manual calculations, as specified in ETSB 11-5. The X/Q value used is the 0-2 hour design basis site boundary value (1.9E-4 s/m^3). The releases to the environment are presented in Table 15.7-2. The result of the dose calculations is presented in Table 15.7-3.

15.7.2 Liquid Radwaste System Failure (Release to Atmosphere)

An analysis to show that the atmospheric release of activity from liquid radwaste tanks complies with 10CFR100 limits is no longer required.

15.7.3 Postulated Radioactive Releases Due to Liquid Radwaste Tank Failure

A detailed analysis of the effects of a liquid radwaste tank failure is not required. As indicated in Sections 11.2.1, 11.3.1, and 11.4.1; the liquid, gaseous, and solid waste management systems are

designed to meet the appropriate requirements of 10CFR50 (including General Design Criterion 60). As indicated in Sections 2.4.1 and 2.4.13, there are no potable water supplies that can be affected by a liquid release at HCGS. Therefore, the acceptance criteria of SRP 15.7.3, Rev. 2, July 1981 are satisfied and no detailed analysis is required.

15.7.4 Fuel Handling Accident

In the reload licensing methodology, the fuel handling accident is re-analyzed for each new fuel design. This event is not re-analyzed for a specific reload unless a modification is made to the fuel handling equipment that can increase the severity of the event.

The analysis presented in this section is representative of the original safety analysis of the HCGS. Section S.2.2.3.5 of GESTAR II (Reference 15.7-6) provides the number of fuel rod failures conservatively evaluated to occur for the fuel handling accident. The number of failures can be determined to be bounding for each fuel design.

15.7.4.1 Identification of Causes

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 assemblies. A variety of events that qualify for the class of accidents termed "fuel handling accidents" has been investigated. The accident that produces the largest number of ruptured spent fuel rods is the drop of a spent fuel assembly into the reactor core when the reactor vessel head is off.

15.7.4.2 Frequency Classification

This event is categorized as a limiting fault.

15.7.4.3 Sequence of Events

The most severe fuel handling accident from a radiological viewpoint is the dropping of a fuel assembly onto the top of the core. The sequence of events is given in Table 15.7-4.

15.7.4.4 Identification of Operator Actions

In the event of a fuel handling accident, no operator action is assumed in the accident analysis. Defense-in-depth measures require that contingency measures be in place to promptly close Secondary Containment openings within 30 minutes of a fuel handling accident to allow ventilation systems to draw the release in the proper direction such that it can be treated and monitored.

15.7.4.5 System Operation

In the event of a fuel handling accident, the accident analysis releases all radioactivity out the open equipment hatch and no ventilation system operation is assumed. Defense-in-depth measures require that prior to handling irradiated fuel the FRVS or Reactor Building Ventilation System (RBVS) is in operation drawing air into the building and exhausting through an operable radiation monitor.

15.7.4.6 Effects of Single Failures and Operator Errors

In the event of a fuel handling accident, no ventilation system operation or operator action is assumed in the accident analysis.

15.7.4.7 Core and System Performance

15.7.4.7.1 Mathematical Model

The analytical methods and associated assumptions used to evaluate the consequences of this accident are considered to provide a 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 1 foot-pound prior to cladding failure. Each rod that fails as a result of gross compression distortion is expected to absorb approximately 250 foot-pounds before cladding failure, based on 1 percent 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 68.6 percent fuel, 16.3 percent cladding, and 15.1 percent other structural materials 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 available for clad deformation is considered to be proportional to the mass ratio given as

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

which, given the weight ratios above, equals 0.519.

5.7.4.7.2 Input Parameters and Initial Conditions

The assumptions used in the analysis of this accident are listed below:

1. The fuel assembly is dropped from a height of 32.95 feet. However, the maximum height allowed by the fuel handling equipment is 32 feet 3.45 inches, which is conservative with regard to the assumption of 32.95 feet.

2. The entire amount of potential energy referenced to the top of the reactor core is available for application to the fuel assemblies involved in the accident. This assumption neglects the dissipation of some of the mechanical energy of the falling fuel assembly in the water above the core.

3. None of the energy associated with the dropped fuel assembly is absorbed by the fuel material, uranium dioxide.

4. The grapple head and three sections of the telescoping mast remain attached to the dropped assembly.

5. The energy of the entire assembly/mast system falling to its side from the vertical position (second impact) is included in the energy available to damage fuel rods.

15.7.4.7.3 Results

15.7.4.7.3.1 Energy Available

Dropping a fuel assembly onto the reactor core from the assumed height of 32.95 feet results in the assembly acquiring, for the first impact, an amount of kinetic energy equivalent to 31,870 foot-pounds. When the assembly falls to its side from the vertical position, the amount of kinetic energy acquired is equal to 8780 foot-pounds.

15.7.4.7.3.2 Energy Loss Per Impact

Based on the fuel geometry in the reactor core, four fuel assemblies are struck by the impacting assembly on the first impact.

The second impact is expected to be less direct. The broad side of the dropped assembly impacts approximately 24 more fuel assemblies. Because the total energy to cause damage to the clad is independent of the number of impacts, multiple impacts do not need to be

considered. Two impacts were considered, because the second will occur at a location different from the first.

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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.

15.7.4.7.3.3 First Impact Fuel Rod Failures

The first impact dissipates 31,870 foot-pounds of energy. It is assumed that 50 percent of this energy is absorbed by the dropped fuel assembly, and that the remaining 50 percent is absorbed by the struck fuel assemblies in the core. Because the fuel rods of the dropped fuel assembly are susceptible to the bending mode of failure, and because 1 foot-pound 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.

The fuel rods of the struck assemblies are assumed to fail by a 1-percent strain in compression. To cause cladding failure of one fuel rod as a result of compression, 250 foot-pounds of energy are required. The energy available for clad deformation was considered proportional to the mass ratio, which is equal to a maximum of 0.519 for this analysis.

The number of fuel rod failures in the four impacted assemblies caused by compression is computed as follows:

$$\frac{0.5 \times 31{,}870 \times 0.519}{250} = 33$$

Thus, during the first impact, the total number of fuel rod failures is 62 rods (bending) for the dropped bundle and 33 rods (compression) for struck bundles, resulting in a total of 95 failed rods.

15.7.4.7.3.4 Second Impact Fuel Rod Failures

The dropped assembly was assumed to tip over and impact horizontally on the top of the core. The remaining available energy was used to predict the number of additional rod failures. The available energy was calculated by assuming a linear weight distribution in the assembly, with a point load at the top of the assembly to represent the fuel grapple weight. The energy available was found to equal 8780 foot-pounds.

The number of fuel rod failures caused by compression on the second impact is computed as follows:

$$\frac{0.5 \times 8780 \times 0.519}{250} = 9$$

Thus, during the second impact, a total of nine fuel rod failures occur.

15.7.4.7.3.5 Total Fuel Rod Failures as a Result of Impacts

For the first impact, 95 rods fail, and for the second impact, 9 rods fail, resulting in 104 total failed rods in the event of a fuel handling accident.

15.7.4.8 Barrier Performance

The reactor coolant pressure boundary (RCPB) and primary containment are assumed to be open. The transport of fission products from the reactor building is discussed in Sections 15.7.4.9.1 and 15.7.4.9.2.

15.7.4.9 Radiological Consequences

A radiological analysis is provided for this accident:

The fission product inventory in the fuel rods that are assumed to be damaged is based on reactor core operation at 4031 MWt and assumed irradiation time periods that allow for radionuclides to reach equilibrium or maximum values. A 24-hour period for decay from the above power condition is assumed, because it is not expected that fuel handling can begin within 24 hours following initiation of reactor shutdown.

The methods, assumptions, and conditions used to evaluate this accident are in accordance with those guidelines set forth in Regulatory Guides 1.183, Appendix B, Rev. 0 (Ref. 15.7-7). The specific models and assumptions used to evaluate this event, based on the above criteria, are presented in Appendix 15A.

Specific values of parameters used in this evaluation are presented in Table 15.7-5.

15.7.4.9.1 Fission Product Release from Fuel

The number of damaged rods is calculated to be 104. An earlier, more conservative analysis assumed the number of failures to be 124, and this value is used in the radiological analysis. A peaking factor of 1.75 is applied to the average rod fission product inventories of the damaged rods. It is assumed that 8 percent of the I-131, 10 percent of the Kr-85, 5 percent of the other noble gases and halogen inventories, and 12 percent of the alkali metal inventory of the damaged rods are released from the rods.

15.7.4.9.2 Fission Product Transport to the Environment

An overall pool decontamination factor of 200 is such that one-half percent of the iodine released from the damaged rods becomes airborne above the fuel pool water surface, and is made up of 57 percent elemental and 43 percent organic iodines. All of the noble gases released from the damaged fuel become airborne above the fuel pool. None of the alkali metals released from the damaged fuel become airborne above the fuel pool.

It is assumed that the activity above the fuel pool is exhausted to the environment over a two hour period. No decay is assumed after the activity is released from the fuel. No filtration of the activity released is credited.

15.7.4.9.3 Radiological Results

Dose conversion factors for iodine are taken from Federal Guidance Report (FGR) 11 (Ref. 15.7-8), and breathing rates during the accident are taken from Regulatory Guide 1.183. The whole body dose is calculated using the dose conversion factors for the semi-infinite cloud model discussed in FGR 12 (Ref. 15.7-9). The calculated doses for the design basis analysis are presented in Table 15.7-8. The licensing basis Fuel Handling Accident radiological consequences are not impacted by the introduction of 12 GE14i assemblies at HCGS (Reference 15.7-1).

15.7.5 Spent Fuel Cask Drop Accident

The spent fuel cask is equipped with redundant sets of lifting lugs and yokes compatible with the single failure proof Reactor Building crane thus preventing a cask drop due to a single failure. Therefore, the analysis of the spent fuel cask drop is not performed. Refer to Section 9.1.4.2.2 for a description of the Reactor Building crane and the interlocks that prevent moving the spent fuel cask over the fuel pool.

15.7.6 References

15.7-1 NRC letter to PSEG Nuclear dated October 7, 2010, "Hope Creek Generating Station - Issuance of Amendment 184 Re: Use of Isotopic Test Assemblies For Cobalt-60 Production (TAC No. ME2949)" Adams Accession No. ML102700263).

15.7-2 Deleted

15.7-3 Deleted

15.7-4 Deleted

15.7-5 U.S. Nuclear Regulatory Commission, "Branch Technical Position ETSB 11-5," Rev. 0, Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure (NUREG-0800, Revision 2), July 1981.

15.7-6 "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (latest approved revision), and "General Electric Standard Application for Reactor Fuel (Supplement for United States)", NEDE-24011-P-A-US (latest approved revision).

15.7-7 U.S. NRC Regulatory Guide 1.183, Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors, July 2000.

15.7-8 Federal Guidance Report 11, EPA-520/1-88-020, September 1988, Limiting Values Of Radionuclide Intake And Air Concentration And Dose Conversion Factors For Inhaled, Submersion, And Ingestion.

15.7-9 Federal Guidance Report 12, EPA-402-R-93-081, September 1993, External Exposure To Radionuclides In Air, Water, And Soil.

TABLE 15.7-1

SEQUENCE OF EVENTS FOR
OFF-GAS TREATMENT SYSTEM FAILURE

Time, s	Event
0	Event begins, system fails
0	Noble gases are released
<60	Ventilation exhaust radiation alarms alert plant personnel
≥0	Operator actions begin with initiation of appropriate system isolations, manual scram initiation, and assurance of reactor shutdown cooling.

TABLE 15.7-2

OFF-GAS TREATMENT SYSTEM FAILURE (DESIGN BASIS ANALYSIS)
ACTIVITY RELEASED TO ENVIRONS

Isotope	Activity, Ci
Kr-83m	2.2E6(1)
Kr-85m	4.5E5
Kr-85	1.8E3
Kr-87	1.2E6
Kr-88	1.5E6
Kr-89	2.2E3
Xe-131m	1.2E3
Xe-133m	2.3E4
Xe-133	6.5E5
Xe-135m	3.9E5
Xe-135	1.8E6
Xe-137	1.1E4
Xe-138	1.5E6

(1) 2.2E6 = 2.2×10^6

TABLE 15.7-3

OFF-GAS TREATMENT SYSTEM FAILURE - RADIOLOGICAL EFFECTS

	Whole Body, rem
Site boundary (901 meters, 2-hour dose)	6.56E-2

TABLE 15.7-4

SEQUENCE OF EVENTS FOR FUEL HANDLING ACCIDENT

Time, min	Event
0	Fuel bundle is being handled by refueling equipment. The bundle drops into the top of the core.
0	Some of the fuel rods in both the dropped bundle and reactor core are damaged, resulting in the release of gaseous fission products to the reactor coolant and eventually to the Reactor Building atmosphere.
≤ 1	The Reactor Building Ventilation or FRVS Radiation Monitoring System alarms to alert plant personnel.
≤ 5	Operator actions begin.
≤ 30	Defense-in-depth measures promptly close Secondary Containment openings to allow ventilation systems to draw the release in the proper direction such that it can be treated and monitored.
≤ 120	All radioactivity is assumed to be released through the open equipment hatch.

TABLE 15.7-5

FUEL HANDLING ACCIDENT - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSES

		Design Basis Assumptions
1.	Data and Assumptions Used to Estimate Radioactive Source from Postulated Accidents	
	a. Power level, MWt	4031
	b. Radial peaking factor	1.75
	c. Number of fuel rods damaged	124
	d. Release of activity from damaged fuel by nuclide, percent	
	(1) Noble gas	5
	(2) Kr-85	10
	(3) I-131	8
	(4) Other halogens	5
	(5) Alkali metals	12
e.	Fractions released to the environment	
	(1) Organic iodine	0.43
	(2) Elemental iodine	0.57
	(3) Particulate iodine	0
	(4) Noble gases	1
	(5) Alkali metals	0
	f. Reactor coolant activity before accident	NA

	Design Basis Assumptions
g. Total number of fuel rods in core	47,368
2. Data and Assumptions Used to Estimate Activity Released	
a. Primary containment leak rate, percent/day	NA
b. Reactor building release rate, cfm	1.535E+05
c. Valve movement times	NA
d. Exhaust system parameters	
(1) Flow rate, cfm	NA
(2) Filter efficiency, percent	
i. Organic iodine	0
ii. Elemental iodine	0
iii. Particulate iodine	0
iv. Particulate fission products	NA
e. Recirculation system parameters	

	Design Basis Assumptions
(1 Flow rate, cfm	NA
(2) Mixing fraction	0
(3) Filter efficiency, percent	-
f. Containment spray parameters (flow rate, drop size, etc)	NA
g. Containment volumes, ft^3	
(1) Reactor building	4.00E+06
(2) Fuel pool water volume	NA
(3) Free air volume above fuel pool	NA
h. All other pertinent data and assumptions	None
3. Dispersion Data (X/Q's calculated using methodology of Section 2.3.4.2.1)	
a. Exclusion Area boundary (EAB)/ low population zone (LPZ) distance, m	901/8047

	Design Basis Assumptions
b. X/Q's (s/m^3) for time intervals of:	
(1) 0-2 h - EAB/LPZ	1.9E-4/1.9E-5
(2) 2-4 h - LPZ	NA
(3) 4-8 h - LPZ	NA
(4) 8-24 h - LPZ	NA
(5) 1-4 days - LPZ	NA
(6) 4-30 days - LPZ	NA

4. Dose Data

a. Method of dose calculation	Appendix 15A
b. Dose conversion assumptions	Appendix 15A
c. Doses	Table 15.7-8

TABLE 15.7-6

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TABLE 15.7-7

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TABLE 15.7-8

FUEL HANDLING ACCIDENT

RADIOLOGICAL EFFECTS[1]

Exclusion Area boundary (2-hour dose)	5.33E-01 rem TEDE
Low population zone (2-hour dose)	5.33E-02 rem TEDE

1. The above results of the radiological consequence evaluations are not impacted by the introduction of 12 GE14i assemblies at HCGS.

TABLE 15.7-9

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TABLE 15.7-10

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TABLE 15.7-11

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TABLE 15.7-12

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HOPE CREEK GENERATING STATION

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15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

An ATWS evaluation is performed for each plant modification that has the potential to challenge the ATWS event acceptance criteria. For GNF fuel, Section 1 of GESTAR II (Reference 15.8-4) contains NRC-approved criteria for evaluating each new fuel design for ATWS. The GNF fuel contained in HCGS has been evaluated on a generic basis in References 15.8-5 and 15.8-9 and determined to meet the GESTAR ATWS new fuel criteria.

The ATWS has been analyzed for HCGS full cores of GE14 and GNF2 fuel. (References 15.8-6, 15.8-7, and 15.8-8)

The following is a generic description of how the Hope Creek Generating Station complies with the ATWS rules.

15.8.1 Requirements

The issue of postulated failure to scram the reactor following an anticipated transient, i.e., an anticipated transient without scram (ATWS), has been under consideration by the NRC. As a result of its assessment, the NRC has required the recirculation pump trip (RPT) feature for the boiling water reactor (BWR).

It should be noted that the NRC has determined that the probability of an ATWS event is acceptably small, and that any additional plant modifications for ATWS need not satisfy the requirements for a design basis accident.

The HCGS emergency operating procedures will be developed from the BWR Owners' Group Generic Emergency Procedure Guidelines. ATWS events are covered in these guidelines. The Hope Creek Generating Station Emergency Operating Procedure, HC.OP-EO.ZZ-101A, ATWS-RPV Control, contains the necessary actions to be taken during an ATWS event. Training programs for reactor operators, senior reactor operators, and shift technical advisers will incorporate the bases and philosophy of the GE/BWR Owners' Group generic emergency operating procedures until such time as the HCGS emergency operating procedures are developed.

15.8.2 Plant Capabilities

The Hope Creek Generating Station (HCGS) design uses diverse, highly redundant, and very reliable scram systems. This includes the normal scram systems, plus the electrically diverse Alternate Rod Insertion (ARI) System. Each of these systems is frequently tested and would insert the control rods even if multiple component failures should occur, thus making the probability of an anticipated transient without scram (ATWS) event extremely remote.

The ATWS recirculation pump trip (RPT) feature prevents reactor vessel overpressure and possible short term fuel damage for the most limiting postulated ATWS event. Subsequent to an ATWS event for which the ARI system does not insert the control rods, the long term shutdown of the reactor can be accomplished by either manual insertion of the control rods, or simultaneous two pump injection of sodium pentaborate solution into the vessel.

PSE&G has voluntarily committed to incorporate in the Hope Creek plant the features described in Section 15.8.3, in order to bound possible future NRC requirements for ATWS. Both PSE&G and GE view these added features as providing acceptable resolution to the ATWS issue.

15.8.3 Equipment Description

This section describes the equipment and control logic added or modified exclusively for ATWS prevention or mitigation. The description covers design and functional requirements and references that contain more detailed information.

15.8.3.1 Redundant Reactivity Control System

The Redundant Reactivity Control System (RRCS) determines that a transient is underway that exceeds expected operating parameters. After deciding that ATWS mitigation is the appropriate action, the RRCS activates ATWS prevention equipment. The RRCS uses transient detection sensors for high vessel dome pressure and low vessel water level, and the actuation logic to initiate ARI, RPT, the Standby Liquid Control (SLC) System, and feedwater runback.

The RRCS consists of two completely redundant divisions. Each division is initiated automatically by the ATWS detection sensors, which are independent of the Reactor Protection System (RPS)

sensors, or manually by switches that require similar type of operator actions as manual scram. The RRCS logic uses APRM signals (not downscale) following a time delay as confirmation of an ATWS event before permitting automatic actuation of the SLC system or feedwater runback.

Additional information on the RRCS is contained in Sections 7.1 and 7.6.

15.8.3.2 Alternate Rod Insertion

The purpose of the ARI function is to blow down the scram discharge air header through valves separate from RPS scram valves, thereby providing a parallel path for control rod insertion. ARI consists of the redundant scram air header valves that are actuated by the ATWS detection sensors. The RRCS logic is designed so that successful ARI performance will avoid subsequent ATWS mitigation action (SLC system initiation and feedwater runback).

Additional information on the ARI system is contained in Sections 7.1 and 7.6.

15.8.3.3 Recirculation Pump Trip

The recirculation pump motors are tripped by the RRCS logic. The purpose of the RPT is to reduce core flow and create core voids to decrease power generation thus limiting any power or pressure disturbance. The RPT function is single failure proof and is provided with inservice test capability, except for the action of the final breakers.

Additional information on the RPT function of the RRCS is contained in Sections 7.1 and 7.6.

15.8.3.4 Feedwater Runback

Upon the receipt of a high pressure signal from the RRCS (not low water level), including confirmation of no scram, feedwater flow is limited, thereby reducing power and steam discharge to the suppression pool. The system provides for manual operation override to allow an increase in feedwater flow, if needed.

Additional information on the feedwater runback function of the RRCS is contained in Sections 7.1 and 7.6.

15.8.3.5 Standby Liquid Control System

The Standby Liquid Control (SLC) System is initiated automatically by the RRCS logic when needed; it can also be initiated by an operator in the main control room in accordance with plant operating procedures. The system is designed to inject sodium pentaborate solution through a core spray sparger. Simultaneous operation of two pumps at full capacity allows adequate margin to bring the reactor to a subcritical state. The system can be periodically tested without affecting its ability to respond to an actuation signal.

Additional information on the SLC system is provided in Sections 3.9, 7.1, 7.6, 9.3.5, and 15.8.3.1.

15.8.3.6 Scram Discharge Volume

The scram discharge volume of the Control Rod Drive (CRD) System minimizes the potential for a common mode failure of the scram function. Redundant instrument volume water level sensors for the CRDs and instrument line piping ensure the availability of sufficient capacity to receive water from a full reactor scram. The design employs redundant Class 1E sensors and redundant vent and drain valves. Performance of the safety functions is ensured in the event of a single active failure or the bypass of the sensors during plant operation.

Additional information on the scram discharge volume is contained in Section 4.6. Instrumentation is described in Section 7.

15.8.4 SRP Rule Review

15.8.4.1 SRP 15.8, Acceptance Criterion II.a.

In SRP 15.8, acceptance criterion II.a. requires that GDC 10 be applied to the anticipated transient without scram (ATWS) event.

GDC 10 is not applied, as the postulated ATWS event is so remote that it is outside the frequency classification range for design basis accidents for which GDC 10 applies. This justificatin results from an NRC meeting on June 16, 1981, for formalizing NRC recommendations on ATWS mitigation.

15.8.4.2 SRP 15.8, Acceptance Criterion II.b.

In SRP 15.8, acceptance criterion II.b. requires that GDC 15 be applied to the ATWS event.

GDC 15 is not applied, as the postulated ATWS event is so remote that it is outside the frequency classification range for design basis accidents for which GDC 15 applies. This justification results from an NRC meeting June 16, 1981, for formalizing NRC recommendations on ATWS mitigation.

15.8.4.3 SRP 15.8, Acceptance Criterion II.c.

In SRP 15.8, acceptance criterion II.c. requires that GDC 27 be applied to the ATWS event.

GDC 26 is not applied, as the postulated ATWS event is so remote that it is outside the frequency classification range for design basis accidents for which GDC 26 applies. This justification results from an NRC meeting June 16, 1981, for formalizing NRC recommendations on ATWS mitigation.

15.8.4.4 SRP 15.8, Acceptance Criterion II.d.

In SRP 15.8, acceptance criterion II.d. requires that GDC 27 be applied to the ATWS event.

GDC 27 is not applied, as the postulated ATWS event is so remote that it is outside the frequency classification range for design basis accidents for which GDC 27 applies. This justification results from an NRC meeting June 16, 1981, for formalizing NRC recommendations on ATWS mitigation.

15.8.4.5 SRP 15.8, Acceptance Criterion II.e.

In SRP 15.8, acceptance criterion II.e. requires that GDC 29 be applied to the ATWS event.

GDC 29 is not applied, as the postulated ATWS event is so remote that it is outside the frequency classification range for design basis accidents for which GDC 29 applies. This justification results from an NRC meeting June 16, 1981, for formalizing NRC recommendations on ATWS mitigation.

15.8.4.6 SRP 15.8, Acceptance Criterion II.f.

In SRP Section 15.8, acceptance criterion II.f. states that the BWR recirculation pump trips (RPT) is acceptable if it meets the criteria provided in Section IV-4 of Volume 2 of Reference 15.8-1.

Following publication of Volume 2 of Reference 15.8-1, the NRC developed a set of design criteria to determine RPT acceptability. These criteria, which are essntially the same as the criteria a. through z. of Section IV-4 in Volume 2 of Reference 15.8-1, were used for evaluating the Monticello and Hatch RPT designs; and the NRC determined the designs to be acceptable. The NRC has not applied criterion j. of Section IV-4 in Volume 2 of Reference 15.8-1.

15.8.5 References

15.8-1 Office of Nuclear Reactor Regulation, "Anticipated Transients Without Scream for Light Water Reactors," NUREG-0460, Volumes 1-4, U.S. Nuclear Regulatory Commission.

15.8-2 Deleted

15.8-3 Deleted

15.8-4 General Electric Company, "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, (latest approved version), and "General Electric Standard Application for Reactor Fuel (Supplement for United States)," NEDE-24011-P-A-US, (latest approved version).

15.8-5 General Electric Company, "GE14 Compliance With Amendment 22 of NEDE-24011-P-A (GESTAR II)," NEDC-32868P Revision 1, September 2000.

15.8-6 "Fuel Transition Report For Hope Creek Generating Station," NEDC-33158P, Supplement 1, Revision 1, April 2005

15.8-7 NEDC-33066P, Revision 2, "Hope Creek Generating Station, APRM/RBM/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA)," [Feb. 2005]

15.8-8 GE Hitachi Nuclear Energy, "GNF2 Fuel Design Cycle-independent Analyses for Hope Creek Generating Station", 002N9723, Revision 0, September 2016.

15.8-9 Global Nuclear Fuel, "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)," NEDC-33270P, Revision 6, March 2016.

15.9 PLANT NUCLEAR SAFETY OPERATIONAL ANALYSIS (A SYSTEM LEVEL/QUALITATIVE PLANT FAILURE MODES AND EFFECTS ANALYSIS)

The HCGS is designed to accommodate limiting transients and postulated safety system failures, as discussed in Sections 15.1 through 15.8. The nuclear safety operational analysis (NSOA) is intended to provide a further assurance that these limiting transients remain binding for all possible transients when the system level failure modes and effects are evaluated.

In this section, the NSOA systematically presents the philosophy behind the formulation of these limiting transients. This analysis is the result of generic developments encompassing the design, calculation, testing, and operating experience of the BWR 4,5, and 6 product lines. Broad examination of these system designs assures that HCGS's design is enveloped.

15.9.1 Objectives

15.9.1.1 Essential Protective Sequences

An objective of the nuclear safety operational analysis (NSOA) is to identify and demonstrate that the essential protection sequences needed to accommodate normal plant operations, anticipated and abnormal operational transients, and design basis accidents (DBAs) are available and adequate. Each event considered in this chapter is further examined and analyzed. Specific essential protective sequences are identified. The appropriate sequence is discussed for all boiling water reactor (BWR) operating modes.

15.9.1.2 Design Basis Adequacy

An objective of the NSOA is to identify and demonstrate that the safety design basis of the various structures, systems, or components needed to satisfy the plant essential protection sequences are appropriate, available, and adequate. Each

protective sequence identifies the specific structures, systems, or components performing safety or power generation functions. Interrelationships between primary systems and secondary (or auxiliary) equipment in providing these functions are shown. The individual design bases, identified throughout the FSAR for each structure, system, or component are brought together by the analysis in this section. In addition to the individual equipment design bases, the plant wide design bases are examined and presented here.

15.9.1.3 Qualitative Failure Modes and Effects Analysis

An objective of the NSOA is to identify a system level/qualitative failure modes and effects analysis (FMEA) of essential protective sequences to show compliance with the single active failure (SAF) or single operator error (SOE) criteria. Each protective sequence entry is evaluated relative to SAF or SOE criteria. Safety classification aspects and interrelationships between systems are also considered.

15.9.1.4 NSOA Criteria Relative to Plant Safety Anaysis

An objective of the NSOA is to identify the systems, equipment, or components' operational conditions and requirements that are essential to satisfy the nuclear safety operational criteria discussed in this section.

15.9.1.5 Technical Specification Operational Basis

An objective of the NSOA is to establish limiting operating conditions, testing, and surveillance bases relative to plant technical specification operational requirements.

15.9.2 Approach of Nuclear Safety Operational Analysis

15.9.2.1 Evaluation of Consequences

The nuclear safety operational analysis (NSOA) is an evaluation of "event consequence" of different classifications. Refer to Figure 15.9-1 for a description of the systematic process by which these consequences are converted so that they conform to safety requirements.

15.9.2.2 NSOA Development

15.9.2.2.1 Scope and Classification Of Plant Events

15.9.2.2.1.1 Normal Operations

Normal operations are those under planned conditions without significant abnormalities. Operations subsequent to an incident, e.g., transient, accident, or special event, are not considered planned unless the procedures followed, or equipment used, are identical to those used during any one of the defined planned operations. Normal operations are listed in Table 15.9-1 as Events 1 to 6, and are further defined in Section 15.9.6.2.2.

15.9.2.2.1.2 Anticipated Operational Transients

Anticipated operational transients are deviations from normal conditions that are expected to occur at a moderate frequency. As such, the design includes the capability to withstand the conditions without operational impairment. Included are incidents that result from a single operator error (SOE), i.e., the set of actions that is a direct consequence of a single erroneous decision of the operator. These incidents are listed in Table 15.9-2 as Events 7 to 29 and are further defined in Section 15.9.6.3.3.

15.9.2.2.1.3 Abnormal Operational Transients

Abnormal operational transients are deviations from normal conditions that occur infrequently. The design includes the capability to withstand these conditions without operational impairment. Refer to Section 15.9.6.4.3 and Events 30 to 39 in Table 15.9-3 for detailed definitions.

15.9.2.2.1.4 Design Basis Accident

A design basis accident (DBA) is a hypothetical event whose characteristics and consequences are used in the design of those systems and components essential to the integrity of radioactive material barriers. The potential radioactive release resulting from a DBA is greater than from any other postulated event. DBAs are listed in Table 15.9-4 as Events 40 to 49 and are further defined in Section 15.9.6.5.3.

15.9.2.2.1.5 Special Events

Special events are postulated to demonstrate some special capability of the plant, in accordance with NRC requirements. These events are listed in Table 15.9-5 and further defined in Section 15.9.6.6.3.

15.9.2.2.2 Safety and Power Generation

15.9.2.2.2.1 Safety

The regulatory safety requirements include:

1. Accommodation of abnormal operational transients and postulated DBAs

2. Maintenance of containment integrity

3. Assurance of Emergency Core Cooling Systems (ECCS) operation

4. Maintenance of reactor coolant pressure boundary (RCPB) integrity.

These requirements are associated with 10CFR50.67 dose limits, infrequent and remote probability occurrences, single active failure (SAF) criteria, worst case operating conditions and initial assumptions, automatic corrective action, dose and environmental effects, and other coincidental (mechanistic or non-mechanistic) plant and environmental situations.

15.9.2.2.2.2 Power Generation

The regulatory requirements for power generation include:

1. Accommodation of planned operations and anticipated operational transients.

2. Minimization of radiological releases to appropriate levels.

3. Assurance of safe and orderly reactor shutdown and/or return to normal operation.

4. Maintenance of plant equipment design conditions to ensure long-term reliable operation.

These requirements are governed by 10CFR20, 10CFR50, Appendix I, moderate and high probability occurrences, normal operating conditions and initial assumptions, allowable immediate operator and manual actions, and environmental effects.

15.9.2.2.3 Frequency of Events

The events in this section are classified per initiating frequency occurrence. The consideration of additional failures, or operator errors, necessitates changing the classification to a lower frequency category.

The introduction of SAFs or SOEs into the examination of planned operation, anticipated operational transients, or abnormal operational transient evaluations has not been previously considered a design basis or evaluation prerequisite. It is included here to demonstrate the plant's capability to accommodate this new requirement.

15.9.2.2.4 Design Margins

Adequate design margins are included such that the consequences established in this section relative to public safety are in strict conformance to regulatory requirements.

15.9.2.2.5 Safety Function Definition

The definition of safety function includes the following:

1. The essential protective sequences shown for an event in this section list the minimum structures and systems that are required to satisfy the SAF or SOE evaluation. Other protective "success paths" also exist.

2. Not all the events involve the same assumptions. For example, a loss-of-coolant accident (LOCA) and a safe shutdown earthquake (SSE) are associated with event 42. In event 40, control rod drop accident (CRDA) is not assumed to be associated with an SSE or operating basis earthquake (OBE). Therefore, seismic requirements are not considered for event 40. The equipment essential to

safety associated with event 40 protective sequences is also capable of withstanding more limiting events, such as event 42.

3. Containment serves a safety function for some events when a unfiltered radiological release is unacceptable; but for other events, the safety function is not applicable, e.g., during refueling. The containment integrity in post-accident recovery is needed to limit doses to less than 10CFR50.67 limits. After radiological sources are depleted with time, further containment is unnecessary. Thus, the "time domain" and "need for" aspects of a function are taken into account in event evaluations.

4. The operation of engineered safety features (ESF) equipment during normal plant operation does not imply that the ESF equipment design capabilities are required for this event category. Also, the use of ESF or SAF proof systems for anticipated operational transients does not imply that equipment design capabilities; e.g., seismic, redundancy, diversity, testability, IEEE; are required for this event category.

15.9.2.2.6 Envelope and Actual Event Analyses

The event analyses presented in this chapter form an "envelope analysis" evaluation based upon expected situations. Study of the actual plant occurrences, frequencies, and their actual impact are reflected in their categorization in this section. This places the plant safety evaluations and impressions into a better perspective by focusing attention on the "envelope analysis."

15.9.2.2.7 Analysis Consistency

Figure 15.9-2 illustrates three inconsistencies in postulating system failures for a typical BWR plant. Panel A shows the possible inconsistency resulting from operational requirements

being placed on separated levels of protection for one event. If the second and sixth levels of protection are important enough to warrant operational requirements, then so are the third, fourth, and fifth levels. Panel B shows the possible inconsistency resulting from operational requirements being arbitrarily placed on some action thought to be important to safety. In the case shown, reactor trip represents different protection levels for two similar events in one category; if the fourth level of protection for Event B is important enough to warrant an operational requirement, then so is the fourth level for Event A. Thus, to simply place operational requirements on all equipment needed for some action, e.g., reactor trip, isolation, etc, could be inconsistent and unreasonable if different protection levels are represented. Panel C shows the possible inconsistency resulting from operational requirements being placed on some arbitrary level of protection for any and all postulated events. Here, the inconsistency is not recognizing and accounting for different event categories based upon cause or expected frequency of occurrence.

Inconsistencies of the types shown on Figure 15.9-2 are avoided in the NSOA by directing the analysis to event consequences oriented aspects. Analytical inconsistencies are avoided by treating all the events of a category under the same set of functional rules, by applying another set of functional rules to another category, and by having a consistent set of rules between categories. Thus, it is valid to compare the results of the analyses of the events in any one category and invalid to compare events of different categories, with different rules. An example of this is the different rules, i.e., limits, assumptions, etc, of accidents compared to anticipated transients.

15.9.2.3 Comprehensiveness of the Analysis

The method of analysis must be sufficiently comprehensive so that all plant hardware and the full range of plant operating conditions are considered. The tendency to be preoccupied with

"worst cases" or those that appear to give the most severe consequences is recognized; however, the protection sequences essential to lesser cases may be different from the worst case sequence. To ensure that operational and design basis requirements are defined and appropriate for all equipment essential to attaining acceptable consequences, all essential protection sequences must be identified for each of the plant safety events examined.

15.9.2.4 Systematic Approach of the Analysis

In summary, the systematic method used in this analysis contributes to both the consistency and comprehensiveness of the analysis mentioned above. The desired characteristics representative of a systematic approach to selecting BWR operational requirements are as follows:

1. Specify measures of unacceptable consequences

2. Consider all normal operations

3. Systematic event selection

4. Common treatment analysis of all events of any one type

5. Systematic identification of plant actions and systems essential to avoid unacceptable consequences

6. Emergence of operational requirements and limits from system analysis.

Figure 15.9-1 illustrates the systematic process by which the operational and design basis nuclear safety requirements and technical specifications are derived. The process involves the evaluation of carefully selected plant events relative to the unacceptable consequences (specified measures of safety). Those limits, actions, systems, and components found to be essential to

achieving acceptable consequences are subject to operational requirements.

Figure 15.9-3 summarizes the systematic treatment of the safety analysis.

15.9.2.5 Relationship of Nuclear Safety Operational Analysis to Safety Analyses

One of the main objectives of the operational analysis is to identify all essential protection sequences and establish the detailed equipment conditions essential to satisfy the nuclear safety operational criteria. The spectrum of events examined in previous sections of this chapter represents a complete set of plant safety considerations. The worst cases are correspondingly analyzed. The NSOA takes into account the frequency of occurrence, unacceptable consequences, assumption categories, etc, to further demonstrate that these worst case analyses assure plant safety.

The detailed discussion relative to each of the events covered in the preceding sections of this chapter is not repeated in this section. Refer to the appropriate section as cross-referenced in Tables 15.9-1 through 15.9-5.

15.9.2.6 Relationship Between Nuclear Safety Operational Analysis and Operational Requirements, Technical Specifications, Design Bases, and Single Active Failure Aspects

By definition, an "operational requirement" is a requirement or limit on either the value of a plant variable or the operability condition associated with a plant system. Such requirements must be observed during all modes of plant operation, not just at full power, to ensure that the plant is operated safely to avoid unacceptable results. There are two kinds of operational requirements for plant hardware:

1. Limiting condition for operation, which is the required condition for a system while the reactor is operating in a specified state

2. Surveillance requirements, which cover the nature and frequency of tests required to ensure that the system is capable of performing its essential functions.

Operational requirements are systematically selected in accordance with one of two criteria:

1. To ensure that unacceptable consequences are mitigated following specified plant events by examining and challenging the system design

2. To ensure that consequences of a transient or accident are acceptable with the existence of an SAF or SOE.

The individual structures and systems that perform a safety function are required to do so under design basis conditions, involving environmental consideration, and under single active component assumptions. The NSOA confirms the previous examination of the individual equipment requirement conformance analyses.

15.9.2.7 Unacceptable Consequences Criteria

Tables 15.9-6 through 15.9-10 identify the unacceptable consequences associated with different event categories. To prevent or mitigate these consequences, they are recognized as the major bases for identifying system operational requirements as well as the bases for all other safety analysis criteria throughout the FSAR.

15.9.2.8 General Nuclear Safety Operational Criteria

The following general nuclear safety operational criteria are used to select operational requirements:

Applicability	Nuclear Safety Operational Criteria
Planned operation, anticipated, abnormal operational transients, DBAs, additional special plant capability events	The plant shall be operated so as to avoid unacceptable consequences.
Anticipated and abnormal operational transients and DBAs	The plant shall be operated such that no SAF can prevent the safety actions essential to avoid the unacceptable consequences associated with anticipated or abnormal operational transients or design basis accidents. However, this requirement is not applicable during structure, system, or component repair if the availability of the safety action is maintained either by restricting the allowable repair time or by more frequent testing of a redundant structure, system, or component.

The unacceptable consequences associated with the different categories of plant operation and events are dictated by:

1. Probability of occurrence

2. Allowable limits (per the probability) related to radiological, structural, environmental, etc, aspects

3. Coincidence of other related or unrelated disturbances

4. Time domain of event and consequences consideration.

15.9.3 Method of Analysis

15.9.3.1 General Approach

The nuclear safety operational analysis (NSOA) is performed on a typical boiling water reactor (BWR) plant. The results of the analysis are the nuclear safety operational requirements and the restrictions on plant hardware and its operation that must be observed both to satisfy the nuclear safety operational criteria and to show compliance of the plant safety and power generation requirements. Figure 15.9-1 shows the process used in the analysis. The following inputs are required for the analysis of specific plant events:

1. Unacceptable consequences criteria (Section 15.9.2.7)
2. General nuclear safety operational criteria (Section 15.9.2.8)
3. BWR operating states (Section 15.9.3.2)
4. Selection of events for analysis (Section 15.9.3.3)
5. Guidelines for event analysis (Section 15.9.3.5).

The essential plant components and limits so identified are then considered to be in agreement with, and subject to, nuclear operational design basis requirements and technical specification restrictions.

15.9.3.2 Boiling Water Reactor Operating States

The four BWR operating states (A, B, C, and D) in which the reactor can exist are defined as follows:

1. State A - In state A, the reactor is in a shutdown condition, the vessel head is off, and the vessel is at atmospheric pressure

2. State B - In state B, the reactor vessel head is off, the reactor is not shut down, and the vessel is at atmospheric pressure

3. State C - In state C, the reactor vessel head is on and the reactor is shut down

4. State D - In state D, the reactor vessel head is on and the reactor is not shut down.

These four states are further defined in Section 15.9.6.2.4 and summarized in Table 15.9-11.

The main objective in selecting operating states is to divide the BWR operating spectrum into sets of initial conditions to facilitate consideration of various events in each state.

Each operating state includes a wide spectrum of values for important plant parameters. Within each state, these parameters are considered over their entire range to determine the limits on their values necessary to satisfy the nuclear safety operational criteria. Such limitations are presented in the sections of the FSAR that describe the systems associated with the parameter limit. The plant parameters to be considered in this manner include the following:

1. Reactor coolant temperature

2. Reactor vessel water level

3. Reactor vessel pressure

4. Reactor vessel water quality

5. Reactor coolant forced circulation flow rate

6. Reactor power level (thermal and neutron flux)

7. Core neutron flux distribution

8. Feedwater temperature

9. Containment temperature and pressure

10. Suppression pool water temperature and level.

15.9.3.3 Selection of Events for Analysis

15.9.3.3.1 Normal Operations

Normal operations refers to operations under predetermined conditions in the absence of significant abnormalities. Normal operations can be defined in the following chronological sequence:

1. Refueling outage - Includes all the planned operations associated with a normal refueling outage, except those tests in which the reactor is taken critical and returned to the shutdown condition. The following planned operations are included in a refueling outage:

   a. Plannned, physical movement of core components (fuel, control rods, etc)

   b. Refueling test operations (except criticality and shutdown margin tests)

   c. Planned maintenance

   d. Required inspection.

2. Achieving criticality (startup) - Includes all plant actions normally accomplished in bringing the plant from a condition in which all control rods are fully inserted to a condition in which nuclear criticality is achieved and maintained.

3. Reactor heatup - Begins after criticality is achieved and includes all plant actions normally accomplished in approaching nuclear system rated temperature and pressure by using nuclear power (reactor critical). Heatup extends through warmup and synchronization of the main turbine generator.

4. Power operation - Begins when heatup ends and includes continued plant operation at power levels in excess of heatup power.

5. Achieving reactor shutdown - Begins when the main generator is unloaded and includes all plant actions normally accomplished in achieving nuclear shutdown (more than one fuel rod subcritical) following power operation.

6. Reactor cooldown - Begins when achieving nuclear shutdown ends and includes all plant actions normal to the continued removal of decay heat and the reduction of reactor temperature and pressure.

The exact point at which some of the planned operations end and others begin cannot be precisely determined. It will be shown later that such precision is not required, because the protection requirements are adequately defined in the passage of one state to another. Dependence of several planned operations on one fuel rod subcritical condition provides an exact point on either side of which protection (especially scram) requirements differ. Thus, where a precise boundary between planned operations is needed, the definitions provide the needed precision.

Operations subsequent to an incident, i.e., transient, accident, or additional plant capability event, are not considered normal operations until the actions taken or equipment used in the plant are identical to those that would be used had the incident not occurred.

Together, BWR operating states and normal operations define the full spectrum of conditions from which transients, accidents, and special events are initiated. The BWR operating states define only the physical condition, e.g., pressure, temperature, etc, of the reactor; normal operations define what the plant is doing. The separation of physical conditions from the operation being performed is deliberate and facilitates careful consideration of all possible initial conditions from which incidents may occur.

15.9.3.3.2 Anticipated Operational Transients

To select anticipated operational transients, eight nuclear system parameter variations are considered potential initiating causes of threats to the fuel and the reactor coolant pressure boundary (RCPB). The parameter variations are as follows:

1. Reactor pressure vessel (RPV) pressure increase

2. Reactor core coolant temperature decrease

3. Control rod withdrawal

4. RPV coolant inventory decrease

5. Reactor core coolant flow decrease

6. Reactor core coolant flow increase

7. Reactor core coolant temperature increase

8. Excess of reactor core coolant inventory.

These parameter variations, if uncontrolled, could result in damage to the reactor fuel, the RCPB, or both. An RPV pressure increase threatens to rupture the RCPB from internal pressure. A pressure increase also collapses voids in the moderator, causing an insertion of positive reactivity that threatens fuel damage as a result of overheating. A reactor core coolant temperature decrease results in an insertion of positive reactivity as density increases. This could lead to fuel overheating. Positive reactivity insertions are possible from causes other than RPV pressure or moderator temperature changes. Such reactivity insertions threaten fuel damage caused by overheating. Both an RPV coolant inventory decrease and a reduction in coolant flow through the core threatens the integrity of the fuel, as the coolant becomes unable to adequately remove the heat generated in the core. An increase in coolant flow through the core reduces the void content of the moderator and results in an insertion of positive reactivity. Core coolant temperature increase, which could be the result of a heat exchanger malfunction during operation in the shutdown cooling mode, threatens the integrity of the fuel. An excess of core coolant inventory could be the result of malfunctioning water level control equipment, which can also result in a turbine trip, causing an increase in RPV pressure and power.

Anticipated operational transients are defined as transients resulting from a single active failure (SAF) or a single operator error (SOE) that can be reasonably expected (moderate probability of occurrence, which is once per day to once in 20 years) during any mode of plant operation. Examples of SAFs or SOEs in this range of probability are:

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

2. Starting or stopping any single component

3. Malfunction or maloperation of any single control device

4. Any single electrical failure

5. Any SOE.

An operator error is defined as an active deviation from nuclear plant standard operating practices. An SOE is the set of actions that is a direct consequence of a single reasonably expected erroneous decision. The set of actions is limited as follows:

1. Those that could be performed by only one person.

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

3. Those that are subsequent to the initial operator error and that affect the designed operation of the plant, but are not necessarily directly related to the operator error.

Examples of SOE are as follows:

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

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

3. An incorrect calibration of an average power range monitor.

4. Manual isolation of the main steam lines caused by operator misinterpretation of an alarm or indication.

Different types of an SAF or SOE are applied to several plant systems, with consideration for a variety of plant conditions, to discover events directly resulting in an undesired parameter

variation. Once discovered, each event is evaluated for the threat it poses to the integrity of the radioactive material barriers.

15.9.3.3.3 Abnormal Operational Transients

To select abnormal operational transients, eight nuclear system parameter variations are considered potential initiating causes of gross core wide fuel failures and threats to the RCPB. The parameter variations are as follows:

1. RPV pressure increase

2. Reactor core coolant temperature decrease

3. Control rod withdrawal

4. Reactor core coolant inventory decrease

5. Reactor core coolant flow decrease

6. Reactor core coolant flow increase

7. Reactor core coolant temperature increase

8. Excess of coolant inventory.

The eight parameter variations listed above include all effects within the nuclear system caused by abnormal operational transients that threaten gross corewide reactor fuel integrity, or seriously affect the RCPB. Variation of any one parameter may cause a change in another listed parameter; however, for analysis purposes, threats to barrier integrity are evaluated by groups according to the parameter variation originating the threat.

Abnormal operational transients are defined as incidents resulting from single or multiple equipment failures and/or single or

multiple operator errors that are not reasonably expected (less than one event in 20 years to one in 100 years) during any mode of plant operation. Examples of single or multiple operational failures and/or single or multiple operator errors are:

1. Failure of major power generation equipment components

2. Multiple electrical failures

3. Multiple operator errors

4. Combinations of equipment failure and an operator error.

Operator error is defined as an active deviation from nuclear plant standard operating practices. A multiple operator error is the set of actions that is a direct consequence of several unexpected erroneous decisions.

Examples of multiple operator errors are as follows:

1. Inadvertent loading of, and operating with, a fuel assembly in an improper position

2. The movement of a control rod during refueling operations.

The various types of single error and/or single malfunction are applied to various plant systems, with consideration for a variety of plant conditions, to discover events directly resulting in an undesired parameter variation. Once discovered, each event is evaluated for the threat it poses to the integrity of the various radioactive material barriers.

15.9.3.3.4 Design Basis Accidents

A design basis accident (DBA) is defined as a hypothesized event that affects the radioactive material barriers and that is not expected during plant operations. These are plant events,

equipment failures, and combinations of initial conditions that are of extremely low probability (once in 100 years to once in 10,000 years). The postulated accident types considered are as follows:

1. Mechanical failure of a single component leading to the release of radioactive material from one or more barriers The components referred to here are not those that act as radioactive material barriers. An example of mechanical failure is the breakage of the coupling between a control rod drive (CRD) and the control rod.

2. Arbitrary rupture of any single pipe up to, and including, complete severance of the largest pipe in the RCPB This kind of accident is considered only under conditions in which the nuclear system is pressurized.

For purposes of analysis, accidents are categorized as those events that result in releasing radioactive material as shown in Tables 15.9-4 and 15.9-5, and as follows:

1. From the fuel with the RCPB and Reactor Building enclosure initially intact (event 40)

2. Directly to the primary containment (event 42)

3. Directly to the reactor, or Turbine Building enclosures, with the primary containment initially intact (events 40, 43, 44, 45)

4. Directly to the Reactor Building enclosure with primary containment not intact (event 41)

5. Directly to the spent fuel containing facilities (event 41)

6. Directly to the Turbine Building enclosure (events 46, 47)

7. Directly to the environs (events 48, 49).

The effects of various accident types are investigated, with consideration for the full spectrum of plant conditions, to examine events that result in the release of radioactive material.

15.9.3.3.5 Special Events

A number of additional events are evaluated to demonstrate plant capabilities relative to special arbitrary nuclear safety criteria. These special events involve extremely low probability situations. As an example, the adequacy of the redundant reactivity control system is demonstrated by evaluating the special event, "reactor shutdown without control rods." Another similar example, the capability to perform a safe shutdown from outside the main control room, is demonstrated by evaluating the special event, "reactor shutdown from outside the main control room."

15.9.3.4 Applicability of Events to Operating States

The first step in performing an operational analysis for a given incident, e.g., transient, accident, or special event, is to determine the operating states in which the incident can occur. An incident is considered applicable within an operating state if the incident can be initiated from physical conditions that characterize the operating state. Applicability of the "normal operations" to the operating states follows from the definitions of planned operations. A planned operation is considered applicable within an operating state if the planned operation can be conducted when the reactor is in one of the four operating states.

15.9.3.5 Guidelines for Event Analysis

Functional guidelines followed in performing SAF, operational, and design basis analyses for the various plant events are as follows:

1. An action, system, or limit shall be considered essential only if it is essential to avoid an unacceptable result or to satisfy the nuclear safety operational criteria.

2. The full range of initial conditions, as defined in item c., below, shall be considered for each event analyzed so that all essential protection sequences are identified. Consideration is not limited to worst cases, because lesser cases sometimes may require more restrictive actions or systems different from the worst cases.

3. The initial conditions for transients, accidents, and special events shall be limited to conditions that would exist during planned operations in the applicable operating state.

4. For normal operations, consideration shall be made only for actions, limits, and systems essential to avoid the unacceptable consequences during operation in that state, as opposed to transients, accidents, and special events that are followed through to completion. Normal operations are treated differently from other events, because the transfer from one state to another during planned operations is deliberate. For events other than normal operations, the transfer from one state to another may be unavoidable.

5. Limits shall be derived only for those essential parameters that are continuously monitored by the operator. Parameter limits associated with the required performance of an essential system are considered to be

included in the requirement for the operability of the system. Limits on frequently monitored process parameters are called "envelope limits," and limits on parameters associated with the operability of a safety system are called "operability limits." Systems associated with the control of the envelope parameters are considered nonessential if it is possible to place the plant in a safe condition without using the system in question.

6. For transients, accidents, and special events, consideration shall be made for the entire duration of the event and aftermath until some planned operation is resumed. Normal operation is considered resumed when the procedures being followed, or equipment being used, are identical to those used during any one of the defined planned operations. Where extended core cooling is an immediate integral part of the event, it will be included in the protection sequence. Where it may be an eventual part of the event, it will not be directly added but can be implied to be available.

7. Credit for operator action shall be taken on a case by case basis depending on the conditions that would exist at the time operator action would be required. Because transients, accidents, and special events are considered through the entire duration of the event until normal operation is resumed, manual operation of certain systems is sometimes required following the more rapid or automatic portions of the event. Credit for operator action is taken only when the operator can reasonably be expected to accomplish the required action under the existing conditions.

8. For transients, accidents, and special events, only those actions, limits, and systems, for which there arises a unique requirement as a result of the event, shall be considered essential. For instance, if a system that was

operating prior to the event, during planned operation, were to be employed in the same manner following the event, and if the event did not affect the operation of the system, then the system would not appear on the protection sequence diagram.

9. The operational analyses shall identify all the support or auxiliary systems essential to the functioning of the front line safety systems. Safety system auxiliaries whose failure results in safe failure of the front line safety systems shall be considered nonessential.

10. A system or action that plays a unique role in the response to a transient, accident, or special event shall be considered essential unless the effects of the system or action are not included in the detailed analysis of the event.

15.9.3.6 Steps in an Operational Analysis

All information needed to perform an operational analysis for each plant event has been presented as shown on Figure 15.9-1. The procedure followed in performing an operational analysis for a given event, selected according to the event selection criteria, is as follows:

1. Determine the BWR operating states in which the event is applicable.

2. Identify all the essential protection sequences, i.e., safety actions and front line safety systems, for the event in each applicable operating state.

3. Identify all the safety system auxiliaries essential to the functioning of the front line safety systems.

The above three steps are performed in Section 15.9.6.

To derive the operational requirements and technical specifications for the individual components of a system included in any essential protection sequence, the following steps are taken:

1. Identify all the essential actions within the system (intrasystem actions) necessary for the system to function to the degree necessary to avoid the unacceptable consequences.

2. Identify the minimum hardware conditions necessary for the system to accomplish the minimum intrasystem actions.

3. If the single failure criterion applies, identify the additional hardware conditions necessary to achieve the plant safety actions (reactor trip, pressure relief, isolation, cooling, etc) in spite of single failures. This step gives the nuclear safety operational requirements for the plant components so identified.

4. Identify surveillance requirements and allowable repair times for the essential plant hardware, as discussed in Section 15.9.5.2.

5. Simplify the operational requirements determined in steps 3, and 4., so that technical specifications may be obtained that encompass the true operational requirements and are easily used by plant operations and management personnel.

15.9.4 Display of Operational Analysis Results

15.9.4.1 <u>General</u>

To fully identify and establish the requirements, restrictions, and limitations that must be observed during plant operation, plant systems and components must be related to the needs for their actions in satisfying the nuclear safety operational criteria. This section displays these relationships in a series of block diagrams.

Tables 15.9-1 through 15.9-5 and Table 15.9-11 indicate the operating states to which each event is applicable. For each event, a block diagram is presented showing the conditions and systems required to achieve each essential safety action. The block diagrams show only those systems necessary to provide the safety actions so that the nuclear safety operational and design basis criteria are satisfied. The total plant capability to provide a safety action generally is not shown; only the minimum capability essential to satisfy the operational criteria is shown. Sufficient protective equipment only is cited in the diagram to provide the necessary action. Many events can ease more paths to success than are shown. Operational analyses involve the minimum equipment needed to prevent or avert an unacceptable consequence. Thus, the diagrams depict all essential protection sequences for each event with the least amount of protective equipment needed. After all of these protection sequences are identified in block diagram form, system requirements are derived by considering all events in which the particular system is employed. The analysis considers the following conceptual aspects:

1. The BWR operating state.

2. Types of operations or events that are possible within the operating state.

3. Relationships of certain safety actions to the unacceptable consequences and to specific types of operations and events.

4. Relationships of certain systems to safety actions, and to specific types of operations and events.

5. Supporting or auxiliary systems essential to the operation of the front line safety systems.

6. Functional redundancy which is the single failure criterion applied at the safety action level. This is, in effect, a qualitative system level, failure modes and effects analysis (FMEA) type analysis.

Each block in the sequence diagrams represents a finding of essentiality for the safety action, system, or limit under consideration. Essentiality in this context means that the safety action, system, or limit is needed to satisfy the nuclear safety operational criteria. Essentiality is determined through an analysis in which the safety action, system, or limit being considered is completely disregarded in the analyses of the applicable operations or events. If the nuclear safety operational criteria are satisfied without the safety action, system, or limits, then the safety action, system, or limit is not essential, and no operational nuclear safety requirement would be indicated. When disregarding a safety action, system, or limit results in violating one or more nuclear safety operational criteria, the safety action, system, or limit is considered essential, and the resulting operational nuclear safety requirements can be related to specific criteria and unacceptable consequences.

15.9.4.2 Protection Sequence and Safety System Auxiliary Diagrams

Block diagrams illustrate essential protection sequences for each event requiring unique safety actions. These protection sequence

diagrams show only the required front line safety systems. The format and conventions used for these diagrams are shown on Figure 15.9-4.

The auxiliary systems essential to the correct functioning of front line safety systems are shown on safety system auxiliary diagrams. The format used for these diagrams is shown on Figure 15.9-5. The diagram indicates that auxiliary systems A, B, and C are required for proper operation of front line safety system X.

Total plant requirements for an auxiliary system, or the relationship of a particular auxiliary system to all other safety systems, i.e., front line and auxiliary, within an operating state are shown on the auxiliary diagrams. The format used for these diagrams is shown on Figure 15.9-6. The convention employed for Figure 15.9-6 indicates that auxiliary system A is required:

1. To be single failure proof relative to system γ in state A for events X, Y; state B for events X, Y; state C for events X, Y, Z; and state D for events X, Y, Z.

2. To be single failure proof relative to the parallel combination of systems α and β in state A for events U, V, W; state B for events V, W; state C for events U, V, W, X; and state D for events U, V, W, X.

3. To be single failure proof relative to the parallel combination of system π and system ϵ, in series with the parallel combination of systems ξ and ψ in state C for events Y, W and state D for events Y, W, Z. As noted, system ξ is part of the combination but does not require auxiliary system A for its proper operation.

4. To be single failure proof relative to system δ in state B for events Q and R; and state D for events Q, R, and S.

With these three types of diagrams, it is possible to determine, for each system, the detailed functional requirements and conditions to be observed regarding system hardware in each operating state. The detailed conditions to be observed regarding system hardware include nuclear safety operational requirements such as test frequencies and the number of components that must be operable.

15.9.5 Bases for Selecting Surveillance Test Frequencies

After the essential nuclear safety systems and engineered safeguards have been identified by applying the nuclear safety operational criteria, surveillance requirements are selected for these systems. In this selection process, the various systems are considered for relative availability, test capability, plant conditions necessary for testing, and engineering experience with the system type.

15.9.6 Operational Analyses

Results of the operational analyses are discussed in the following paragraphs and displayed on Figures 15.9-7 through 15.9-12 and Tables 15.9-1 through 15.9-5.

15.9.6.1 Safety System Auxiliaries

Figures 15.9-7 and 15.9-8 show the safety system auxiliaries essential to the functioning of each front line safety system.

Commonality of auxiliary diagrams is shown on Figures 15.9-52 through 15.9-57.

15.9.6.2 Normal Operations

15.9.6.2.1 General

Requirements for the normal or planned operations normally involve limits (L) on certain key process variables and restrictions (R) on certain plant equipment. The control block diagrams for each operating state, as shown on Figures 15.9-9 through 15.9-12, show only those controls necessary to avoid unacceptable safety consequences 1-1 through 1-4 in Table 15.9-6. Table 15.9-1 summarizes additional information for normal operations.

15.9.6.2.2 Event Definitions

Following is a description of the planned operations, events 1 through 6, as they pertain to each of the four operating states. The description of each operating state contains a definition of that state, a list of the planned operations that apply to that state, and a list of safety actions that are required to avoid the unacceptable safety consequences.

1. Event 1, refueling outage - Refueling outage includes all the planned operations associated with a normal refueling outage, except those tests in which the reactor is made critical and returned to the shutdown condition. The following planned operations are included in refueling outage:

    a. Planned, physical movement of core components (fuel, control rods, etc)

    b. Refueling test operations, except criticality and shutdown margin tests

c. Planned maintenance

d. Required inspection.

2. Event 2, achieving criticality (startup) - Achieving criticality includes all the plant actions normally accomplished in bringing the plant from a condition in which all control rods are fully inserted to a condition in which nuclear criticality is achieved and maintained.

3. Event 3, reactor heatup - Heatup begins where achieving criticality ends and includes all plant actions normally accomplished in approaching nuclear system rated temperature and pressure by using nuclear power (reactor critical). Heatup extends through warmup and synchronization of the main turbine generator.

4. Event 4, power operation (electric generation) - Power operation begins where heatup ends and includes plant operation at power levels in excess of heatup power or steady state operation. It also includes plant maneuvers such as:

    a. Daily electrical load reduction and recoveries

    b. Electrical grid frequency control adjustment

    c. Control rod movements

    d. Power generation surveillance testing involving:

        (1) Main stop valve closing

        (2) Turbine control valve adjustments

        (3) Main steam isolation valve (MSIV) exercising.

5. Event 5, achieving reactor shutdown - Achieving shutdown begins where the main generator is unloaded and includes all plant actions normally accomplished in achieving nuclear shutdown (more than one fuel rod subcritical) following power operation.

6. Event 6, reactor cooldown - Cooldown begins where achieving shutdown ends and includes all plant actions normal to the continued removal of decay heat and the reduction of nuclear system temperature and pressure.

15.9.6.2.3 Required Safety Actions/Related Unacceptable Consequences

The following paragraphs describe the safety actions for planned operations. Each description includes a selection of the operating states that apply to the safety action, the plant system affected by limits or restrictions, and the unacceptable consequence that is avoided. The four operating states are defined in Table 15.9-11. The unacceptable consequences criteria are tabulated in Table 15.9-6.

15.9.6.2.3.1 Radioactive Material Release Control

Radioactive materials may be released to the environs in any operating state; therefore, radioactive material release control is required in all operating states. The gaseous radwaste radiation monitoring system provides indication for gaseous release through the north vent stack and the offgas exhaust vent. Gaseous release through other vents are monitored by the ventilation monitoring system. All liquid wastes are monitored by process liquid radiation monitors and batch sampling before a controlled release. Limits are expressed on the gaseous radwaste system, liquid radwaste system, and solid radwaste system, so that the planned releases of radioactive materials comply with the

limits given in 10CFR20, 10CFR50, and 10CFR71 (related unacceptable safety result 1-1).

15.9.6.2.3.2 Core Coolant Flow Rate Control

In state D, when above approximately 10 percent nuclear boiler rated (NBR) power, the core coolant flow rate must be maintained above certain minimums, i.e., limited, to maintain the integrity of the fuel cladding (1-2) and ensure the validity of the plant safety analysis (1-4).

15.9.6.2.3.3 Core Power Level Control

The plant safety analyses of accidental positive reactivity additions have assumed as an initial condition that the neutron source level is above a specified minimum. Because a significant positive reactivity addition can only occur when the reactor is less than one fuel rod subcritical, the assumed minimum source level need be observed only in states B and D. The minimum source level assumed in the analyses has been related to the count/s readings on the source range monitors (SRM); thus, this minimum power level limit on the fuel is expressed as a required SRM count level. Observing the limit ensures the validity of the plant safety analysis (1-4). Maximum core power limits are also expressed for operating state B and D to maintain fuel integrity (1-2) and remain below the maximum power levels assumed in the plant safety analysis (1-4).

15.9.6.2.3.4 Core Neutron Flux Distribution Control

Core neutron flux distribution must be limited in state D; otherwise, core power peaking could result in fuel failure (1-2). Additional limits are expressed in this state, because the core neutron flux distribution must be maintained within the envelope of conditions considered by plant safety analysis (1-4).

15.9.6.2.3.5 Reactor Pressure Vessel Water Level Control

In any operating state, the reactor pressure vessel (RPV) water level could, unless controlled, drop to a level that will not provide adequate core cooling; therefore, RPV water level control applies to all operating states. Observation of the RPV water level limits protects against fuel failure (1-2) and ensures the validity of the plant safety analysis (1-4).

15.9.6.2.3.6 Reactor Pressure Vessel Pressure Control

RPV pressure control is not needed in state A and B, because vessel pressure cannot be increased above atmospheric pressure. In state C, a limit is expressed on the RPV to ensure that it is not hydrostatically tested until the temperature is above the nil ductility transition temperature plus 60°F; this prevents excessive stress (1-3). Also, in states C and D, a limit is expressed on the Residual Heat Removal (RHR) System to ensure that it is not operated in the shutdown cooling mode when the RPV pressure is greater than approximately 150 psig; this prevents excessive stress (1-2). In states C and D, a limit on the RPV pressure is necessitated by the plant safety analysis (1-4).

15.9.6.2.3.7 Reactor Pressure Vessel Temperature Control

In operating states C and D, a limit is expressed on the RPV to prevent the reactor vessel head bolting studs from being in tension when the temperature is less than 70°F to avoid excessive stress (1-3) on the reactor vessel flange. This limit does not apply in states A and B, because the head will not be bolted in place during criticality tests or refueling. In all operating states, a limit is expressed on the reactor vessel to prevent an excessive rate of change of the reactor vessel temperature to avoid excessive stress (1-3). In states C and D, where it is a planned operation to use the feedwater system, a limit is placed on the reactor fuel, so that the feedwater temperature is maintained within the envelope of conditions

considered by the plant safety analysis (1-4). For state D, a limit is placed on the temperature difference between the recirculation system and the RPV to prevent the starting of the recirculation pumps. This operating restriction and limit prevents excessive stress in the reactor vessel (1-3).

15.9.6.2.3.8 Reactor Pressure Vessel Water Quality Control

In all operating states, water of improper chemical quality could produce excessive stress as a result of chemical corrosion (1-3). Therefore, a limit is placed on reactor coolant chemical quality in all operating states. For all operating states where the nuclear system can be pressurized (states C and D), an additional limit on reactor coolant activity ensures the validity of the analysis of the main steam line break accident (1-4).

15.9.6.2.3.9 Reactor Pressure Vessel Leakage Control

Excessive RPV leakage could occur only while the RPV is pressurized; thus, limits are applied only to the reactor vessel in states C and D. Observing these limits prevents vessel damage due to excessive stress (1-3) and ensures the validity of the plant safety analysis (1-4).

15.9.6.2.3.10 Core Reactivity Control

In state A, during refueling outage, a limit is imposed on core loading (fuel) to ensure that core reactivity is maintained within the envelope of conditions considered by the plant safety analysis (1-4). In all states, limits are imposed on the control rod drive (CRD) system to ensure adequate control of core reactivity, so that core reactivity remains within the envelope of conditions considered by the plant safety analysis (1-4).

15.9.6.2.3.11 Control Rod Worth Control

Any time the reactor is not shut down and is generating less than 30 percent power (state D), a limit is imposed on the control rod pattern to ensure that control rod worth is maintained within the envelope of conditions considered by the analysis of the control rod drop accident (1-4).

15.9.6.2.3.12 Refueling Restriction

By definition, planned operation event 1 (refueling outage) applies only to state A. Observing the restrictions on the reactor fuel and on the operation of the CRD system within the specified limit maintains plant conditions within the envelope considered by the plant safety analysis (1-4).

15.9.6.2.3.13 Drywell and Reactor Enclosure Pressure and Temperature Control

In states C and D, limits are imposed on the drywell and suppression pool to maintain temperature and pressure within the envelope considered by plant safety analysis (1-4). These limits ensure an environment in which instruments and equipment can operate correctly within the drywell. Limits on the pressure suppression pool apply to the water temperature and water level to ensure that the pool has the capability of absorbing the energy discharged during a safety/relief valve blowdown or a loss-of-coolant accident (LOCA).

15.9.6.2.3.14 Stored Fuel Shielding, Cooling, and Reactivity Control

Both new and spent fuel will be stored during all operating states, therefore, stored fuel shielding, cooling, and reactivity control apply to all operating states. Limits are imposed on the spent fuel pool storage positions, water level, fuel handling procedures, and water temperature. Observing the limits on fuel

storage positions ensures that spent fuel reactivity remains within the envelope of conditions considered by the plant safety analysis (1-4). Observing the limits on water level ensures shielding in order to maintain conditions within the envelope of conditions considered by the plant safety analysis (1-4), and provides the fuel cooling necessary to avoid fuel damage (1-2). Observing the limit on water temperature avoids excessive fuel pool stress (1-3).

15.9.6.2.4 Operational Safety Evaluations

1. State A - In state A, the applicable events for planned operations are refueling outage, achieving criticality, and cooldown (events 1, 2, and 6, respectively).

   Figure 15.9-9 shows the necessary safety actions for planned operations, corresponding plant systems, and the event for which these actions are necessary. As indicated in the diagram, the required safety actions are as follows:

   a. Radioactive material release control

   b. RPV water level control

   c. RPV temperature control

   d. RPV water quality control

   e. Core reactivity control

   f. Refueling restrictions

   g. Stored fuel shielding, cooling, and reactivity control.

2. State B - In state B, the applicable planned operations are achieving criticality and shutdown (events 2 and 5, respectively).

   Figure 15.9-10 relates the necessary safety actions for planned operations, plant systems, and the event for which the safety actions are necessary. The required safety actions for planned operation in state B are as follows:

   a. Radioactive material release control

   b. Core power level control

   c. RPV water level control

   d. RPV temperature control

   e. RPV water quality control

   f. Core reactivity control

   g. Rod worth control

   h. Stored fuel shielding, cooling, and reactivity control.

3. State C - In state C, the applicable planned operations are achieving criticality and cooldown (events 2 and 6, respectively).

   Sequence diagrams relating safety actions for planned operations, plant systems, and applicable events are shown on Figure 15.9-11. The required safety actions for planned operation in state C are as follows:

   a. Radioactive material release control

b. RPV water level control

c. RPV pressure control

d. RPV temperature control

e. RPV water quality control

f. RPV leakage control

g. Core reactivity control

h. Containment and reactor enclosure temperature control

i. Stored fuel shielding, cooling, and reactivity control.

4. State D - In state D, the applicable planned operations are achieving criticality, heatup, power operation, and shutdown (events 2, 3, 4, and 5, respectively).

   Figure 15.9-12 relates safety actions for planned operations, corresponding plant systems, and events for which the safety actions are necessary. The required safety actions for planned operation in state D are as follows:

   a. Radioactive material release control

   b. Core coolant flow rate control

   c. Core power level control

   d. Core neutron flux distribution control

   e. RPV water level control

f. RPV pressure control

g. RPV temperature control

h. RPV water quality control

i. RPV leakage control

j. Core reactivity control

k. Rod worth control

l. Drywell and reactor enclosure pressure and temperature control

m. Stored fuel shielding, cooling, and reactivity control.

15.9.6.3 <u>Anticipated Operational Transients</u>

15.9.6.3.1 General

The safety requirements and protection sequences for anticipated operational transients are described in the following paragraphs for events 7 through 29. The protection sequence block diagrams show the sequence of front line safety systems, as shown on Figures 15.9-13 through 15.9-35. The auxiliaries for the front line safety system are indicated in the auxiliary diagrams on Figures 15.9-7 and 15.9-8, and the commonality of auxiliary diagrams on Figures 15.9-52 through 15.9-57.

15.9.6.3.2 Required Safety Actions/Related Unacceptable Consequences

Safety actions for anticipated operational transients that mitigate or prevent the unacceptable safety consequences are

listed below. Refer to Table 15.9-7 for the unacceptable consequences criteria.

Safety/Action	Related Unacceptable Consequence Criteria	Reason Action Required
Reactor trip and/or recirculation pump trip (RPT)	2-2 2-3	To prevent fuel damage and limit RPV pressure rise
Pressure relief	2-3	To prevent excessive nuclear system pressure rise
Core and containment cooling	2-1 2-2 2-4	To prevent fuel and containment damage in the event that normal cooling is interrupted
Reactor vessel isolation	2-2	To prevent fuel damage by reducing the outflow of steam and water from the reactor vessel, thereby limiting the decrease in reactor vessel water level
Restore ac power	2-2	To prevent fuel damage by restoring ac power to systems essential to other safety actions
Prohibit rod motion	2-2	To prevent exceeding fuel limits during transients
Containment isolation	2-1 2-4	To minimize radiological effects

15.9.6.3.3 Event Definitions and Operational Safety Evaluations

1. Event 7, manual and inadvertent reactor trip The deliberate manual or inadvertent automatic reactor trip due to a single operator error (SOE) is an event that can occur under any operating condition. Although assumed to occur here for examination purposes, multiple operator error or action is necessary to initiate such an event.

   While all the safety criteria apply, no unique safety actions are required to control the planned operation like event after effects of the subject initiation actions. In all operating states, therefore, the safety criteria are met through the basic design of the plant systems. Figure 15.9-13 identifies the protection sequences for this event.

2. Event 8, loss of plant instrument/service air system Loss of all plant instrument or service air system causes reactor shutdown and the closure of isolation valves. Although these actions occur, they are not a requirement to prevent unacceptable consequences in themselves. Multiple equipment failures would be necessary to cause the deterioration of the subject system to the point that the components supplied with instrument or service air would cease to operate normally and/or fail safe. The resulting actions are identical to event 14, described later.

   Isolation of the main steam lines can result in a transient for which some degree of protection is required only in operating states C and D. In operating states A and B, the main steam lines are continuously isolated.

   Isolation of all main steam lines is most severe and rapid in operating state D during power operation.

Figures 15.9-14 and 15.9-21 show how reactor trip is accomplished by main steam line isolation through the actions of the Reactor Protection System (RPS) and CRD system. The RPV pressure relief system provides pressure relief. Pressure relief, combined with loss of feedwater flow, causes reactor vessel water level to fall. Either high pressure coolant injection (HPCI) or reactor core isolation cooling (RCIC) supplies water to maintain water level and protect the core until normal steam flow or other planned operation is established.

Adequate reserve service air supplies are maintained exclusively for the continual operation of the automatic depressurization system safety relief valves (ADS SRVs) until reactor shutdown is accomplished.

3. Event 9, inadvertent HPCI pump start (moderator temperature decrease) An inadvertent pump start, i.e., temperature decrease, is defined as an unintentional start of any Nuclear Steam Supply System (NSSS) pump that adds sufficient cold water to the reactor coolant inventory to cause a measurable decrease in moderator temperature. This event is considered in all operating states, because it can potentially occur under any operating condition. Since the HPCI pump operates over nearly the entire range of the operating states and delivers by far the greatest amount of cold water from the condensate storage tank (CST) to the RPV, the following analysis will describe its inadvertent operation rather than other NSSS pumps, e.g., RCIC, RHR, and core spray.

While all the safety criteria apply, no unique safety actions are required to control the adverse effects of such a pump start, i.e., pressure increase and temperature decrease in states A and C. In these operating states, the safety criteria are met through the

basic design of the plant systems, and no safety action is specified. In states B and D, where the reactor is not shut down, the operator or the plant's normal control system can control any power changes in the normal manner of power control.

Figure 15.9-15 illustrates the protection sequence for the subject event. Failure of the plant's normal control system pressure regulator or single failure of the feedwater controller systems will result in further protection sequences. These are shown in events 22 and 23. The single failure aspects of their protection sequences will not be required.

4. Event 10, startup of idle recirculation pump The cold startup of an idle recirculation pump can occur in any state and is most severe and rapid for those operating states in which the reactor may be critical (states B and D). When the transient occurs in the range of 10 to 60 percent power operation, no safety action response is required. Reactor power is normally limited to approximately 60 percent design power, because of core flow limitations while operating with one recirculation loop working. Above about 60 percent power, a high neutron flux reactor trip is initiated. If the event occurs when the reactor is in operating state D, not at power operation, but critical (5 percent to 10 percent), the resulting transient may produce a high level neutron flux reactor trip from the intermediate range monitor (IRM). No safety actions are required in state B, since the power would be less than 5 percent.

   As shown on Figure 15.9-16, the reactor trip action is accomplished through the combined actions of the neutron monitoring, reactor protection, and CRD systems. At power operation (10 to 60 percent), the high level IRM

reactor trip is not initiated, because the core flux monitoring has been shifted to the average power range monitors (APRM).

5. Event 11, recirculation flow control failure (increasing flow) A recirculation flow control failure causing increased flow is applicable in states C and D. In state D, the resulting increase in power level is limited by a reactor trip. As shown on Figure 15.9-17, the reactor trip safety action is accomplished through the combined actions of the neutron monitoring, reactor protection, and CRD systems.

6. Event 12, recirculation flow control failure (decreasing flow) This recirculation flow control malfunction causes a decrease in core coolant flow. This event is not applicable to states A and B, because the reactor vessel head is off and the recirculation pumps normally would not be in use.

   The number and type of flow controller failure modes determine the protection sequence for the event. Motor generator set flow control system failures of one or both of the master flow controllers will result in a transient equivalent to one or two RPTs, as shown on Figure 15.9-18.

7. Event 13, trip of one or both recirculation pumps - The trip of one recirculation pump produces a milder transient than does the simultaneous trip of two recirculation pumps.

   The transient resulting from the two RPTs (both loops) is not severe enough to require any unique safety action. The transient is compensated for by the inherent nuclear stability of the reactor. This event is not applicable in states A and B, because the reactor vessel head is off

and the recirculation pumps normally would not be in use. The trip could occur in states C and D; however, the reactor can accommodate the transient with no unique safety action requirement. Figure 15.9-19 provides the protection sequence for the event for one or both RPT actuations.

In fact, this event constitutes an acceptable operational technique to reduce or minimize the effects of other event conditions. To this end, an engineered RPT capability is included in the plant operational design to reduce pressure and thermohydraulic transient effects. Operating states C and D are involved in this event.

A single RPT requires no protection system operation.

Two RPTs result in a high water level trip of the main turbine, which further causes a main stop valve closure and its subsequent reactor trip actuation. Main steam line isolation soon occurs and is followed by RCIC/HPCI systems initiation on low water level. SRV actuation will follow.

8. Event 14, isolation of one or all main steam lines - Isolation of the main steam lines due to inadvertent MSIV closure, can result in a transient for which some degree of protection is required only in operating states C and D. In operating states A and B, the main steam lines are continuously isolated.

   Isolation of all main steam lines is most severe and rapid in operating state D during power operation.

   Figure 15.9-20 shows how reactor trip is accomplished by main steam line isolation through the actions of the RPS and CRD systems. The RPV pressure relief system provides pressure relief. Pressure relief, combined with loss of

feedwater flow, causes reactor vessel water level to fall and either HPCI or RCIC supply water to maintain water level and protect the core until normal steam flow or other planned operation is established.

Isolation of one main steam line causes a significant transient only in state D during high power operation. Reactor trip is the only unique action required to avoid fuel damage and nuclear system overpressure. Because the feedwater system and main condenser remain in operation following the event, no unique requirement arises for core cooling.

As shown on Figure 15.9-21, the reactor trip safety action is accomplished through the combined actions of the neutron monitoring, reactor protection, and CRD systems.

9. Event 15, inadvertent opening of a steam main safety/relief valve (SRV) - The inadvertent opening of a SRV is possible in any operating state. The protection sequences are shown in Figure 15.9-22. In states A, B, and C, the water level cannot be lowered far enough to threaten fuel damage; therefore, no safety actions are required.

   In state D, there is a slight decrease in reactor pressure following the event. The pressure regulator closes the main turbine control valves enough to stabilize pressure at a level slightly below the initial value. There are no unique safety system requirements for this event.

   If the event occurs when the feedwater system is not active in state D, a loss in the coolant inventory results in a reactor vessel isolation. The low water level signal initiates reactor vessel isolation. The

nuclear system pressure relief system provides pressure relief.

Core cooling is accomplished by the RCIC and HPCI systems, which are automatically initiated by the incident detection circuitry (IDC). Either ADS, or the manual actuation of the SRVs remain as the backup depressurization system if needed. After the vessel has depressurized, long term core cooling is accomplished by LPCI and core spray, which are initiated on low water level by the IDC system or are manually operated. Containment suppression pool cooling is manually initiated.

10. Event 16, control rod withdrawal error for refueling and startup operations - A control rod withdrawal error resulting in an increase of positive reactivity can occur under any operating condition; therefore, it must be considered in all operating states. For this specific event situation, only states A and B apply.

    a. Refueling - no unique safety action is required in operating state A for the withdrawal of one control rod, because the core is more than one control rod subcritical. Withdrawal of more than one control rod is precluded by the protection sequence shown on Figure 15.9-23. During core alterations, the mode switch is normally in the "refuel" position, which allows the refueling equipment to be positioned over the core and also inhibits control rod withdrawal. Therefore, this transient applies only to operating state A.

       No safety action is required, because the total worth (positive reactivity) of one fuel assembly or control rod is not adequate to cause criticality. Moreover, mechanical design of the control rod

assembly prevents physical removal without removing the adjacent fuel assemblies.

b. Startup - During low power operation (state B), the neutron monitoring system via the RPS will initiate reactor trip if necessary, as shown on Figure 15.9-23.

11. Event 17, control rod withdrawal error (during power operation) - A control rod withdrawal error resulting in an increase of positive reactivity can occur under any operating condition; thus, it must be considered in all operating states. For this specific event situation, only statesC and D apply.

During power operation (power range state D), a number of plant protective devices of various designs prohibit the control rod motion before critical levels are reached, as shown on Figure 15.9-24.

Systems in the power range, 0 to 100 percent NBR, prevent the selection of an out of sequence rod movement by using the rod worth minimizer (RWM) which uses either banked position or group withdrawal sequences. In addition, the movement of the rod is monitored and limited within acceptable intervals either by neutronic effects or actual rod motion. Rod block monitor (RBM) provides movement surveillance. Beyond these rod motion control limits are the fuel/core reactor trip protection systems.

In state C, no protective action is needed.

12. Event 18, loss of RHR system shutdown cooling - The loss of RHR system shutdown cooling can occur only during the low pressure portion of a normal reactor shutdown and cooldown.

As shown on Figure 15.9-25, for most single failures that could result in primary loss of shutdown cooling capabilities, no unique safety actions are required; in these cases, shutdown cooling is simply reestablished using redundant shutdown cooling equipment. In the cases where the RHR system shutdown cooling suction line becomes inoperative or supply and return valves isolate due to a loss of offsite power, a unique arrangement for cooling arises. In states A and B, in which the reactor vessel head is off, the low pressure coolant injection (LPCI) and core spray systems can be used to maintain RPV water level. In states C and D, in which the reactor vessel head is on and the system can be pressurized, the ADS or manual actuation of SRVs, in conjunction with any of the Emergency Core Cooling Systems (ECCSs) and the RHR suppression pool cooling mode (both manually operated), can be used to maintain water level and remove decay heat. Suppression pool cooling is initiated to remove heat energy from the suppression pool system.

13. Event 19, RHR system increased shutdown cooling - An RHR system shutdown cooling malfunction that causes a moderator temperature decrease must be considered in all operating states. However, this event is not considered in states C and D if RPV pressure is too high to permit operation of the RHR system, as shown on Figure 15.9-26. No unique safety actions are required to avoid the unacceptable safety consequences for transients as a result of a reactor coolant temperature decrease induced by misoperation of the RHR heat exchangers.

    In states B and D, where the reactor is at or near critical, the slow power increase resulting from the lower coolant temperature would be controlled by the operator in the same manner normally used to control power in the source or intermediate power ranges.

14. Event 20, loss of feedwater flow - A loss of feedwater flow results in a net decrease in the coolant inventory available for core cooling. A loss of feedwater flow can occur in states C and D. Appropriate responses to this transient include a reactor trip on low water level and restoration of RPV water level by HPCI and RCIC.

As shown on Figure 15.9-27, the reactor protection and CRD systems effect a reactor trip on low water level. The Primary Containment and Reactor Vessel Isolation Control System (PCRVICS) and the MSIVs act to isolate the reactor vessel. After the MSIVs close, decay heat slowly raises system pressure to the lowest safety/relief valve setting. Pressure is relieved by the RPV pressure relief system. Either the RCIC or HPCI system can maintain adequate water level for initial core cooling and to restore and maintain water level. For long term shutdown and extended core coolings, containment and suppression pool cooling systems are manually initiated.

The requirements for operating state C are the same as for state D, except that the reactor trip action is not required in state C.

15. Event 21, loss of feedwater heating - Loss of feedwater heating must be considered with regard to the nuclear safety operational criteria only in operating state D, because significant feedwater heating does not occur in any other operating state.

The neutron flux increase associated with this event might reach the reactor trip setpoint. As shown on Figure 15.9-28, the reactor trip safety action is accomplished through actions of the neutron monitoring, reactor protection, and CRD systems.

16. Event 22, feedwater controller failure (maximum demand) - A feedwater controller failure, causing an excess of coolant inventory in the reactor vessel, is possible in all operating states. Feedwater controller failures considered are those that would give failures of automatic flow control, manual flow control, or feedwater bypass valve control. In operating states A and B, no safety actions are required, since the vessel head is removed and the moderator temperature is low. In operating state D, any positive reactivity effects responses by the reactor caused by cooling of the moderator can be mitigated by a reactor trip. As shown on Figure 15.9-29, the accomplishment of the reactor trip safety action is satisfied through the combined actions of the neutron monitoring, reactor protection, and CRD systems. Pressure relief is required in States C and D and is achieved through the operation of the RPV pressure relief system. Initial restoration of the core water level is by the RCIC and HPCI systems. Prolonged isolation may require extended core cooling and containment/suppression pool cooling.

17. Event 23, pressure regulator failure (open direction) - A pressure regulator failure in the open direction, causing the opening of a turbine control or bypass valve, applies only in operating states C and D, because in other states the pressure regulator is not in operation. A pressure regulator failure is most severe and rapid in operating State D at low power. This event requires the failure of two of the three pressure regulator channels, or can be initiated by personnel error.

    The various protection sequences giving the safety actions are shown on Figure 15.9-30. Depending on plant conditions existing prior to the event, reactor trip will be initiated either on main steam line isolation, main turbine trip, reactor vessel high pressure, or reactor vessel low water level. The sequence resulting in reactor vessel isolation also depends on initial

conditions. With the mode switch in "run," isolation is initiated when main steam line pressure decreases to approximately 800 psig. Under other conditions, isolation is initiated by reactor vessel low water level. After isolation is completed, decay heat will cause reactor vessel pressure to increase until limited by the operation of the safety relief valves. Core cooling following isolation can be provided by RCIC or HPCI. Shortly after reactor vessel isolation, normal core cooling can be reestablished via the main condenser and feedwater systems or, if prolonged isolation is necessary, extended core and containment cooling will be manually initiated.

18. Event 24, pressure regulator failure (closed direction) - A pressure regulator failure in the closed direction (or downscale), causing the closing of turbine control valves, applies only in operating states C and D, because the pressure regulator is not in operation in other states.

    A single pressure regulator failure downscale would result in little or no effect on the plant operation. The second and third pressure regulators would provide turbine/reactor control. Failure of the second unit, which would result in the worst situation, is much less severe than events 25, 27, 30, and 31. The dual pressure regulator failures are most severe and rapid in operating state D at high power.

    The various protection sequences giving the safety actions are shown on Figure 15.9-31. Upon failure of one pressure regulator downscale, normally the remaining two regulators will maintain the plant in the normal status. An additional failure of a regulator will result in

a high neutron flux or pressure reactor trip, system isolation, and subsequent extended isolation core cooling system actuations.

19. Event 25, main turbine trips (with bypass system operation) - A main turbine trip can occur only in operating state D (during heatup or power operation). A turbine trip during heatup is not as severe as a trip at full power, because the initial power level is less than 30 percent, thus minimizing the effects of the transient and enabling return to planned operations via the bypass system operation. For a turbine trip above 30 percent power, a reactor trip, as well as an RPT, will occur via main stop valve closure. Subsequent safety relief valve actuation will occur. Eventual main steam line isolation and RCIC and HPCI system initiation will result from low water level. Figure 15.9-32 depicts the protection sequences required for main turbine trips. Main turbine trip and main generator trip are similar anticipated operational transients and, although main turbine trip is a more severe transient than main generator trip due to the rapid closure of the turbine stop valves, the required safety actions are the same.

20. Event 26, loss of main condenser vacuum (turbine trip) - A loss of vacuum in the main turbine condenser can occur any time steam pressure is available and the condenser is in use; it is applicable to operating states C and D. This nuclear system pressure increase transient is the most severe of the pressure increase transients. However, reactor trip protection in state C is not needed, since the reactor is not coupled to the turbine system.

    For state D, above 30 percent power, loss of condenser vacuum will initiate a turbine trip with its attendant stop valve closures (which leads to reactor trip) and an

RPT will also initiate RPV isolation, SRV actuation, and RCIC and HPCI initial core cooling. A reactor trip is initiated by MSIV closure to prevent fuel damage and is accomplished with the actions of the reactor protection and CRD systems. Below 30 percent power (state D), reactor trip is initiated by a high neutron flux signal. Figure 15.9-33 shows the protection sequences. Decay heat will necessitate extended core and suppression pool cooling. When the RPV depressurizes sufficiently, the low-pressure core cooling systems provide core cooling until a planned shutdown operation via RHR shutdown cooling is achieved.

21. Event 27, main generator trip (with bypass system operation) - A main generator trip with turbine bypass system operation can occur only in operating state D during heatup or power operation. Fast closure of the main turbine control valves is initiated whenever an electrical grid disturbance occurs that results in significant loss of electrical load on the generator. The turbine control valves are required to close as rapidly as possible to prevent excessive overspeed of the main turbine generator rotor. Closure of the turbine control valves will cause a sudden reduction in steam flow, which results in an increase in system pressure. Above 30 percent power, reactor trip will occur as a result of fast control valve closure. Turbine tripping will actuate the RPT. Subsequently, main steam line isolation will result, and pressure relief and initial core cooling by RCIC and HPCI will take place. Prolonged shutdown of the turbine generator unit will necessitate extended core and containment cooling. A generator trip during heatup (less than 30 percent) is not severe, because the Turbine Bypass System can accommodate the decoupling of the reactor and the turbine generator unit, thus minimizing the effects of the transient and enabling return to planned operations. Figure 15.9-34 depicts the

protection sequences required for a main generator trip. Main generator trip and main turbine trip are similar anticipated operational transients. Although the main generator trip is a less severe transient than a turbine trip due to the rapid closure of the main stop valves, the required safety actions for both are the same sequence.

22. Event 28, loss of normal onsite power - See event 29, loss of offsite power. Electrical design prevents loss of normal onsite power as discussed in Section 8.3.

23. Event 29, loss of offsite power - grid loss - There are a variety of plant grid electrical component failures that can affect reactor operation. The total loss of offsite power (LOP) is the most severe. The loss of offsite auxiliary power sources results in a sequence of events similar to that resulting from a loss of feedwater flow as in event 20. The most severe case occurs in state D during power operation. Figure 15.9-35 shows the safety actions required for a total loss of offsite power in all states, A, B, C, and D.

    The reactor protection and CRD systems affect a reactor trip from main turbine trip or loss of RPS power sources. The turbine trip will initiate an RPT. The PCRVICS and the MSIVs act to isolate the reactor vessel. After the MSIVs close, decay heat slowly raises system pressure to the lowest safety relief valve setting. Pressure is relieved by the RPV pressure relief system. After the reactor is isolated and feedwater flow has been lost, decay heat continues to increase RPV pressure, periodically lifting SRVs and causing RPV water level to decrease. The core and containment cooling sequence shown on Figure 15.9-35 shows the short and long term sequences for achieving adequate cooling.

15.9.6.4 Abnormal Operational Transients

15.9.6.4.1 General

The safety requirements and protection sequences for abnormal operational transients are described in the following paragraphs for events 30 through 39. The protection sequence block diagrams, on Figures 15.9-36 through 15.9-40, show the sequence of front line safety systems. The auxiliaries for the front line safety systems are indicated in the auxiliary diagrams on Figures 15.9-7 and 15.9-8, and the commonality of auxiliary diagrams on Figures 15.9-52 through 15.9-57.

15.9.6.4.2 Required Safety Actions/Related Unacceptable Consequences

The following list relates the safety actions for abnormal operational transients to mitigate or prevent the unacceptable safety consequences cited in Table 15.9-8.

Safety Action	Related Unacceptable Consequence	Reason Action Required
Reactor trip and/or RPT	3-2 3-3	To limit gross core wide fuel damage and to limit nuclear system pressure rise
Pressure relief	3-3	To prevent excessive nuclear system pressure rise
Core suppression pool and containment cooling	3-2 3-4	To limit further fuel and containment damage if normal cooling is interrupted

Safety Action	Related Unacceptable Consequence	Reason Action Required
Reactor vessel isolation	3-2	To limit further fuel damage by reducing the outflow of steam and water from the reactor vessel, thereby limiting the decrease in reactor vessel water level
Restore ac power	3-2	To limit initial fuel damage by restoring ac power to systems essential to other safety actions
Containment isolation	3-1	To limit radiological effects

15.9.6.4.3 Event Definition and Operational Safety Evaluation

1. Event 30, main generator trip (with bypass system failure) - A main generator trip with bypass system failure can occur only in operating state D (during heatup or power operation). A generator trip during heatup with bypass failure results in the same situation as the power operation case. Figure 15.9-36 depicts the protection sequences required for a main generator trip. The event is basically the same as that described in event 27 at power levels above 30 percent. A reactor trip, RPT, isolation, SRV, and RCIC and HPCI operation will immediately result in prolonged shutdown, which will follow the same pattern as event 27.

   The thermohydraulic and thermodynamic effects on the core, of course, are more severe than with the bypass

operating. Since the event is of lower probability than event 27, the unacceptable consequences are less limiting.

The load rejection and turbine trip are similar abnormal operational transients and, although main generator trip is a less severe transient than a turbine trip due to the rapid closure of the main stop valves, the required safety actions are the same.

2. Event 31, main turbine trip (with bypass system failure) - A main turbine trip with bypass failure can occur only in operating state D, during heatup or power operation. Figure 15.9-37 depicts the protection sequences required for main turbine trips. Plant operation with bypass system operation above or below 30 percent power, due to bypass system failure, will result in the same transient effects: a reactor trip, an RPT, an isolation, subsequent SRV actuation, and immediate RCIC and HPCI actuation. After initial shutdown, extended core and containment cooling will be required, as noted previously in event 25.

   Turbine trip with bypass system failure results in very severe thermohydraulic impacts on the reactor core. The allowable limit or acceptable calculational techniques for this event are less restrictive, since the event has a lower probability of occurrence than the turbine trip with a bypass operation event.

3. Event 32, inadvertent loading and operation with fuel assembly in improper position - Operation with a fuel assembly in the improper position is shown on Figure 15.9-38 and can occur in all operating states. No protection sequences are necessary relative to this event. Calculated results of worst fuel bundle loading

error will not cause fuel cladding integrity damage. It requires three independent equipment/operator errors to allow this situation to develop.

4. Events 33 through 37 (not used)

5. Event 38, recirculation pump seizure - A recirculation pump seizure event considers the instantaneous stoppage of the pump motor shaft of the recirculation pump in one loop. The case involves operation at design power in state D. A main turbine trip will occur as RPV water level swell exceeds the turbine trip setpoint. This results in a reactor trip and an RPT when the main stop valves close. SRV opening will occur to control pressure level and temperatures. RCIC or HPCI systems will maintain vessel water level. Prolonged isolation will require core and containment cooling and possibly some radiological effluent control.

   The protection sequence for this event is given on Figure 15.9-39.

6. Event 39, recirculation pump shaft break A recirculation pump shaft break event considers the degraded, delayed stoppage of the pump motor shaft of the recirculation pump in one loop. The case involves operation at design power in state D. A main turbine trip will occur as RPV water level swell exceeds the turbine trip setpoint. This results in a reactor trip and an RPT when the main stop valves close. Safety relief valve opening will occur to control pressure, level, and temperature. RCIC or HPCI systems will maintain reactor vessel water level. Prolonged isolation will require core and containment cooling and, possibly, some radiological effluent control.

The protection sequence for this event is given on Figure 15.9-40.

15.9.6.5 Design Basis Accidents

15.9.6.5.1 General

The safety requirements and protection sequences for accidents are described in the following paragraphs for events 40 through 49. The protection sequence block diagrams show the safety actions and the sequence of front line safety systems used for the accidents, as shown on Figures 15.9-41 through 15.9-48. The auxiliaries for the front line safety systems are indicated in the auxiliary diagrams shown on Figures 15.9-7 and 15.9-8, and the commonality of auxiliary diagrams shown on Figures 15.9-52 through 15.9-57.

15.9.6.5.2 Required Safety Actions/Unacceptable Consequences

The following list relates the safety actions for a design basis accident (DBA) to mitigate or prevent the unacceptable consequences cited in Table 15.9-9:

Safety Action	Related Unacceptable Consequence	Reason Action Required
Reactor trip	4-2 4-3	To prevent fuel cladding failure and excessive nuclear system pressures; Failure of fuel barrier includes fuel cladding fragmentation (LOCA) and excessive fuel enthalpy (control rod drop accident)

Safety Action	Related Unacceptable Consequence	Reason Action Required
Pressure relief	4-3	To prevent excessive nuclear system pressure
Core cooling	4-2	To prevent fuel cladding failure
Reactor vessel isolation	4-1	To limit radiological effect to not exceed the guideline values of 10CFR50.67
Establish reactor containment	4-1	To limit radiological effects to not exceed the guideline values of 10CFR50.67
Containment cooling	4-4	To prevent excessive pressure in the containment when containment is required
Stop control rod ejection	4-2	To prevent fuel cladding failure
Restrict loss of reactor coolant (passive)	4-2	To prevent fuel cladding failure
Main control room environmental control	4-5	To prevent overexposure to radiation of plant personnel in the main control room
Limit reactivity insertion rate (passive)	4-2 4-3	To prevent fuel cladding failure and excessive nuclear system pressure

15.9.6.5.3 Event Definition and Operational Safety Evaluations

1. Event 40, control rod drop accident (CRDA) - The CRDA results from an assumed failure of the control rod to drive mechanism coupling after the control rod (very reactive rod) becomes stuck in its fully inserted position. It is assumed that the CRD is then fully withdrawn before the stuck rod falls out of the core. The control rod velocity limiter, an engineered safeguard, limits the control rod drop velocity. The resultant radioactive material release is maintained far below the guideline values of 10CFR100.

   The CRDA is applicable only in operating state D. The CRDA cannot occur in state B, because rod coupling integrity is checked on each rod to be withdrawn if more than one rod is to be withdrawn. No safety actions are required in states A or C where the plant is in a shutdown state by more than the reactivity worth of one control rod prior to the accident.

   Figure 15.9-41 presents the different protection sequences for the CRDA. The reactor is automatically tripped and isolated for all design basis cases except the MSLRM initial case. The neutron monitoring, reactor protection, and CRD systems will provide a reactor trip from high neutron flux. The main steam line radiation monitoring system will initiate the isolation of the reactor water sample valves and a mechanical vacuum pump trip on high high radiation in the main steam lines. Following a valid high-high MSLRM signal indicating high MSL radiation the reactor will be manually scrammed and the MSIV's will be manually closed in that order. Scramming the reactor first prevents further fuel damage due to the reactor pressure spike that occurs if the MSIV's are manually closed without scramming the reactor.

   After the reactor has been tripped and isolated, the RPV pressure relief system allows the steam, produced by decay heat, to be directed to the suppression pool. Initial core cooling is accomplished by the RCIC, HPCI, or normal feedwater system. With prolonged isolation, as indicated on Figure 15.9-41, the reactor operator initiates the RHR/suppression pool cooling mode and

depressurizes the vessel with the manual mode of the ADS or via manual SRV operation. The LPCI or core spray maintain the vessel water level and accomplish extended core cooling. Isolation of turbine condenser fission product releases will also be maintained.

2. Event 41, fuel handling accident - A fuel handling accident can potentially occur any time the fuel assemblies are being manipulated, either over the reactor core or in a spent fuel pool; thus, this accident is considered in all operating states. Considerations include mechanical fuel damage caused by drop impact and a subsequent release of fission products. The protection sequences pertinent to this accident are shown on Figure 15.9-42. Containment and Reactor Building enclosure isolation and Filtration, Recirculation and Ventilation System (FRVS) operation are automatically initiated by the respective enclosure, pool, or ventilation radiation monitoring systems.

3. Event 42 - loss-of-coolant accident (LOCA) resulting from postulated piping breaks within the RCPB inside containment (DBA LOCA) - Pipe breaks inside the containment are considered only when the nuclear system is significantly pressurized (states C and D). The result is a release of steam and water into the containment. Consistent with NSOA criteria, the protection requirements consider all size line breaks, from large liquid recirculation loop piping breaks down to small steam instrument line breaks. The most severe cases are the circumferential break of the largest (liquid) recirculation system pipe and the circumferential break of the largest main steam line.

   As shown on Figure 15.9-43, in operating state C (reactor shutdown, but pressurized), a pipe break accident up to the DBA can be accommodated within the nuclear safety

operational criteria through the various operations of the MSIVs; ECCSs (HPCI, ADS, LPCI, and core spray); PCRVICS; containment, Reactor Building enclosure; FRVS; Control Room Heating, Cooling and Ventilation System; emergency and RHR systems; Safety Auxiliaries Cooling System (SACS); Station Service Water System (SSWS); Hydrogen Control System; and the IDC. For small pipe breaks inside the containment, pressure relief is effected by the Nuclear System Pressure Relief System, which transfers decay heat to the suppression pool. For large breaks, depressurization takes place through the break itself. In state D (reactor not shut down, but pressurized), the same equipment is required as in state C but, in addition, the RPS and the CRD system must operate to trip the reactor. The limiting items, upon which the operation of the above equipment is based, are the allowable fuel cladding temperature and the containment pressure capability. The CRD housing supports are considered necessary whenever the system is pressurized to prevent excessive control rod movement through the RPV following the postulated rupture of one CRD housing, a lesser case of the design basis LOCA related prevention of a postulated control rod ejection and an accident.

After completion of the automatic action of the above equipment, manual operation of the RHR system (suppression pool cooling mode) and SRVs through ADS or manual actuation (controlled depressurization) is required to maintain containment pressure and fuel cladding temperature within limits during extended core cooling.

4. Events 43, 44, 45, LOCA resulting from postulated pipe breaks outside containment - Pipe breaks outside the containment are assumed only to occur any time the nuclear system is pressurized (states C and D). This accident is most severe during operation at high power

(state D). In state C, this accident becomes a subset of the state D sequence.

The protection sequences for the various possible pipe breaks outside the containment are shown on Figure 15.9-44. The sequences also show that for small breaks, the control room operator can use a large number of process indications to identify and isolate the break.

In operating state D (reactor not shut down, but pressurized), reactor trip is accomplished through operation of the RPS and the CRD system. Reactor vessel isolation is accomplished through operation of the MSIVs and the containment and reactor vessel isolation control system.

For a main steam line break, initial core cooling is accomplished by HPCI or ADS/manual SRV operation in conjunction with core spray and LPCI. These systems provide parallel paths to effect initial core cooling, thereby satisfying the single failure criterion. Extended core cooling is accomplished by the single failure proof, parallel combination of core spray and LPCI. The ADS or manual SRV system operation, and the RHR suppression pool cooling mode (both manually operated), are required to maintain containment pressure and fuel cladding temperature within limits during extended core cooling.

5. Event 46, main condenser air removal system leak or failure - It is assumed that the line leading to the steam jet air ejector (SJAE) fails near the main condenser. This results in activity normally processed by the Off-gas Treatment System being discharged directly to the turbine enclosure and, subsequently, through the ventilation system to the environment. This failure results in a loss of flow signal to the gaseous radwaste

system. This event can be considered only under states C and D and is shown on Figure 15.9-45.

The control room operator initiates an emergency shutdown of the reactor to reduce the gaseous activity being discharged. A loss of main condenser vacuum will result (timing depending on leak rate) in a main turbine trip and ultimately a reactor shutdown. Refer to event 26 for the reactor protection sequence shown on Figure 15.9-33.

6. Event 47, augmented Off-gas Treatment System failure - An evaluation of those events that could cause a gross failure in the Off-gas Treatment System has resulted in the identification of a postulated seismic event, more severe than the one for which the system is designed, as the only conceivable event that could cause significant damage.

   The detected gross failure of this system will result in manual isolation of the system from the main condenser. The isolation results in high main condenser pressure and ultimately a reactor trip. Protection sequences for this event are shown on Figure 15.9-46.

   The undetected postulated failure soon results in a system isolation necessitating reactor shutdown because of loss of vacuum in the main condenser. This transient has been analyzed in event 26 as shown on Figure 15.9-33.

7. Event 48, liquid radwaste system leak or failure - Releases that could occur inside and outside of the containment, not covered by events 40 through 47, include small spills and equipment leaks of radioactive materials inside structures housing the subject process equipment. Conservative values for leakage have been assumed and evaluated in the plant under routine releases, as discussed in Section 2.4. The offsite dose

that results from any small spill that could occur outside containment will be negligible in comparison to the dose resulting from the accountable (expected) plant leakages.

The protective sequences for this event are provided on Figure 15.9-47.

8. Event 49, Liquid radwaste system - storage tank failure - Refer to Section 2.4 for a discussion of liquid tank failures.

   The protective sequences for this event are provided on Figure 15.9-48.

15.9.6.6 Special Events

15.9.6.6.1 General

Additional special events are postulated to demonstrate that the plant is capable of accommodating off design occurrences, such as events 50 through 53. As such, these events are beyond the safety requirements of the other event categories. The safety actions shown in the sequence diagrams on Figures 15.9-49 through 15.9-51 for the additional special events follow directly from the requirements cited in the demonstration of the plant's capability.

Auxiliary system support analyses are shown on Figures 15.9-7 and 15.9-52 through 15.9-57.

15.9.6.6.2 Required Safety Action/Unacceptable Consequences

The following list relates the safety actions for special events to prevent the unacceptable consequences cited in Table 15.9-10:

Safety Action	Related Unacceptable Consequence	Reason For Action Available
Main Control Room Considerations		
Manually initiate all shutdown controls from local panels	5-1 5-2	Local panel control has been provided and is available outside main control room
Manually initiate Standby Liquid Control (SLC) System	5-3	SLC system to control reactivity to cold shutdown is available

15.9.6.6.3 Event Definitions and Operational Safety Evaluation

1. Event 50, shipping cask drop - The spent fuel cask is equipped with redundant sets of lifting lugs and yokes compatible with the reactor building crane main hook, thus preventing a cask drop due to a single active failure.

2. Event 51, reactor shutdown anticipated transient without scram (ATWS) - Reactor shutdown from a plant transient occurrence e.g., turbine trip, without the use of control rods is an event currently being evaluated to determine the capability of the plant to be safely shut down. The event is applicable in operating states C and D. Figure 15.9-49 shows the protection sequence for this extremely improbable and demanding event in each operating state.

State D is the most limiting case. Upon initiation of the plant transient situation, on turbine trip, a reactor trip will be initiated, but no control rods are assumed to move. The recirculation pumps will be tripped by the initial turbine trip signal. If the nuclear system becomes isolated from the main condenser, low power neutron heat can be transferred from the reactor to the suppression pool via the SRVs. The IDC initiates operation of the HPCI on low water level, which maintains reactor vessel water level. The SLC system will be automatically initiated from either high reactor vessel pressure or low reactor water level, and the transition from low power neutron heat to decay heat will occur. The RHR suppression pool cooling mode is used to remove the low power neutron and decay heat from the suppression pool as required. When RPV pressure falls to approximately the 100 psig level, the RHR shutdown cooling mode is started and continued to cold shutdown.

3. Event 52, reactor shutdown from outside the main control room - Reactor shutdown from outside the main control room is an event investigated to evaluate the capability of the plant to be safely shut down and cooled to the cold shutdown state from outside the main control room. The event is applicable in operating states A, B, C, and D.

   Figure 15.9-50 shows the protection sequences for this event in operating states B, C, and D. In state A, no sequence is shown, because the reactor is already in the condition finally required for the event. In state C, only cooldown is required, since the reactor is already shut down.

   A reactor trip from outside the main control room can be achieved by opening the ac supply breakers for the RPS. If the RPV becomes isolated from the main condenser,

decay heat is transferred from the reactor to the suppression pool via the SRVs. The incident detection circuitry initiates operation of the RCIC and HPCI systems on low water level, which maintains reactor vessel water level, and the RHR system suppression pool cooling mode is used to remove the decay heat from the suppression pool if required. When the RPV pressure falls below 100 psig level, the RHR system shutdown cooling mode is then started.

4. Event 53, reactor shutdown without control rods - Reactor shutdown without control rods is an event requiring an alternate method of reactivity control: the SLC system. By definition, this event can occur only when the reactor is not already shut down. Therefore, this event is considered only in operating State D.

   The SLC system must operate to avoid unacceptable consequence criteria 5-3. The design bases for the SLC system result from these operating criteria when applied under the most severe conditions (state D at rated power). As indicated on Figure 15.9-51, the SLC system is manually initiated and controlled in state D.

15.9.7 Remainder of Nuclear Safety Operational Analysis

With the information presented in the protection sequence block diagrams, the auxiliary diagrams, and the commonality of auxiliary diagrams, it is possible to determine the exact functional and hardware requirements for each system. This is done by considering each event in which the system is employed and deriving a limiting set of operational requirements. This limiting set of operational requirements establishes the lowest acceptable level of performance for a system or component, or the minimum number of components or portions of a system that must be operable so that plant operation may continue.

The operational requirements derived using the above process may be complicated functions of operating states, parameter ranges, and hardware conditions. The final step is to simplify these complex requirements into technical specifications that encompass the operational requirements and can be used by plant operations and management personnel.

15.9.8 Conclusions

It is concluded that the nuclear safety operational and plant design basis criteria are satisfied when the plant is operated in accordance with the nuclear safety operational requirements determined by the method presented in this section.

15.9.9 References

15.9-1 M. M. Hirsch, "Methods for Calculating Safe Test Intervals and Allowable Repair Times for Engineered Safeguard Systems," NEDO-10739, January 1973.

TABLE 15.9-1

NORMAL OPERATION

NSOA Event Number	Event Description	NSOA Event Figure Number	FSAR Section Number	BWR Operating State[1] A	B	C	D
1	Refueling - Initial - Reload	15.9-9	-	X			
2	Achieving criticality	15.9-9,10,11,12	-	X	X	X	X
3	Heatup	15.9-12					X
4	Power operation, generation - Steady state - Daily load reduction & recovery - Grid frequency control response - Control rod sequence exchanges - Power generation surveillance testing Main stop valve surveillance tests Turbine control valve surveillance tests MSIV surveillance tests	15.9-12	-				X
5	Achieving shutdown	15.9-10,12	-		X		X
6	Cooldown	15.9-9,11	-	X		X	

(1) The BWR operating states are defined in Section 15.9.6.2.4 and summarized in Table 15.9-11.

TABLE 15.9-2

ANTICIPATED OPERATIONAL TRANSIENTS

NSOA Event Number	Event Description	NSOA Event Figure Number	FSAR Section Number	BWR Operating State A	B	C	D
7	Manual or inadvertent reactor trip	15.9-13	7.2	X	X	X	X
8	Loss of plant instrument/ service air systems	15.9-14	9.3.1	X	X	X	X
9	Inadvertent startup of HPCI pump	15.9-15	15.5.1	X	X	X	X
10	Inadvertent startup of idle recirculation pump	15.9-16	15.4.4	X	X	X	X
11	Recirculation flow control failure with increasing flow	15.9-17	15.4.5			X	X
12	Recirculation flow control failure with decreasing flow	15.9-18	15.3.2			X	X
13	Recirculation pump trip - With one pump - With two pumps	15.9-19	15.3.1			X	X
14	Isolation of main steam lines		15.2.4				
	- All main steam lines	15.9-20				X	X
	- One main steam line	15.9-21				X	X
15	Inadvertent opening of a main steam relief valve	15.9-22	15.6.1	X	X	X	X
16	Control rod withdrawal error	15.9-23	15.4.1				
	- During refueling			X			
	- During startup				X		
17	Control rod withdrawal rod error at power	15.9-24	15.4.2			X	X
18	RHR system, loss of shutdown cooling	15.9-25	15.2.9	X	X	X	X
19	RHR system, increased shutdown cooling	15.9-26	15.1.6	X	X	X	X
20	Loss of feedwater flow	15.9-27	15.2.7			X	X
21	Loss of feedwater heating	15.9-28	15.1.1				X

TABLE 15.9-2 (Cont)

NSOA Event Number	Event Description	NSOA Event Figure Number	FSAR Section Number	BWR Operating State A	B	C	D
22	Feedwater controller failure maximum demand	15.9-29	15.1.2	X	X	X	X
23	Pressure regulator failure, open	15.9-30	15.1.3			X	X
24	Pressure regulator failure, closed	15.9-31	15.2.1			X	X
25	Main turbine trip with bypass system operation	15.9-32	15.2.3				X
26	Loss of main condenser vacuum	15.9-33	15.2.5			X	X
27	Main generator trip with bypass system operation	15.9-34	15.2.2				X
28	Loss of normal onsite power	----	15.2.6				
29	Loss of plant normal offsite power, grid loss	15.9-35	15.2.6	X	X	X	X

TABLE 15.9-3

ABNORMAL OPERATIONAL TRANSIENTS

NSOA Event Number	Event Description	NSOA Event Figure Number	FSAR Section Number	BWR Operating State A	B	C	D
30	Main generator trip with bypass system failure	15.9-36	15.2.2				X
31	Main turbine trip with bypass system failure	15.9-37	15.2.3				X
32	Inadvertent loading and operation with fuel assembly in improper position	15.9-38	15.4.7	X	X	X	X
33	Not used	----	----				
34	Not used	----	----				
35	Not used	----	----				
36	Not used	----	----				
37	Not used	----	----				
38	Recirculation pump seizure	15.9-39	15.3.3				X
39	Recirculation pump shaft break	15.9-40	15.3.4				X

TABLE 15.9-4

DESIGN BASIS ACCIDENTS

NSOA Event Number	Event Description	NSOA Event Figure Number	FSAR Section Number	BWR Operating State A	B	C	D
40	Control rod drop accident	15.9-41	15.4.9				X
41	Fuel handling accident	15.9-42	15.7.4	X	X	X	X
42	Loss-of-coolant accident (LOCA), piping breaks within the RCPB inside containment	15.9-43	15.6.5			X	X
43	LOCA, piping breaks outside containment	15.9-44	15.6.4			X	X
44	Instrument line break outside drywell	15.9-44	15.6.2			X	X
45	Feedwater line break outside containment	15.9-44	15.6.6			X	X
46	Main condenser Air Removal System leak or failure	15.9-45	15.7.1			X	X
47	Augmented Gaseous Radwaste System failure	15.9-46	15.7.1	X	X	X	X
48	Liquid Radwaste System leak or failure	15.9-47	15.7.2	X	X	X	X
49	Liquid Radwaste System storage tank failure	15.9-48	15.7.3	X	X	X	X

TABLE 15.9-5

SPECIAL EVENTS

NSOA Event Number	Event Description	NSOA Event Figure Number	FSAR Section Number	BWR Operating State A	B	C	D
50	Shipping cask drop	----	15.7.5				
51	Reactor shutdown from anticipated transient without scram (ATWS)	15.9-49	15.8			X	X
52	Reactor shutdown from outside control room	15.9-50	7.5	X	X	X	X
53	Reactor shutdown without control rods	15.9-51	9.3.5				X

TABLE 15.9-6

UNACCEPTABLE CONSEQUENCES CRITERIA
PLANT EVENT CATEGORY
NORMAL OPERATION

Unacceptable Consequences

1-1 Release of radioactive material to the environs that exceeds the limits of either 10CFR20 or 10CFR50

1-2 Fuel failure to such an extent that, were the freed fission products released to the environs via the normal discharge paths for radioactive material, the limits of 10CFR20 would be exceeded

1-3 Nuclear system stress in excess of that allowed for planned operation by applicable industry codes

1-4 Existence of a plant condition not considered by plant safety analyses

TABLE 15.9-7

UNACCEPTABLE CONSEQUENCES CRITERIA

PLANT EVENT CATEGORY

ANTICIPATED OPERATIONAL TRANSIENTS

Unacceptable Consequences

2-1	Release of radioactive material to the environs that exceeds the limits of 10CFR20
2-2	Any fuel failure calculated as a direct result of the transient analyses
2-3	Nuclear system stress exceeding that allowed for transients by applicable industry codes
2-4	Containment stresses exceeding that allowed for transients by applicable industry codes when containment is required

TABLE 15.9-8

UNACCEPTABLE CONSEQUENCES CRITERIA
PLANT EVENT CATEGORY
ABNORMAL OPERATIONAL TRANSIENTS

Unacceptable Consequences

3-1	Radioactive material release exceeding the guideline values of a small fraction of Regulatory Guide 1.183
3-2[1]	Failure of the fuel barrier as a result of exceeding mechanical or thermal limits
3-3	Nuclear system stresses exceeding that allowed for transients by applicable industry codes
3-4	Containment stresses exceeding that allowed for accidents by applicable industry codes when containment is required

---

(1) Failure of the fuel barrier means gross core wide fuel cladding perforations.

TABLE 15.9-9

UNACCEPTABLE CONSEQUENCES CRITERIA
PLANT EVENT CATEGORY
DESIGN BASIS ACCIDENTS

	Unacceptable Consequences
4-1	Radioactive material release exceeding the guideline values of Regulatory Guide 1.183
4-2[1]	Failure of the fuel barrier as a result of exceeding mechanical or thermal limits
4-3	Nuclear system stresses exceeding that allowed for accidents by applicable industry codes
4-4	Containment stresses exceeding that allowed for accidents by applicable industry codes when containment is required
4-5	Overexposure to radiation of plant main control room personnel

(1) Failure of the fuel barrier includes fuel cladding fragmentation (loss-of-coolant accident) and excessive fuel enthalpy (control rod drop accident).

TABLE 15.9-10

UNACCEPTABLE CONSEQUENCES CONSIDERATIONS
PLANT EVENT CATEGORY
SPECIAL EVENTS

Special Events Considered

A. Reactor shutdown from outside main control room

B. Reactor shutdown without control rods

C. Reactor shutdown with anticipated transient without scram (ATWS)

D. Shipping cask drop

Capability Demonstration

5-1 Ability to shut down reactor by manipulating controls and equipment outside the main control room

5-2 Ability to bring the reactor to the cold shutdown condition from outside the main control room

5-3 Ability to shut down the reactor independent of control rods

5-4 Ability to contain radiological contamination

5-5 Ability to limit radiological exposure

TABLE 15.9-11

BWR OPERATING STATES[1]

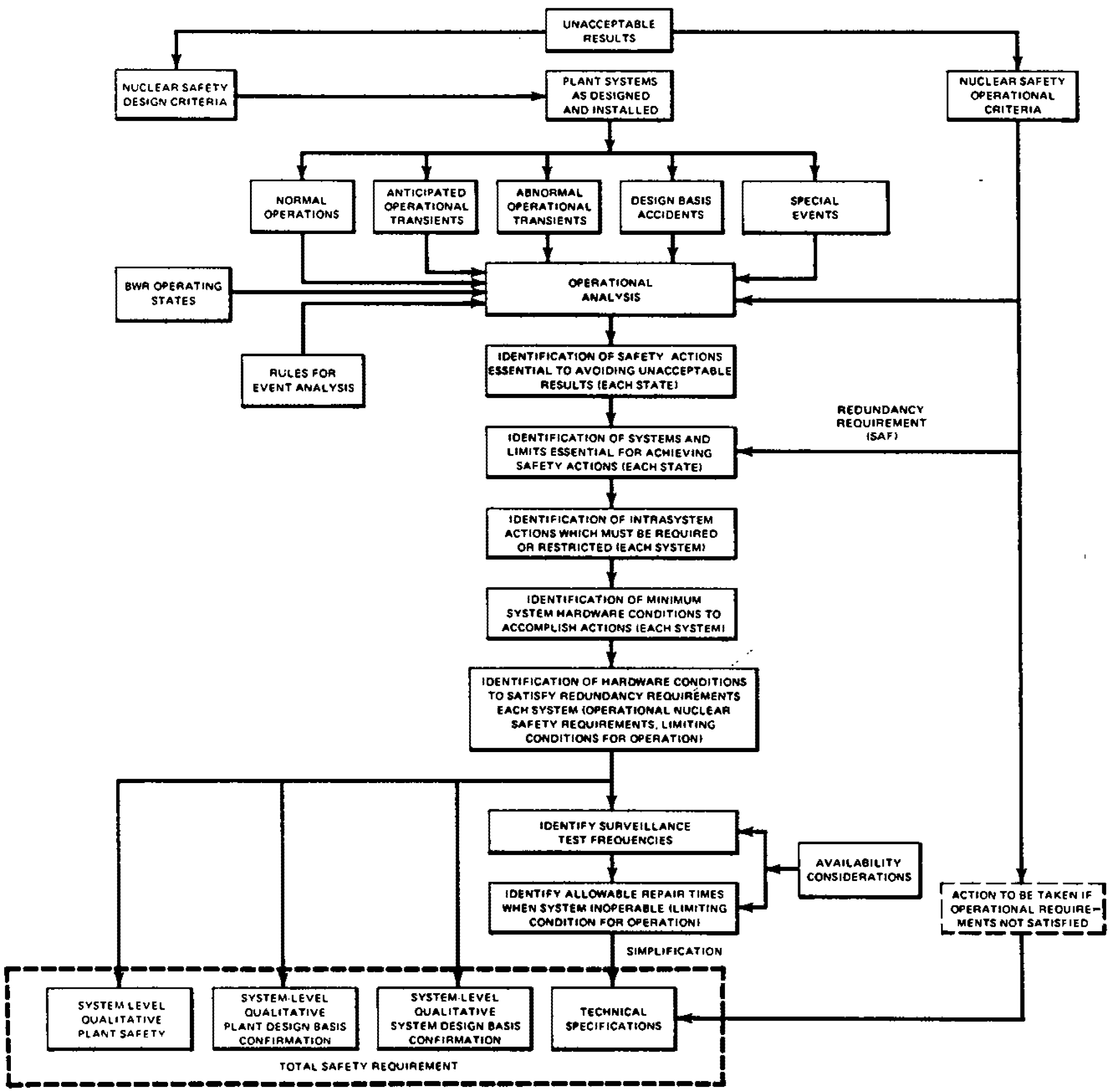
Conditions	States A	B	C	D
Reactor vessel head off	X	X		
Reactor vessel head on			X	X
Shutdown	X		X	
Not shut down		X		X

Definition

Shutdown: K_{eff} sufficiently less than 1 that the full withdrawal of any one control rod could not produce criticality under the most restrictive potential conditions of temperature, pressure, core age, and fission product concentrations.

---

(1) Further discussion is provided in Section 15.9.6.2.4.



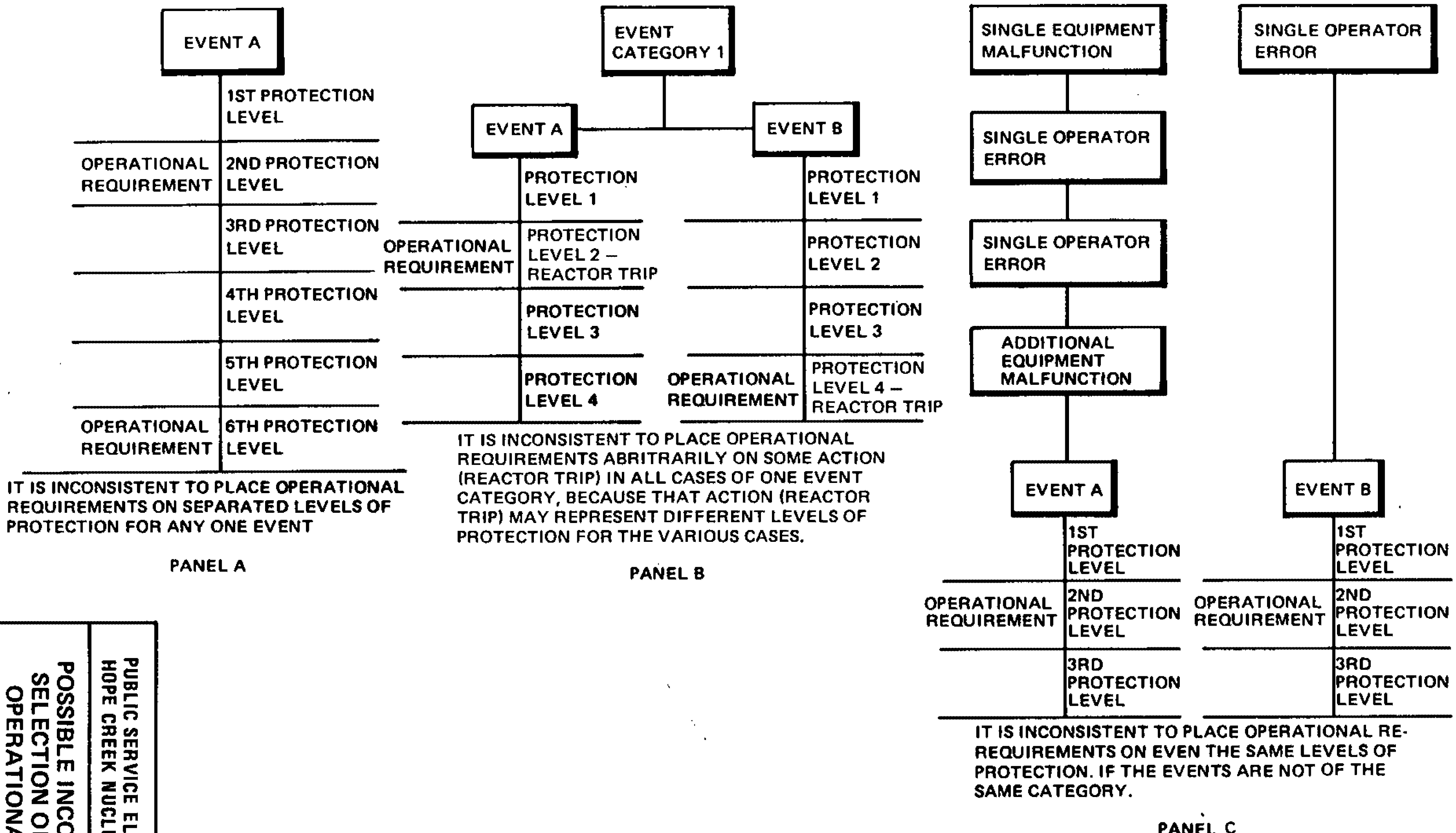
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

BLOCK DIAGRAM OF METHOD USED TO DERIVE NUCLEAR SAFETY OPERATIONAL REQUIREMENTS SYSTEM, LEVEL QUALITATIVE FMEA, DESIGN BASIS CONFIRMATION AUDITS AND TECHNICAL SPECIFICATIONS

UPDATED FSAR

FIGURE 15.9-1



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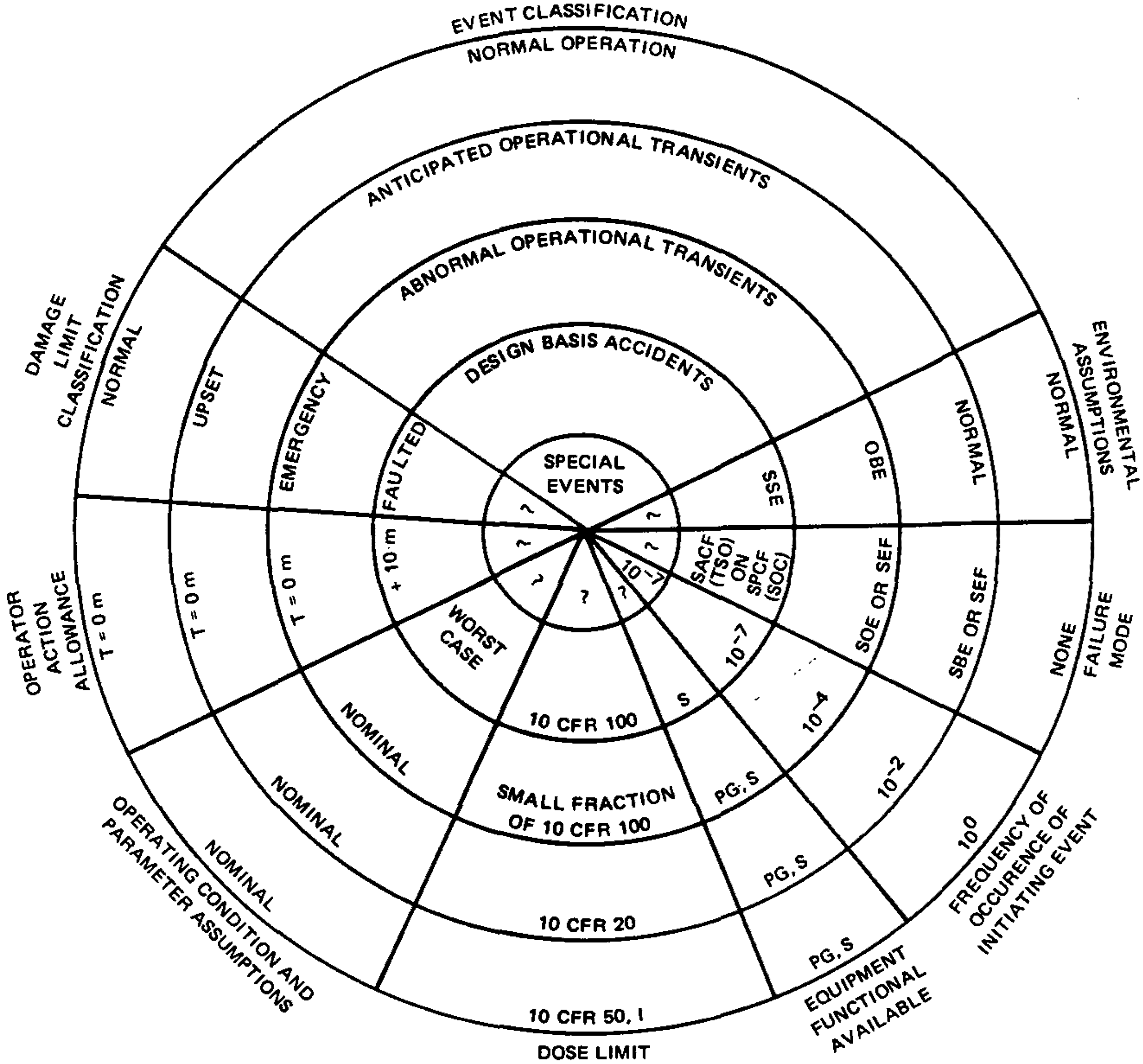
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

POSSIBLE INCONSISTENCIES IN THE SELECTION OF NUCLEAR SAFETY OPERATIONAL REQUIREMENTS

UPDATED FSAR

FIGURE 15.9-2



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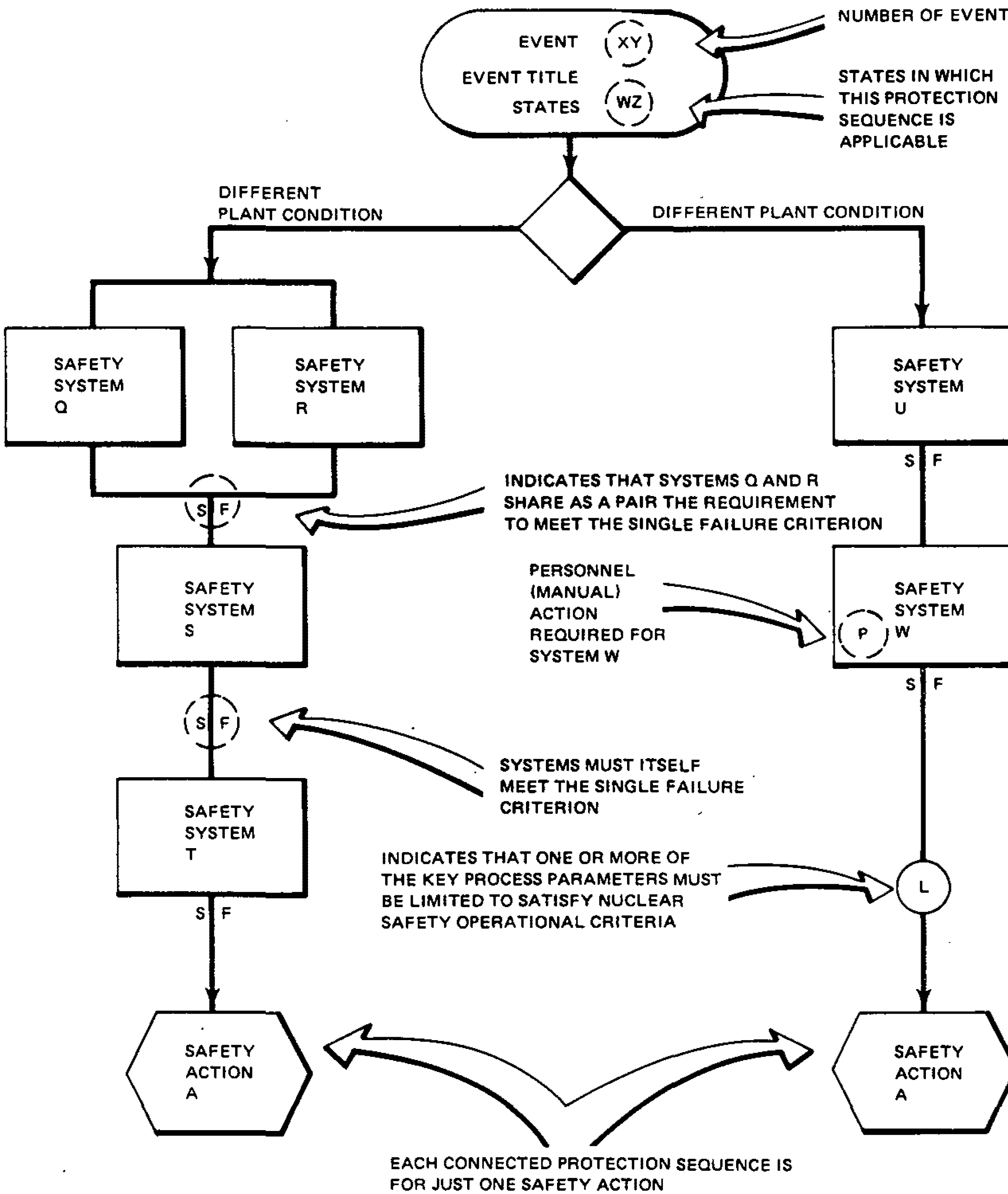
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

SIMPLIFIED NSOA CLASSIFICATION INTERRELATIONSHIPS

UPDATED FSAR

FIGURE 15.9-3



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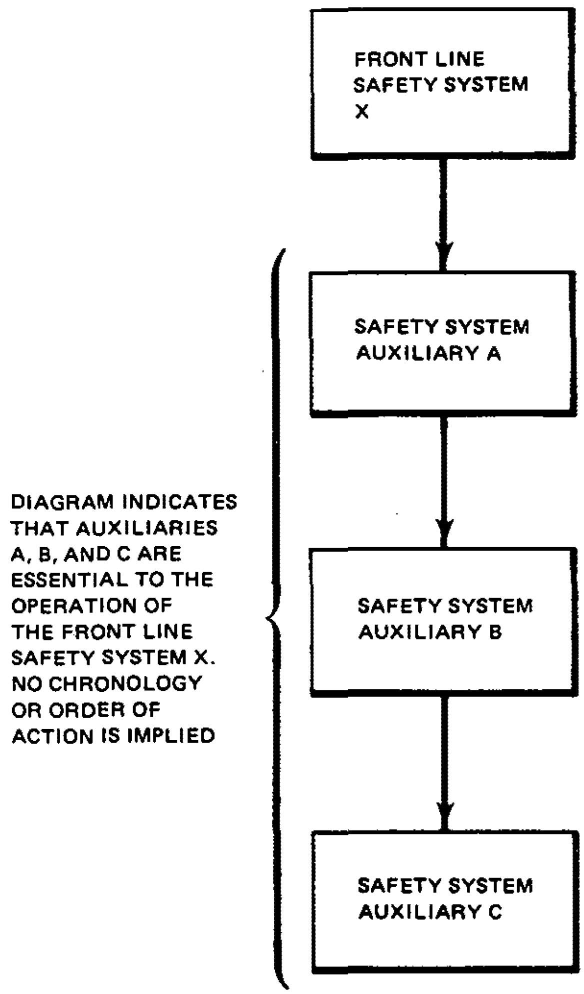
HOPE CREEK NUCLEAR GENERATING STATION

FORMAT FOR PROTECTION SEQUENCE DIAGRAMS

UPDATED FSAR

FIGURE 15.9-4



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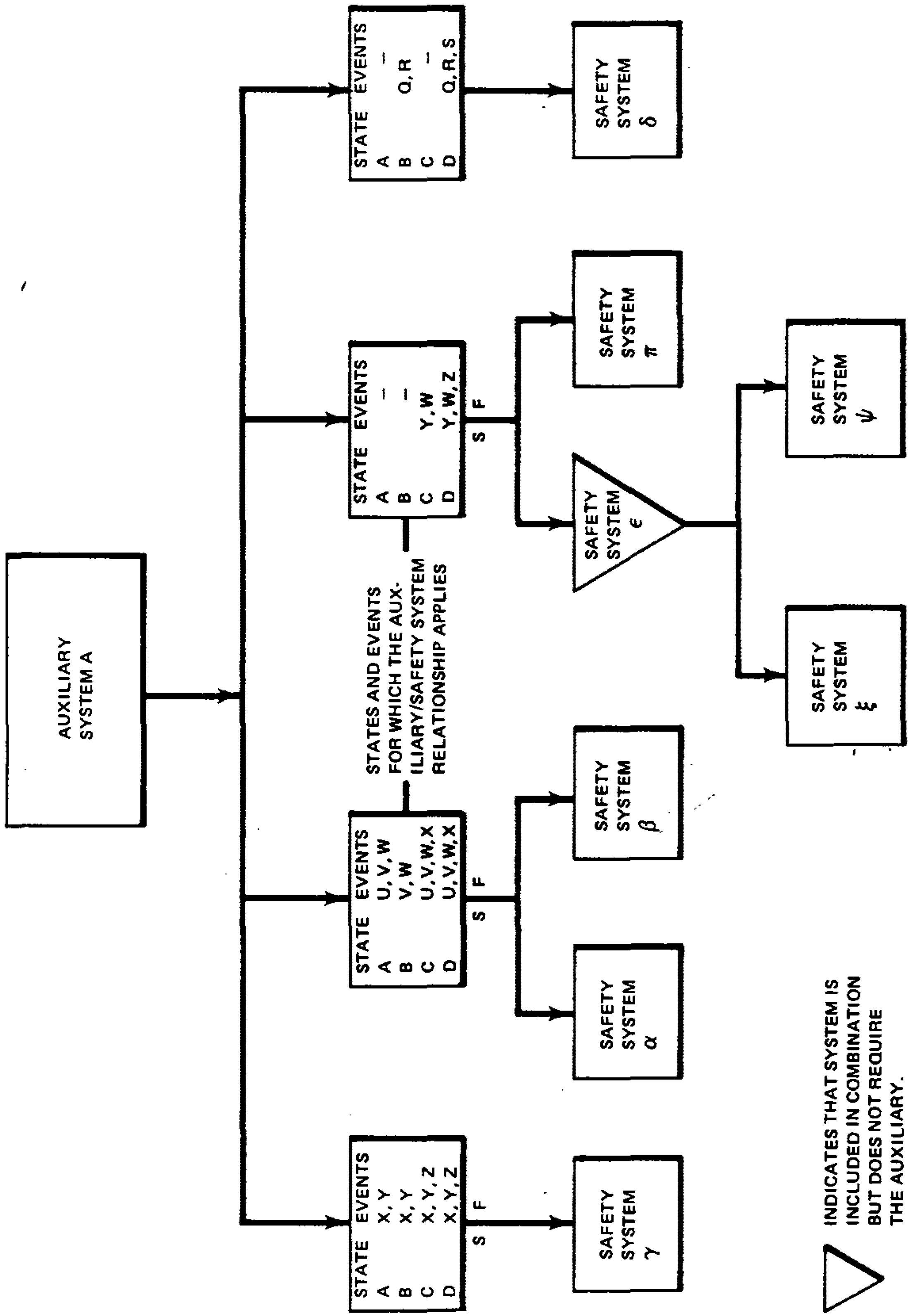
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FORMAT FOR SAFETY SYSTEM AUXILIARY DIAGRAMS

UPDATED FSAR

FIGURE 15.9-5



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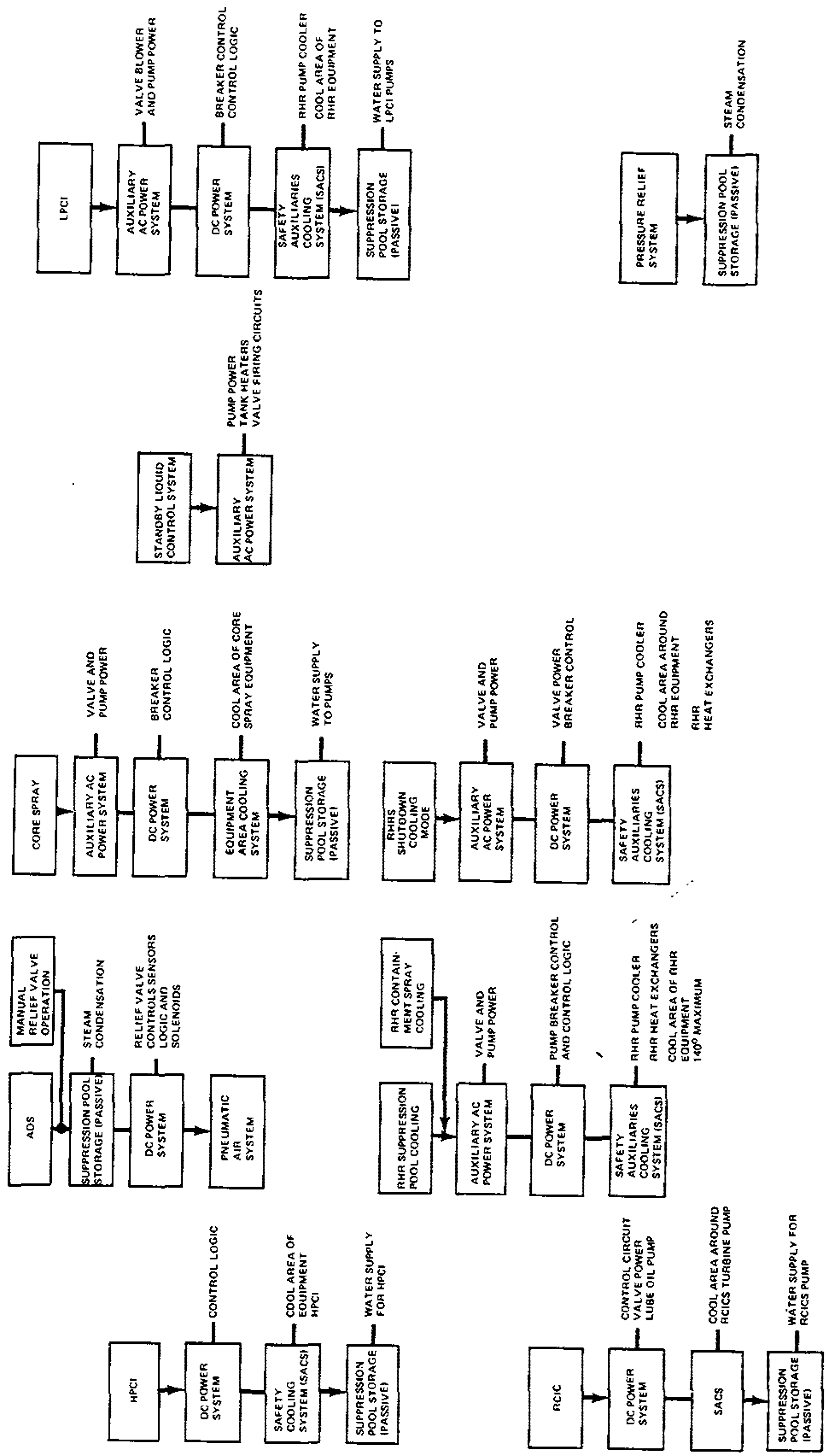
HOPE CREEK NUCLEAR GENERATING STATION

FORMAT FOR COMMONALITY OF AUXILIARY DIAGRAMS

UPDATED FSAR

FIGURE 15.9-6



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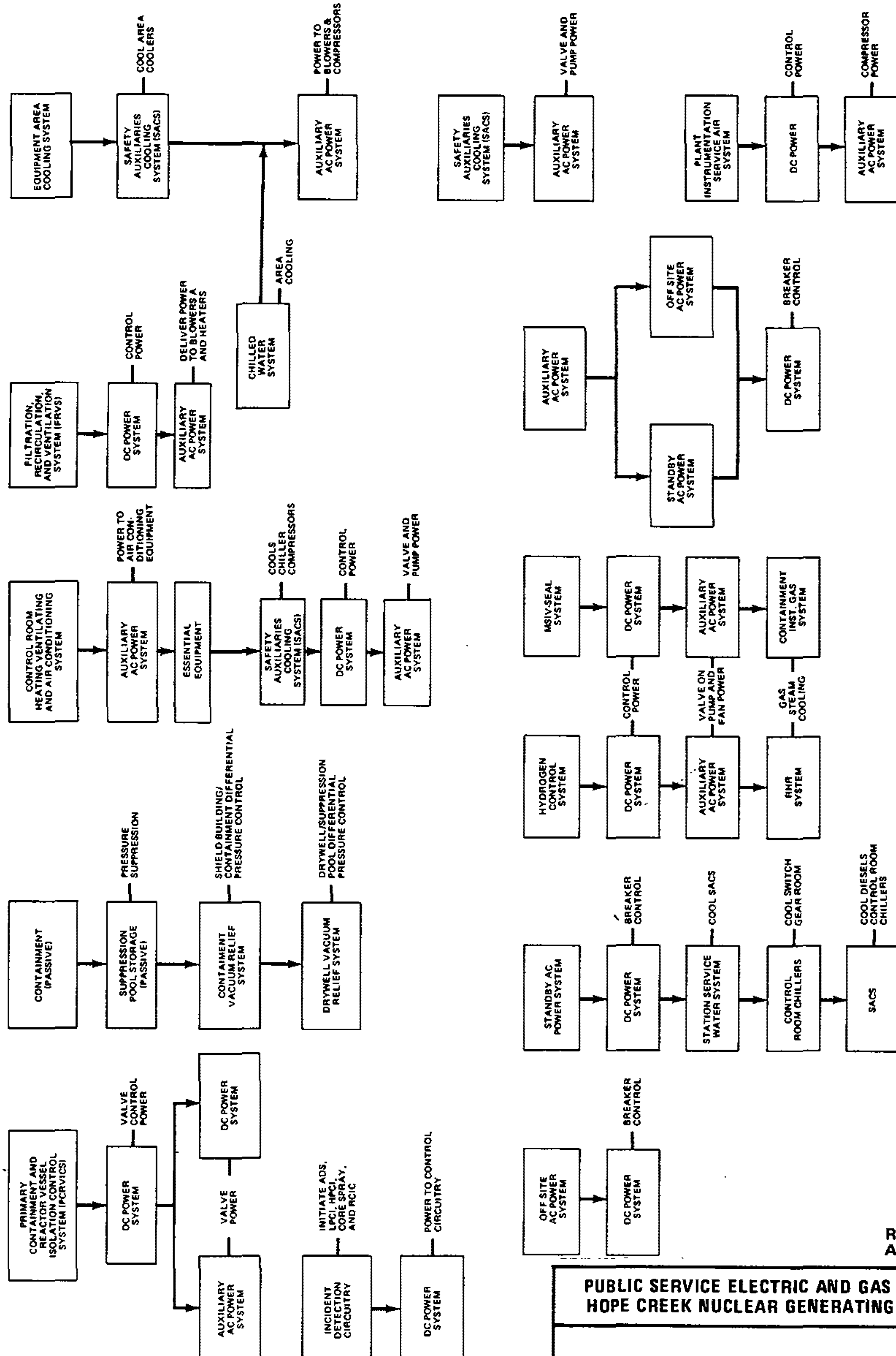
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

SAFETY SYSTEM AUXILIARIES

UPDATED FSAR

FIGURE 15.9-7



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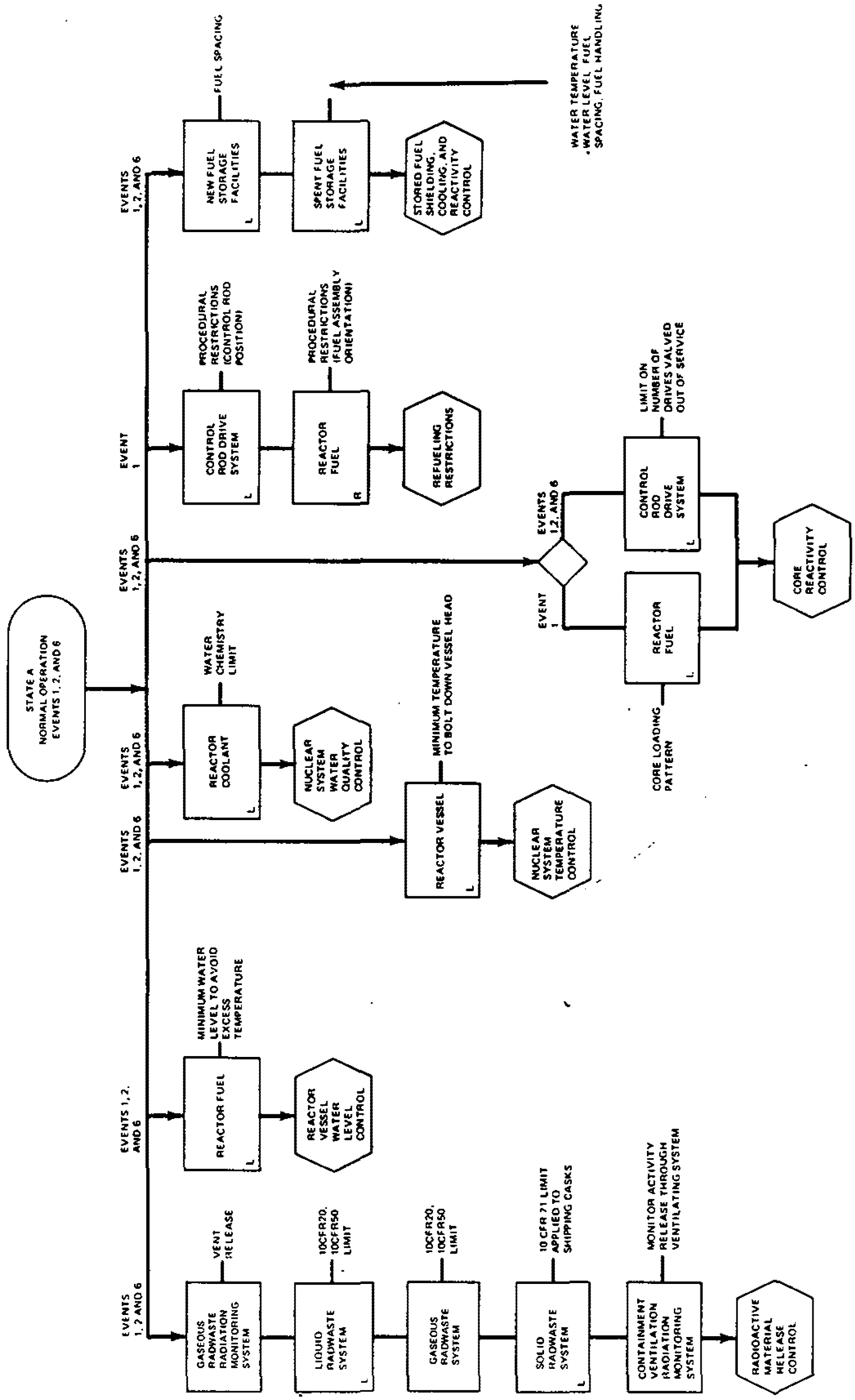
HOPE CREEK NUCLEAR GENERATING STATION

SAFETY SYSTEM AUXILIARIES

UPDATED FSAR

FIGURE 15.9-8



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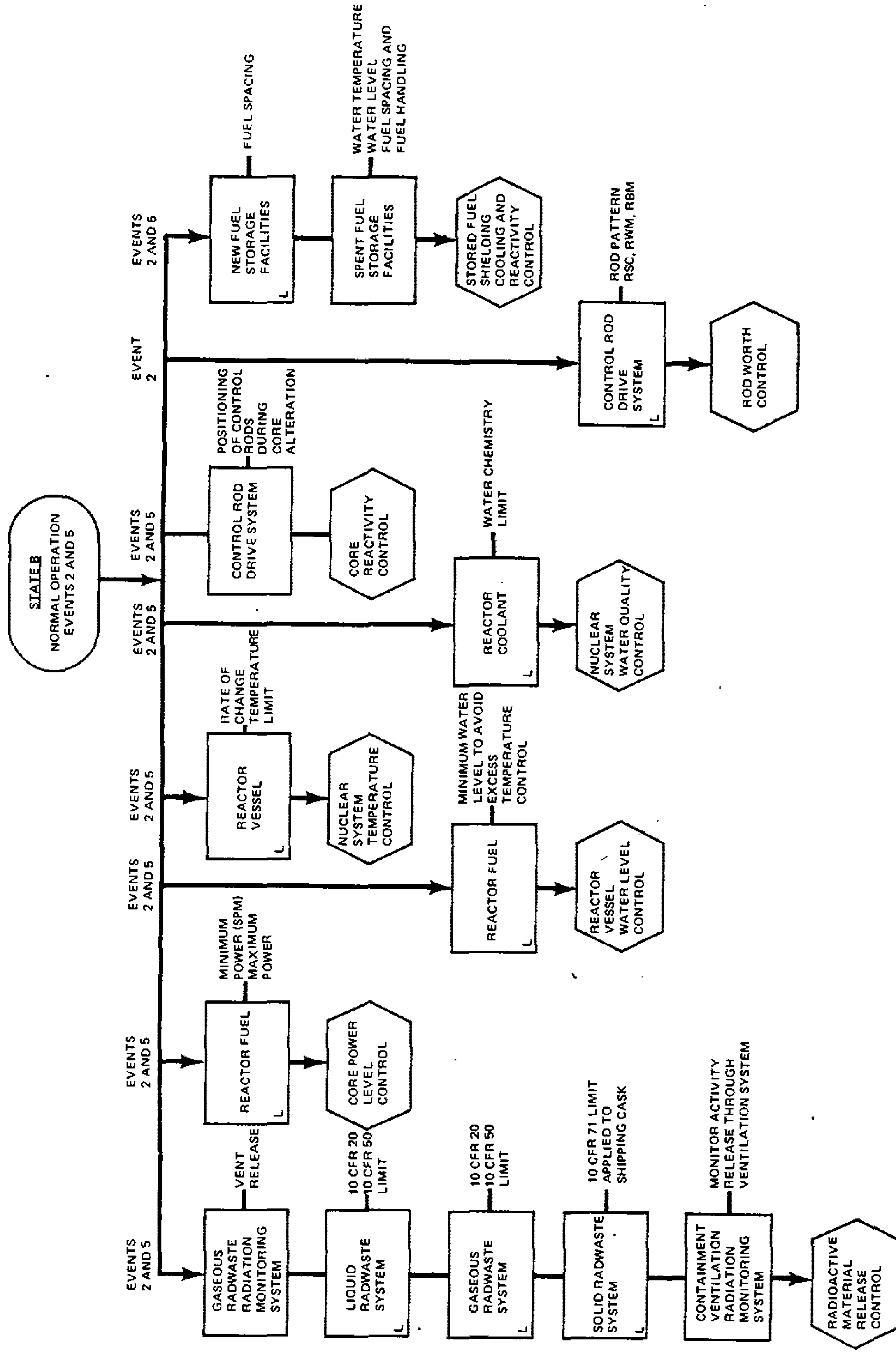
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

SAFETY ACTION SEQUENCES FOR NORMAL OPERATION IN STATE A

UPDATED FSAR

FIGURE 15.9-9



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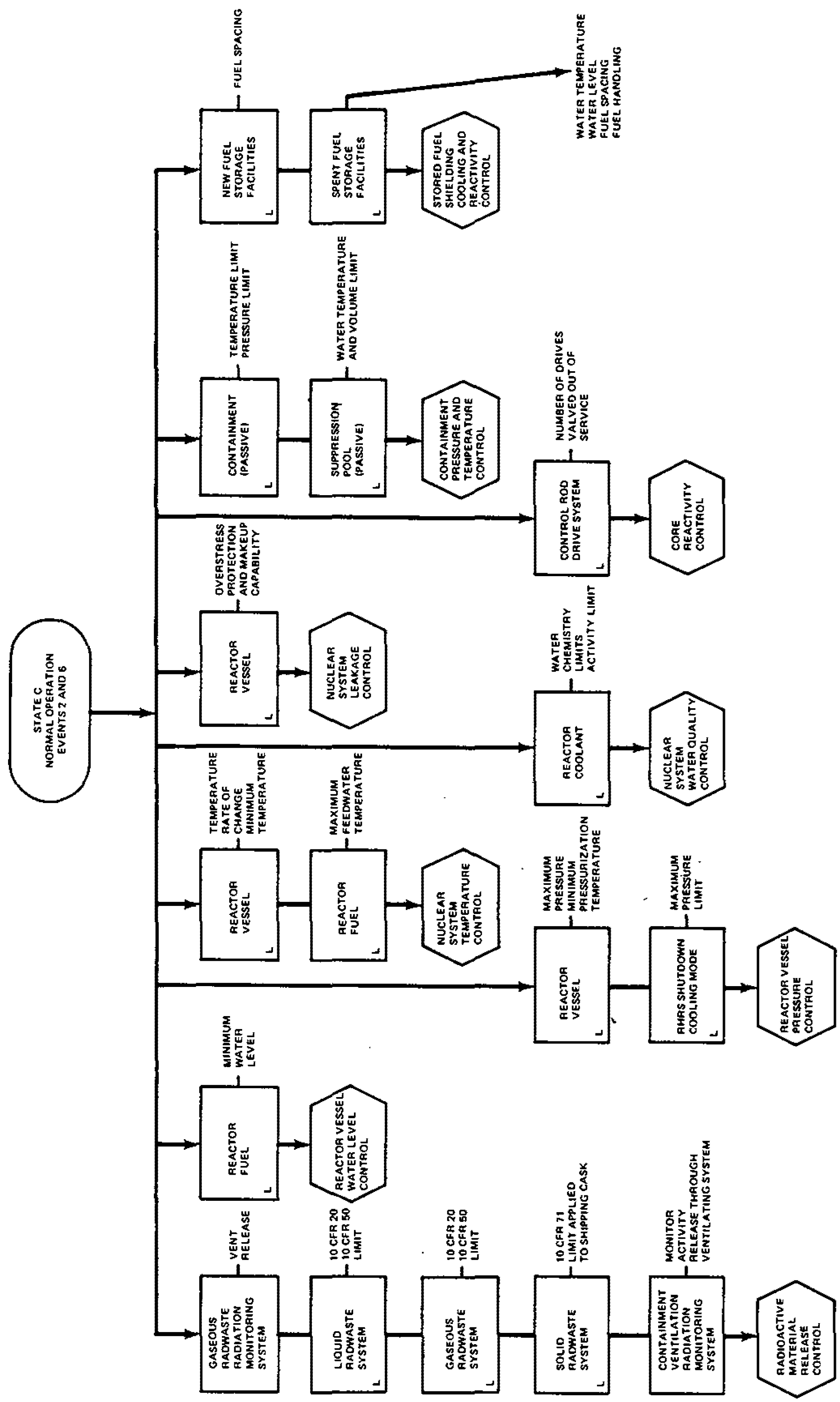
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

SAFETY ACTION SEQUENCES FOR NORMAL OPERATION IN STATE B

UPDATED FSAR

FIGURE 15.9-10



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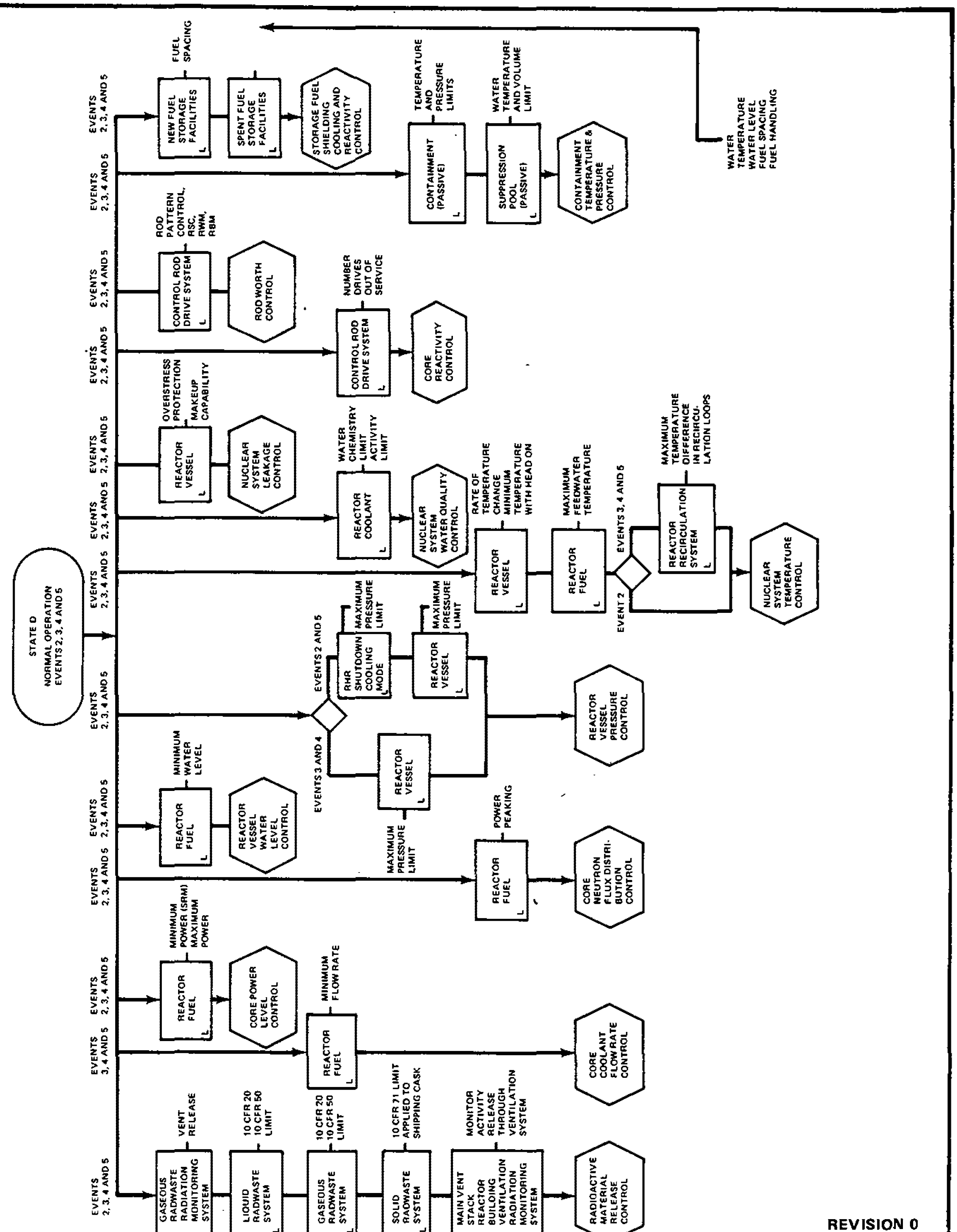
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

SAFETY ACTION SEQUENCES FOR NORMAL OPERATION IN STATE C

UPDATED FSAR

FIGURE 15.9-11



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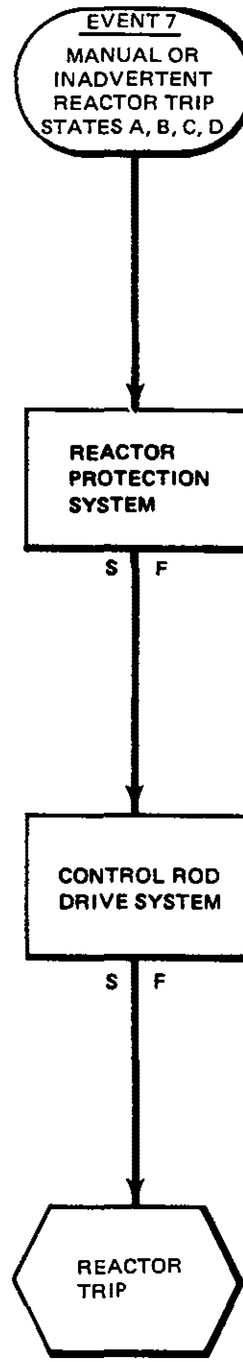
HOPE CREEK NUCLEAR GENERATING STATION

SAFETY ACTION SEQUENCES FOR NORMAL OPERATION IN STATE D

UPDATED FSAR

FIGURE 15.9-12



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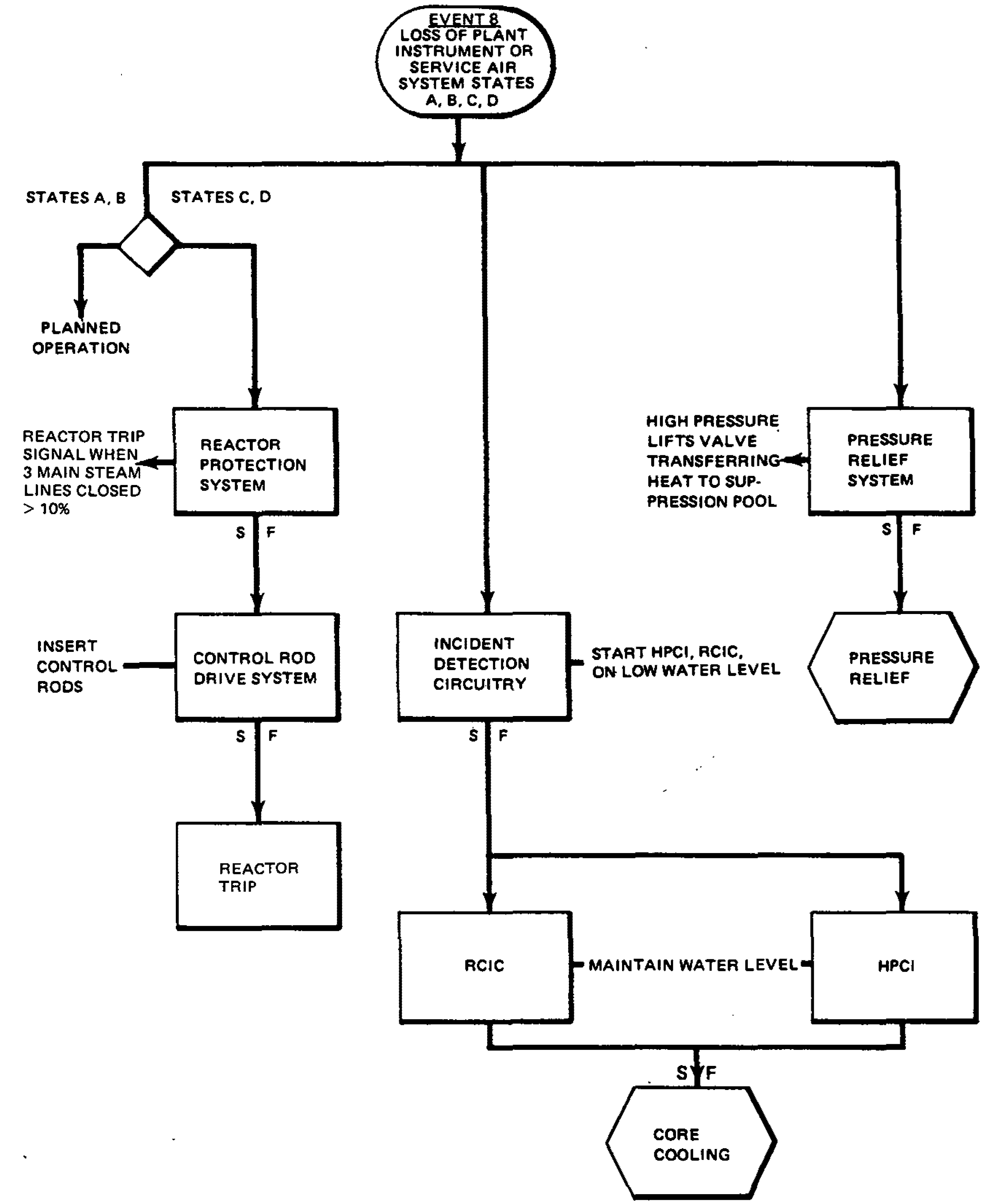
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCE FOR
MANUAL OR INADVERTENT SCRAM

UPDATED FSAR

FIGURE 15.9-13



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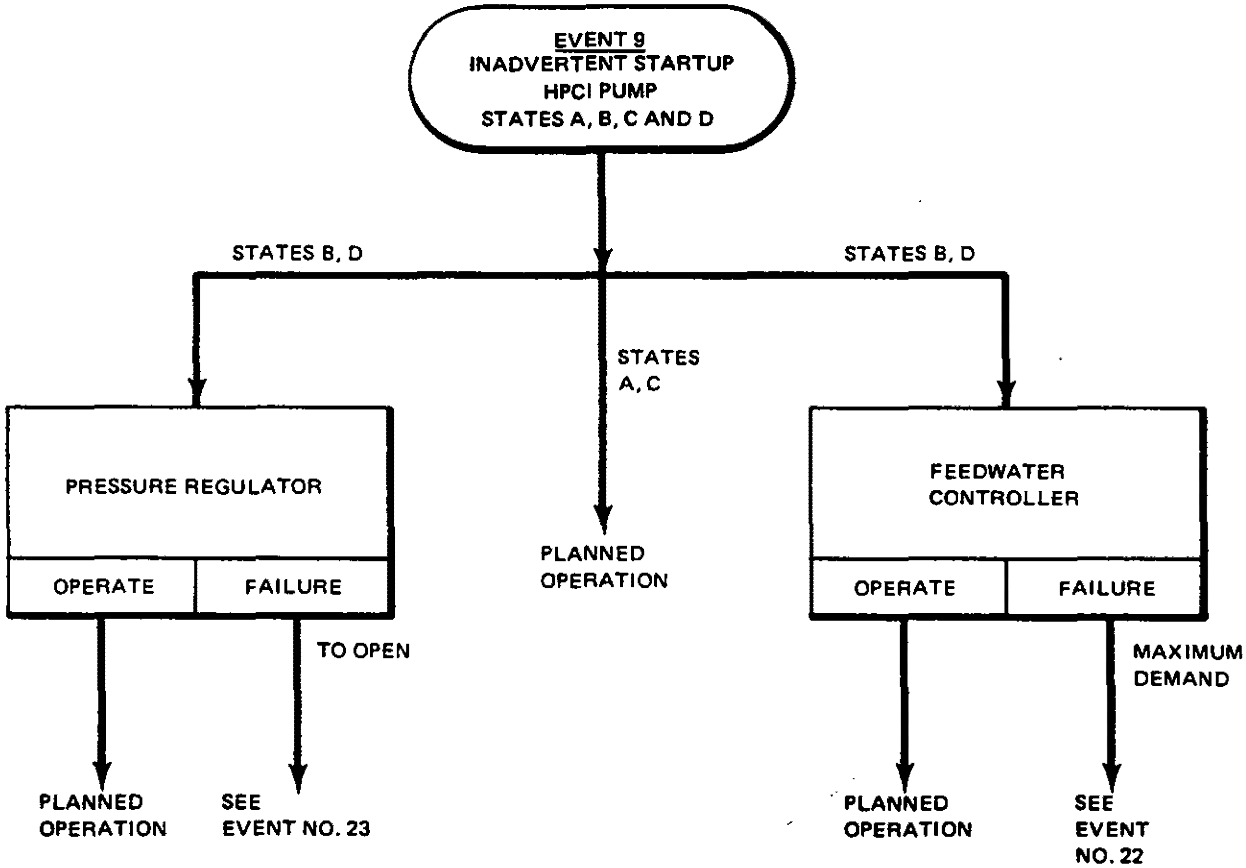
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR LOSS OF PLANT INSTRUMENT OR SERVICE AIR SYSTEM

UPDATED FSAR

FIGURE 15.9-14



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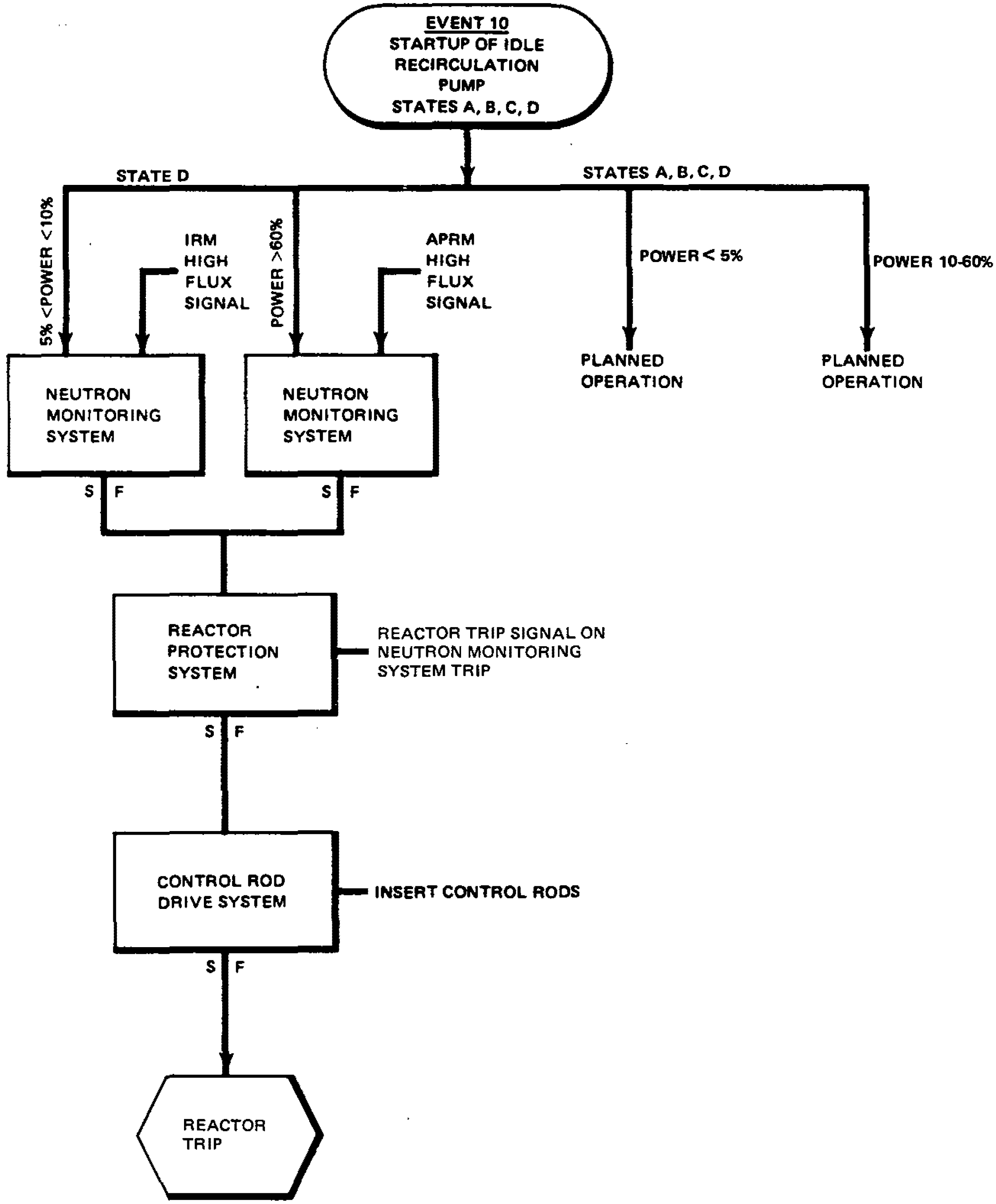
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCE FOR INADVERTENT STARTUP OF HPCI PUMP

UPDATED FSAR

FIGURE 15.9-15



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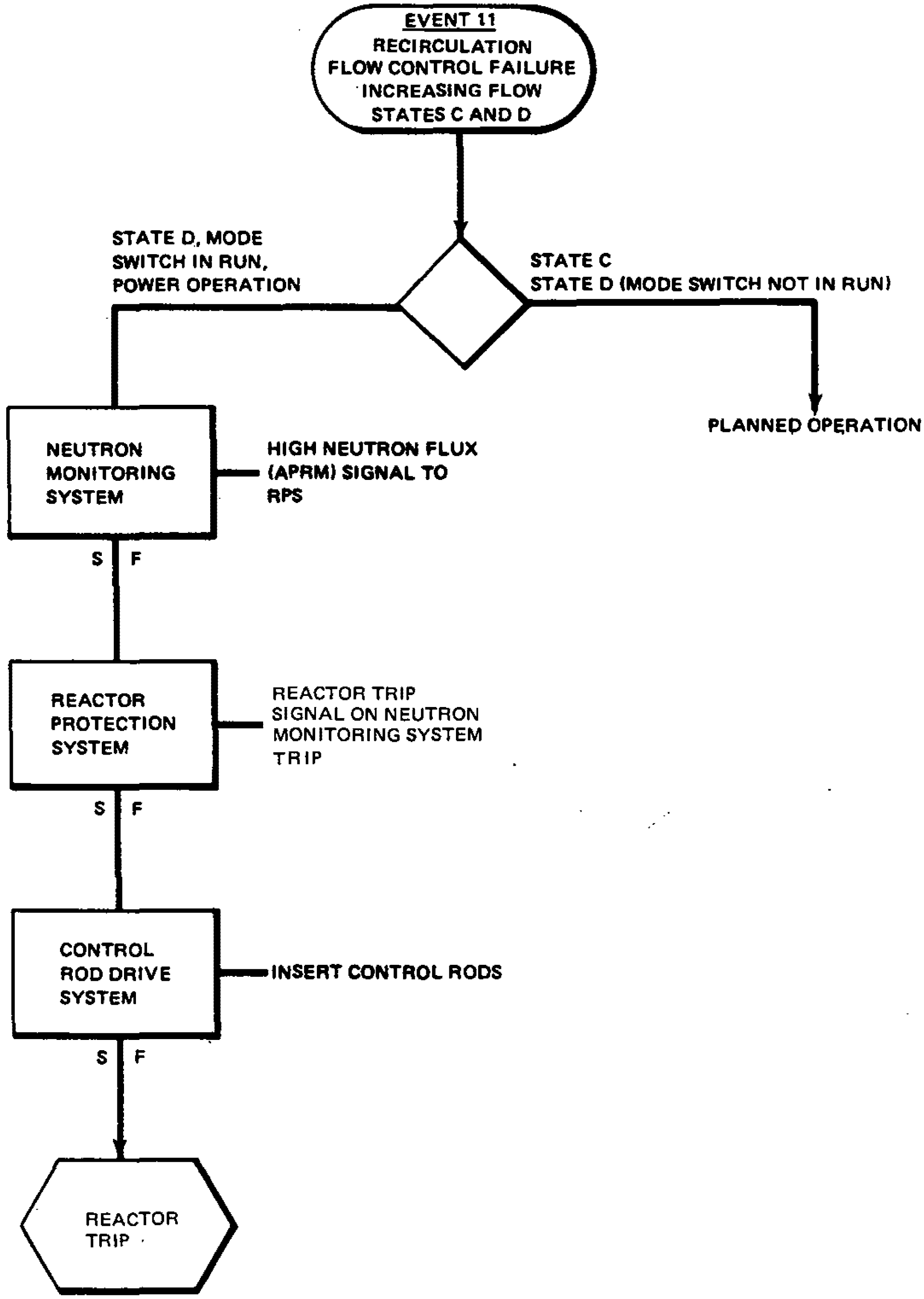
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR INADVERTENT STARTUP OF IDLE RECIRCULATION PUMP

UPDATED FSAR

FIGURE 15.9-16



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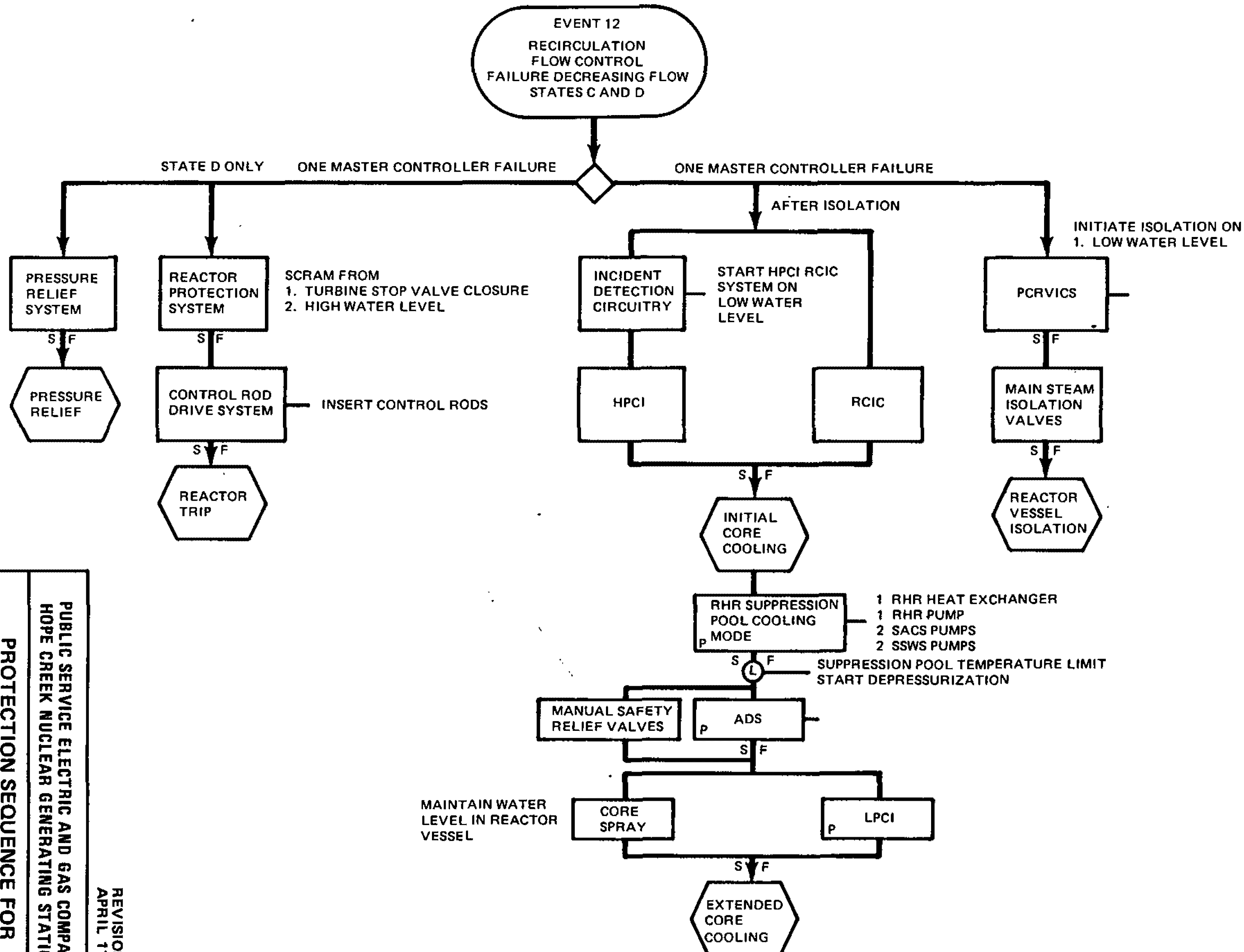
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCE FOR
RECIRCULATION FLOW CONTROL
FAILURE – INCREASING FLOW

UPDATED FSAR

FIGURE 15.9-17



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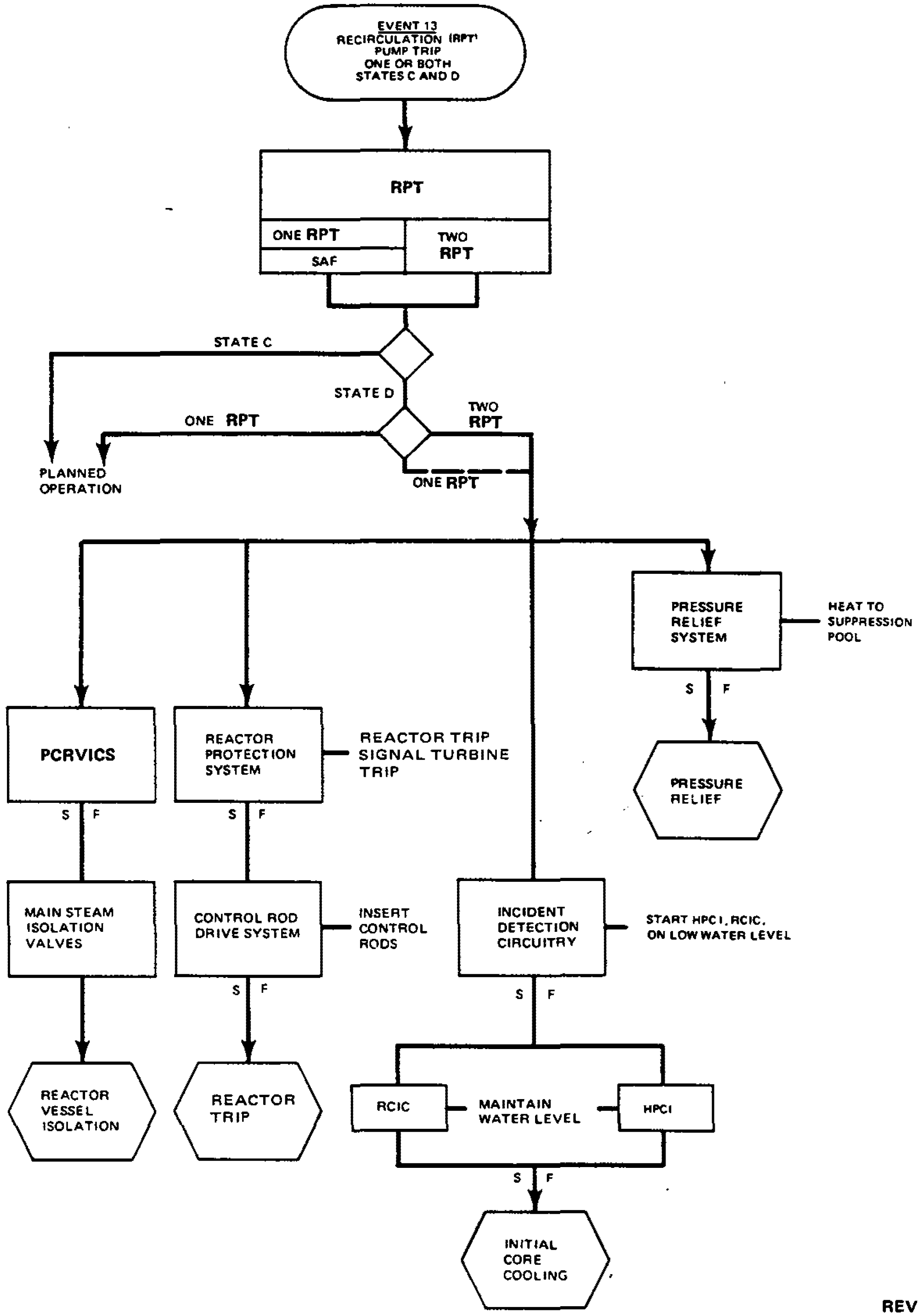
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCE FOR RECIRCULATION FLOW CONTROL FAILURE – DECREASING FLOW

UPDATED FSAR

FIGURE 15.9-18



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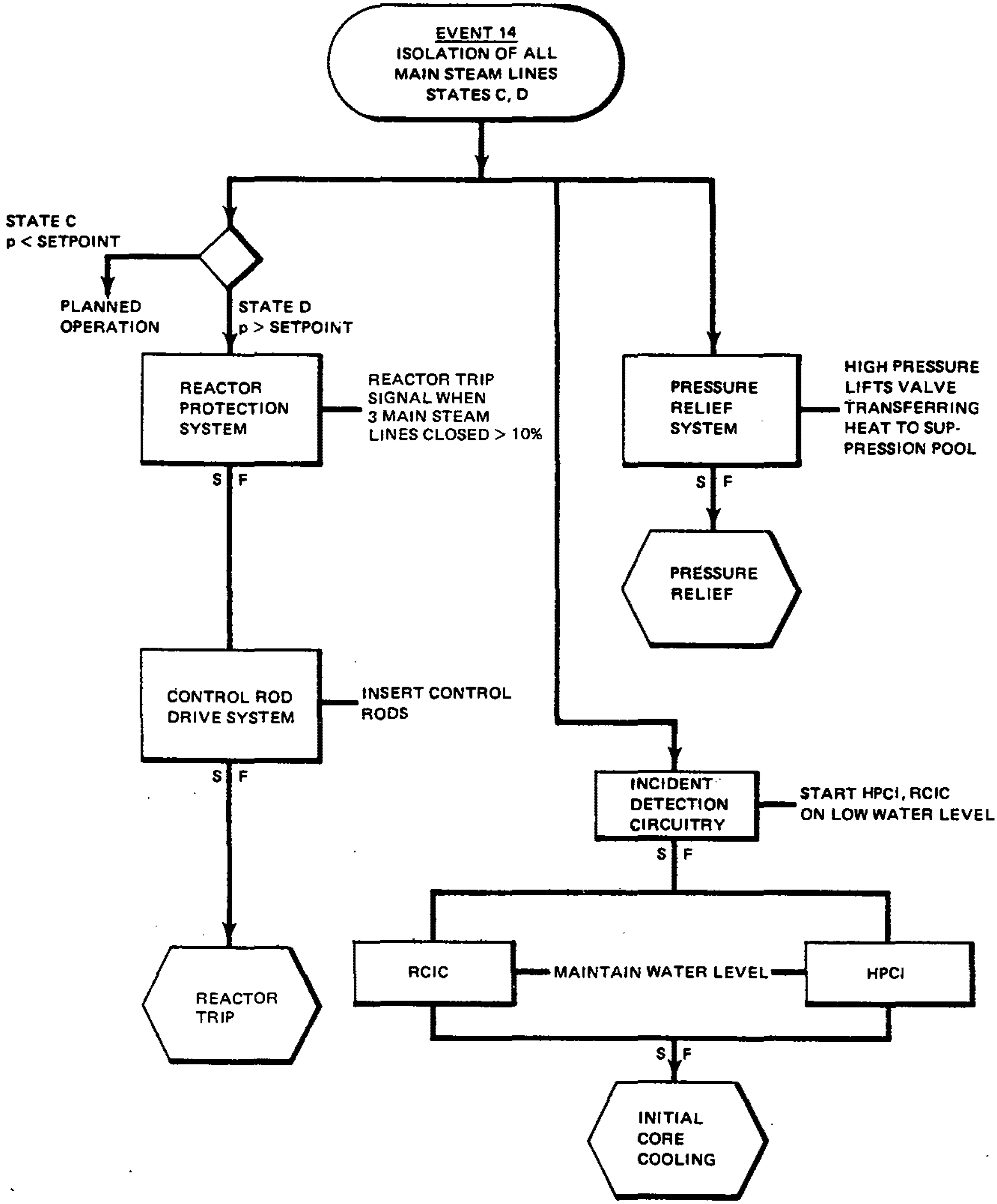
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR RECIRCULATION PUMP TRIP (ONE OR BOTH)

UPDATED FSAR

FIGURE 15.9-19



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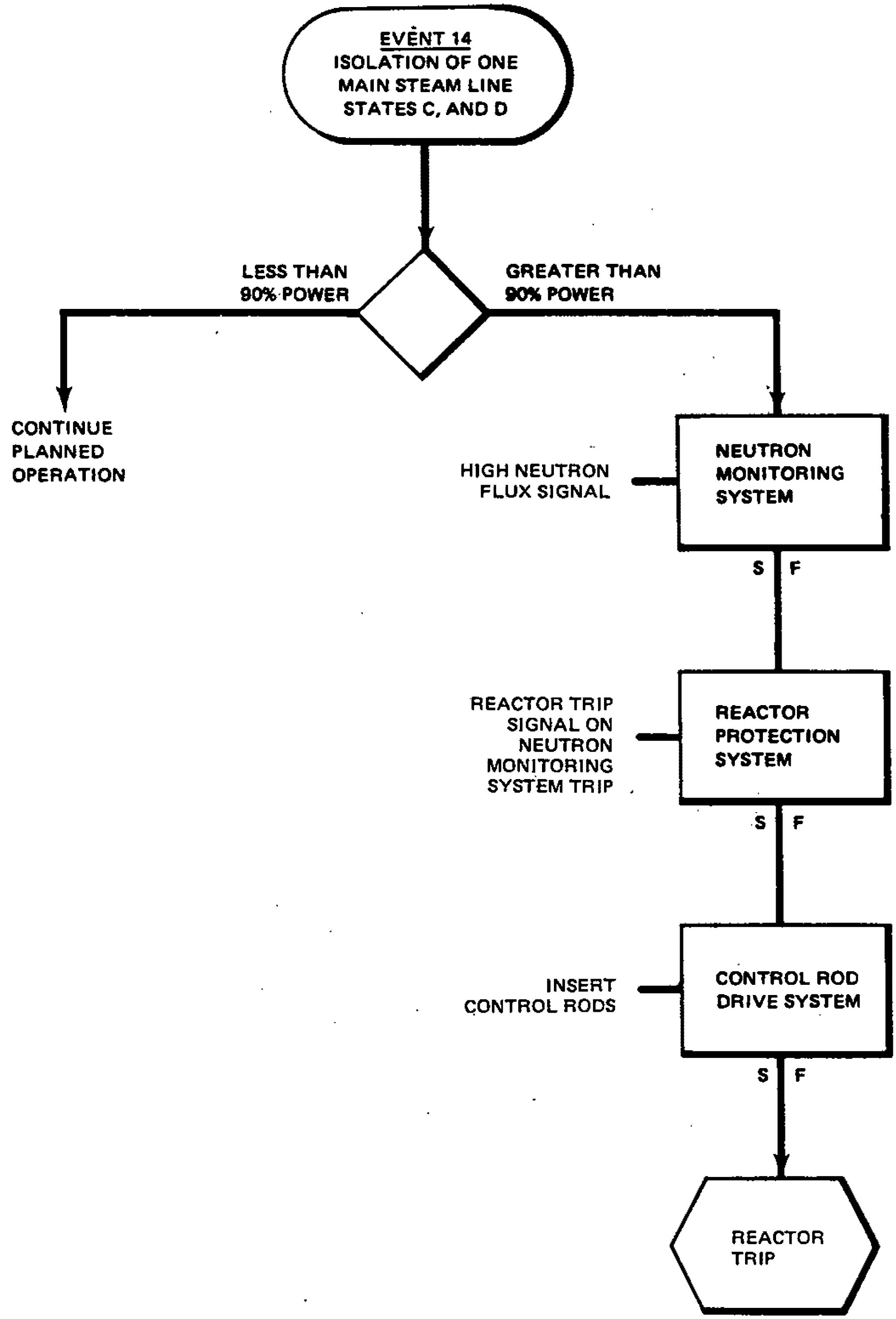
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR
ISOLATION OF ALL
MAIN STEAM LINES

UPDATED FSAR

FIGURE 15.9-20



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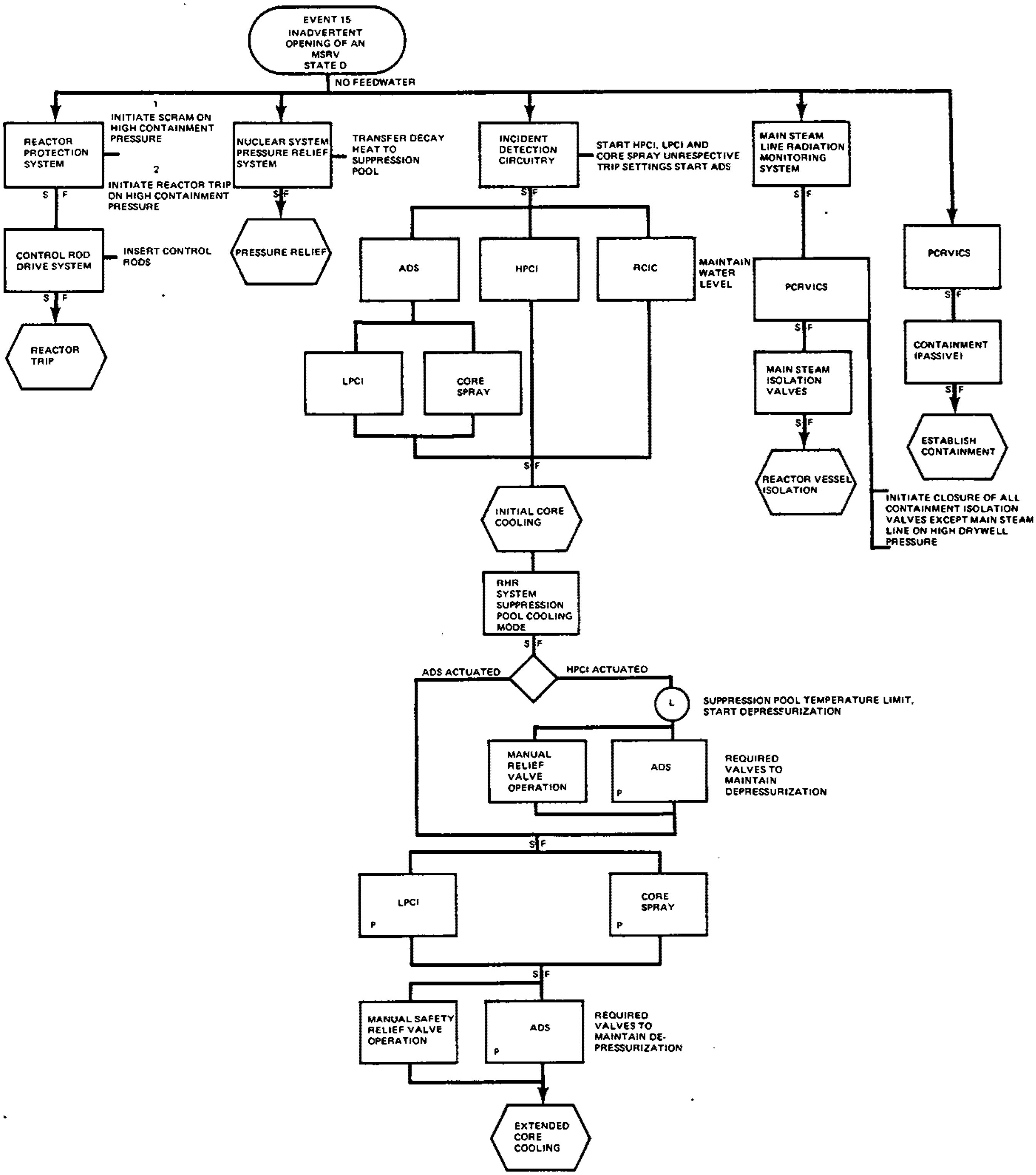
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR
ISOLATION OF ONE
MAIN STEAM LINE

UPDATED FSAR

FIGURE 15.9-21



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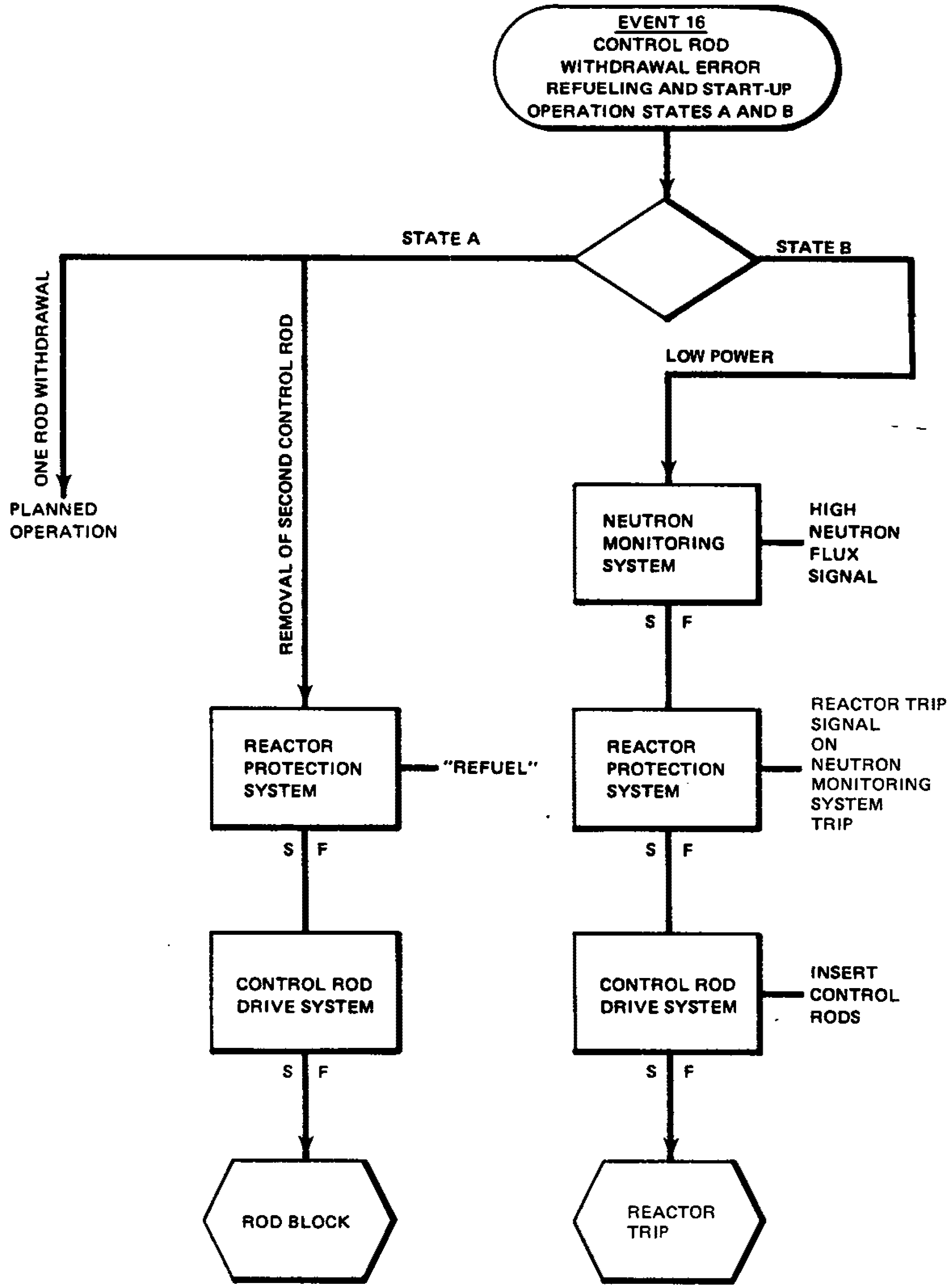
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR INADVERTENT OPENING OF AN MSRV

UPDATED FSAR

FIGURE 15.9-22



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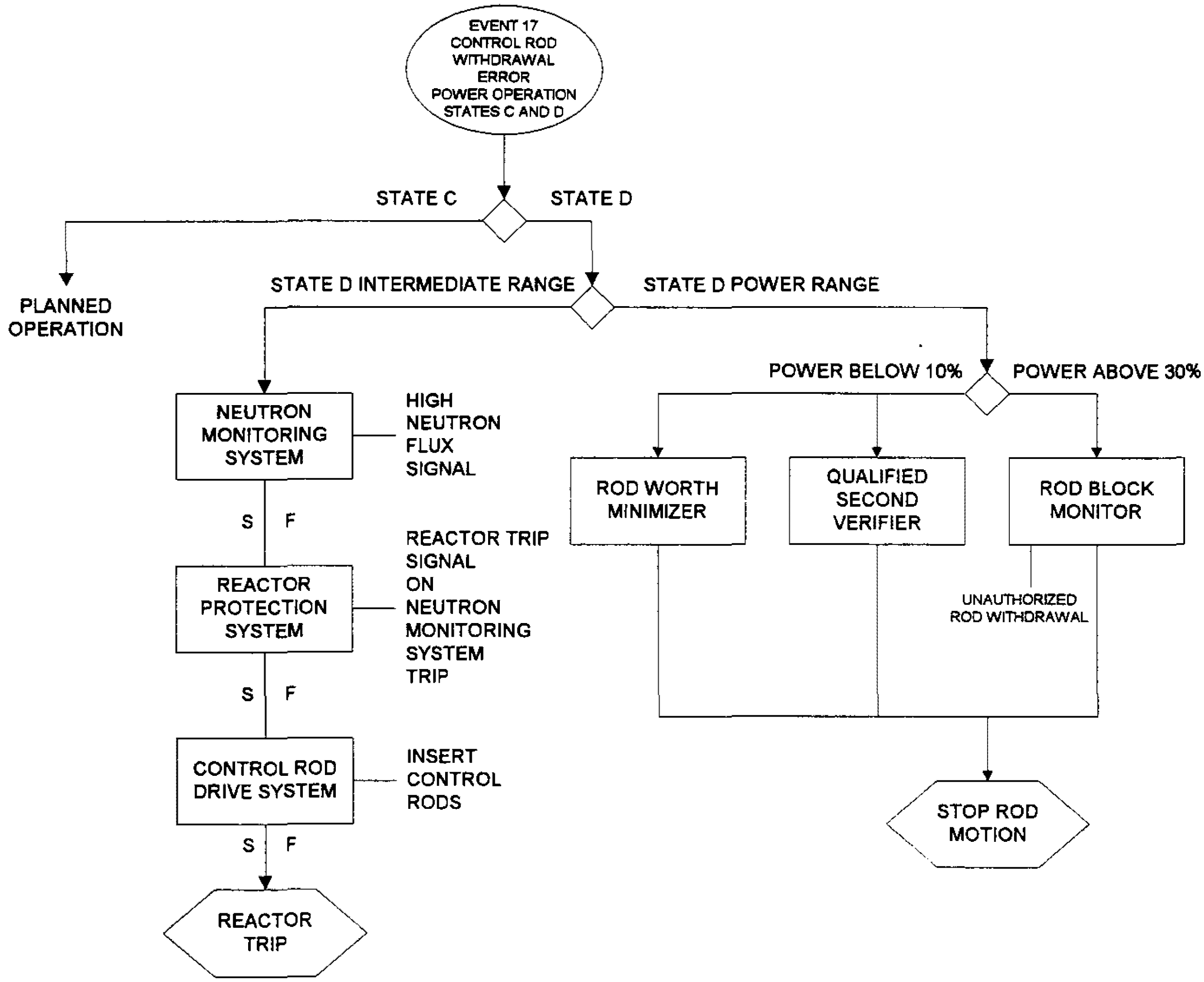
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCE FOR
CONTROL ROD WITHDRAWAL ERROR
FOR REFUELING AND STARTUP
OPERATIONS

UPDATED FSAR

FIGURE 15.9-23

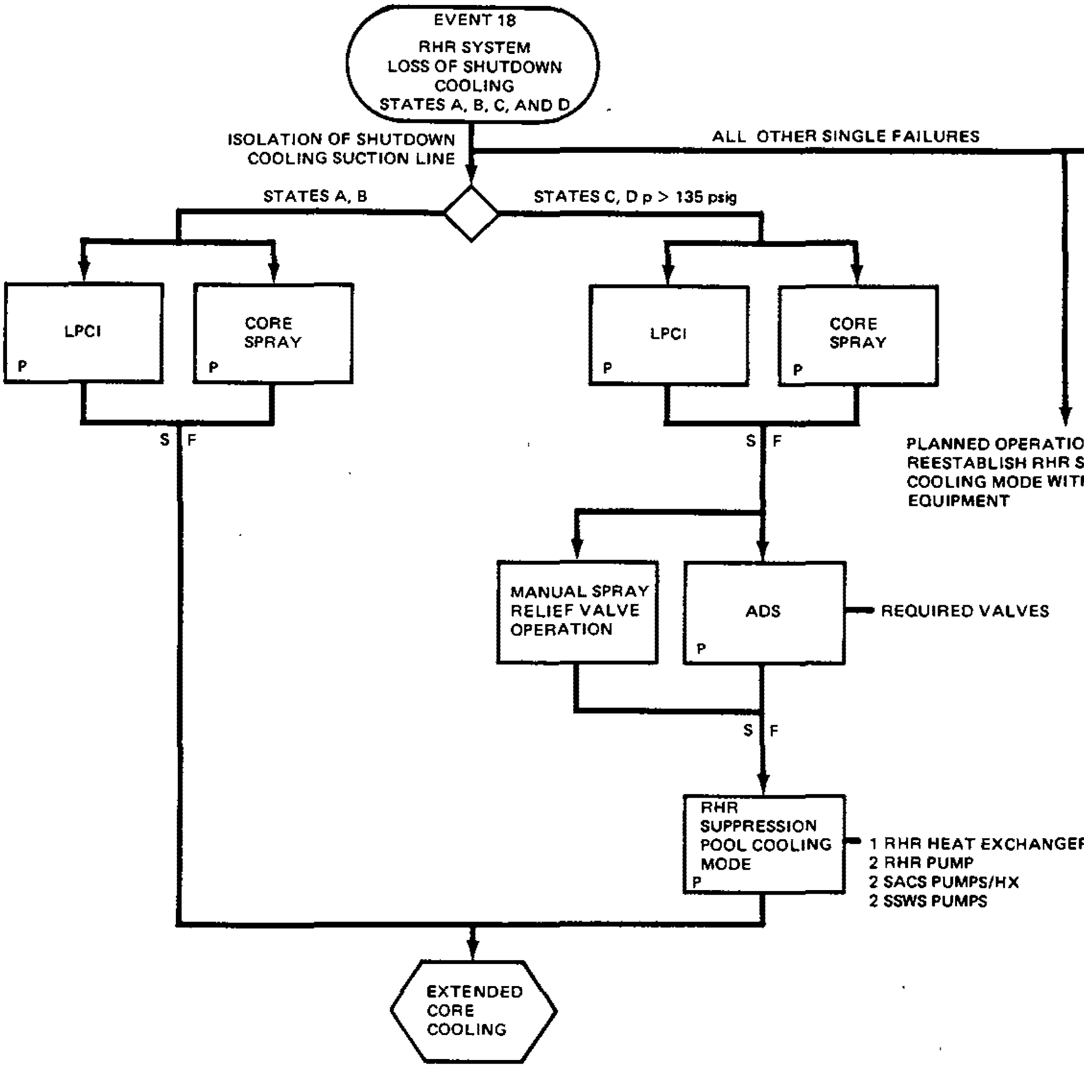


PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCE FOR
CONTROL ROD WITHDRAWAL ERROR -
POWER OPERATION

Updated FSAR
Revision 9, June 13, 1998

Figure 15.9-24



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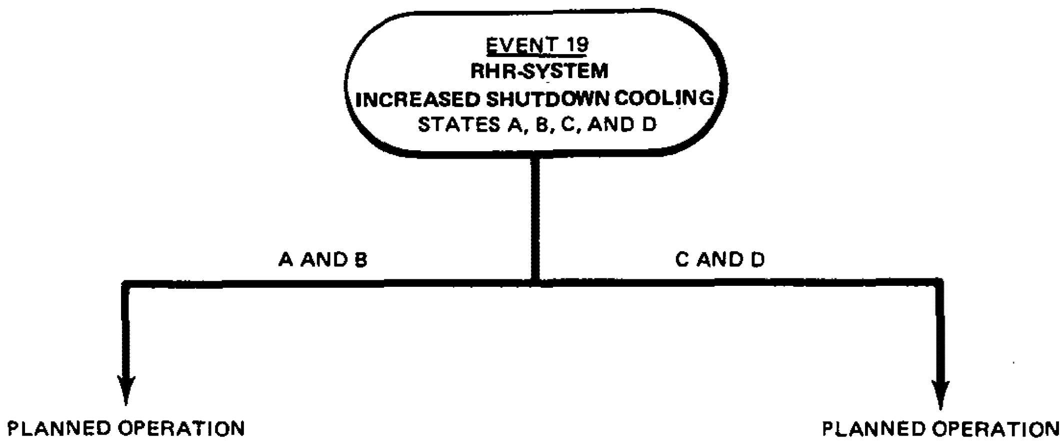
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR
RHR SYSTEM – LOSS OF
SHUTDOWN COOLING

UPDATED FSAR

FIGURE 15.9-25



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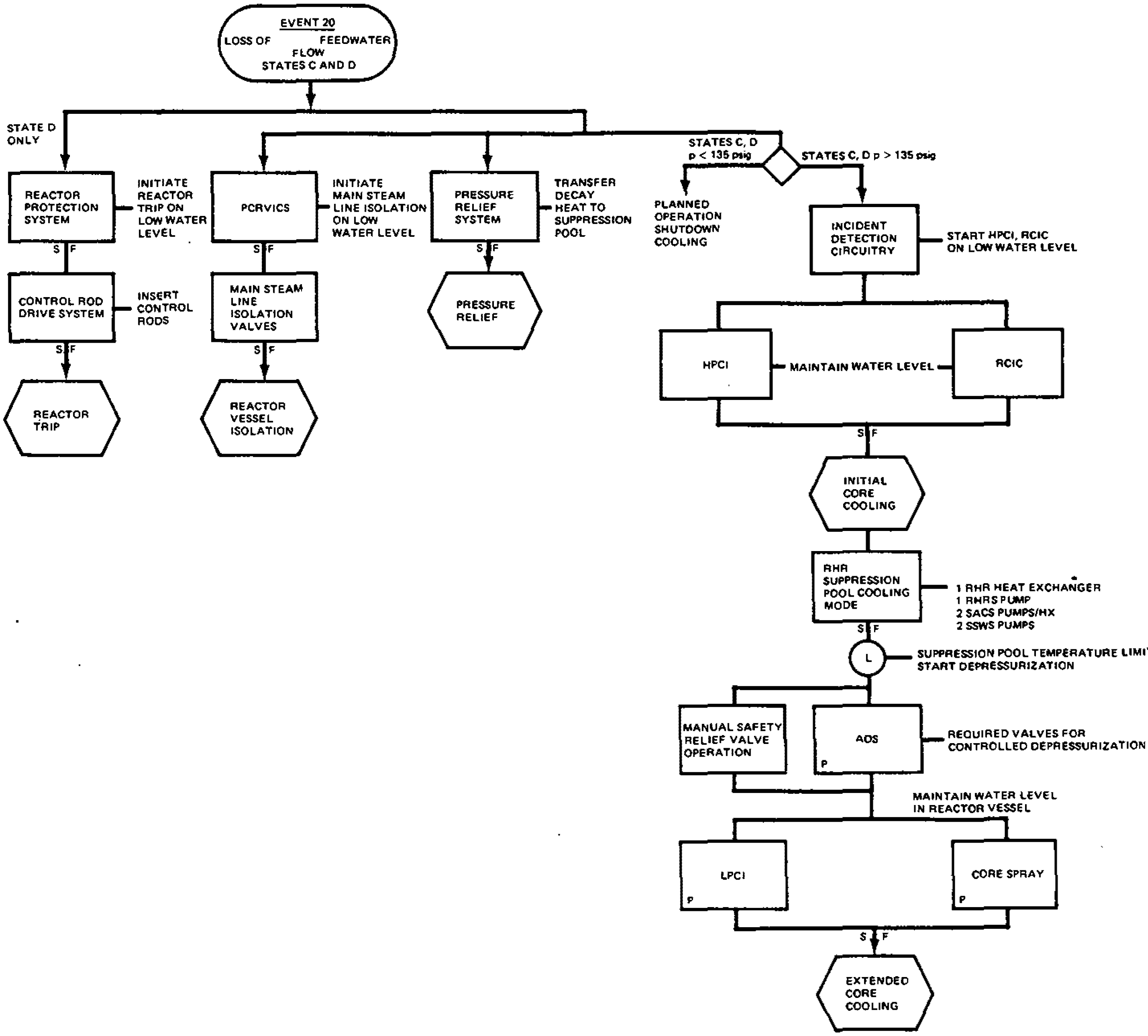
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR RHR SYSTEM INCREASED SHUTDOWN COOLING

UPDATED FSAR

FIGURE 15.9-26



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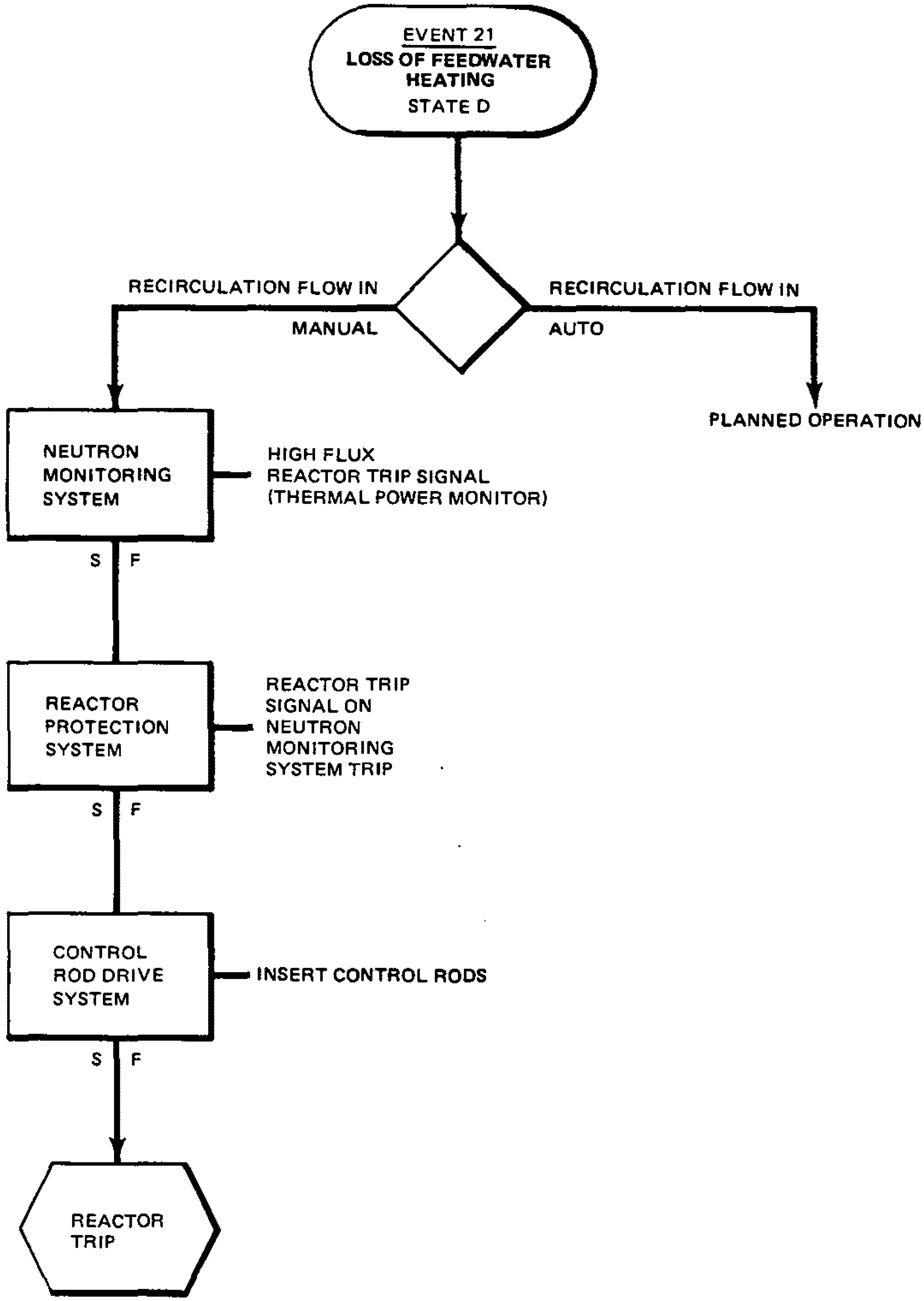
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR LOSS OF FEEDWATER FLOW

UPDATED FSAR

FIGURE 15.9-27



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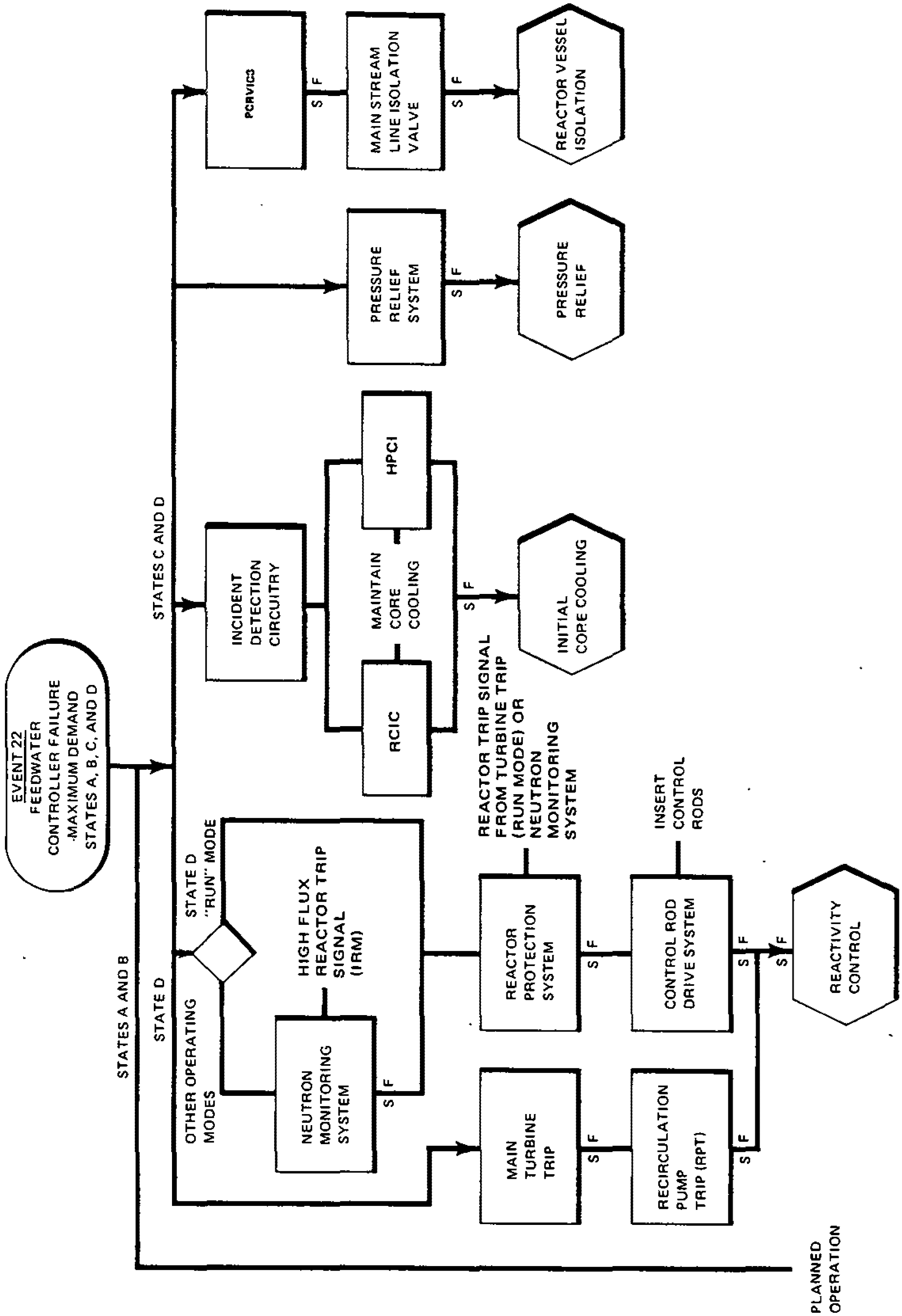
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCE FOR
LOSS OF FEEDWATER HEATING

UPDATED FSAR

FIGURE 15.9-28



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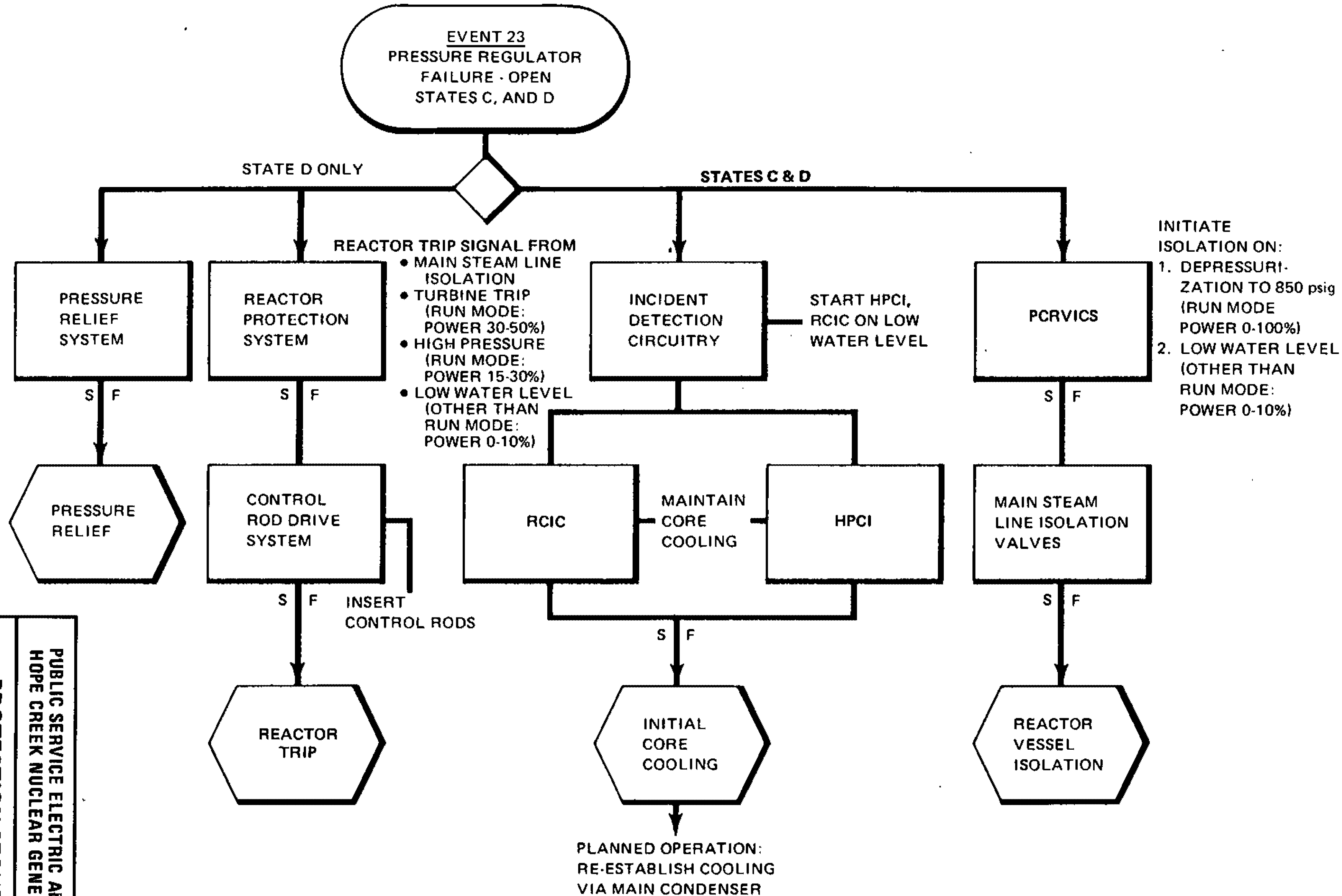
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR FEEDWATER CONTROLLER FAILURE (MAXIMUM DEMAND)

UPDATED FSAR

FIGURE 15.9-29



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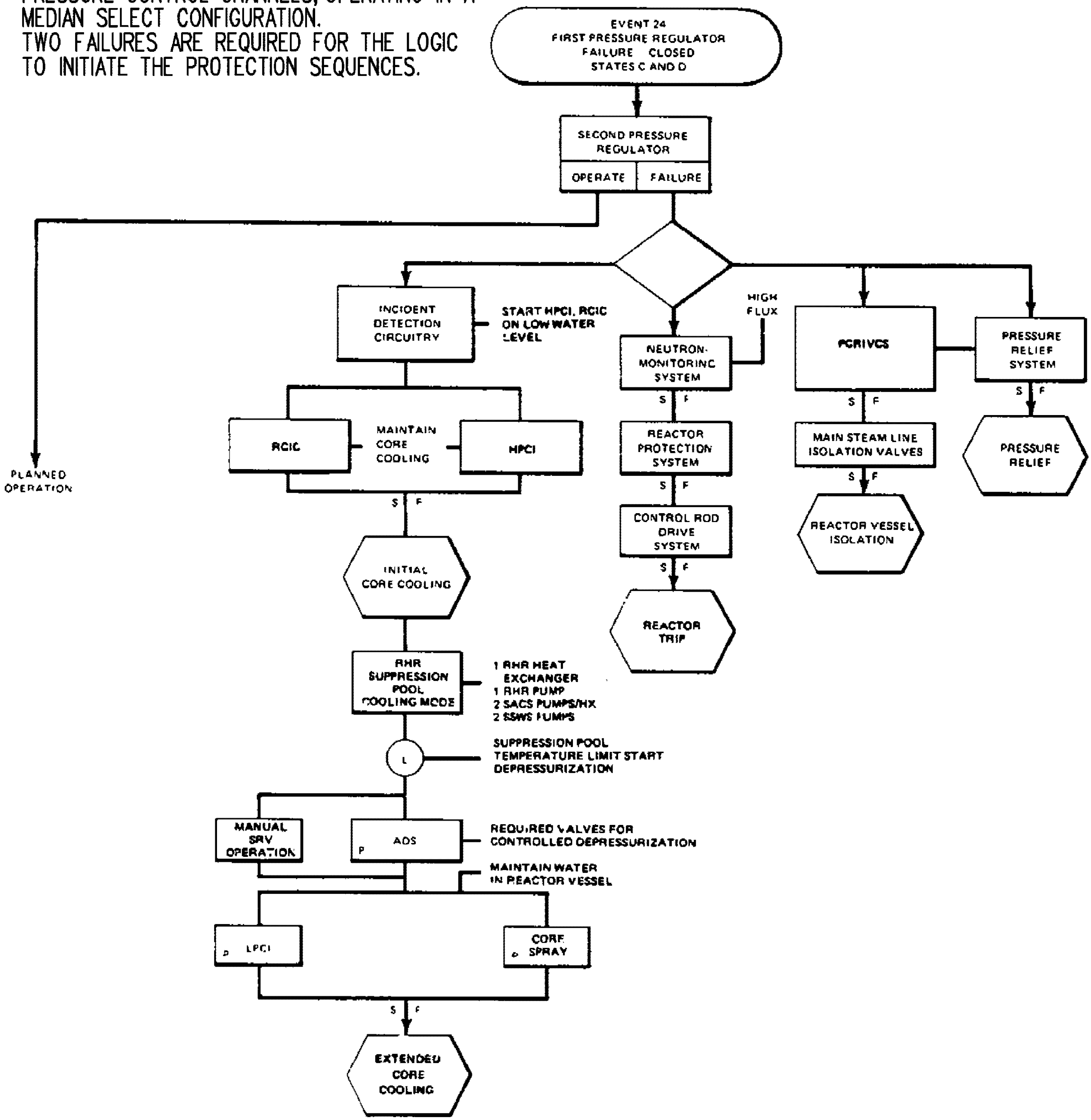
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR
PRESSURE REGULATOR FAILURE
(OPEN)

UPDATED FSAR

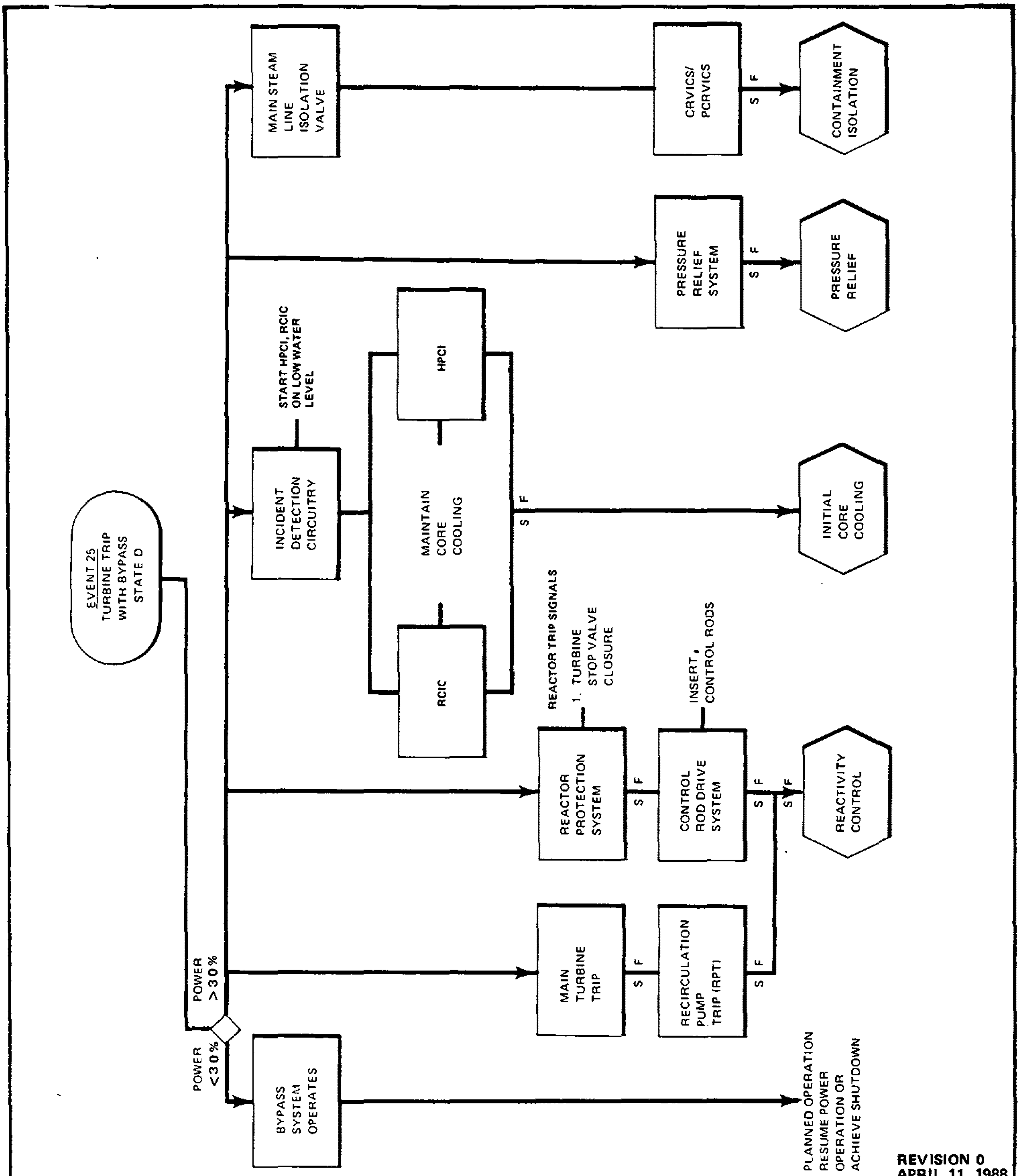
FIGURE 15.9-30

THE DIGITAL EHC SYSTEM UTILIZES THREE PRESSURE CONTROL CHANNELS, OPERATING IN A MEDIAN SELECT CONFIGURATION.
TWO FAILURES ARE REQUIRED FOR THE LOGIC TO INITIATE THE PROTECTION SEQUENCES.



Revision 14, July 26, 2005

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station PROTECTION SEQUENCES FOR PRESSURE REGULATOR FAILURE (CLOSED)
	Updated FSAR Figure 15.9-31



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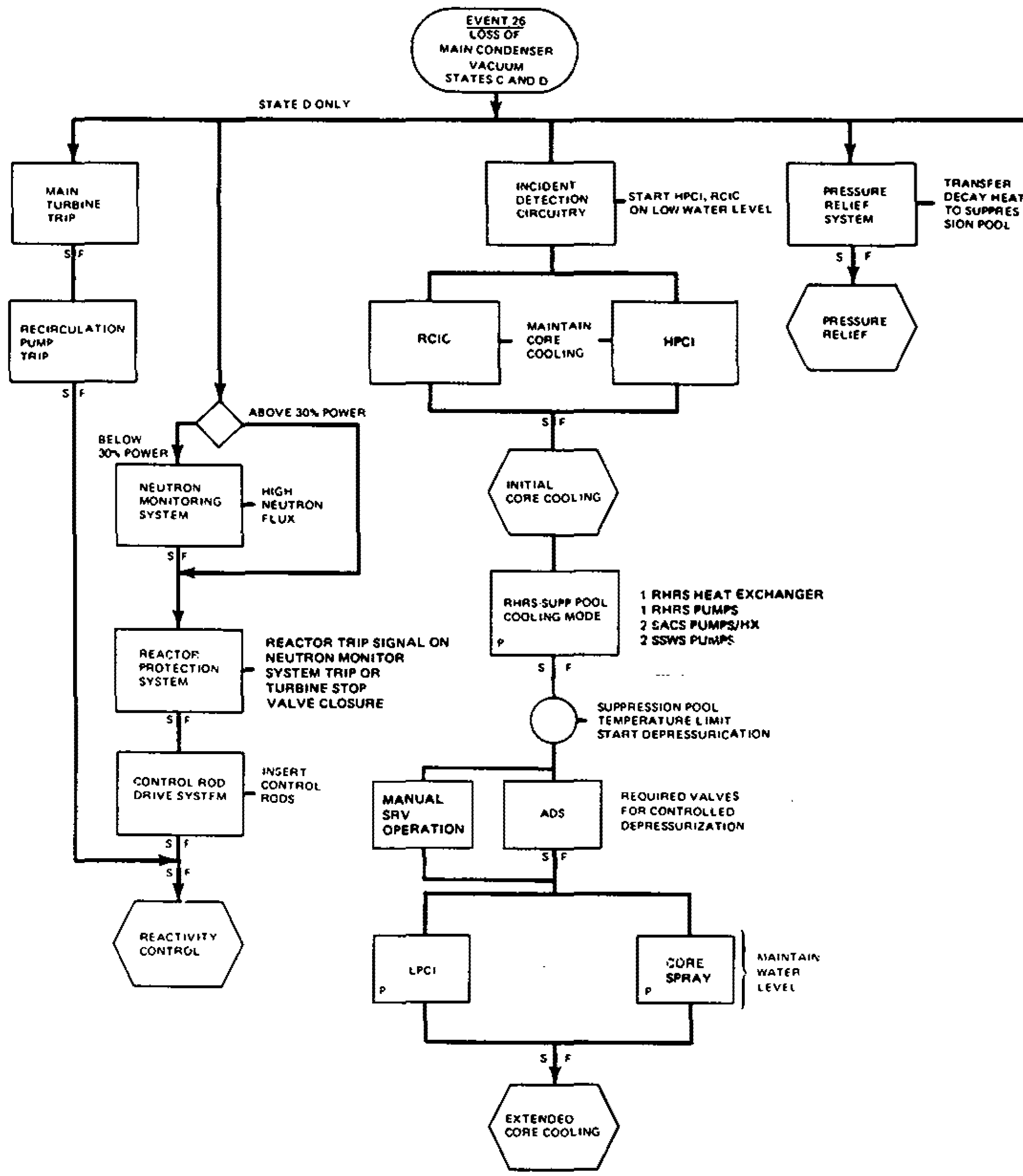
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR
MAIN TURBINE TRIP WITH BYPASS

UPDATED FSAR

FIGURE 15.9-32



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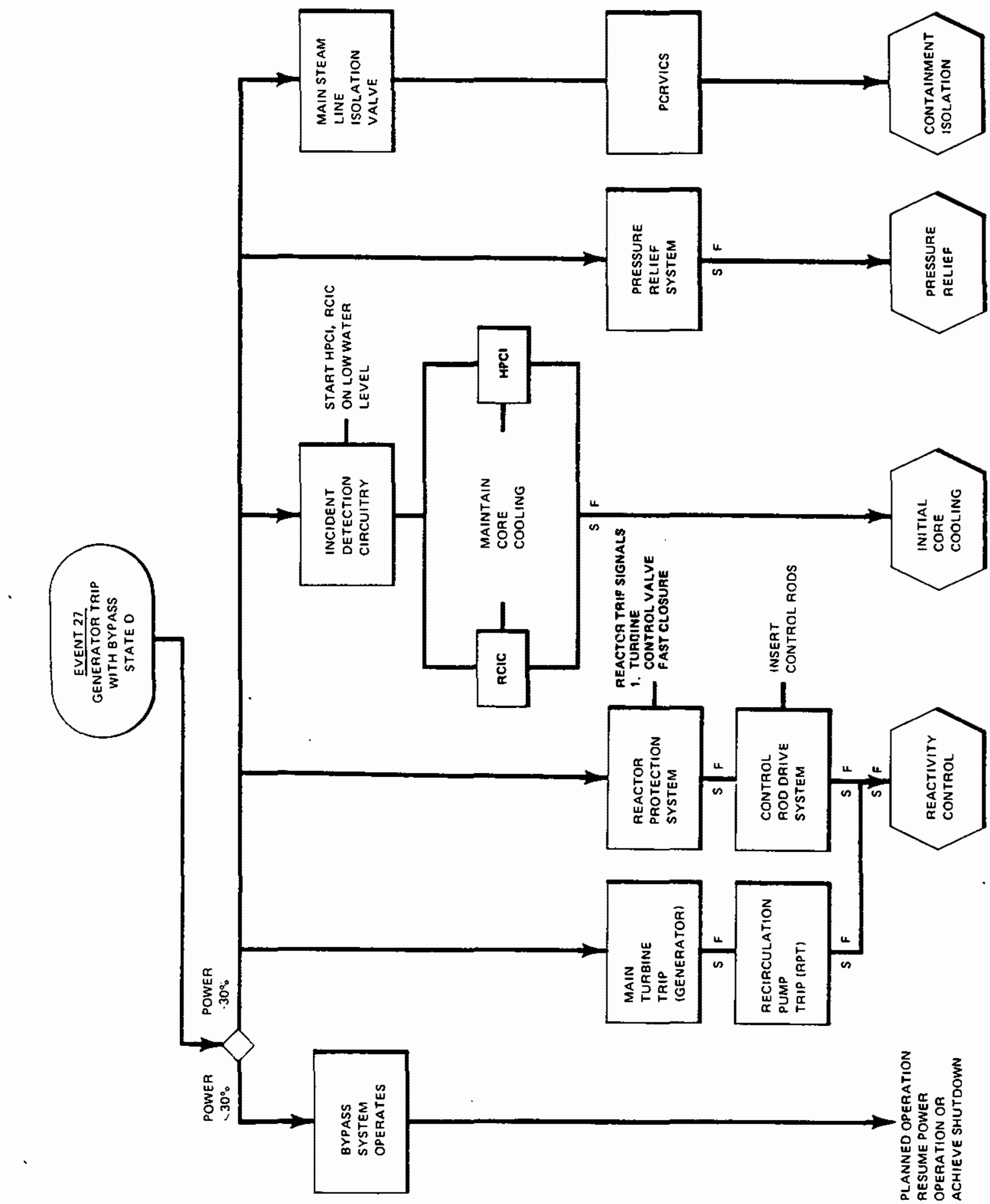
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR
LOSS OF MAIN CONDENSER VACUUM

UPDATED FSAR

FIGURE 15.9-33



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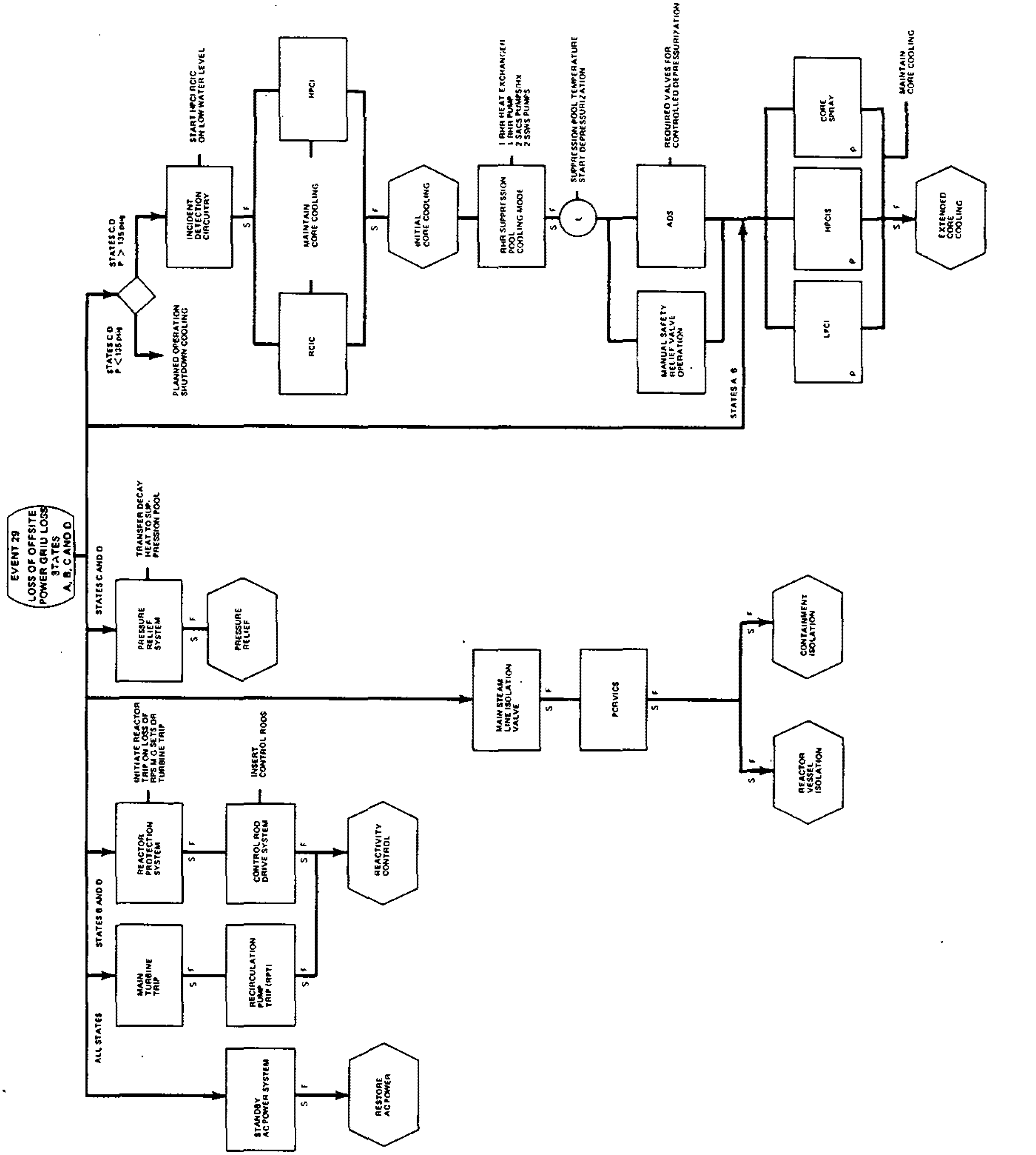
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR MAIN GENERATOR TRIP WITH BYPASS SYSTEM OPERATION

UPDATED FSAR

FIGURE 15.9-34



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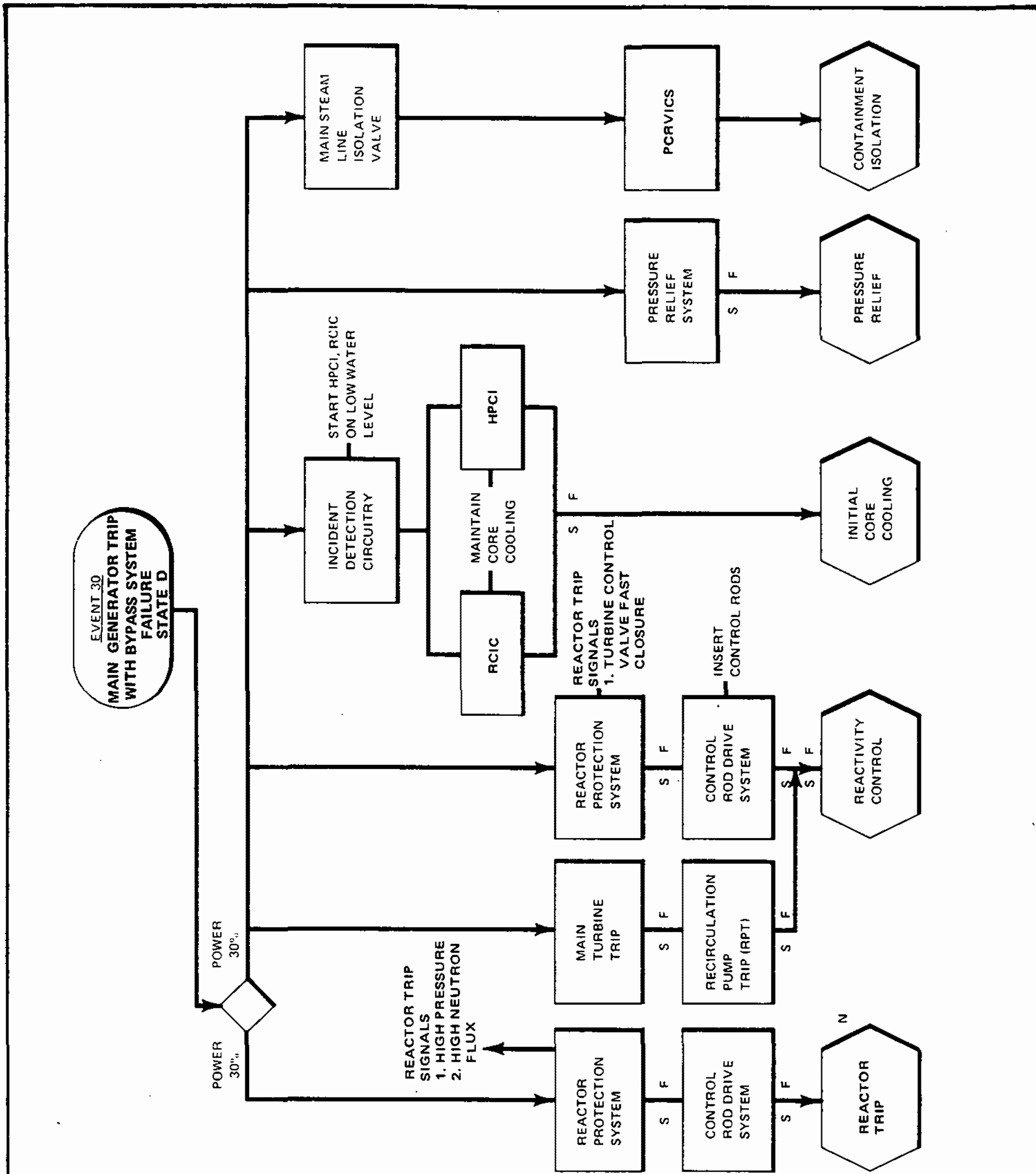
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR LOSS OF OFFSITE POWER (GRID LOSS)

UPDATED FSAR

FIGURE 15.9-35



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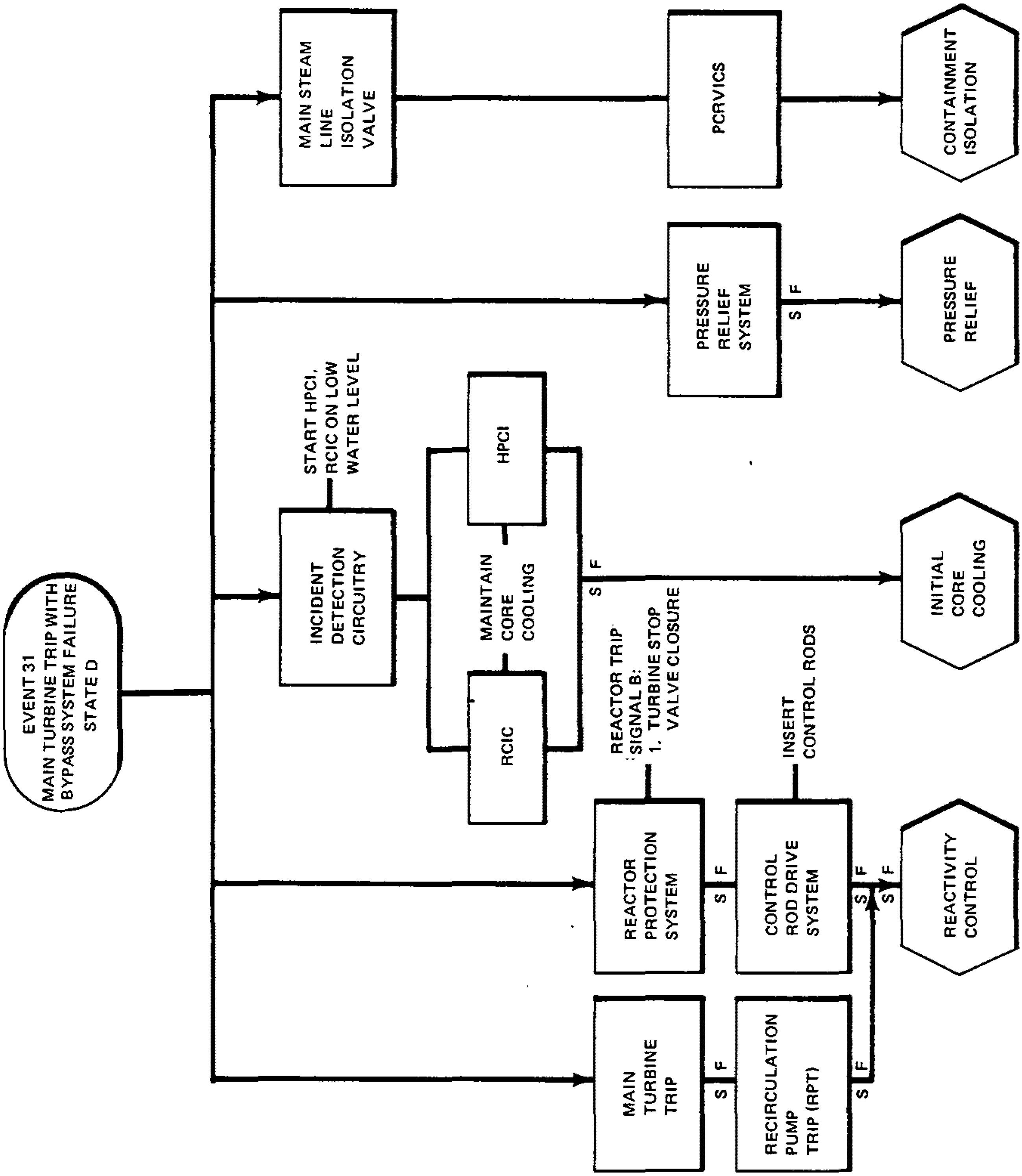
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR MAIN GENERATOR TRIP WITH BYPASS SYSTEM FAILURE

UPDATED FSAR

FIGURE 15.9-36



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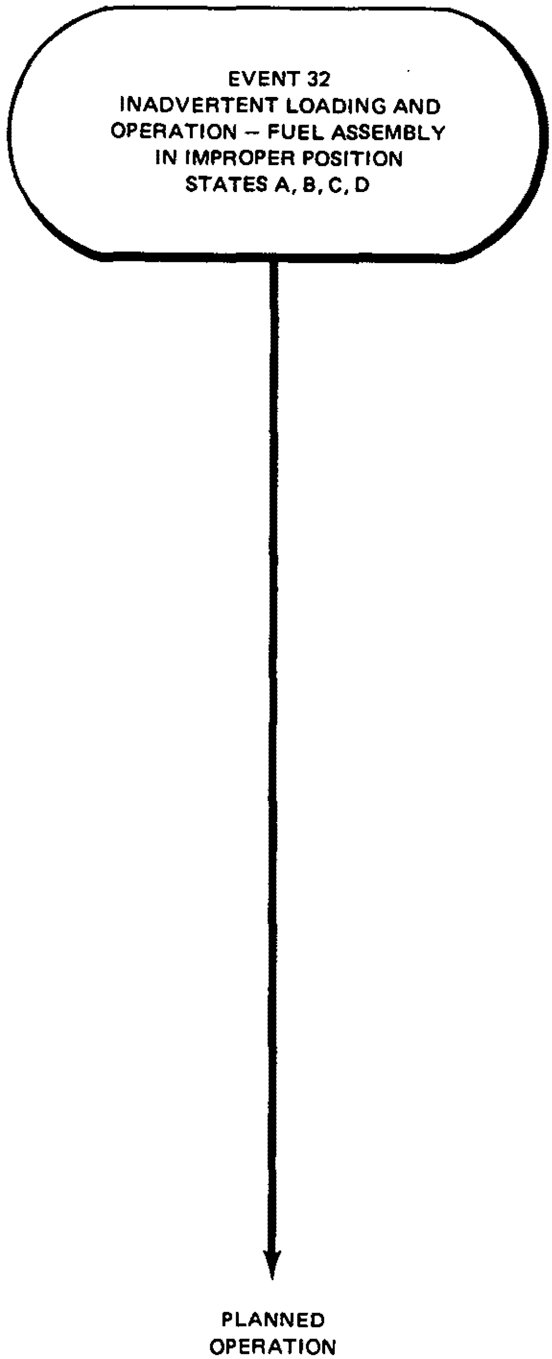
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR
MAIN TURBINE TRIP WITH
BYPASS SYSTEM FAILURE

UPDATED FSAR

FIGURE 15.9-37



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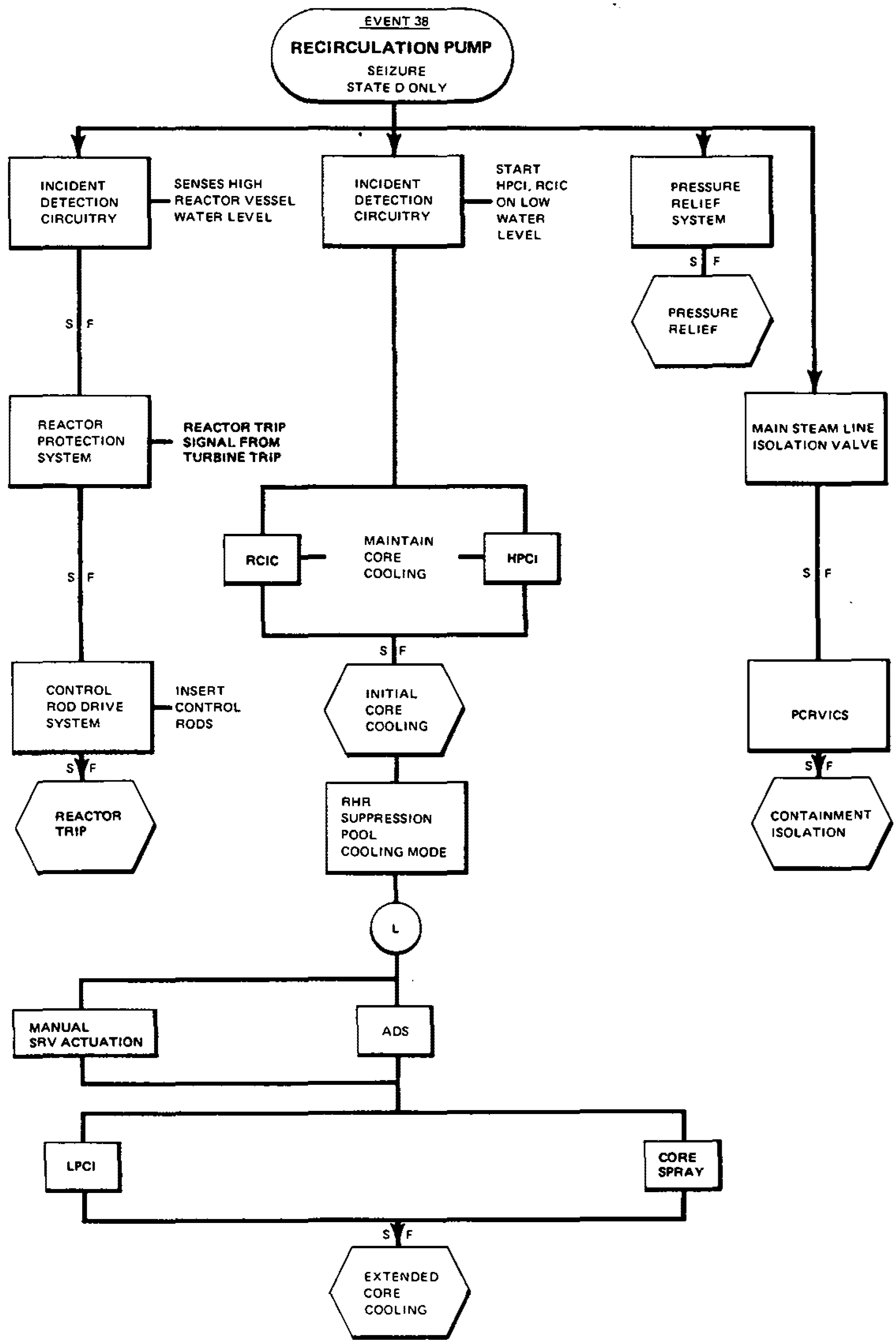
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCE FOR
INADVERTENT LOADING AND
OPERATION OF FUEL ASSEMBLY
IN IMPROPER POSITION

UPDATED FSAR

FIGURE 15.9-38



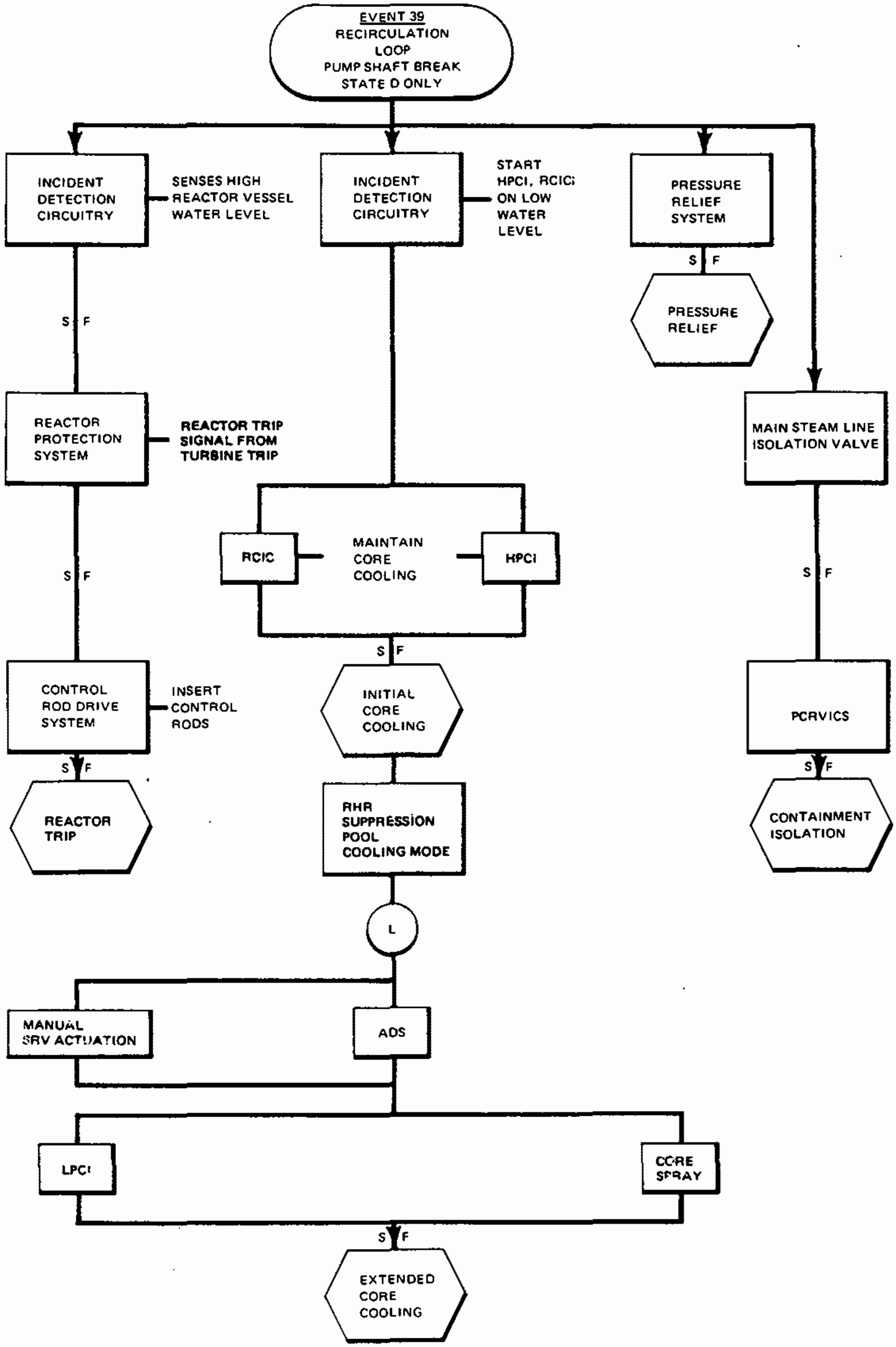
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR RECIRCULATION PUMP SEIZURE

UPDATED FSAR

FIGURE 15.9-39



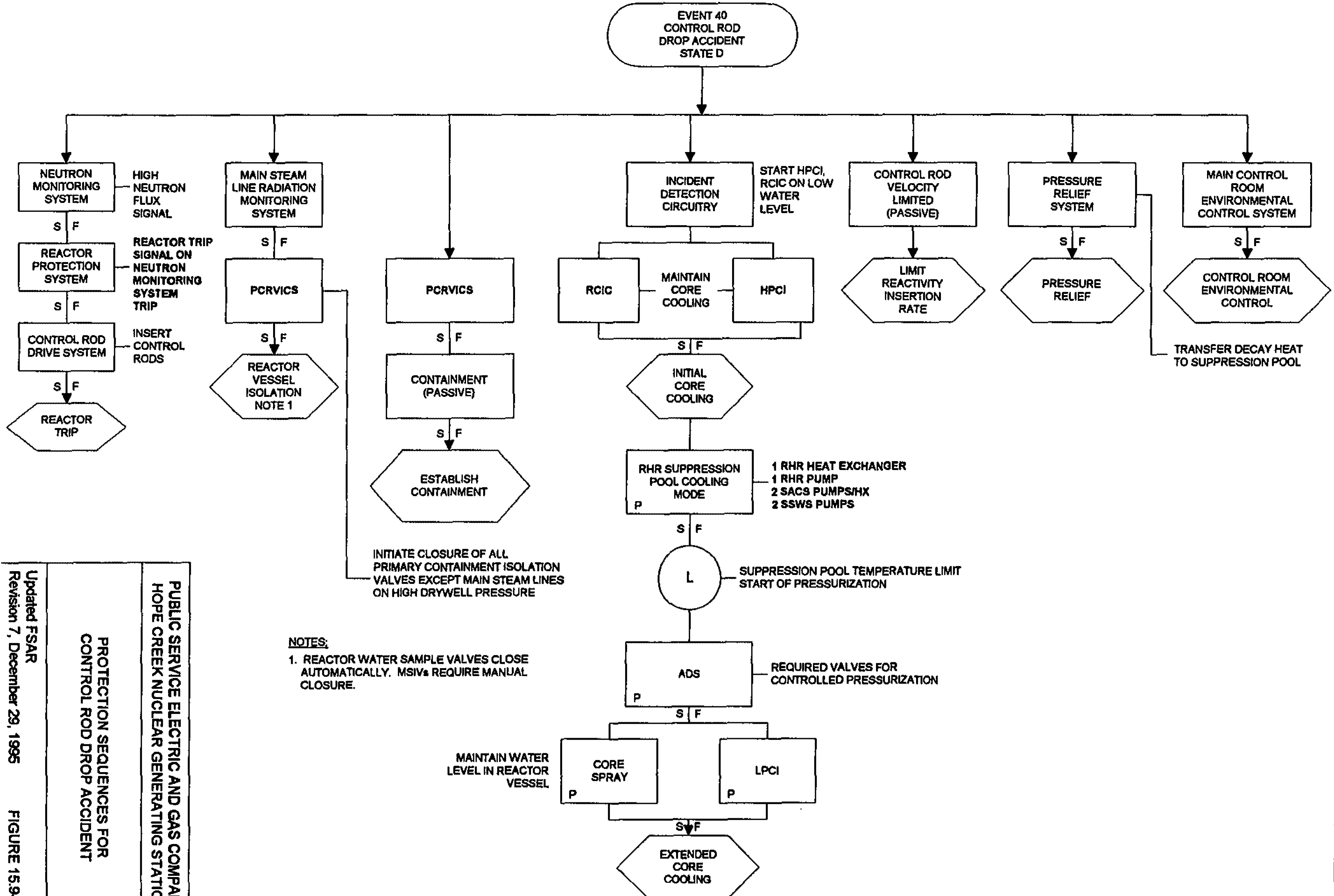
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCE FOR RECIRCULATION PUMP SHAFT BREAK

UPDATED FSAR

FIGURE 15.9-40



NOTES:

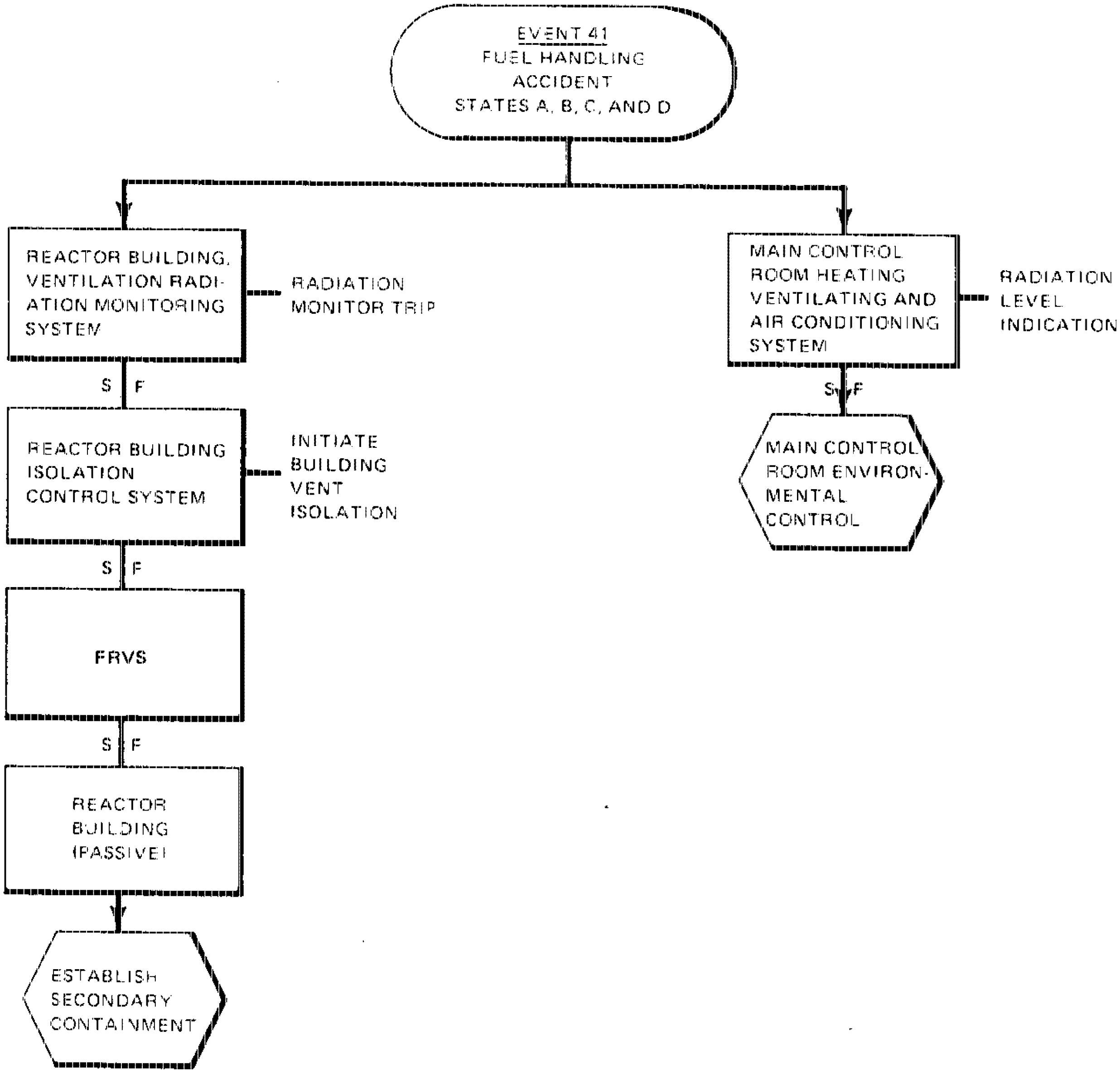
1. REACTOR WATER SAMPLE VALVES CLOSE AUTOMATICALLY. MSIVs REQUIRE MANUAL CLOSURE.

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR
CONTROL ROD DROP ACCIDENT

Updated FSAR
Revision 7, December 29, 1995

FIGURE 15.9-41



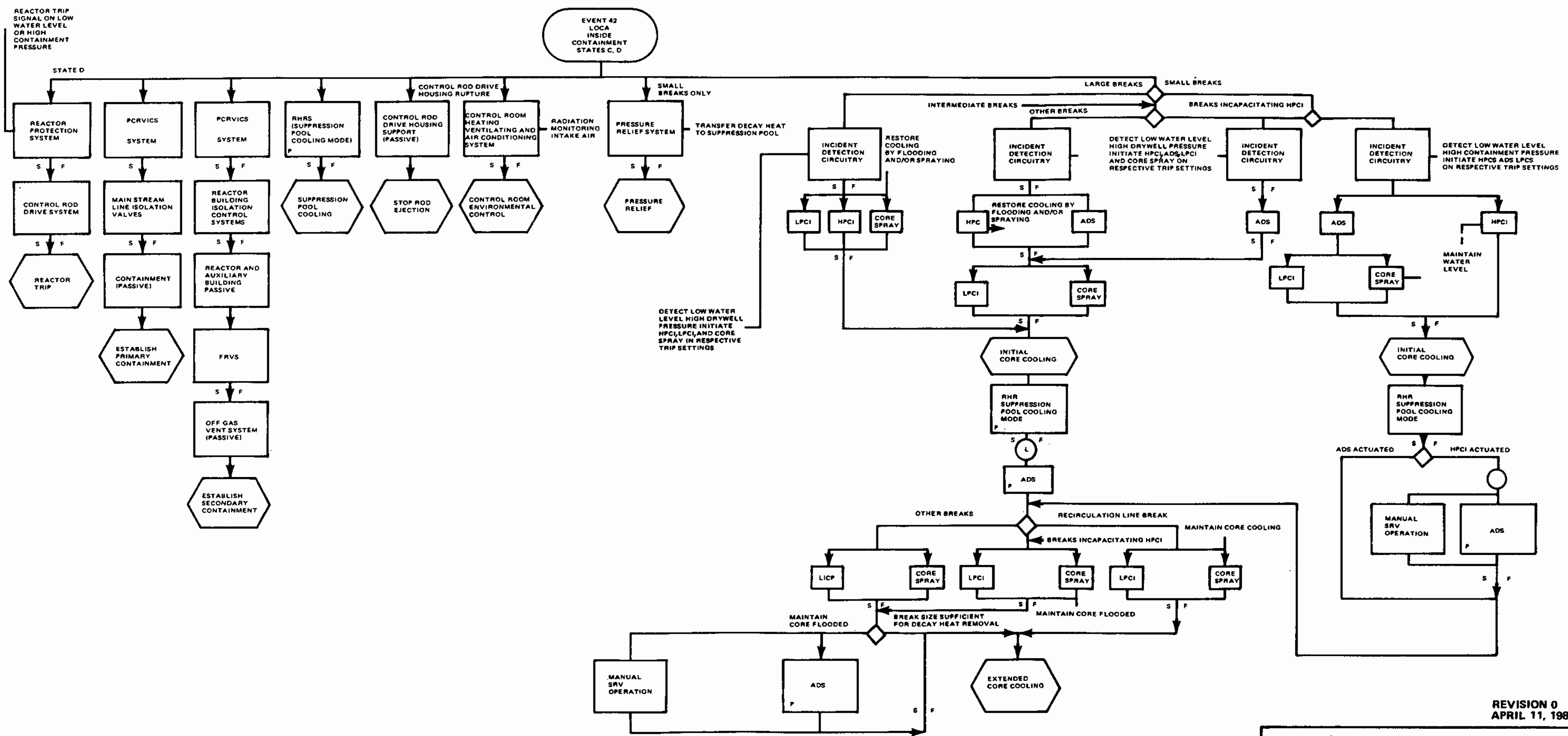
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR FUEL HANDLING ACCIDENT

UPDATED FSAR

FIGURE 15.9-42



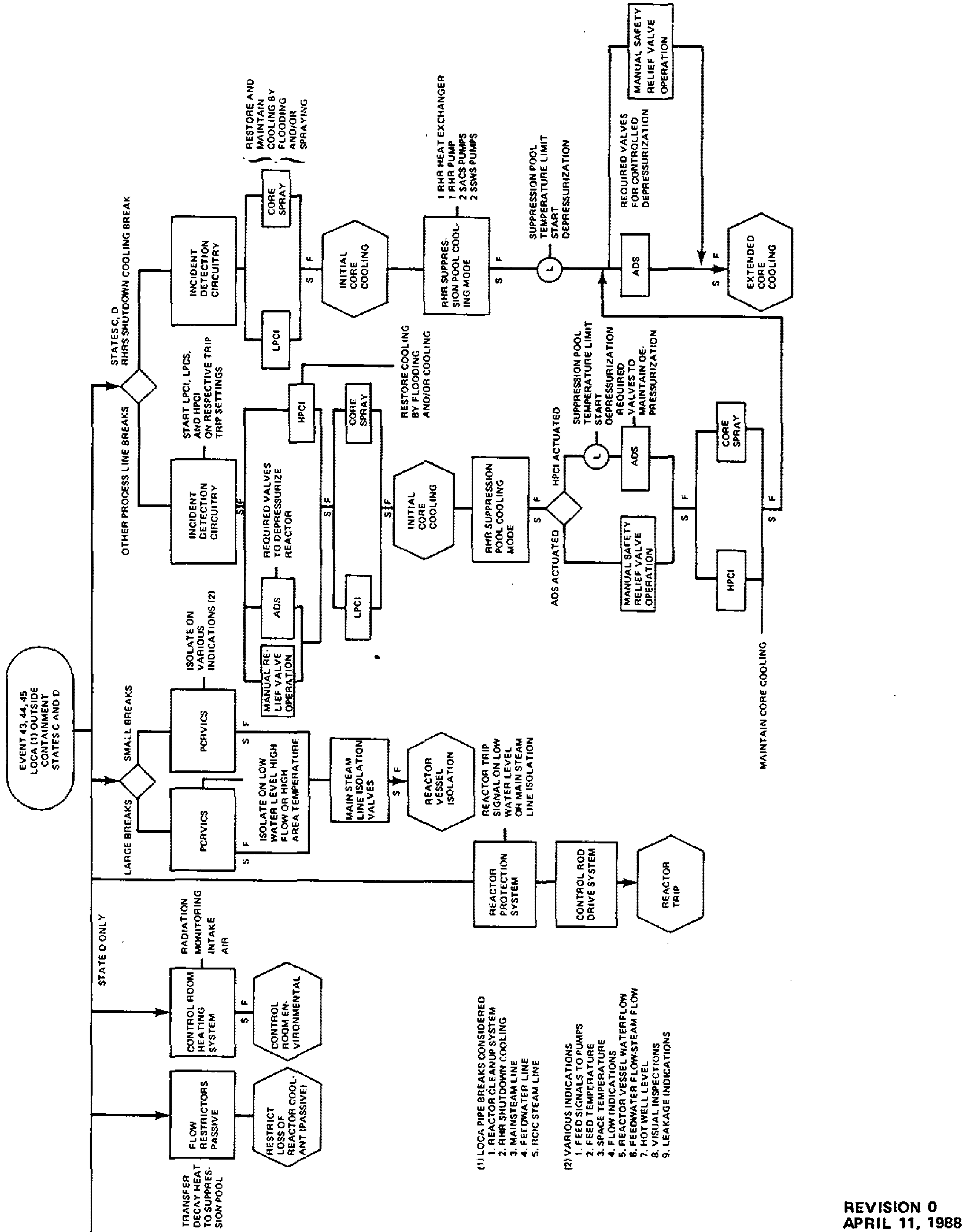
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR LOSS-OF-COOLANT ACCIDENT PIPING BREAKS WITHIN RCPB INSIDE CONTAINMENT

UPDATED FSAR

FIGURE 15.9-43



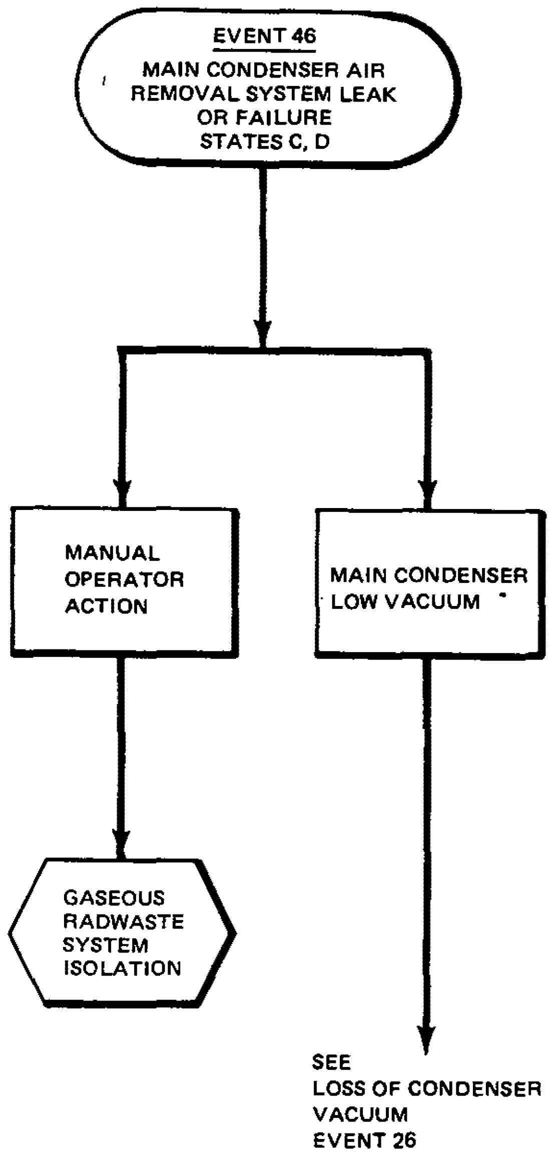
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR LOSS-OF-COOLANT ACCIDENT PIPING BREAKS OUTSIDE CONTAINMENT

UPDATED FSAR

FIGURE 15.9-44



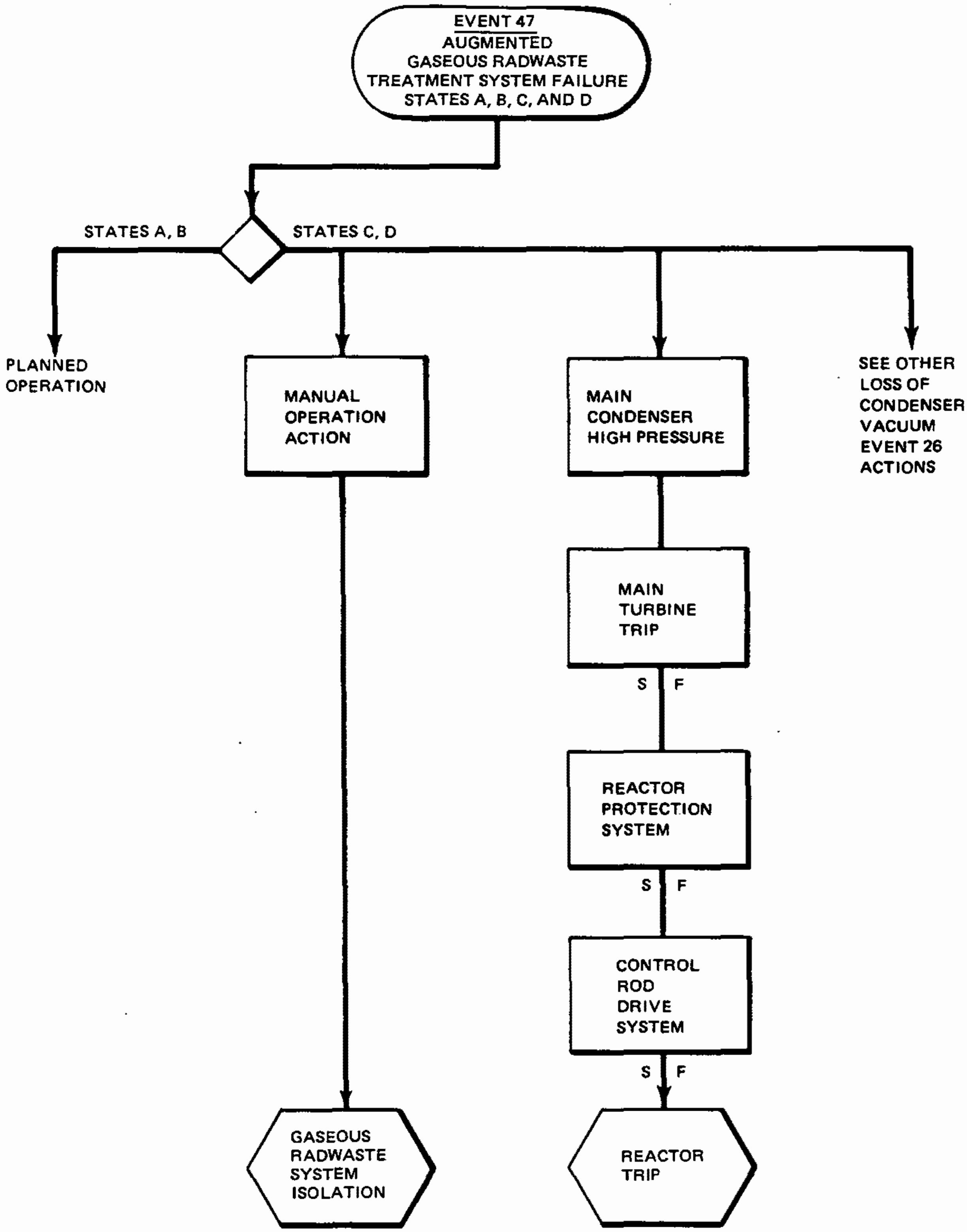
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR MAIN CONDENSER AIR REMOVAL SYSTEM LEAK OR FAILURE

UPDATED FSAR

FIGURE 15.9-45



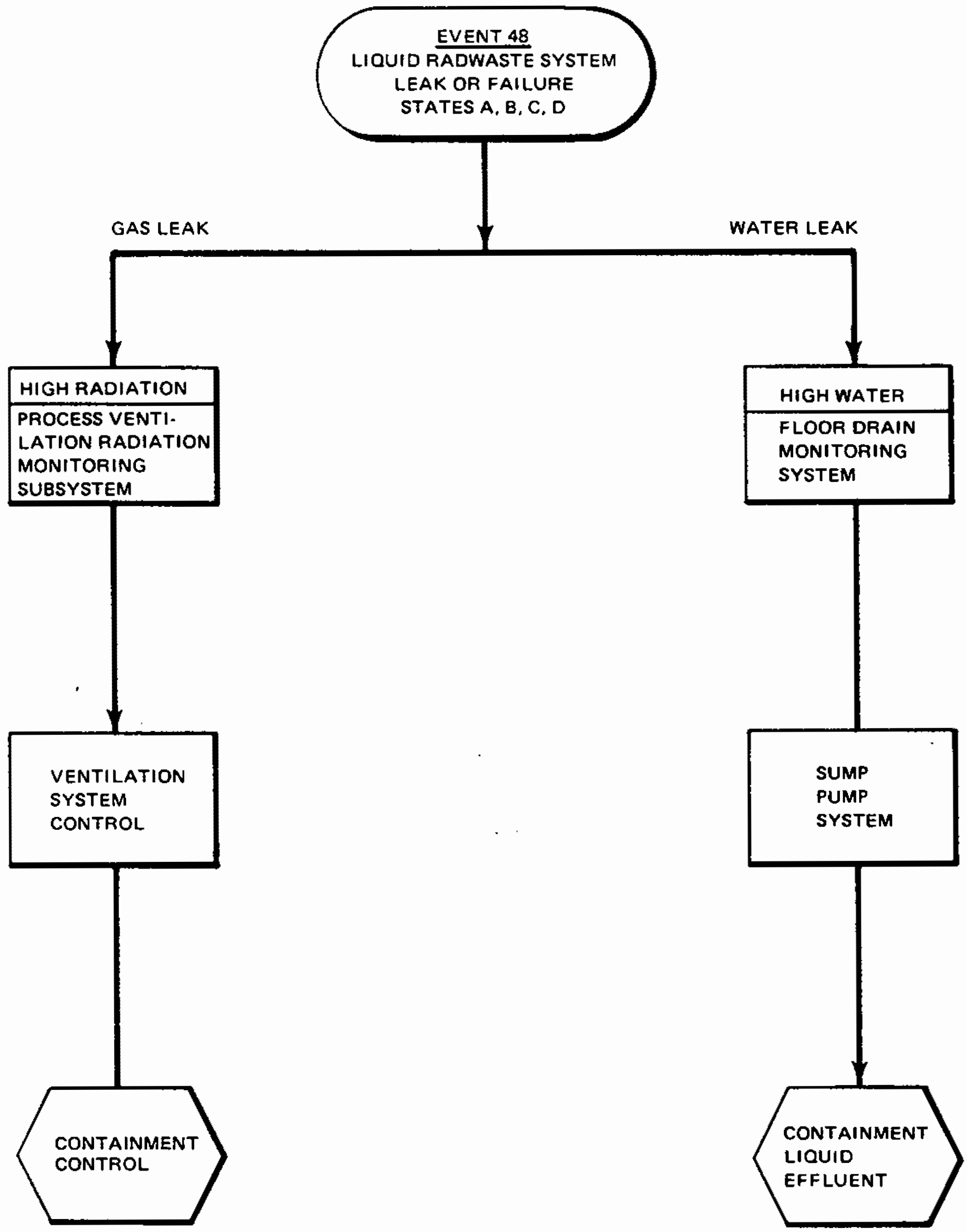
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR
AUGMENTED OFFGAS
TREATMENT SYSTEM FAILURE

UPDATED FSAR

FIGURE 15.9-46



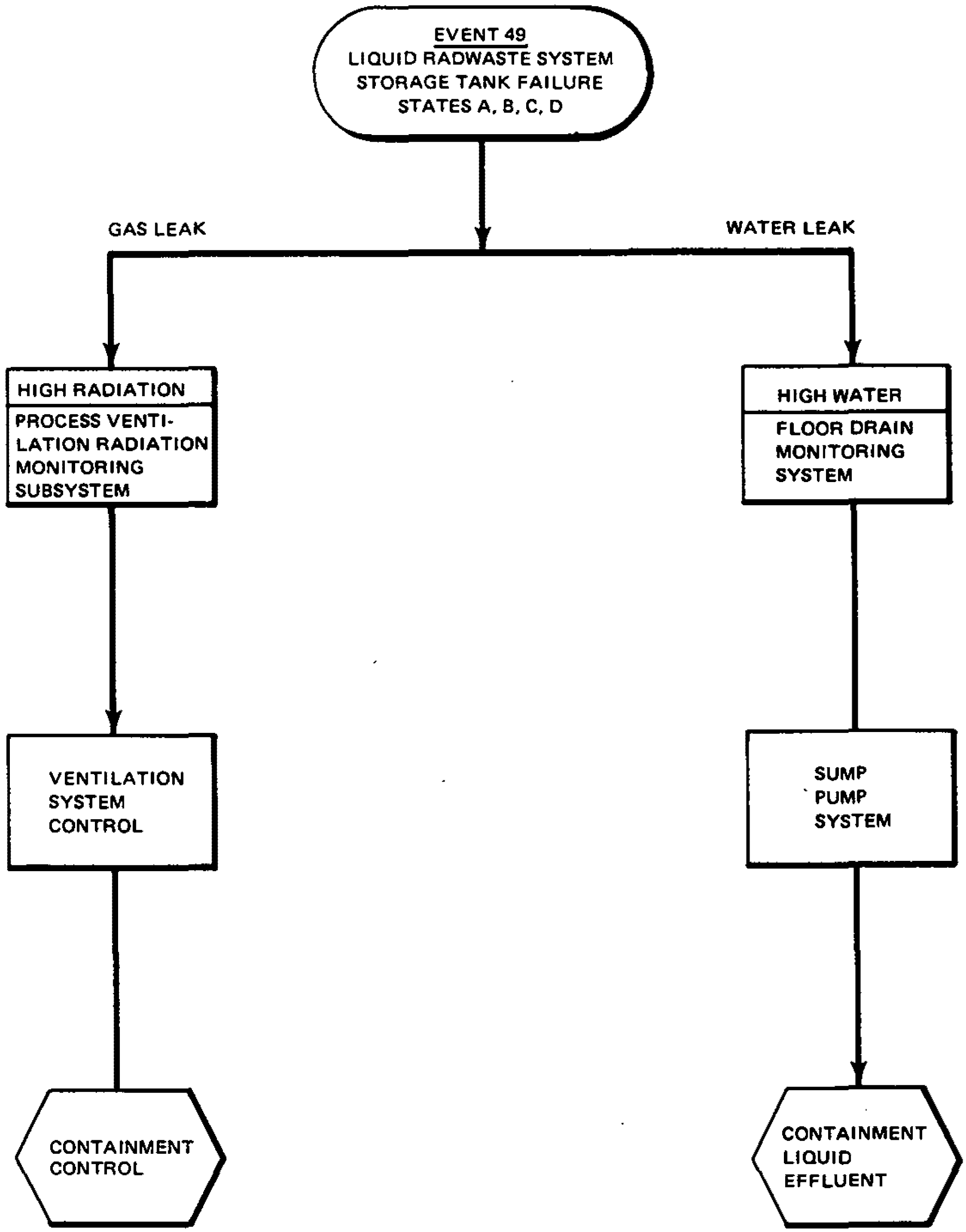
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR
LIQUID RADWASTE SYSTEM
LEAK OR FAILURE

UPDATED FSAR

FIGURE 15.9-47



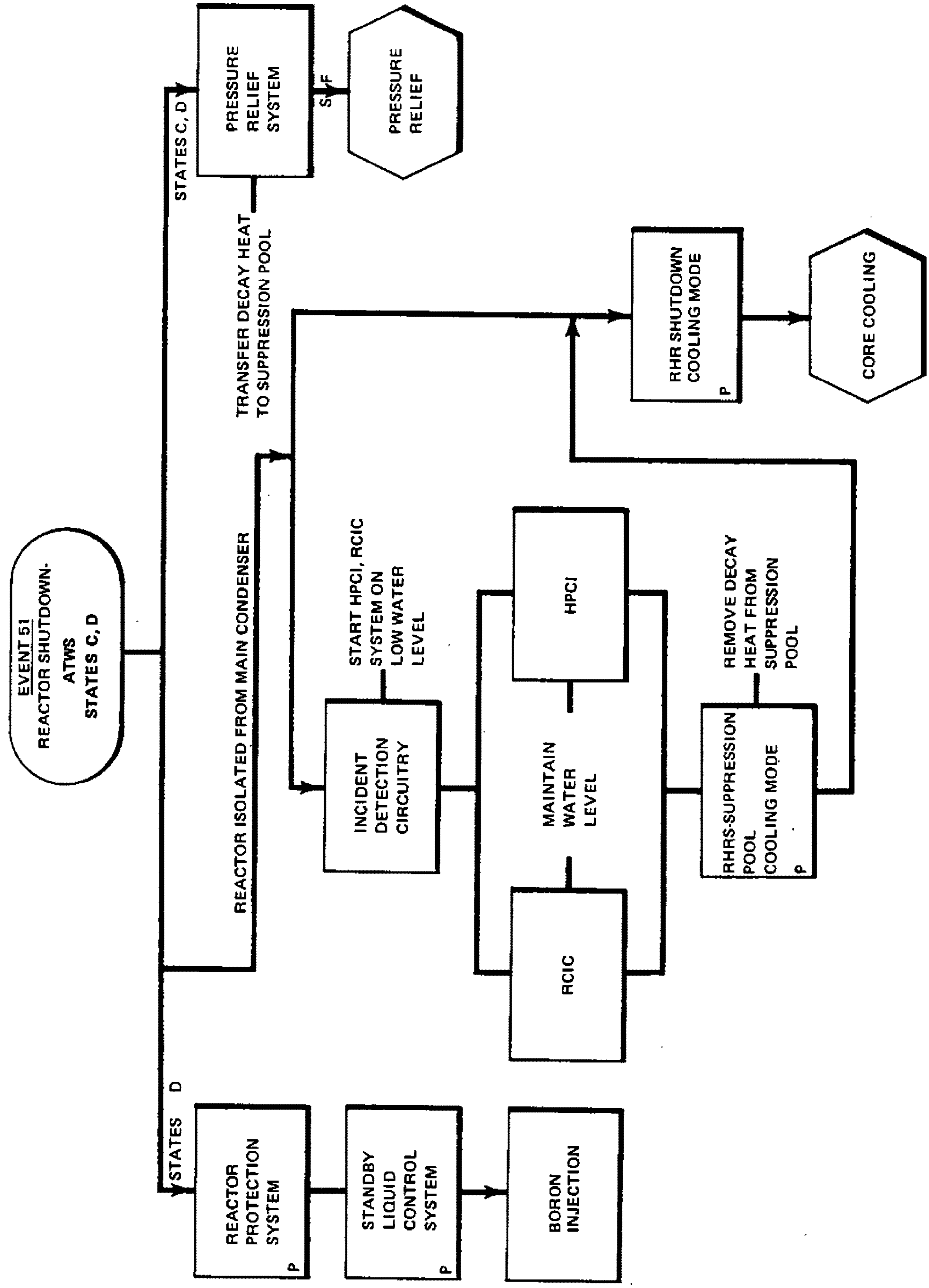
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR LIQUID RADWASTE SYSTEM – STORAGE TANK FAILURE

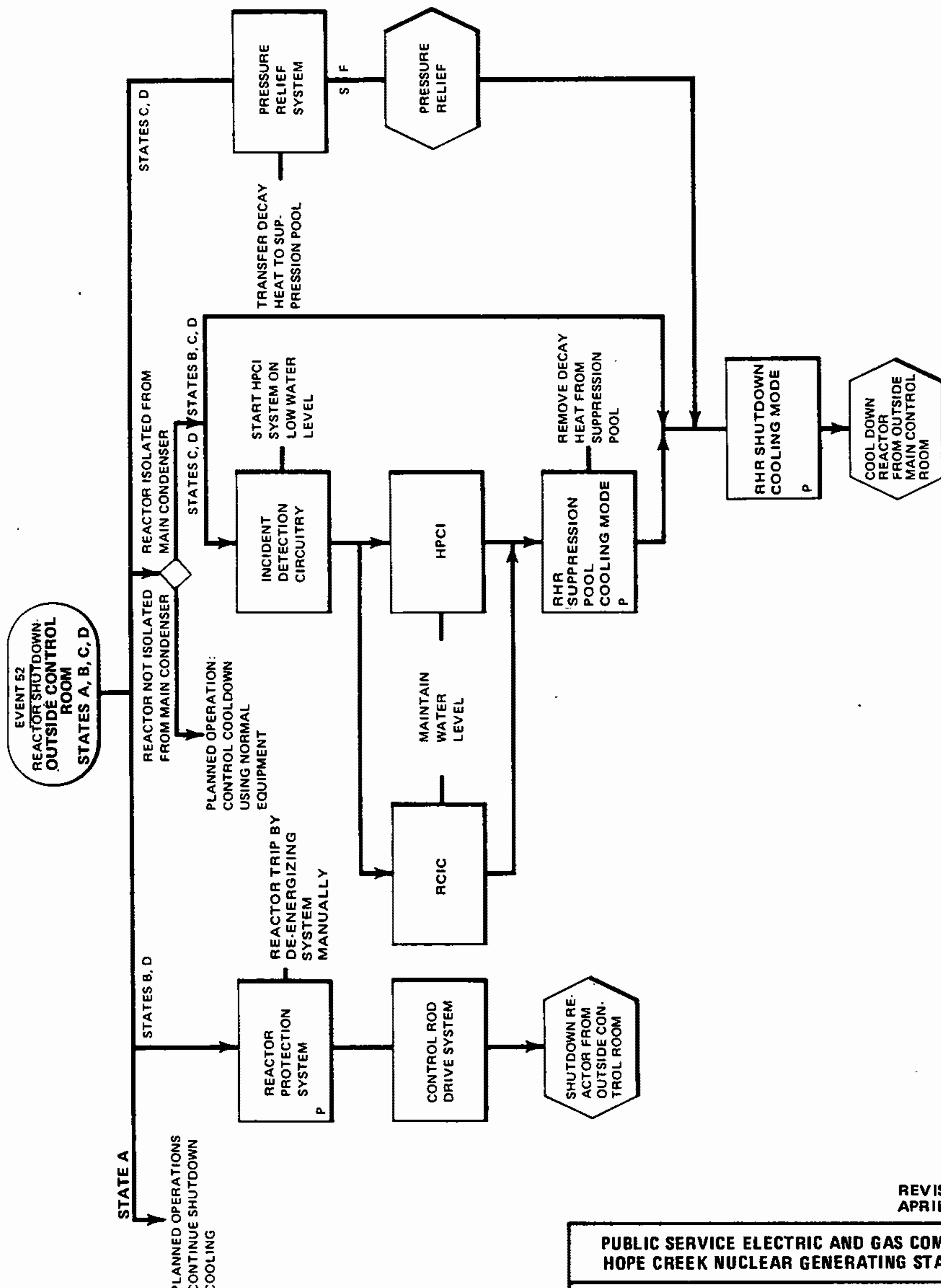
UPDATED FSAR

FIGURE 15.9-48



Revision 15, October 27, 2006

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station PROTECTION SEQUENCES FOR REACTOR SHUTDOWN ATWS
	Updated FSAR Figure 15.9-49



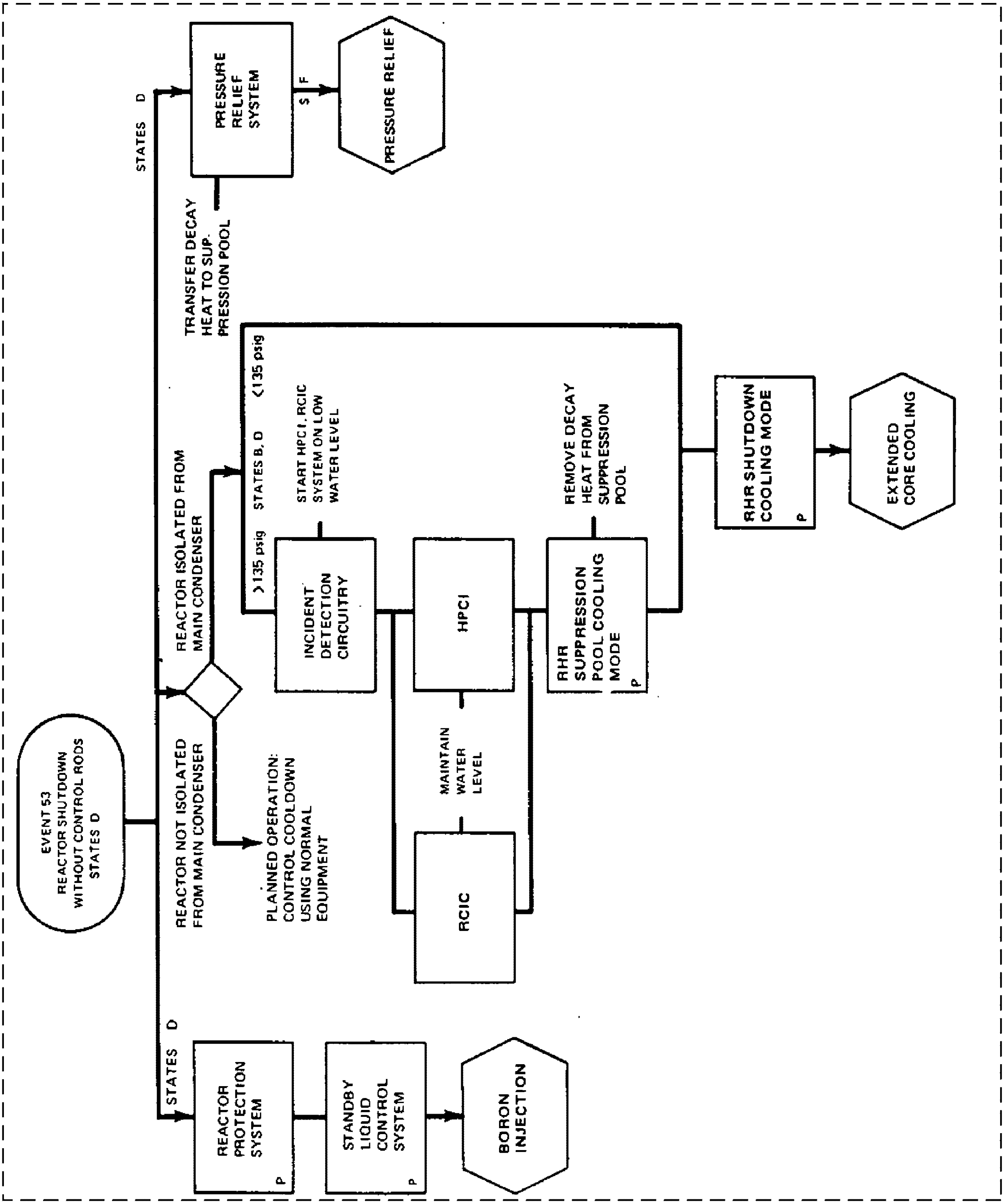
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

PROTECTION SEQUENCES FOR
REACTOR SHUTDOWN
FROM OUTSIDE CONTROL ROOM

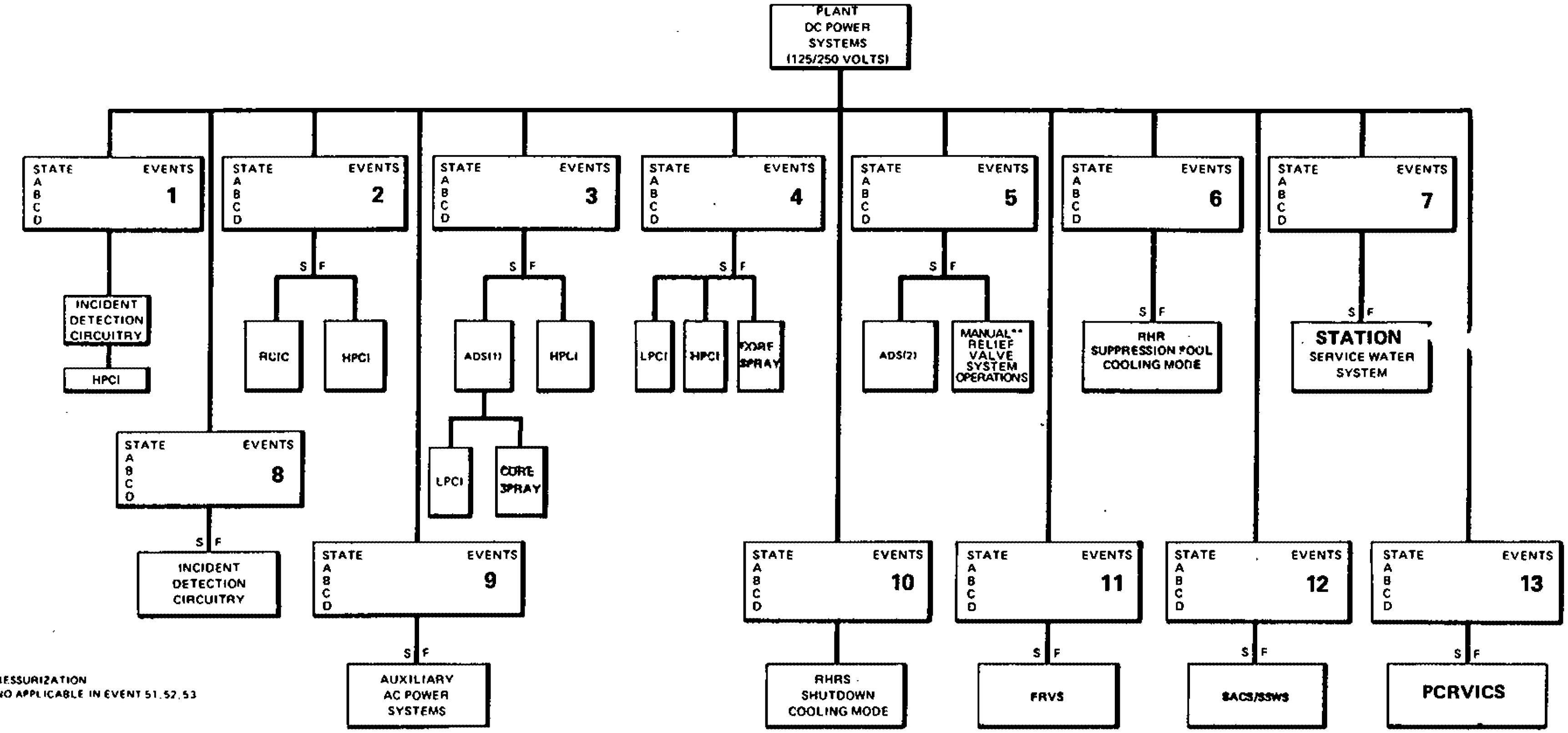
UPDATED FSAR

FIGURE 15.9-50



Revision 15, October 27, 2006

PSEG Nuclear, LLC HOPE CREEK NUCLEAR GENERATING STATION	Hope Creek Nuclear Generating Station PROTECTION SEQUENCES FOR REACTOR SHUTDOWN WITHOUT CONTROL RODS
	Updated FSAR Figure 15.9-51



(1) BLOWDOWN
(2) CONTROLLED DEPRESSURIZATION
(3) SF REQUIREMENT NO APPLICABLE IN EVENT 51,52,53

ITEM	STATE	EVENTS
1	A	
	B	
	C	51,52,53
	D	51,52,53
2	A	
	B	
	C	14,26,23,20,29
	D	14,26,23,20,29,40
3	A	
	B	
	C	15,42,43,44,45
	D	15,42,43,44,45
4	A	29,18
	B	29,18
	C	26,15,20,29,18,42,43,44,45
	D	26,15,20,29,40,18,42,43,44,45
5	A	
	B	
	C	26,15,20,29,18,42,43,44,45
	D	26,15,20,29,40,18,42,43,44,45
6	A	
	B	
	C	26,15,20,29,18,42,43,44,45,51,52,53
	D	26,15,20,29,18,40,42,43,44,45,51,52,53
7	A	29,18,41
	B	29,18,41
	C	26,23,15,20,29,18,41,42,43,44,45,51,52,53
	D	26,23,15,20,29,18,40,41,42,43,44,45,51,52,53
8	A	
	B	
	C	14,15,20,23,26,29,42,43,44,45,51,52,53
	D	14,15,20,23,26,29,40,42,43,44,45,51,52,53
9	A	29,18,41
	B	29,18,41,51,52,53
	C	14,23,26,15,20,29,18,42,41,43,44,45,51,52,53
	D	14,23,26,15,20,19,18,40,42,41,43,44,45,51,52,53
10	A	52,53
	B	52,53
	C	51,52,53
	D	51,52,53
11	A	41
	B	41
	C	41,42
	D	41,42
12	A	
	B	52,52
	C	14,15,20,29,18,42,43,44,45,51,52,53
	D	14,15,20,29,18,40,42,43,44,45,51,52,53
13	A	
	B	
	C	23,15,20,42,43,44,45
	D	23,15,20,40,42,43,44,45

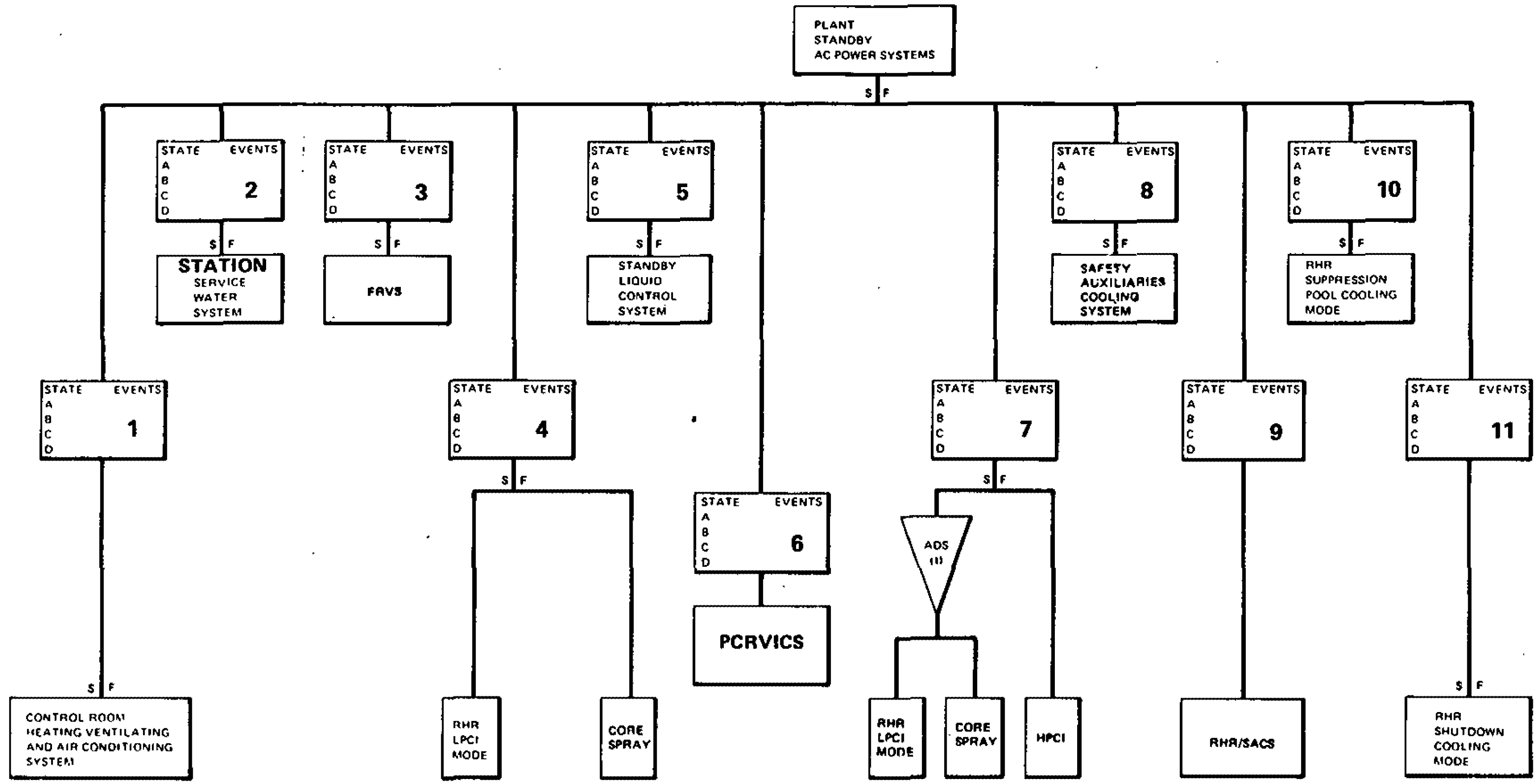
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

COMMONALITY OF AUXILIARY SYSTEMS – DC POWER SYSTEMS (125/250 VOLTS)

UPDATED FSAR

FIGURE 15.9-52



(1) BLOWDOWN
(2) SF REQUIREMENT NOT APPLICABLE IN EVENT 38

INDICATES THAT SYSTEM IS INCLUDED IN COMBINATION BUT DOES NOT REQUIRE THE AUXILIARY POWER

ITEM	STATE	EVENTS
1	A	41
	B	41
	C	41,42,43,44,45
	D	41,42,43,44,45
2	A	18,41,51,52,53
	B	18,41,51,52,53
	C	14,26,23,15,20,29,18,42,43,44,45,51,52,53
	D	14,26,23,15,20,29,18,40,42,43,44,45,51,52,53
3	A	41
	B	41
	C	41,42
	D	41,42
4	A	
	B	
	C	26,15,20,29,18,42,43,44,45
	D	26,15,20,29,18,40,42,43,44,45
5	A	
	B	
	C	51,52,53
	D	51,52,53
6	A	
	B	
	C	23,15,20,42,43,44,45
	D	23,15,20,40,42,43,44,45
7	A	
	B	
	C	15,42,43,44,45
	D	15,42,43,44,45
8	A	29,18
	B	29,18
	C	14,26,23,15,20,29,18,42,43,44,45,51,52,53
	D	14,26,23,15,20,29,18,40,42,43,44,45,51,52,53
9	A	
	B	52,53
	C	26,15,20,29,18,42,43,44,45,51,52,53
	D	26,15,20,29,18,40,42,43,44,45,51,52,53
10	A	
	B	
	C	26,15,20,29,18,42,43,44,45,51,52,53
	D	26,15,20,29,18,40,42,43,44,45,51,52,53

ITEM	STATE	EVENTS
11	A	52,53
	B	52,53
	C	51,52,53
	D	51,52,53

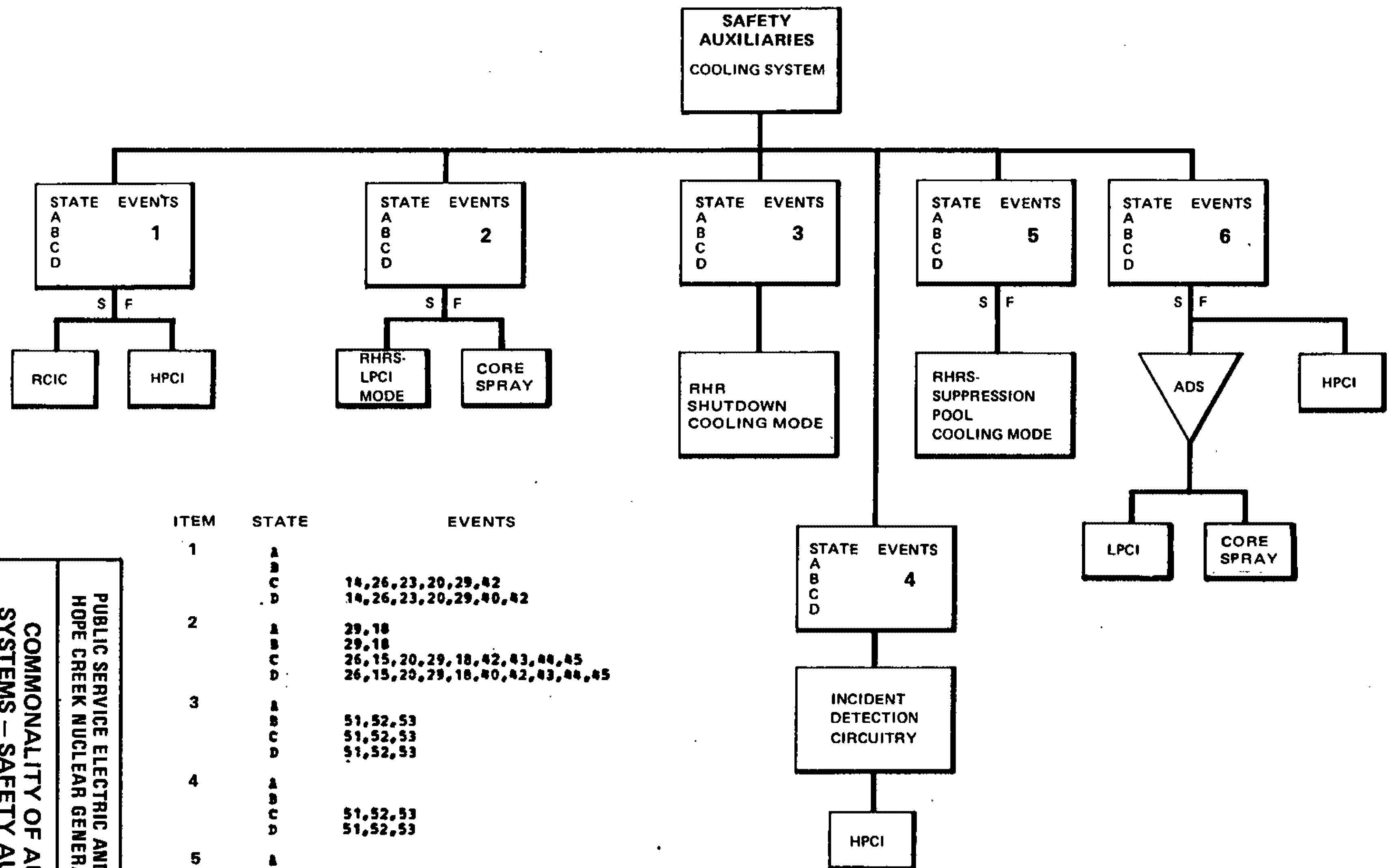
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

COMMONALITY OF STANDBY
AC POWER SYSTEMS
(125/480/4160 VOLTS)

UPDATED FSAR

FIGURE 15.9-53



ITEM	STATE	EVENTS
1	A	
	B	
	C	14,26,23,20,29,42
	D	14,26,23,20,29,40,42
2	A	29,18
	B	29,18
	C	26,15,20,29,18,42,43,44,45
	D	26,15,20,29,18,40,42,43,44,45
3	A	
	B	51,52,53
	C	51,52,53
	D	51,52,53
4	A	
	B	
	C	51,52,53
	D	51,52,53
5	A	
	B	
	C	26,15,20,29,18,42,43,44,45,51,52,53
	D	26,15,20,29,18,40,42,43,44,45,51,52,53
6	A	
	B	
	C	15,42,43,44,45
	D	15,42,43,44,45

NOTE: SF REQUIREMENT NOT APPLICABLE IN EVENTS 51, 52, 53

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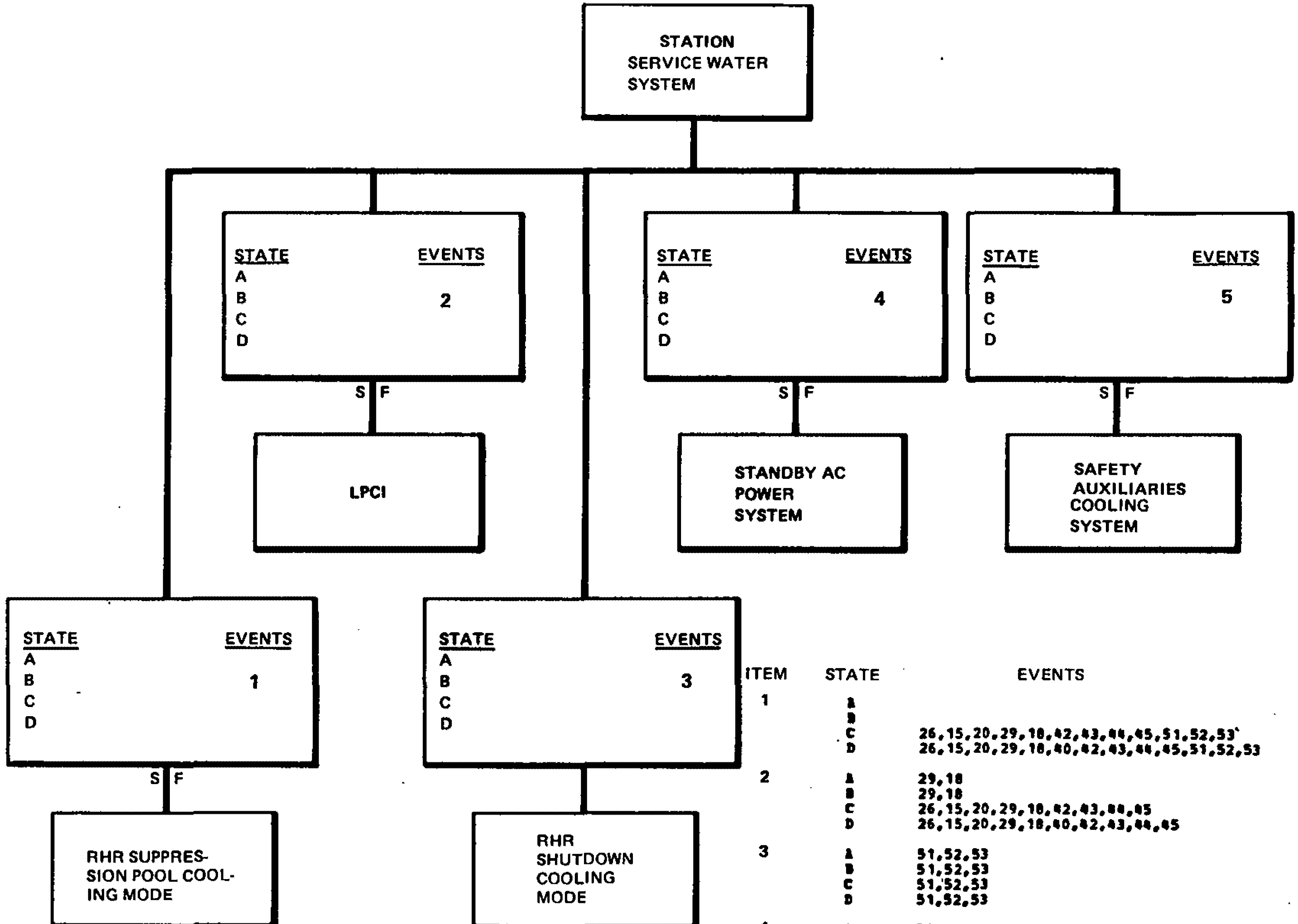
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

COMMONALITY OF AUXILIARY SYSTEMS – SAFETY AUXILIARIES COOLING SYSTEM

UPDATED FSAR

FIGURE 15.9-54



NOTE: SF REQUIREMENT NOT APPLICABLE IN EVENTS 51, 52, 53

ITEM	STATE	EVENTS
1	A	
	B	
	C	26,15,20,29,18,42,43,44,45,51,52,53
	D	26,15,20,29,18,40,42,43,44,45,51,52,53
2	A	29,18
	B	29,18
	C	26,15,20,29,18,42,43,44,45
	D	26,15,20,29,18,40,42,43,44,45
3	A	51,52,53
	B	51,52,53
	C	51,52,53
	D	51,52,53
4	A	29
	B	29
	C	29,42,43,44,45
	D	29,42,43,44,45
5	A	29,18
	B	29,18
	C	14,26,23,15,20,29,18,42,43,44,45,51,52,53
	D	14,26,23,15,20,29,18,40,42,43,44,45,51,52,53

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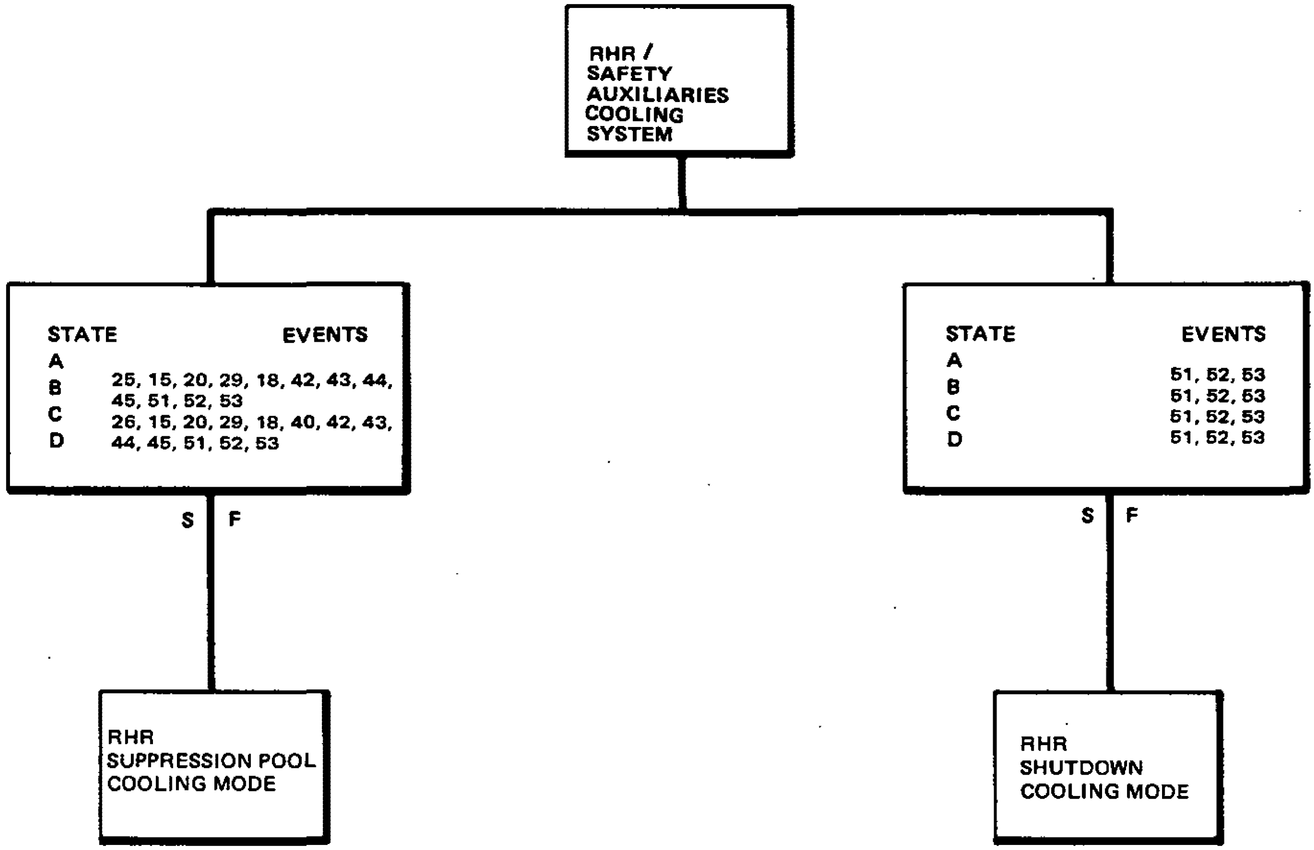
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

COMMONALITY OF
AUXILIARY SYSTEMS –
STATION SERVICE WATER SYSTEM

UPDATED FSAR

FIGURE 15.9-55



NOTE: SF REQUIREMENT NOT APPLICABLE IN EVENTS 51, 52, 53

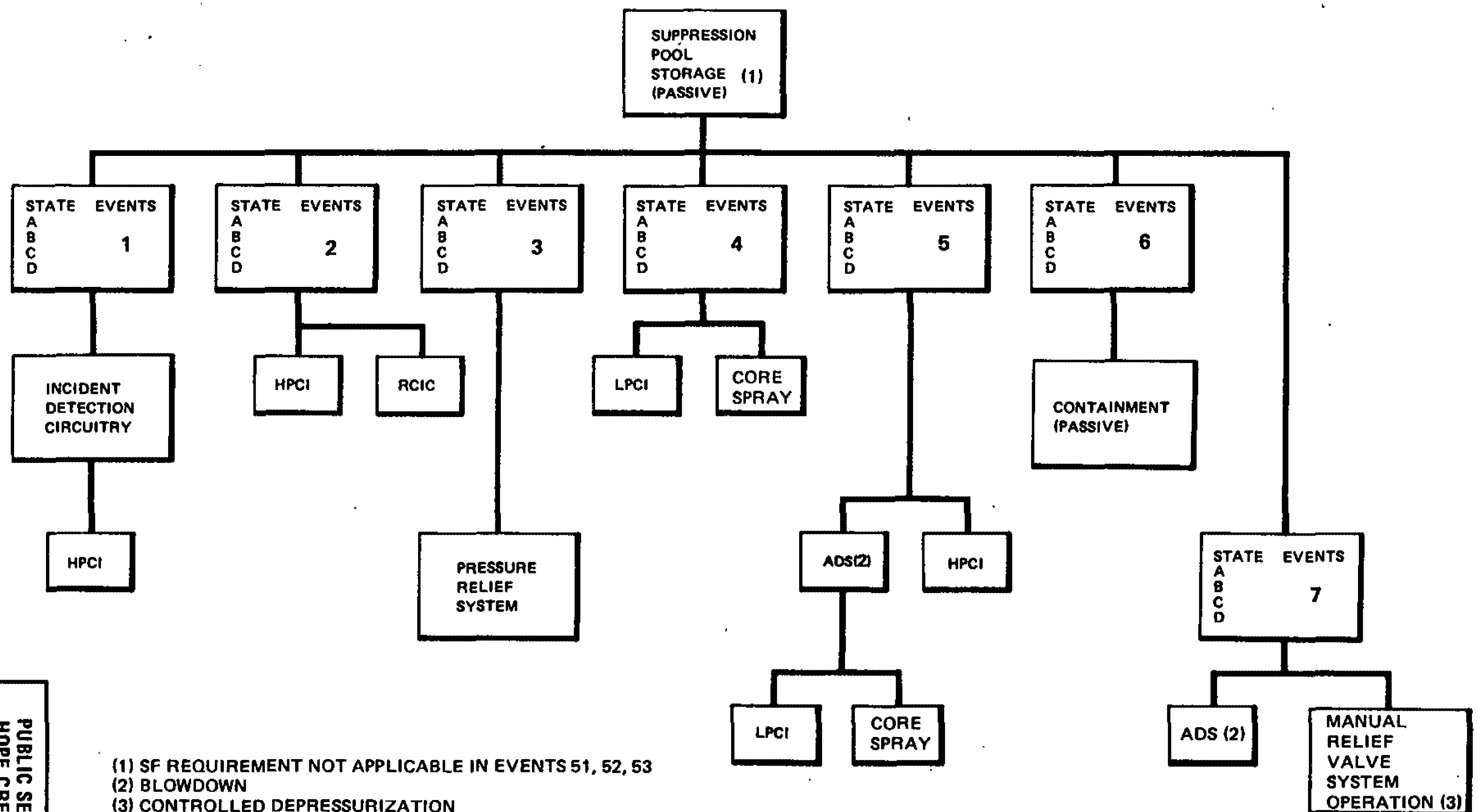
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

COMMONALITY OF AUXILIARY SYSTEMS – RHR/SAFETY AUXILIARIES COOLING SYSTEM

UPDATED FSAR

FIGURE 15.9-56



(1) SF REQUIREMENT NOT APPLICABLE IN EVENTS 51, 52, 53
(2) BLOWDOWN
(3) CONTROLLED DEPRESSURIZATION

ITEM	STATE	EVENTS
1	A	
	B	
	C	51,52,53
	D	51,52,53
2	A	
	B	
	C	14,26,23,20,29
	D	14,26,23,20,29,40
3	A	
	B	
	C	14,26,23,20,29,22,42,43,44,45,51,52,53
	D	30,25,14,26,23,20,29,22,40,42,43,44,45,51,52,53,31,27
4	A	29,18
	B	29,18
	C	26,15,20,29,18,42,43,44,45
	D	26,15,20,29,18,40,42,43,44,45
5	A	
	B	
	C	15,42,43,44,45
	D	15,42,43,44,45
6	A	
	B	23,18
	C	26,23,15,20,18,42,43,44,45,51,52,53
	D	26,23,15,20,18,40,42,43,44,45,51,52,53
7	A	
	B	
	C	26,15,20,29,18,42,43,44,45
	D	26,15,20,29,18,40,42,43,44,45

REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

COMMONALITY OF AUXILIARY SYSTEMS WITH RESPECT TO SUPPRESSION POOL STORAGE

UPDATED FSAR

FIGURE 15.9-57