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15.0 GENERAL

This chapter discusses the effects of anticipated process disturbances and postulated component failures, their consequences, and the capabilities built into the plant to control or accommodate such failures and events. The analyses have been reviewed and revised, as needed, for the:

- Reactor power uprate,
- Installation of the adjustable speed drive for the reactor recirculation pumps,
- Implementation of an alternative source term, and
- Cycle specific changes.

These changes to the plant licensing and design basis and their impact are discussed in this chapter.

The scope of the situations analyzed includes anticipated (expected) operational occurrences, off-design abnormal (unexpected) transients that induce system operating condition disturbances, postulated accidents of low probability, and hypothetical events of extremely low probability. For each reload, the events are evaluated by the fuel vendor(s). The events identified as limiting during the evaluation are analyzed and the sections are revised.

The plant was originally licensed at $\boxed{3323}$ MWt. In 1995, an amendment to the plant Operating License authorized an increase in power to $\boxed{3486}$ MWt. The power uprate analysis was performed in accordance with the NRC-approved General Electric Company (GE) generic power uprate program for boiling water reactors (BWRs).

The postulated events in this chapter have been analyzed for power uprate conditions. The only exceptions to using uprated power are some non-limiting single loop operation (SLO) transients that were not reanalyzed as part of the GE power uprate transient analysis. Their text and figures are clearly marked with ORIGINAL POWER designation. Limiting events in terms of setting the fuel operating limits (e.g., Loss of Feedwater Heating, Generator Load Rejection Without Bypass) are reanalyzed on a cycle specific basis and therefore, may include fuel vendor results and references.

The events in this chapter have been analyzed for application of the adjustable speed drives (ASD) in place of the former reactor recirculation control system that used flow control valves (FCV). The uprated power for Columbia Generating Station is 3486 MWt which is 4.9% higher than the original licensed power of 3323 MWt. All transient initial conditions are specified in Table 15.0-2, 15.0-2A, 15.0-2B or the individual transient event description sections. Several performance improvement packages have been included in the analysis:

1. Final Feedwater Temperature Reduction (FFWTR) and Feedwater Heaters Out-Of-Service (FWHOOS) - Analyses were performed at a reduced feedwater temperature at rated thermal power for operations at end-of-cycle and during the cycle for limiting transients.

2. Increased Core Flow (ICF) – Increased core flow allows for operation at 106% of the rated core flow. The limiting transients were performed for the end-of-cycle with control rods fully withdrawn. This envelopes the operation at increased core flow condition throughout the cycle.

3. Extended Load Line Limit Analysis (ELLLA) - The consequences of the transients were evaluated to determine if operating limit adjustments are necessary for operation in the extended operating domain and compared with the evaluation at rated thermal power and increased core flow region. This comparison ensures bounding of the results at the extended operating domain.

4. Single Loop Operation (SLO) - Limiting transients were re-analyzed for operation at SLO. Using adjustable speed drives (ASD), GE determined the maximum active loop's recirculation flow at 105% of rated pump speed with resultant analyzed power and core flow conditions of 75% UP and 57% of rated core flow for SLO. Prior to power uprate, a comprehensive SLO analysis was performed. For non-limiting events, this analysis has been retained for completeness and historical purposes. These analyses are clearly marked with ORIGINAL POWER designation.

5. End-of-Cycle RPT Out-of-Service (RPT OOS) - The recirculation pump trip (RPT) mitigates several transients that are more severe at end-of-cycle. The limiting transients were re-analyzed with RPT OOS at various power/flow conditions.

6. Turbine Bypass Out-of-Service - Limiting transient events have been analyzed with turbine bypass valves out-of-service at limiting power and flow conditions.

Additional updates address the implementation of the use of alternative source terms (AST) as described in the Regulatory Guide 1.183, July 2000, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." This methodology is based on the advances that have been made in understanding the timing, magnitude, and the chemical form of fission product releases from severe nuclear power plant accidents. The accidents that were reanalyzed by Energy Northwest with the AST:

- Loss of coolant accident (LOCA)
- Fuel handling accident (FHA)
- Control rod drop accident (CRDA)
- Main steam line break, outside containment, accident (MSLB)

The radiological consequences of these accidents were determined based on AST approved for use under 10 CFR 50.67. In accordance with the guidance provided in Regulatory Guide 1.183, the licensing and design basis are revised to reflect the application of full scope AST methodology (with the exception that the TID-14844 will continue to be used as the basis for equipment qualification (EQ) and radiation zone maps/shielding calculations). The accidents analyzed as part of the implementation of AST are subject to the limits specified in 10 CFR 50.67 and guidelines of Regulatory Guide 1.183.

15.0.1 ANALYTICAL OBJECTIVE

The spectrum of postulated initiating events is divided into categories based on the type of disturbance and the expected frequency of the initiating occurrence. The limiting events in each combination of category and frequency are quantitatively analyzed.

15.0.2 ANALYTICAL CATEGORIES

Transient and accident events contained in this report are provided in individual categories as specified by Regulatory Guide 1.70, Revision 2. The results of the events are summarized in Table 15.0-1. Events evaluated are assigned to one of the following applicable categories:

a. Decrease in reactor coolant temperature:

   Reactor vessel water (moderator) temperature reduction results in an increase in core reactivity. This could lead to fuel-cladding damage.

b. Increase in reactor pressure:

   Nuclear system pressure increases threaten to rupture the reactor coolant pressure boundary (RCPB). Increasing pressure also collapses the voids in the core-moderator, thereby increasing core reactivity and power level that could threaten fuel cladding due to overheating.

c. Decrease in reactor coolant system flow rate:

   A reduction in the core coolant flow rate could overheat the cladding as the coolant becomes unable to adequately remove the heat generated by the fuel.

d. Reactivity and power distribution anomalies:

   Transient events included in this category are those that could cause rapid increases in power due to increased core flow disturbance events. Increased

core flow reduces the void content of the moderator increasing core reactivity and power level.

e. Increase in reactor coolant inventory:

Increasing coolant inventory could result in excessive moisture carryover to components such as the main turbine, feedwater turbines, etc.

f. Decrease in reactor coolant inventory:

Reductions in coolant inventory could threaten the fuel as the coolant becomes less able to remove heat generated in the core.

g. Radioactive release from subsystems and components:

Loss of integrity of a radioactive containment component is postulated.

h. Anticipated transients without scram:

To determine the capability of plant design to accommodate an extremely low probability event, a multi-system maloperation situation is postulated.

15.0.2.1 Single Loop Operation (SLO)

Operation with one recirculation loop results in a maximum power output that is 20% to 30% below that which is attainable for two-pump operation. Therefore, the consequences of abnormal operation transients from one-loop operation will be considerably less severe than those analyzed from a two-loop operational mode because of the associated reduction in operation power level.

For pressurization, flow decrease, and cold water increase transients, results presented bound both the thermal and overpressure consequences of one-loop operation. The consequences of flow decrease transients are also bounded by the full power analysis. A single pump trip from one-loop operation is less severe than a two-pump trip from full power because of the reduced initial power level.

Cold water increase transients can result from either recirculation flow controller failure or introduction of colder water into the reactor vessel by events such as loss of feedwater heater. For the former, the flow-dependent minimum critical power ratio (MCPR) values are derived assuming both recirculation loop controllers fail. This condition produces the maximum possible power increase and, hence, maximum ΔMCPR for transients initiated from less than rated power and flow. When operating with only one recirculation loop, the flow and power increase associated with this failure with only one recirculation loop will be less than that

associated with both loops; therefore, the MCPR values derived with the two-pump assumption are conservative for SLO. The latter event, loss of feedwater heating, is generally the most severe cold water increase event with respect to increase in core power. This event is caused by positive reactivity insertion from core inlet subcooling and it is relatively insensitive to initial power level. A generic statistical loss of feedwater heater analysis using different initial power levels and other core design parameters concluded one-pump operation with lower initial power level is conservatively bounded by the full power two-pump analysis. Inadvertent restart of the idle recirculation pump has been analyzed and is applicable for SLO.

From the above discussions, the transient consequence from SLO is bounded by previously submitted full power analyses. The maximum power level that can be attained with one-loop operation is only restricted by the MCPR and overpressure limits established from a full-power analysis.

The following most limiting transients of coldwater increase, pressurization and flow decrease events are analyzed for SLO and the results are shown in Table 15.0-1A:

a. Feedwater flow controller failure (maximum demand),
b. Generator load rejection with bypass failure, and
c. One pump seizure accident.

15.0.3 EVENT EVALUATION

15.0.3.1 Identification of Causes and Frequency Classification

Situations and causes that lead to the initiating event analyzed are described within the analytical categories. The frequency of occurrence of each event is summarized based on operating plant history for the transient event. Events for which inconclusive data exist are discussed separately within each event section.

Each initiating event within the major groups is assigned to one of the following frequency groups:

a. Incidents of moderate frequency - these are incidents that may occur during a calendar year to once per lifetime. This event is referred to as an “anticipated (expected) operational transient.”

b. Infrequent incidents - these are incidents that may occur during the life of the particular plant. This event is referred to as an “abnormal (unexpected) operational transient.”

c. Limiting faults - these are occurrences that are not expected to occur but are postulated because their consequences may result in the release of significant

amounts of radioactive material. This event is referred to as a "design basis (postulated) accident."

15.0.3.1.1 Unacceptable Results for Incidents of Moderate Frequency [Anticipated (Expected) Operational Transients]

The following are unacceptable safety results for incidents of moderate frequency:

a. Release of radioactive material to the environs that exceeds the limits of 10 CFR 20,

b. Reactor operation induced fuel cladding failure,

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

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

15.0.3.1.2 Unacceptable Results for Infrequent Incidents [Abnormal (Unexpected) Operational Transients]

The following are unacceptable safety results for infrequent incidents:

a. Release of radioactivity that results in dose consequences that exceed a small fraction of 10 CFR 50.67 values,

b. Fuel damage that would preclude resumption of normal operation after a normal restart,

c. Generation of a condition that results in consequential loss of function of the reactor coolant system, and

d. Generation of a condition that results in a consequential loss of function of a necessary containment barrier.

15.0.3.1.3 Unacceptable Results for Limiting Faults [Design-Basis (Postulated) Accidents]

The following are unacceptable safety results for limiting faults:

a. Radioactive material release that results in dose consequences that exceed the requirements of 10 CFR 50.67,

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

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

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

e. Radiation exposure to plant operations personnel in the main control room in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident.

15.0.3.2 Sequence of Events and Systems Operation

Each transient or accident is discussed and evaluated in terms of

a. A step-by-step sequence of events from initiation to final stabilized condition (e.g., termination of the accident),

b. The extent to which normally operating plant instrumentation and controls are assumed to function,

c. The extent to which plant and reactor protection systems are required to function,

d. The credit taken for the functioning of normally operating plant systems,

e. The operation of engineered safety systems that is required, and

f. The effect of a single failure or an operator error on the event.

The transient or accident discussion is specific to the event in that it is limited to the events and system operations related to the reactor core performance and postulated damage. In general, the step-by-step description ends when the analysis has demonstrated that the core performance results are within established limits. The stabilized condition does not imply that all actions to stabilize plant parameters or to recover from the transient or accident have been completed by plant personnel. In the stabilized condition, either the core has demonstrated compliance with requirements or the postulated or deterministic damage is complete. At this point, the transient or accident is terminated. After termination of the event, the operator actions or system operations are not event specific. The required actions and expected system operations, needed to establish cold shutdown or to initiate recovery actions, are symptom based and described in

procedures. The events associated with a radiological release and radiological consequences of the transient or accident are also discussed.

15.0.3.2.1 Single Failures or Operator Errors

15.0.3.2.1.1 General. The events considered in this section were evaluated and are provided in this chapter in accordance with Regulatory 1.70, Revision 2.

15.0.3.2.1.2 Initiating Event Analysis.

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

b. The undesired starting or stopping of any single component,

c. The malfunction or maloperation of any single control device,

d. Any single electrical component failure, or

e. Any single operator error.

Operator error is defined as an active deviation from written operating procedures or nuclear plant standard operating practices. The set of actions is limited as follows:

a. Those actions that could be performed by one person,

b. Those actions that would have constituted a correct procedure had the initial decision been correct, and

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

Examples of single operator errors are as follows:

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

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

c. An incorrect calibration of an average power range monitor (APRM), and

d. Manual isolation of the main steam lines as a result of operator misinterpretation of an alarm or indication.

15.0.3.2.1.3 Single Active Component Failure or Single Operator Error Analysis.

a. The undesired action or maloperation of a single active component, or

b. Any single operator error where operator errors are defined as in Section 15.0.3.2.1.2.

15.0.3.3 Core and System Performance

Fuel thermal and hydraulic design are described in Section 4.4.

The fuel cladding integrity safety limit is set so that no fuel damage is calculated to occur if the limit is not violated. Exceeding unacceptable results criteria for fuel cladding integrity for anticipated operational transients is avoided by meeting the following criteria provided in the NRC Standard Review Plan (NUREG-0800) Section 4.4:

a. The expected number of fuel rods in boiling transition should not exceed 0.1% of the fuel rods in the core. This criterion is met by ensuring that the MCPR for any anticipated operational transient is calculated to be not less than the safety limit MCPR values given in the cycle-specific Core Operating Limits Report (COLR).

b. No fuel centerline melting nor uniform total cladding strain in excess of 1% will occur. This criterion is met by compliance with the operating limits for linear heat generation rate (LHGR) given in the cycle-specific COLR.

The operating limit for MCPR is developed as follows:

The MCPR calculated during the transient is compared to the safety limit. The MCPR safety limit is established using the critical power evaluation methods and includes consideration of the operating domain and manufacturing uncertainties and a conservative core power distribution as inputs. The operating limit MCPR is established such that the transient ΔCPR for the dynamic anticipated operational occurrences and quasi steady-state anticipated operational occurrences are included in the evaluation. Thus, the operating limit MCPR is specified to maintain an adequate margin to boiling transition.

The MCPR operating limit is the maximum of (a) the applicable exposure dependent, full power and full flow MCPR limit, (b) the applicable exposure and power dependent MCPR limit, and (c) the flow dependent MCPR limit as specified in the cycle-specific COLR. This stipulation ensures that the safety limit MCPR will not be violated throughout the CGS

operating regime. Full power MCPR limits are specified to define operating limits at rated power and a range of flow conditions that support extended load line operation. Power dependent MCPR limits are specified to define operating limits at other than rated power conditions. A flow dependent MCPR limit is specified to define operating limits at other than rated flow conditions.

Extended load line limit analysis (ELLLA) operation extends the power and flow operating regime for CGS above the rated rod line. The COLR defines the maximum allowable rod line for ELLLA operation. The cycle specific Supplemental Reload Licensing Report (Reference 15.0-1) documents the reload analyses in support of ELLLA operation.

The CGS cycle-specific COLR provides the average planar linear heat generation rate (APLHGR) limits, the MCPR limits, and the linear heat generation rate (LHGR) limits as required by the Technical Specifications.

15.0.3.3.1 Mathematical Model

Unless otherwise stated in the Mathematical Model description for the event being discussed, the following mathematical model was used to perform the Chapter 15 transient analyses.

Transients are analyzed using one of two transient analysis models described in Reference 15.0-3. The one-dimensional transient analysis model ODYN was used to analyze transients involving significant reactor pressurization (i.e., limiting events). The point-kinetics transient analysis model REDY was used for transients not involving significant reactor pressurization (i.e., non-limiting events). The transient analysis model determines the transient pressure, power, heat flux, and average core flow which are required as input to both the ISCOR hot channel analysis and to the TASC transient critical power methodology. The ODYN transient analysis model also calculates the transient peak reactor vessel pressure to demonstrate conformance to the reactor pressure vessel safety limit, which is based on the reactor pressure vessel design pressure.

The overall system model consists of a one-dimensional representation (ODYN) or point kinetics representation of the core (REDY), and representation of the nuclear steam supply system including the reactor vessel, steamline, recirculation, feedwater system, recirculation control system, feedwater control system, and pressure regulator. The main steamline model incorporates mass and momentum balances over multiple nodes allowing for the modeling of the acoustic wave phenomena present in the steamline during transients. In addition, the model provides the capability for simulating the high pressure flooding system, the reactor core isolation cooling system, and the standby liquid control system as necessary for the event to be simulated.

The input data to the transient analysis model come from two sources: (1) the plant model or base deck, and (2) the BWR three-dimensional simulator PANACEA. The plant model

provides the necessary input data for the simulation of the plant, including the plant and control systems performance characteristics. The BWR three-dimensional simulator supplies the core state, the neutron kinetics cross sections, and other data necessary to characterize the reactor core.

For the event to be evaluated, a steady-state initialization is performed and then the parameter changes during the transient are calculated. The steady state initialization includes the recirculation loops and reactor vessel internal pressure drops, core exit pressure, and core inlet flow and enthalpy inputs to the reactor core model. The values are used in the reactor core model to calculate the neutron kinetics, thermal-hydraulics and fuel temperature at steady-state conditions. During the transient, the system model calculates the time dependent response of pressure, flow, neutron flux and heat flux. Plant control responses for systems such as turbine control and recirculation flow control and feedwater flow control are also calculated.

The hot channel analysis is performed using ISCOR to determine the flow distribution in the core during the transient, and to establish the flow to the limiting channels of each type in the core to be analyzed using the transient critical power methodology. The hot channel analysis is based on the transient parameter changes provided by the transient analysis model.

The TASC transient critical power calculation methodology is used to calculate the change in CPR from the initial CPR assumed for the transient being evaluated. This defines the delta-CPR during the transient. The GEXL transient critical power calculation method is used to calculate CPR.

The above mathematical models describe the models used in the power uprate analysis. Cycle specific analyses are performed using vendor specific models for the vendor supplying the reload fuel for the current cycle as described elsewhere in this chapter.

15.0.3.3.2 Input Parameters and Initial Conditions for Analyzed Events

This section discusses the important input parameters used in the analysis for the event discussed. In some cases, the discussion references Table 15.0-2 (or 2A or 2B).

15.0.3.3.3 Consideration of Uncertainties

Except for total core flow and TIP reading, the uncertainties used in the statistical analysis to determine the MCPR fuel cladding integrity safety limit are not dependent on whether coolant flow is provided by one or two recirculation pumps. Uncertainties used in the two-loop operation analysis are documented in the FSAR. A 6% core flow measurement uncertainty has been established for single loop operation (compared to 2.5% for two-loop operation). This value conservatively reflects the one standard deviation (one sigma) accuracy of the core flow measurement system documented in Reference 15.0-2. In SLO, measurement and prediction uncertainties for radial power distribution and axial power distribution also increase. In the

current methodology, axial power uncertainty is not an important parameter. The net effect of these revised uncertainties is an incremental increase in the required MCPR fuel cladding integrity safety limit. The MCPR safety limit for SLO is given in the cycle specific COLR.

15.0.3.3.3.1 Core Flow Uncertainty Analysis

The uncertainty analysis procedure used to establish the core flow uncertainty for one-pump operation is essentially the same as for two-pump operation, with some exceptions. The core flow uncertainty analysis is described in Reference 15.0-2. The analysis of one-pump core flow uncertainty is summarized below.

For SLO, the total core flow can be expressed as follows (see Figure 15.0-2):

$$W_C = W_A - W_I$$

where:

W_C = total core flow

W_A = active loop flow, and

W_I = inactive loop (true) flow.

By applying the "propagation of errors" method to the above equation, the variance of the total flow uncertainty can be approximated by:

$$\sigma^2_{W_C} = \sigma^2_{W_{sys}} + \left[\frac{1}{1-a}\right]^2 \left(\sigma^2_{W_{A_{rand}}}\right) + \left[\frac{a}{1-a}\right]^2 \left(\sigma^2_{W_{I_{rand}}} + \sigma^2_C\right)$$

where:

σ_{W_C} = uncertainty of total core flow;

$\sigma_{W_{sys}}$ = uncertainty systematic to both loops;

$\sigma_{W_{A_{rand}}}$ = random uncertainty of active loop only;

$\sigma_{W_{I_{rand}}}$ = random uncertainty of inactive loop only;

σ_C = uncertainty of "C" coefficient; and

a = ratio of inactive loop flow (WI) to active loop flow (WA).

From an uncertainty analysis, the conservative, bounding values of

$$\sigma_{W_{sys}}, \sigma_{W_{A_{rand}}}, \sigma_{W_{I_{rand}}}, \text{ and } \sigma_C$$

are 1.6%, 2.6%, 3.5% and 2.8% respectively. Based on the above uncertainties and a bounding value of 0.36* for "a", the variance of the total flow uncertainty is approximately:

$$\sigma^2_{W_C} = (1.6)^2 + \left[\frac{1}{1-0.36}\right]^2 (2.6)^2 + \left[\frac{0.36}{1-0.36}\right]^2 \left((3.5)^2 + (2.8)^2\right) = (5.0\%)^2$$

When the effect of 4.1% core bypass flow split uncertainty at 12% (bounding case) bypass flow fraction is added to the above total core flow uncertainty, the active coolant flow uncertainty is:

$$\sigma^2_{\text{active coolant}} = (5.0\%)^2 + \left[\frac{0.12}{1-0.12}\right]^2 (4.1\%)^2 = (5.1\%)^2$$

which is less than the 6% core flow uncertainty assumed in the statistical analysis.

In summary, core flow during one-pump operation is measured in a conservative way and its uncertainty has been conservatively evaluated.

15.0.3.3.4 Results

This section discusses the results, in terms of core and system performance, of the event analyzed. The COLR provides operating limits that are the results of analytical evaluations that impact core operating parameters for the current cycle. In addition, critical parameters for the complete set of transients analyzed are shown in Table 15.0-1. From the data in Table 15.0-1, an evaluation of the limiting event for that particular category and parameter can be made. The limiting events are reanalyzed for the current operating cycle. Table 15.0-3 provides a summary of accidents that may have radiological consequences.

* This flow split ratio varies from about 0.13 to 0.36. The 0.36 value is a conservative bounding value. The analytical expected value of the flow split ratio for CGS is ~0.23.

15.0.3.4 Barrier Performance

This section addresses the performance of the RCPB and the containment system during transients and accidents.

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

Piping systems within the secondary containment structure (i.e., the reactor building) have been analyzed for pipe break effects including jet impingement, jet reaction, pipe whip, and subcompartment pressurization. Where necessary, these loads were included in the design of the structure to ensure that the secondary containment can perform its required functions as defined in Section 6.2.3.

15.0.3.5 Radiological Consequences

This section addresses the radiological release consequences during the incidents of moderate frequency (anticipated operational transients), infrequent incidents (abnormal operational transients), and limiting faults (design basis accidents [DBA]) events. For all events where consequences are limiting a detailed quantitative evaluation is presented. For nonlimiting events, a qualitative evaluation is presented or the results are referenced from a more limiting or enveloping case or event.

For limiting faults (DBA), conservative assumptions considered to be acceptable to the NRC for the purpose of worst case bounding of the event and determining the adequacy of the plant design to meet 10 CFR 50.67 requirements are assumed. This is referred to as the "design basis analysis."

The atmospheric dispersion coefficients are presented in Tables 15.0-4 and 15.0-5. Reference will be made to these tables in the discussion of the analyses.

15.0.4 REFERENCES

15.0-1 Supplemental Reload Licensing Report for Columbia (most recent version referenced in COLR).

15.0-2 General Electric Company, General Electric BWR Thermal Analysis Basis (GETAB); Data, Correlation, and Design Application, NEDO-10958-A, January 1977.

15.0-3 GE Nuclear Energy, "WNP-2 Power Uprate Transient Analysis Task Report," GE-NE-208-08-0393, Revision 0, September 1993 (Proprietary).

15.0-4 GE-Hitachi Report, "Evaluation of Steam Flow Induced Error Impact on the L3 Setpoint Analysis Limit," GEH-NE-0000-0077-4603, December 2007.

Table 15.0-1

Results Summary of Transient Events Applicable to Columbia Generating Station[a]

Paragraph I.D.	Figure I.D.	Description	Maximum Neutron Flux (%NBR)	Maximum Dome Pressure (psig)	Maximum Vessel Pressure (psig)	Maximum Steam Line Pressure (psig)	Maximum Core Average Surface Heat Flux (% of Initial)	DCPR[b,c,e]	Frequency Category
15.1		DECREASE IN REACTOR COOLANT TEMPERATURE							
15.1.1		Loss of Feedwater Heating, Manual Flow Control						0.14	(d)
15.1.2	15.1-1	Feedwater Controller Failure, Max Demand	210	1146	1168	1145	114	0.27	(d)
15.1.3	15.1-2	Pressure Regulator Fail-Open	131	1151	1172	1151	100	<0.01	
15.1.4		Inadvertent Opening of Safety or Relief Valve	104	1020	1061	1012	100	<0.01	(d)
15.1.6		RHR Shutdown Cooling Malfunction Decreasing Temperature	See text						(d)
15.2		INCREASE IN REACTOR PRESSURE							
15.2.1		Pressure Regulator Fail-Closed	163	1188	1220	1187	106	n/a	(d)
15.2.2	15.2-1	Generator, Load Rejection, Bypass-On[f]	See text						(d)
15.2.2	15.2-2	Generator Load Rejection, Bypass-Off	275	1238	1260	1235	111	0.30	(d)
15.2.3	15.2-3	Turbine Trip, Bypass-On[f]	See text						(d)
15.2.3	15.2-4	Turbine Trip, Bypass-Off	278	1238	1260	1235	111	0.30	(d)
15.2.4	15.2-5	Inadvertent MSIV Closure	206	1200	1234	1198	100	0.022	(d)
15.2.5	15.2-6	Loss of Condenser Vacuum	256	1173	1199	1166	111	0.12	(d)
15.2.6	15.2-7	Loss of Auxiliary Power Transformers	106[h]	1169	1185	1166	100	<0.01	(d)
15.2.6	15.2-8	Loss of All Grid Connections	196	1173	1196	1166	106	0.079	(d)
15.2.7	15.2-9	Loss of all Feedwater Flow	106[h]	1142	1152	1142	100	<0.01	(d)
15.2.8		Feedwater Piping Break	See Section 15.6.6						
15.2.9		Failure of RHR Shutdown Cooling	See text						

Table 15.0-1

Results Summary of Transient Events Applicable to Columbia Generating Station[a] (Continued)

Paragraph I.D.	Figure I.D.	Description	Maximum Neutron Flux (%NBR)	Maximum Dome Pressure (psig)	Maximum Vessel Pressure (psig)	Maximum Steam Line Pressure (psig)	Maximum Core Average Surface Heat Flux (% of Initial)	DCPR[b,c,e]	Frequency Category
15.3		DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE							
15.3.1	15.3-1	Trip of One Recirculation Pump Motor	106[h]	1020	1059	1012	100	< 0.01	(d)
15.3.1	15.3-2	Trip of Both Recirculation Pump Motors	106[h]	1077	1088	1076	100	< 0.01	(d)
15.3.2	15.3-3	Speed Decrease of One Main Recirc Motor	106[h]	1020	1059	1012	100	< 0.01	(d)
15.3.2	15.3-4	Speed Decrease of Two Main Recirc Motors	106[h]	1061	1072	1061	100	< 0.01	(d)
15.3.3	15.3-5	Seizure of One Recirculation Pump	106[h]	1099	1110	1098	100	< 0.01	(i)
15.3.4		Recirc Pump Shaft Break	See 15.3.3						
15.4		REACTIVITY AND POWER DISTRIBUTION ANOMALIES							
15.4.1.1		RWE - Refueling	See text						(j)
15.4.1.2		RWE - Startup	See text						(j)
15.4.2		RWE - At Power	See text						(d)
15.4.3		Control Rod Misoperation	See Sections 15.4.1 and 15.4.2						
15.4.4	15.4-1	Abnormal Startup of Idle Recirculation Loop	124[c]	1004	1026	998	190	0.53	(d)
15.4.5	15.4-2	Speed Increase of One Main Recirc Motor	136[c]	990	1009	986	127	0.15	(d)
15.4.5		Speed Increase of Both Main Recirc Motors	153[c]	1006	1033	1001	149	0.27	(d)
15.4.7		Misplaced Bundle Accident	See text						(j)
15.4.9		Rod Drop Accident							(i)
15.5		INCREASE IN REACTOR COOLANT INVENTORY							
15.5.1	15.5-1	Inadvertent HPCS Pump Start[f]	102[h]	1020	1052	1012	100	< 0.01[g]	(d)
15.5.3		BWR Transients	See appropriate events in Sections 15.1 and 15.2						

[a] Results reflect GE14 fuel introduction, some of which are dependent on fuel design and core loading pattern. Compliance with the event acceptance criteria is demonstrated by cycle-dependent analysis of potentially limiting events just prior to the operation of that cycle. The results are reported in the Supplemental Reload Licensing Report (Reference 15.0-1).

Table 15.0-1

Results Summary of Transient Events Applicable to Columbia Generating Station[a] (Continued)

[b] MCPR operating limits are based on the delta-CPR (DCPR) results from the limiting transient event and the MCPR safety limit defined in the Technical Specifications.

[c] Option B DCPR results are reported.

[d] Moderate frequency.

[e] This value is only for the more limiting GE14 fuel.

[f] Non-limiting event under power uprate conditions (event not reanalyzed).

[g] ODYN results without the adjustment factors delineated in the ODYN Report NEDO-24154, NEDE-24154P.

[h] No increase from initial value.

[i] Limiting fault.

[j] Infrequent incident.

Table 15.0-1A

Summary of Transient Peak Value Results
Single-Loop Operation

Paragraph/ Figure	Description	Maximum Neutron Flux (% NBR)	Maximum System Pressure (psig)	Frequency Category
	Initial condition	75	1020	N/A
15.1.2/ 15.1-3	Feedwater flow controller failure (maximum demand) uprated power	89	1118	(a)
15.2.2	Generator load rejection - uprated power	131	1184	(a)
15.3.3/ 15.3-6	Seizure of active recirculation pump	75	1014	(b)

[a] Moderate frequency incident.
[b] Limiting fault.

Table 15.0-2

Input Parameters and Initial Conditions for Transients

		REDY (ASD Events)	REDY[a]	ODYN
1.	Thermal power level, MWt			
	Licensed value	3486	3323	3486
	Analysis value	3702	3464	3629
2.	Steam flow, lbs/hr analysis value	16.09×10^6	14.98×10^6	15.73×10^6
3.	Core flow, lbs/hr	108.5×10^6	108.36×10^6	$95.5\text{-}115.0 \times 10^6$
4.	FW flow rate, lb/sec analysis value	4471	4161	4362
5.	Feedwater temperature, °F	426	424	426
6.	Vessel dome pressure, psig	1020	1020	1020
7.	Vessel core pressure, psig	1031	1031	1031
8.	Turbine bypass capacity, %NBR	22.7	25	22.7
9.	Core coolant inlet enthalpy, Btu/lb	528.3	529.3	529.6
10.	Turbine inlet pressure, psig	992	975	997
11.	Fuel lattice	8 x 8/9 x 9	8x8	Simulated 8x8/9x9
12.	Core average fuel cladding gap conductance, Btu/sec-ft^2-°F	0.3608	0.1667	Fuel specific
13.	Core leakage flow, %	10.20	11.84	Cycle specific
14.	Required MCPR operating limit	(b)	1.24	(c)
15.	MCPR safety limit	(b)	1.06	(c)
16.	Doppler coefficient (-)¢/°			
	Nominal EOC-1	0.311	0.227	(d)
	Analysis data ASD events			
	1. Increase power	0.295	0.215	
	2. Decrease power	0.327		
17.	Void coefficient (-)¢/% Rated			
	Nominal EOC-1		7.48	(d)
	Analysis data for power increase events	15.93	12.70	(d)
	Analysis data for power decrease events	12.10	7.065	(d)
18.	Core average rated void fraction, % (Steady state)	41.24	41.32	43.1
19.	Scram reactivity, $k analysis data	Figure 15.0-1	Figure 15.0-1	(d)
20.	Control rod drive speed, position versus time	Figure 15.0-1	Figure 15.0-1	Figure 15.0-1
21.	Jet pump ratio, M	2.36	2.41	2.39

Table 15.0-2

Input Parameters and Initial Conditions for Transients (Continued)

		REDY (ASD Events)	REDY[a]	ODYN
22.	Safety/relief valve capacity, % NBR safety valve capacity @ 1241 psig	108.6	111.5	108.6
	Relief valve capacity @ setpoint values in item 25 of this table	@ 1121 psig 98.3 @ 1131 psig 99.1 @ 1141 psig 100.0 @ 1151 psig 100.9 @ 1161 psig 101.7	101.8 102.8 103.7 104.6 105.5	98.3 99.1 100 100.9 101.7
	Manufacturer		Crosby	Crosby
	Quantity installed		18	18
23.	Relief function delay, sec	0.4	0.4	0.4
24.	Relief function response, sec	0.15	0.1	0.15
25.	Setpoints for safety/relief valves Safety function, psig	1200, 1210 1221, 1231 1241	1177, 1187, 1197, 1207, 1217	1200, 1210 1221, 1231 1241
	Relief function, psig	1121, 1131 1141, 1151 1161	1091, 1101, 1111, 1121, 1131	1121, 1131 1141, 1151 1161
26.	Number of valve groupings simulated			
	Safety function, number		5	5
	Relief function, number		5	5
27.	High flux trip analysis setpoint (123 x 1.041), % NBR	128.0	126.20	128[e]
28.	High pressure scram setpoint, psig	1086	1071	1086
29.	Vessel level trips, inches with respect to dryer skirt bottom			
	Level 8 - (L8), in.	59.5	55.5	59.5
	Level 4 - (L4), in.		31.5	30
	Level 3 - (L3), in.	7.5[i]	12.5[i]	(f)
	Level 2 - (L2), in.		(-38)	(f)
30.	APRM thermal trip analysis setpoint (117 x 1.041)% NBR @ 100% core flow	121.8	122.030	121.8[e]
31.	Recirculation pump trip delay, sec	0.190	0.140	0.190
32.	Recirculation pump trip inertia time constant for analysis, sec	6[g]	6[g]	6[g]
33.	RPS response time delay	(h)	(h)	(h)

Table 15.0-2

Input Parameters and Initial Conditions for Transients (Continued)

[a] REDY values reflect the pre-uprate initial conditions. Only limiting events were analyzed for power uprate conditions with FCV.

[b] See COLR.

[c] Not applicable to reload 7/cycle 8 simulation.

[d] ODYN values are calculated within the code.

[e] The thermal multiplier (1.041 = 3629/3486) is used to give a conservative margin that is proportional to the core power.

[f] Parameter not used in the analysis.

[g] The inertia time constant is defined by the expression:

$$t = \frac{2\,\pi\,J_O\,n}{g\,T_O}$$

where

t = inertia time constant (sec)
J_o = pump motor inertia (lb-ft^2)
n = rated pump speed (rps)
g = gravitational constant (ft/sec^2)
T_o = pump shaft torque (lb-ft)

[h] The "maximum overall response time" as addressed in the LCS is utilized for each scram encountered in the Chapter 15 events.

[i] The impact of steam flow induced error on the analytical limit does not impact event descriptions or conclusions (Reference 15.0-4).

Table 15.0-2A

Input Parameters and Initial Conditions for Transients and Accidents for Single-Loop Operation

		Original Rated Power	Uprated Power
1.	Thermal power analysis value (MWt)	2596.9	2615
2.	Flow		
	Steam (1b/hr)	10.79×10^6	10.76×10^6
	Core (lb/hr)	59.0×10^6	61.85×10^6
	Core bypass (lb/hr)	5.88×10^6	6.22×10^6
	Feedwater (lb/hr)	10.79×10^6	10.76×10^6
	Turbine bypass (lb/hr)	5.88×10^6	N/A
	Turbine bypass (% rated)	N/A	23%
3.	Core Inlet Enthalpy (Btu/lb)	510.8	513.7
4.	Pressure		
	Vessel dome (psia)	1020	1008
	Vessel core (psia)	1029.7	1017.7
	Turbine inlet (psia)[a]	960.5	1000
5.	Jet pump ratio (M)	3.2	3.4
6.	Safety/relief valve capacity		
	% NBR @ 1,164 psig	107.1	N/A
	Manufacturer	Crosby	Crosby
	Quantity installed	18	18
	% NBR @ 1241 psig	N/A	108.6
7.	Relief function		
	Delay (sec)	0.4	0.4
	Response (sec)	0.1	0.15

Table 15.0-2A

Input Parameters and Initial Conditions for Transients and Accidents for Single-Loop Operation (Continued)

		Original Rated Power	Uprated Power
8.	Setpoints for safety/relief valves		
	Safety function (psig)	1177, 1187, 1197, 1207, 1217	1200, 1210, 1221, 1231, 1241
	Relief function (psig)	1106, 1116, 1126, 1136, 1146	1121, 1131, 1141, 1151, 1161
9.	Number of valve groupings simulated		
	Safety function (number)	5	5
	Relief function (number)	5	5
10.	Setpoints		
	High flux trip analysis (1.21 x 1.043) (% NBR)	126.2	128
	High pressure scram (psig)	1071	1086
	APRM thermal trip (% NBR @ 100% core flow)	122.03	121.8
11.	Vessel level trips (ft above instrument zero)		
	Level 8 - (L8) (ft)	4.542	
	Level 4 - (L4) (ft)	2.625	
	Level 3 - (L3) (ft)	1.083[b]	
	Level 2 - (L2) (ft)	(-)4.167	
12.	RPT delay (sec)	0.19	0.19
13a.	RPT inertia for analysis (lb/ft^2)	24,500	N/A
13b.	RPT inertia time constant (sec)	N/A	6

[a] Pressure specified at rated power condition. Off-rated power pressure drop is calculated by transient analysis code.

[b] The impact of steam flow induced error on the analytical limit does not impact event descriptions or conclusions (Reference 15.0-4).

Table 15.0-2B

Input Parameters and Initial Conditions for GNF Reload Transient Analyses

	Parameter	Value
1.	Thermal power level, MWt	3486
2.	Steam flow, lbs/hr analysis value	15.01×10^6
3.	Core flow, lbs/hr	$95.5 - 115.0 \times 10^6$
4.	FW flow rate, lb/sec analysis value	4161.6
5.	Feedwater temperature, °F	421.2
6.	Vessel dome pressure, psig	1020
7.	Vessel core pressure, psig	1032
8.	Turbine bypass capacity, %NBR	23.75
9.	Core coolant inlet enthalpy, Btu/lb (rated flow)	527.2
10.	Turbine inlet pressure, psig	990
11.	Fuel lattice	10 x 10 mixed core
12.	Required MCPR operating limit	See COLR
13.	MCPR safety limit	See COLR
14.	Control rod drive speed, position versus time[a]	
15.	Jet pump ratio, M	2.285
16.	Safety/relief valve capacity, % NBR safety valve capacity	See Table 5.2-3
	Manufacturer	Crosby
	Quantity installed	18
17.	Relief valve function	
	delay, sec	0.4
	response, sec	0.15
18.	Safety valve function	
	delay, sec	0.0
	response, sec	0.3

Table 15.0-2B

Input Parameters and Initial Conditions for GNF Reload Transient (Continued)

	Parameter	Value	
19.	Setpoints for safety/relief valves	# in Group	Setpoint
	Safety function, psig	2[b]	1200
		4[b]	1210
		4	1221
		4	1231
		4	1241
	Relief function, psig	2[b]	1156
		4[b]	1166
		4	1176
		4	1186
		4	1196
20.	Number of valve groupings simulated		
	Safety function, number (actual/credited)	5 / 3	
	Relief function, number (actual/credited)	5 / 3	
21.	High flux trip analysis setpoint, % NBR	123.0	
22.	High pressure scram setpoint, psig	1086	
23.	Vessel level trips, inches with respect to instrument zero		
	Level 8 – (L8), in.	59.5	
	Level 4 – (L4), in.	30	
	Level 3 – (L3), in.	2.5[c]	
	Level 2 – (L2), in.	-90 LOCA -70 Non LOCA	
24.	APRM thermal trip analysis setpoint	Not credited	
25.	Recirculation pump trip delay, sec	0.200	
26.	RPS response time delay	See LCS	

[a] See Section 4.6.1.1.2.5.3.
[b] Valve group function not credited in safety analyses.
[c] This allows the correction due to steam flow induced error (Reference 15.0-4).

Table 15.0-3

Summary of Accidents

Paragraph I.D.	Title	Failed Fuel Calculated Value
15.3.3	Seizure of one recirculation pump	None
15.3.4	Recirculation pump shaft break	None
15.4.9	Rod drop accident	850 rods
15.6.2	Instrument line break	None
15.6.4	Steam system pipe break outside containment	None
15.6.5	Loss-of-coolant accident within RCPB	100%
15.6.6	Feedwater line break	None
15.7.1	Main condenser gas treatment system failure	N/A
15.7.3	Liquid radwaste tank failure	N/A
15.7.4	Fuel handling accident	250 rods
15.8	Anticipated transients without scram	None

Table 15.0-4

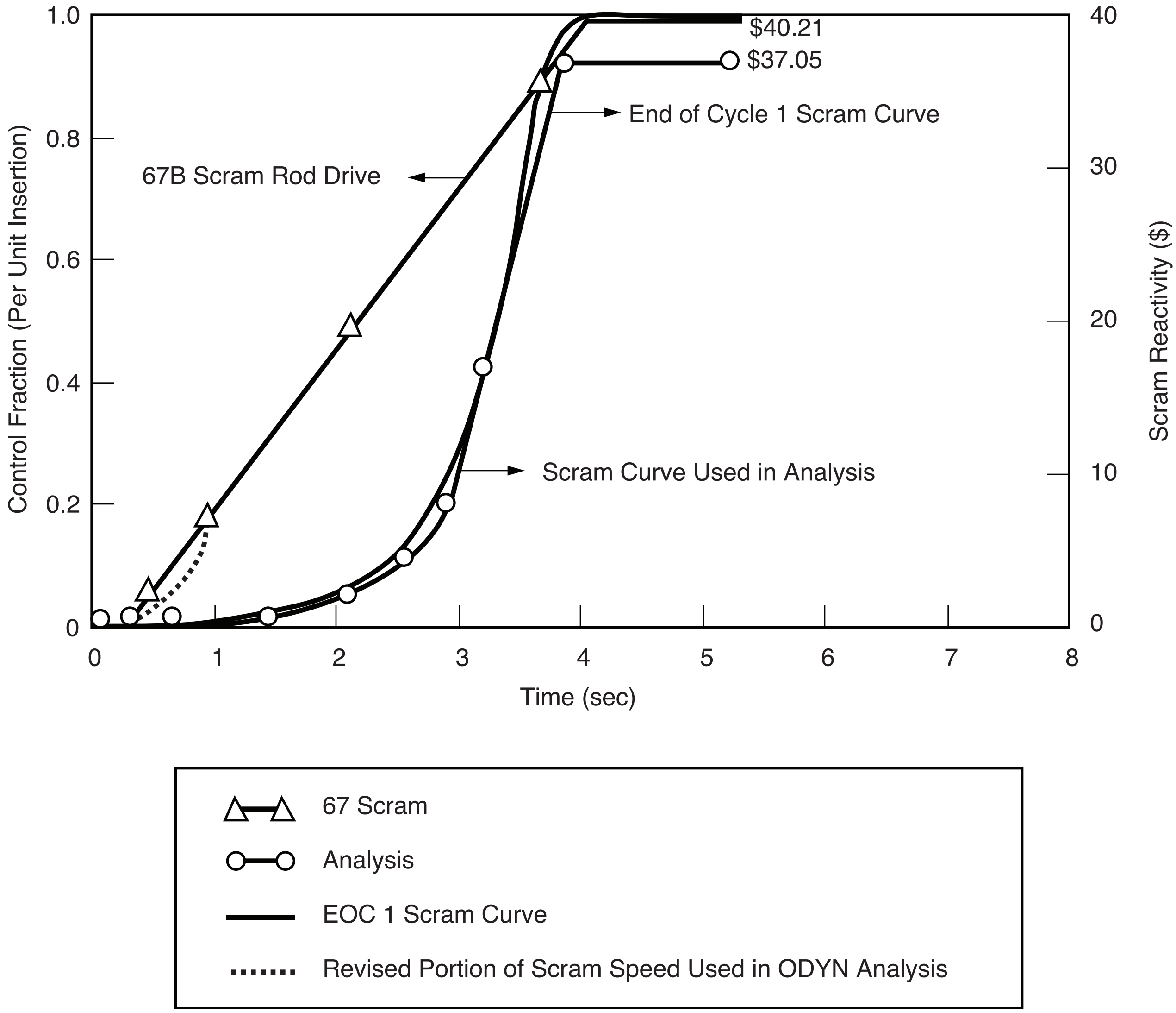
χ/Q (s/m^3) values for the EAB and LPZ

Time Period	EAB χ/Q (s/m^3)	LPZ χ/Q (s/m^3)
0 - 2 hrs	1.81 E-4	4.95 E-5
2 - 8 hrs		4.95 E-5
8 - 24 hrs		3.69 E-5
1 - 4 d		1.95 E-5
4 - 30 d		7.81 E-6

Table 15.0-5

Control Room Atmospheric Dispersion Factors
(sec/m^3)

	Hours	Turbine Building	Secondary Containment	SGT System Release
Filtered Intake Release Path	0 - 2	8.81E-4	2.82E-4	1.43E-4
	2 - 8	3.75E-4	2.17E-4	1.05E-4
	8 - 24	1.93E-4	8.77E-5	4.14E-5
	24 - 96	1.50E-4	7.42E-5	3.52E-5
	96 - 720	1.44E-4	6.40E-5	3.03E-5
Unfiltered Intake Release Path	0 - 2	4.70E-3	7.02E-4	6.95E-4
	2 - 8	2.00E-3	3.19E-4	3.36E-4
	8 - 24	1.03E-3	1.30E-4	1.28E-4
	24 - 96	8.01E-4	1.05E-4	9.72E-5
	96 - 720	7.69E-4	9.00E-5	7.69E-5



Columbia Generating Station Final Safety Analysis Report	Scram Position and Reactivity Characteristics		
	Draw. No. 900547.61	Rev.	Figure 15.0-1

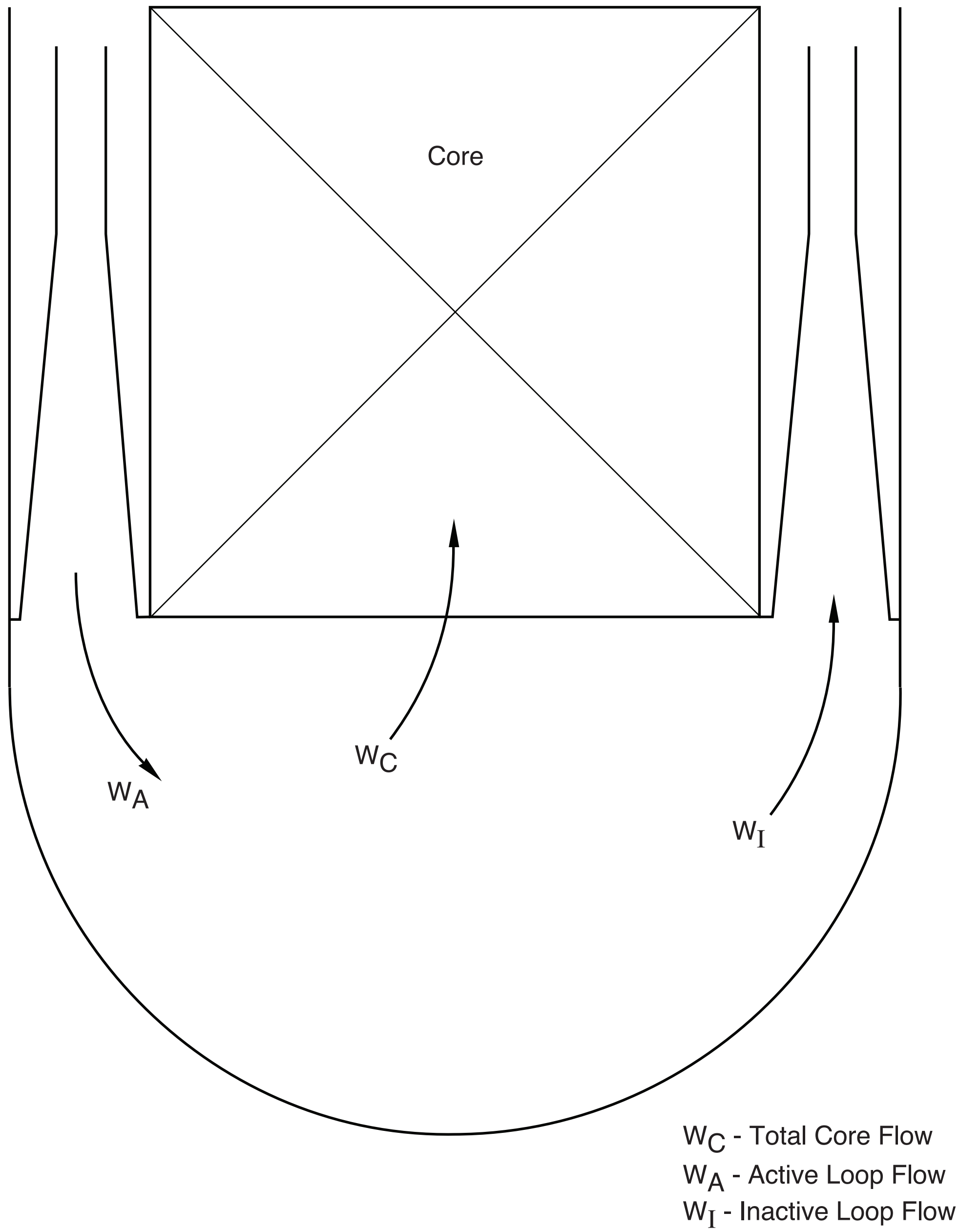


Illustration of Single Recirculation Loop Operation Flows

15.1 DECREASE IN REACTOR COOLANT TEMPERATURE

15.1.1 LOSS OF FEEDWATER HEATING

15.1.1.1 Identification of Causes and Frequency Classification

15.1.1.1.1 Identification of Causes

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

a. Steam extraction line to heater is closed, and
b. Steam is bypassed around heater.

The first case produces a gradual cooling of the feedwater. In the second case, the steam bypasses the heater and no heating of that feedwater occurs. In either case, the reactor vessel receives cooler feedwater. The maximum number of feedwater heaters that can be tripped or bypassed by a single event represents the most severe transient for analysis considerations. This event has been conservatively estimated to incur a loss of up to 100°F of the feedwater heating capability of the plant and causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient.

15.1.1.1.2 Frequency Classification

This event is considered to be an incident of moderate frequency and is analyzed under worst case conditions of a 100°F loss at full power.

15.1.1.2 Sequence of Events and Systems Operation

The loss of feedwater heating leads to a gradual decrease in the temperature of the feedwater entering the reactor vessel. The decrease in feedwater temperature results in an increase in the core inlet subcooling which collapses voids, and increases the core average power. The gradual power change allows fuel thermal response to maintain pace with the increase in neutron flux. For this analysis, it was assumed that the initial feedwater temperature dropped 100°F.

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. Engineered safety feature (ESF) system initiation is not anticipated or required to prevent or mitigate the transient.

The Average Power Range Monitor (APRM) Flow Biased Simulated Thermal Power trip setpoint provides protection against transients such as the Loss of Feedwater Heating where thermal power increases slowly. While the sequence of events may produce sufficiently high

flux levels to initiate an APRM reactor protection system trip, no credit is taken for a reactor trip in the analysis of the event. A description of the APRM system and operation is provided in Sections 7.2.1.1.1.2 and 7.6.1.4.3.

15.1.1.2.1 The Effect of Single Failures and Operator Errors

The loss of feedwater heating generally leads to an increase in reactor power level. The APRM system is the mitigating system and is designed to be single failure proof. Therefore, single failures are not expected to result in a more severe event than analyzed.

15.1.1.3 Core and System Performance

15.1.1.3.1 Mathematical Model

The analytical dynamic behavior has been determined using the steady state boiling water reactor (BWR) simulator code PANACEA (Reference 15.1-2). This code does not provide plots of the dynamic behavior of basic parameters as a function of time nor does it provide information for a sequence of events table. Therefore, no figures or tables are available. Reference 15.1-6 approves the use of PANACEA for modeling the Loss of Feedwater Heating event.

The loss of feedwater heating (LFWH) event analysis supports an assumed 100°F decrease in the feedwater temperature. The result is an increase in core inlet subcooling, which collapses voids, thereby, increasing the core power and shifting the axial power distribution toward the bottom of the core. As a result of the axial power shift and increased core power, voids begin to build up at the bottom of the core, acting as negative feedback to the void collapse process. The negative feedback moderates the core power increase. The PANACEA code is used to determine the change in minimum critical power ratio (MCPR) during the event. Analyses were performed for a range of cycle exposures to ensure that appropriate limits are set. Although there is a substantial increase in core thermal power during the event, the increase in steam flow is much less because a large part of the added power is used to overcome the increase in inlet subcooling. The increase in steam flow is accommodated by the pressure control system by the turbine control (governor) valves or the turbine bypass valves, so no pressurization occurs (Reference 15.1-3).

15.1.1.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2B.

15.1.1.3.3 Results

The LFWH transient is analyzed for each reload core to quantify the reduction in thermal margins. The results of the analysis are provided in the cycle specific Supplemental Reload Licensing Report (Reference 15.1-3).

15.1.1.3.4 Considerations of Uncertainties

Factors such as exposure and magnitude of feedwater temperature change are assumed to be at the worst configuration so that any deviations seen in the actual plant operation reduce the severity of the event.

15.1.1.4 Barrier Performance

The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed. Therefore, barrier integrity and function is maintained.

15.1.1.5 Radiological Consequences

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

15.1.2 FEEDWATER CONTROLLER FAILURE - MAXIMUM DEMAND

15.1.2.1 Identification of Causes and Frequency Classification

15.1.2.1.1 Identification of Causes

This event is postulated on the basis of a single failure of a control device, specifically one that can directly cause an increase in coolant inventory by increasing the feedwater flow. The most severe applicable event is a feedwater controller failure (FWCF) during maximum flow demand. The feedwater controller is forced to its upper limit at the beginning of the event. The event is evaluated for both single and two reactor recirculation loop operations. Because the two-loop operation event is bounding, the core performance analysis is limited to the feedwater controller failure during two-loop operation. However, the MCPR operating limit for single loop operation (SLO) is obtained by adding the ΔCPR from two-loop operation to the MCPR safety limit (SLMCPR) for SLO.

15.1.2.1.2 Frequency Classification

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

15.1.2.2 Sequence of Events and Systems Operation

The increase in feedwater flow, due to a failure of the feedwater control system to maximum demand, results in an increase in the water level and a decrease in the coolant temperature at the core inlet. The increase in core inlet subcooling causes an increase in core power. As the feedwater flow continues at maximum demand, the water level continues to rise and eventually reaches the high water level trip setpoint. The initial water level is conservatively assumed to be at the low level normal operating range of 30 inches above instrument zero to delay the high-level trip and maximize the core inlet subcooling that results from the FWCF. The high water level trip causes the turbine throttle (stop) valves to close in order to prevent damage to the turbine from excessive liquid inventory in the steam line. The valve closures create a compression wave that travels to the core causing a void collapse and subsequent rapid power excursion. In addition to the turbine throttle valve closure, the turbine governor valves also close in the fast closure mode. The closure of the governor (control) turbine valves initiates a reactor scram and a recirculation pump trip. Because of the partially opened initial position of the governor valves, they will close faster than the throttle valves and initiate the pressurization portion of the event. The turbine bypass valves are assumed operable and provide some pressure relief. The core power excursion is mitigated in part by the pressure relief, but the primary mechanisms for termination of the event are reactor scram and revoiding of the core.

The high-pressure core spray (HPCS) system and reactor core isolation cooling (RCIC) system initiate on a low reactor water level (L2) to maintain long-term water level control following tripping of feedwater pumps. The analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems.

Table 15.1-1 lists the sequence of events for Figure 15.1-1. The figure shows the changes in variables during this transient.

15.1.2.2.1 Sequence of Events and Systems Operation – Single Loop Operation

The simulated feedwater controller transient is shown in Figure 15.1-3 for the case of 75% power, 57% core flow. The high-water level turbine trip and feedwater pump trip are initiated at approximately 8.4 sec. A scram occurs simultaneously with the turbine trip and limits the neutron flux peak and fuel thermal transient so no fuel damage occurs.

Table 15.1-1A lists the sequence of events for Figure 15.1-3. The figures show the changes in important variables during this transient.

Identification of Operator Actions

a. Observe high feedwater pump trip has terminated the failure event,

b. Switch the feedwater controller from auto to manual control to try to regain a correct output signal, and

c. Conduct follow-up assessment.

15.1.2.2.2 The Effect of Single Failures and Operator Errors

The first sensed event to initiate 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 L8 setpoint. At this point in the logic, a single failure will 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, but not impacting, the pressurization "signature."

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

15.1.2.3 Core and System Performance

15.1.2.3.1 Mathematical Model

The predicted dynamic behavior has been determined using a computer simulated, analytical model of a direct-cycle BWR. This model is described in detail in Reference 15.1-4. Results from the two-loop operation bound the SLO event. Therefore, the discussion of core and system performance is limited to the description of the analysis for two-loop operation.

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

a. An integrated one-dimensional core model is assumed which includes a detailed description of hydraulic feedback effects, axial power shape changes, and reactivity feedbacks;

b. The fuel is represented by an average cylindrical fuel and cladding model for each axial location in the core;

c. The steam lines are modeled by eight pressure nodes incorporating mass and momentum balances which will predict any wave phenomena present in the steam line during a pressurization transient;

d. The core average axial water density and pressure distribution is calculated using a single channel to represent the heated active flow and a single channel to represent the bypass flow. A model, representing liquid and vapor mass and

energy conservation and mixture momentum conservation, is used to describe thermal-hydraulic behavior. Changes in the flow split between the bypass and active channel flow are accounted for during transient events;

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

f. The ability to simulate necessary reactor protection system functions is provided.

15.1.2.3.2 Input Parameters and Initial Conditions

These analyses have been performed with the plant conditions in Table 15.0-2B.

All rods out scram characteristics are assumed. The safety/relief valve (SRV) action is conservatively assumed to occur with higher than nominal setpoints. The transient is simulated by programming an upper limit failure in the feedwater system such that 139% feedwater flow occurs at the nominal reactor operating pressure of 1035 psia.

An increase in feedwater flow will cause a corresponding drop in feedwater temperature. However, the relatively large time constant of the feedwater heaters (order of minutes) plus the flow transport time (10 sec from heaters to vessel and 3 sec from sparger to core) would preclude any effect of temperature reduction on the transient since the transient is essentially over in about 20 sec. Therefore, feedwater temperature is assumed to remain constant.

15.1.2.3.3 Results

The simulated feedwater controller transient is shown in Figure 15.1-1. The high water level turbine trip and feedwater pump trip are initiated as stated in Table 15.1-1. Results reflect GE14 fuel introduction, some of which are dependent on fuel design and core loading pattern. Compliance with the event acceptance criteria is demonstrated by cycle-dependent analysis of potentially limiting events just prior to the operation of that cycle. The results are reported in the Supplemental Reload Licensing Report (Reference 15.1-3).

Because the total change in feedwater flow is greatest from reduced power conditions, the feedwater controller failure (FWCF) transient was analyzed for several reduced power states. The power dependent MCPR limits are established to protect the fuel during the FWCF event.

15.1.2.3.4 Consideration of Uncertainties

All systems used for protection in this event were assumed to have the most conservative response characteristics. Therefore, actual plant behavior is expected to lead to a less severe transient.

15.1.2.4 Barrier Performance

The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed. Therefore, barrier integrity and function is maintained.

15.1.2.5 Radiological Consequences

The consequence of this event does not result in fuel failure. It does result in the discharge of normal coolant activity to the suppression pool by means of SRV operation, which is contained in the primary containment. This event does not result in an uncontrolled release to the environment, so the plant operator can choose to hold the activity in containment or discharge it when conditions permit. If purging of the containment is chosen, the release would be in accordance with established requirements.

15.1.3 PRESSURE REGULATOR FAILURE - OPEN

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 in the Digital Electro-Hydraulic (DEH) control system is limited by a maximum flow limiter imposed at the turbine controls. This limiter is set to limit maximum steam flow demand to approximately 130% NBR.

If the triple redundant DEH control system fails such that the turbine control (governor) valves fully open and the turbine bypass valves partially open, the maximum steam flow is established.

15.1.3.1.2 Frequency Classification

This event 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 15.1-2 lists the sequence of events for Figure 15.1-2. Figure 15.1-2 depicts how the high water level turbine trip and isolation valve closure stops vessel depressurization and produces a normal shutdown of the reactor.

15.1.3.2.2 Systems Operation

Depressurization results in formation of voids in the reactor coolant and causes a decrease in reactor power almost immediately. In this simulation, the depressurization rate is large enough such that water level swells to the sensed level trip setpoint (L8), initiating main turbine and feedwater turbine trips. Position switches on the turbine stop (throttle) valves initiate a reactor scram and RPT and shut down the reactor. After the turbine trip, the failed DEH control system signals the bypass to open to full bypass flow of 25% NBR steam flow. After the pressurization resulting from the turbine stop (throttle) valve closure, the pressure increase opens the relief valves and pressure drops and continues to drop until turbine inlet pressure is below the low turbine pressure isolation setpoint when main steam line isolation limits the duration and severity of the depressurization.

In order to properly simulate the expected sequence of events, the analysis of this event assumed normal functioning of plant instrumentation and controls, plant protection, and reactor protection systems except as otherwise noted.

Initiation of HPCS and RCIC system functions will occur when the vessel water level reaches the L2 setpoint although this is not included in the analysis. Normal startup and actuation can take up to 30 sec before effects are realized. If these events occur, they will follow some time after the primary concerns of fuel thermal margin and overpressure effects have occurred and are expected to be 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 be single failure proof for initiation of MSIV closure.

Reactor scram sensing, originating from limit switches on the MSIVs, is designed to be single failure proof. It is, therefore, concluded that the basic phenomenon of pressure decay is adequately terminated.

15.1.3.3 Core and System Performance

15.1.3.3.1 Mathematical Model

The point-kinetics REDY model described in Section 15.0.3.3.1 is used to simulate this event.

15.1.3.3.2 Input Parameters and Initial Conditions

This transient is simulated by setting the DEH control system demand signal to a high value, which causes the turbine control (governor) valves to open fully and the turbine bypass valves to open partially. A DEH control failure with 130% steam flow demand signal was simulated as a worst case since 130% is the normal maximum flow limit in order to conform with Table 15.1-2.

A 5-sec isolation valve closure is assumed when the turbine pressure decreases below the turbine inlet low pressure setpoint for main steam line isolation initiation.

Reactor scram is initiated when the isolation valves reach the 10% closed position. This is the maximum travel from the full open position allowed by specification.

This analysis has been performed, unless otherwise noted, with the plant conditions listed in Table 15.0-2.

15.1.3.3.3 Results

Results are summarized in Table 15.0-1.

No significant reductions of fuel thermal margins occur. No significant thermal stresses are imposed on the reactor coolant pressure boundary (RCPB).

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 would result. The rate of depressurization may be limited by the bypass capacity, but it is unlikely. For example, the turbine valves will open to the valves-wide-open state admitting slightly more than the rated steam flow, and with the limiter in this analysis set to fail at 130%, it is expected that less than 25% would be bypassed. This is, therefore, not a limiting factor for the plant. If the rate of depressurization does change, it will be terminated by the low turbine inlet pressure trip setpoint.

Depressurization rate has a proportional effect upon the voiding action in the core and the flashing in the vessel bulk water regions. If the rate is low enough, the water level may not swell to the high water level trip setpoint and the isolation will occur earlier when pressure at

the turbine decreases below 850 psia. The reactor will scram as a result of the MSIV closure. Since power is being depressed as the pressure decreases (due to additional voiding in the core), this transient is less severe when a slower depressurization rate is assumed. Therefore, the assumed L8 trip provides the most restrictive margins on MCPR and peak vessel pressure.

15.1.3.4 Barrier Performance

Barrier performance analyses were not required since the consequences of this event 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 is below the ASME code upset limit for the RCPB.

15.1.3.5 Radiological Consequences

The consequence of this event does not result in fuel failure. It does result in the discharge of normal coolant activity to the suppression pool by means of SRV operation, which is contained in the primary containment. This event does not result in an uncontrolled release to the environment, so the plant operator can choose to hold the activity in containment or discharge it when conditions permit. If purging of the containment is chosen, the release would be in accordance with established requirements.

15.1.4 INADVERTENT SAFETY/RELIEF VALVE OPENING

The event is defined as the inadvertent opening of an SRV which stays in the "open" position. It was determined that this event is not limiting from a core performance standpoint.

15.1.4.1 Identification of Causes and Frequency Classification

15.1.4.1.1 Identification of Causes

Cause of inadvertent 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 impact. It is therefore simply postulated that a failure occurs and the event is analyzed accordingly. Detailed discussion of the valve is provided in Section 5.2.2.

15.1.4.1.2 Frequency Classification

This event is categorized as an incident of moderate frequency.

15.1.4.2 Sequence of Events and Systems Operation

15.1.4.2.1 Sequence of Events

Table 15.1-3 lists the sequence of events.

15.1.4.2.2 Systems Operation

In this transient, the analysis assumes normal functioning of plant instrumentation and controls, specifically, the relief valve discharge line temperature sensors and the suppression pool temperature sensors and reactor pressure vessel level control systems. The opening of an SRV allows steam to be discharged into the suppression pool. 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 within a few seconds closes the turbine control (governor) valve far enough to stabilize reactor vessel pressure at a slightly lower value and reactor power settles at nearly the initial power level. Additionally, although not credited in the analysis to mitigate the consequences of this transient, minimum reactor and plant protection systems, emergency core cooling system flow, and RHR suppression pool cooling, are available.

15.1.4.2.3 The Effect of Single Failures and Operator Errors

From a core performance standpoint, a single failure or operator error would simply activate the reactor protection system resulting in a plant shutdown. A single failure or operator error cannot increase the severity of this event.

The instrumentation which detects and audibly alarms the resulting suppression pool temperature rise, and the RHR containment heat removal system are designed to meet the single failure criteria. The operator must manually initiate suppression pool cooling.

15.1.4.3 Core and System Performance

15.1.4.3.1 Mathematical Model

The one-dimensional ODYN model described in Section 15.0.3.3.1 is used to simulate this event.

15.1.4.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2, the ODYN column. A discussion of the SRV is provided in Section 5.2.2.

15.1.4.3.3 Results

Thermal margins decrease only slightly through the transient, and no fuel damage results from the transient. The MCPR is essentially unchanged and, therefore, the safety limit margin is unaffected.

15.1.4.4 Barrier Performance

The transient resulting from a stuck open relief valve is a mild depressurization which is within the range of normal load following. Therefore, there is no significant effect on RCPB and containment design pressure limits.

Since quenchers are used as steam discharge devices on the steam relief lines, no unstable condensation oscillations are expected which could damage the containment vessel. This is discussed in Appendix 3A.

Therefore, barrier integrity and function is maintained.

15.1.4.5 Radiological Consequences

While the consequence of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool by means of SRV operation. Since this activity is contained in the primary containment there will be no exposures to operating personnel. Since this event does not result in an uncontrolled release to the environment the plant operator can choose to hold the activity in containment or discharge it to the environment when conditions permit. If purging of the containment is chosen, the release would be in accordance with established requirements.

15.1.5 SPECTRUM OF STEAM PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT IN A PRESSURIZED WATER REACTOR

This event is not applicable to BWR plants.

15.1.6 INADVERTENT RESIDUAL HEAT REMOVAL SHUTDOWN COOLING OPERATION

This transient is classified as a nonlimiting event for both original and uprated power conditions. Therefore, no further analysis has been performed.

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 could cause temperature reduction.

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

15.1.6.1.2 Frequency Classification

This event is categorized as an incident 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 will occur before any thermal limits are reached if the operator does not take action. The sequence of events for this event is shown in Table 15.1-4.

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 event is not considered while at power operation since pressure is too high to permit operation of RHR shutdown cooling.

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 would be 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 event to be more severe.

If the operator takes action, the slow power rise will be controlled in the normal manner. If no operator action is taken, a scram will terminate the power increase before thermal limits are reached.

15.1.6.3 Core and System Performance

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

15.1.6.4 Barrier Performance

The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed. Therefore, barrier integrity and function is maintained.

15.1.6.5 Radiological Consequences

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

15.1.7 REFERENCES

15.1-1 For Power Uprate: GE Nuclear Energy, "WNP-2 Power Uprate Transient Analysis Task Report," GE-NE-208-08-0393, September 1993 (Proprietary).

15.1-2 NEDE-30130-P-A, "Steady State Nuclear Methods," April 1985.

15.1-3 Supplemental Reload Licensing Report for Columbia (most recent version referenced in COLR).

15.1-4 NEDC-24154-P-A, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," Volumes 1, 2, 3 and 4, February 2000.

15.1-5 GE Nuclear Energy, "WNP-2 Power Uprate Project NSSS Engineering Report," GE-NE-208-17-0993, Revision 1, December 1994.

15.1-6 "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A and "Supplement for United States," NEDE-24011-P-A-US (most recent approved revision referenced in COLR).

Table 15.1-1

Sequence of Events for Figure 15.1-1

Feedwater Controller Failure
100% Reactor Power / 106% Core Flow

Time (sec)	Event
0	Initiate simulated failure of 139% upper limit on feedwater flow.
10.06	L8 vessel level setpoint trips main turbine and feedwater pumps. Turbine bypass operation initiated.
10.16	Turbine control (governor) or stop (throttle) valves fully closed.
10.08	Reactor scram trip actuated from main turbine control (governor) valve fast closure.
10.16	Turbine bypass valves start to open.
10.27	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation.

Table 15.1-1A

Sequence of Events for Figure 15.1-3

Feedwater Controller Failure
Single Loop Operation
75% Power / 57% Flow

Time (sec)	Event
0	Initiate an upper limit failure of 146% of rated feedwater flow.
8.39	L8 vessel level setpoint trips main turbine and feedwater pumps.
8.39	Recirculation pump trip (RPT) actuated by stop valve position switches.
8.40	Reactor scram trip actuated from main turbine stop valve position switches.
8.49	Turbine stop valves closed and main turbine bypass valves start to open.
8.58	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation.

Table 15.1-2

Sequence of Events for Figure 15.1-2

Pressure Regulator Failure - Open Uprated Power

Time (sec)	Event
0	Simulate maximum limit on steam flow, (130%) to main turbine.
0.2	Main turbine bypass valves open.
3.31	Vessel water level (L8) trip initiates turbine and feedwater trips.
3.32	Main turbine stop valves reach 90% open position initiating a reactor scram.
3.50	Both recirculation pumps trip.
6.15	Feedwater recirculation valves trip.
6.95	Group 3 relief valves actuated.
7.40	Group 4 relief valves actuated.
10[a]	Pressure relief valves closed.
57.98[a]	Main steam line isolation valves closed on turbine inlet pressure (approximately 850 psia).
77	High-pressure core spray and RCIC system initiation on low level (L2).

[a] Estimated.

Table 15.1-3

Sequence of Events for Inadvertent
Safety/Relief Valve Opening

Uprated Power

Time	Event
0	Initiate opening of one SRV which remains open throughout the event.
1[a]	Reactor dome pressure decreases.
3[a]	DEH turbine control system pressure regulator initiates closure of the turbine control (governor) valves to stabilize reactor vessel pressure.
8+	Reactor power settles near the initial power level.

[a] Approximately.

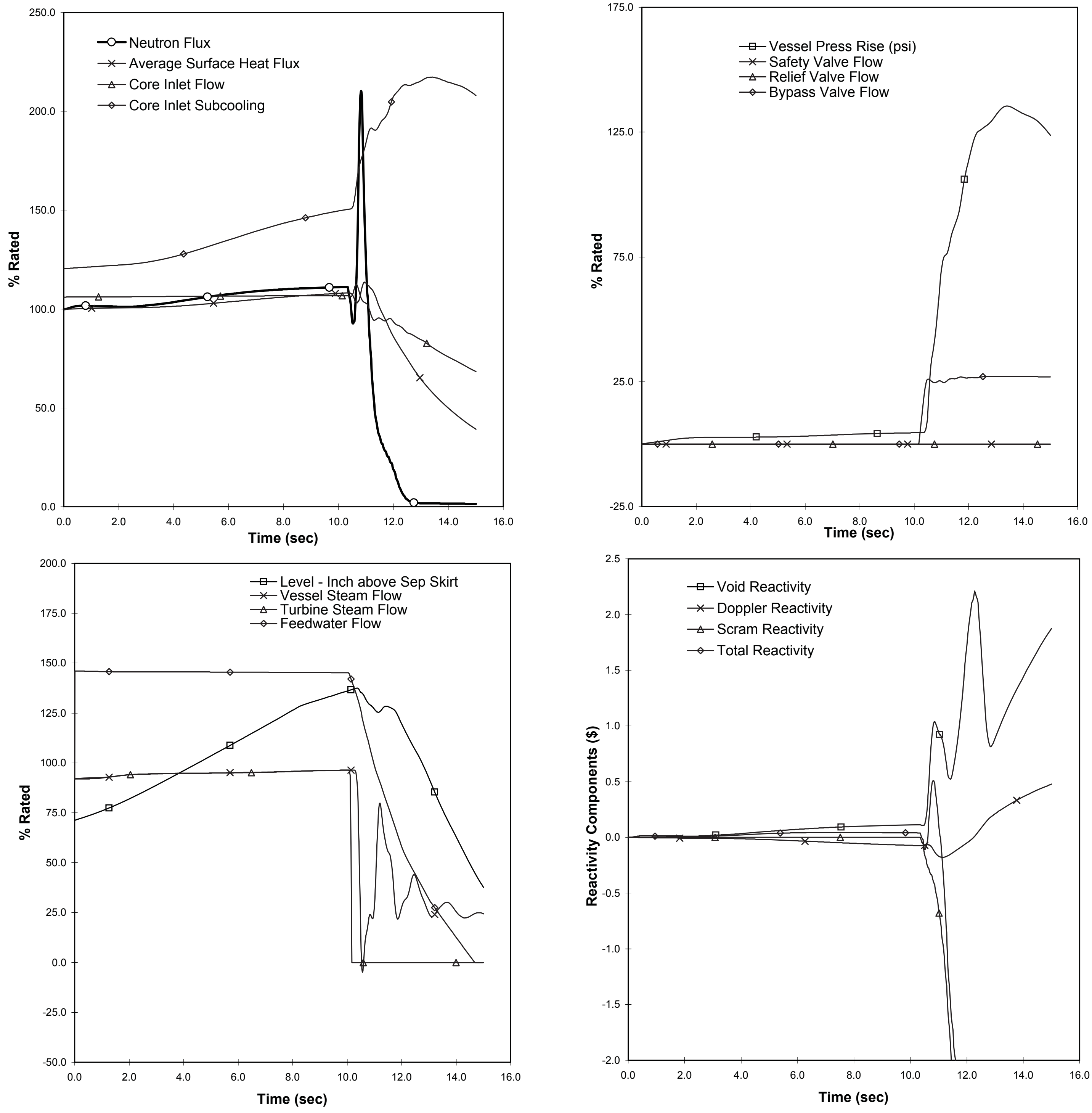
Table 15.1-4

Sequence of Events for Inadvertent
Residual Heat Removal Shutdown Cooling Operation

Original Rated Power

Time[a]	Event
0	Residual heat removal shutdown cooling inadvertently activated.
0-10 minutes	Slow rise in reactor power.
+10 minutes	Operator may take action to limit power rise. Flux scram will occur if no action is taken.

[a] Approximately.



Columbia Generating Station Final Safety Analysis Report	Feedwater Controller Failure, Maximum Demand System Response 100% Power, 106% Flow
	Draw. No. 910402.34 — Rev. — Figure 15.1-1

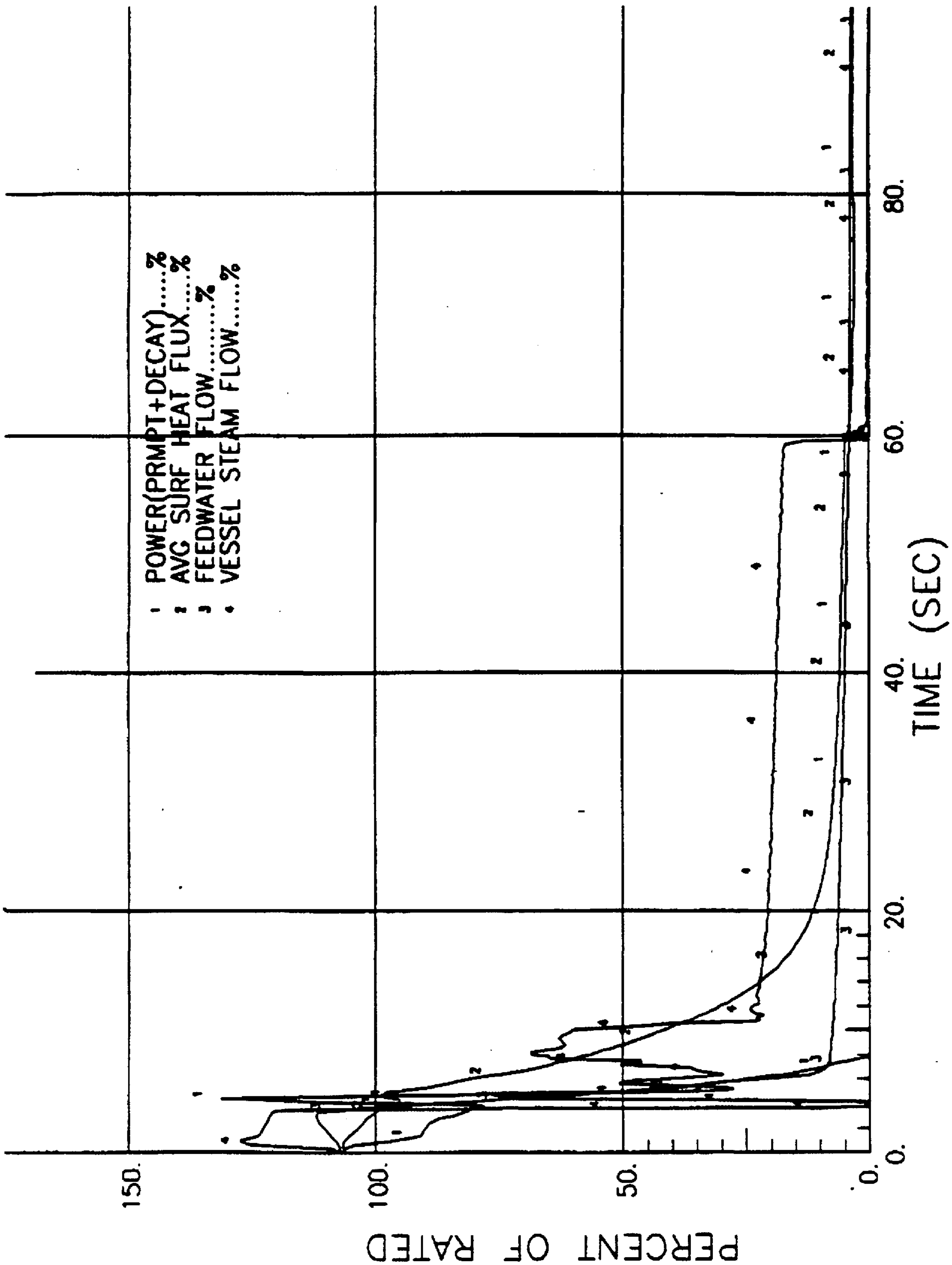
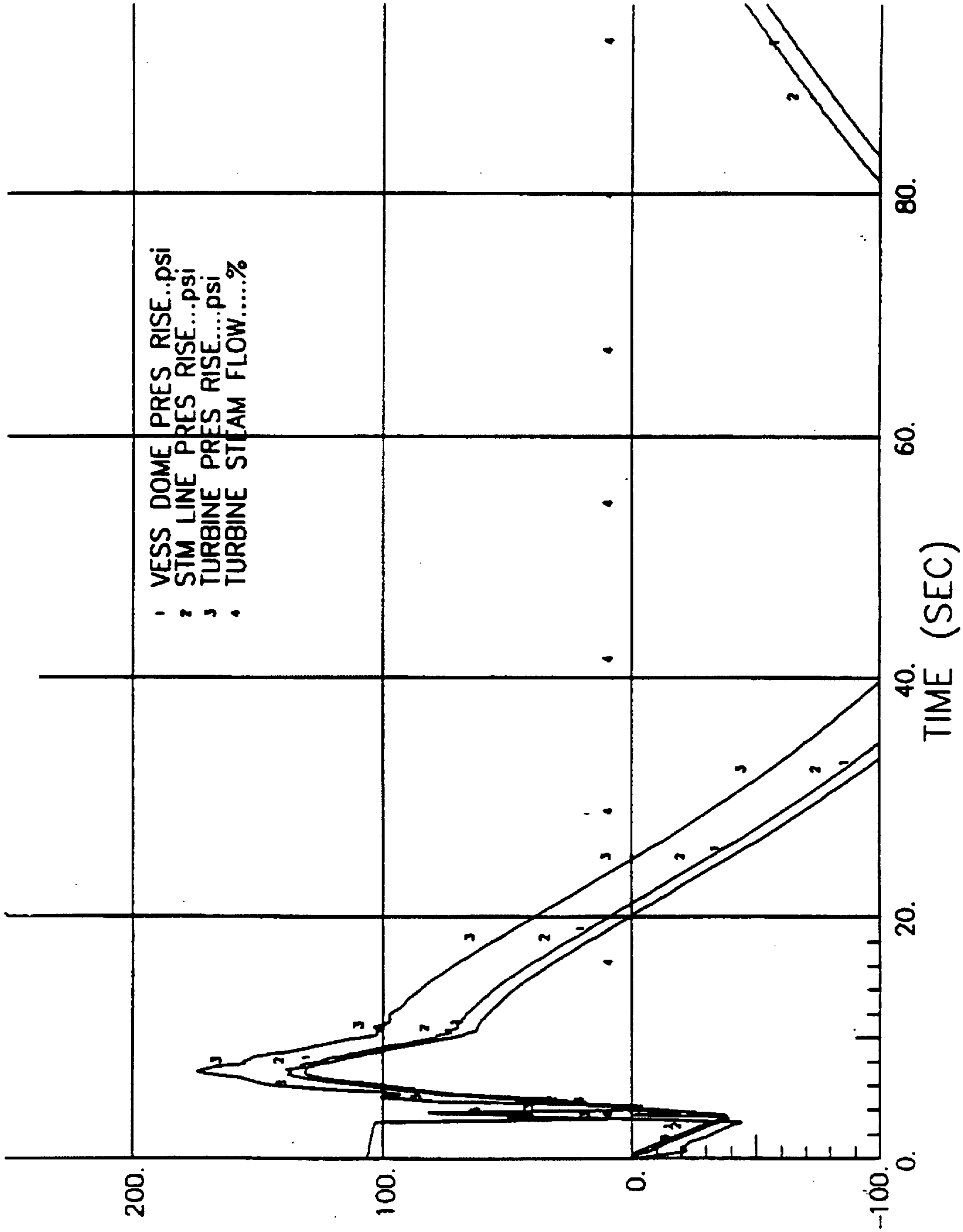


Figure 4-2a. Pressure Regulator Failure - Open at 106.2% Uprated Power, 100% Flow

Amendment 53
November 1998



Columbia Generating Station Final Safety Analysis Report	Pressure Regulator Failure - Open at 106.2% Uprated Power, 100% Flow Draw. No. 020361.58 · Rev. · Figure 15.1-2.2

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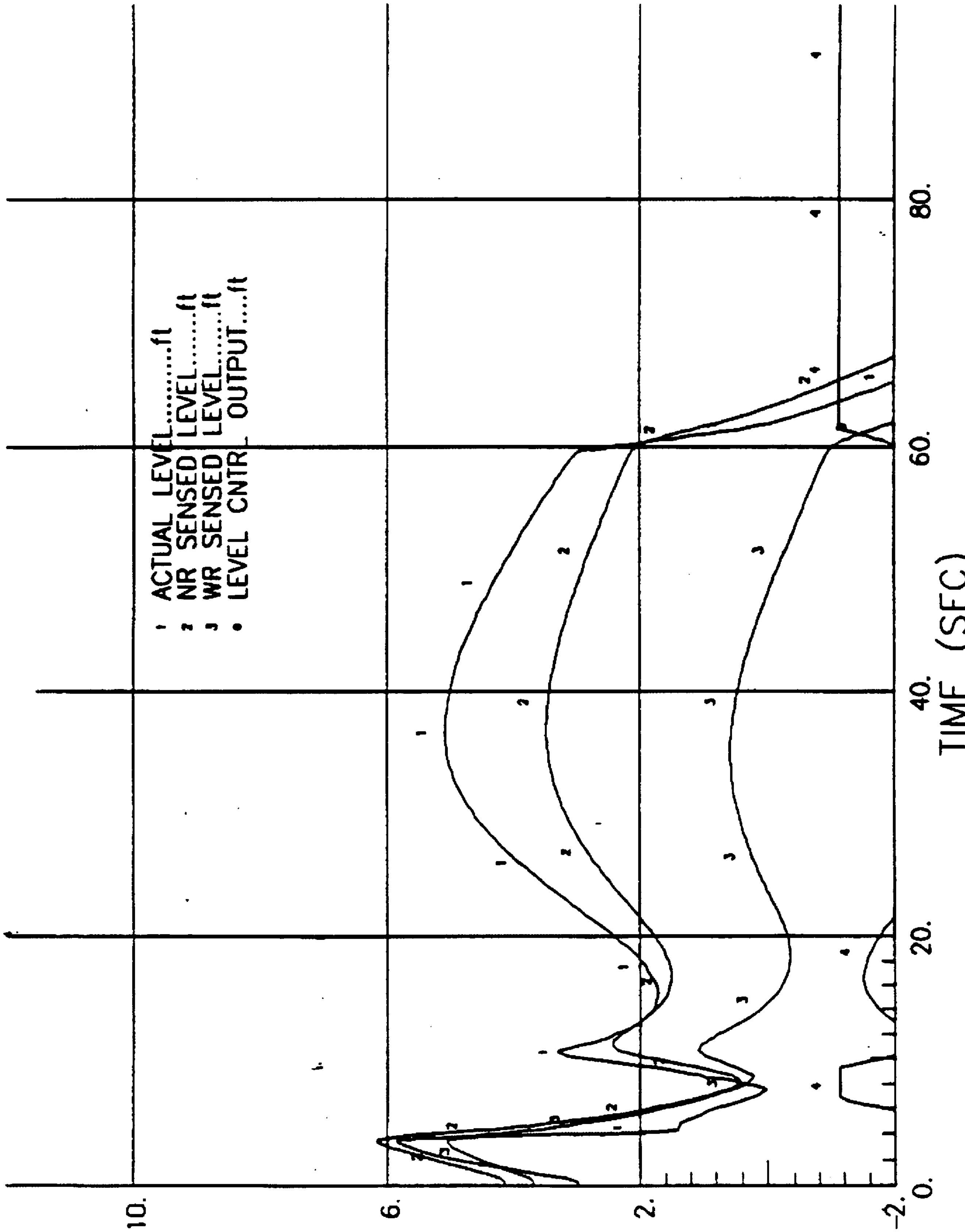
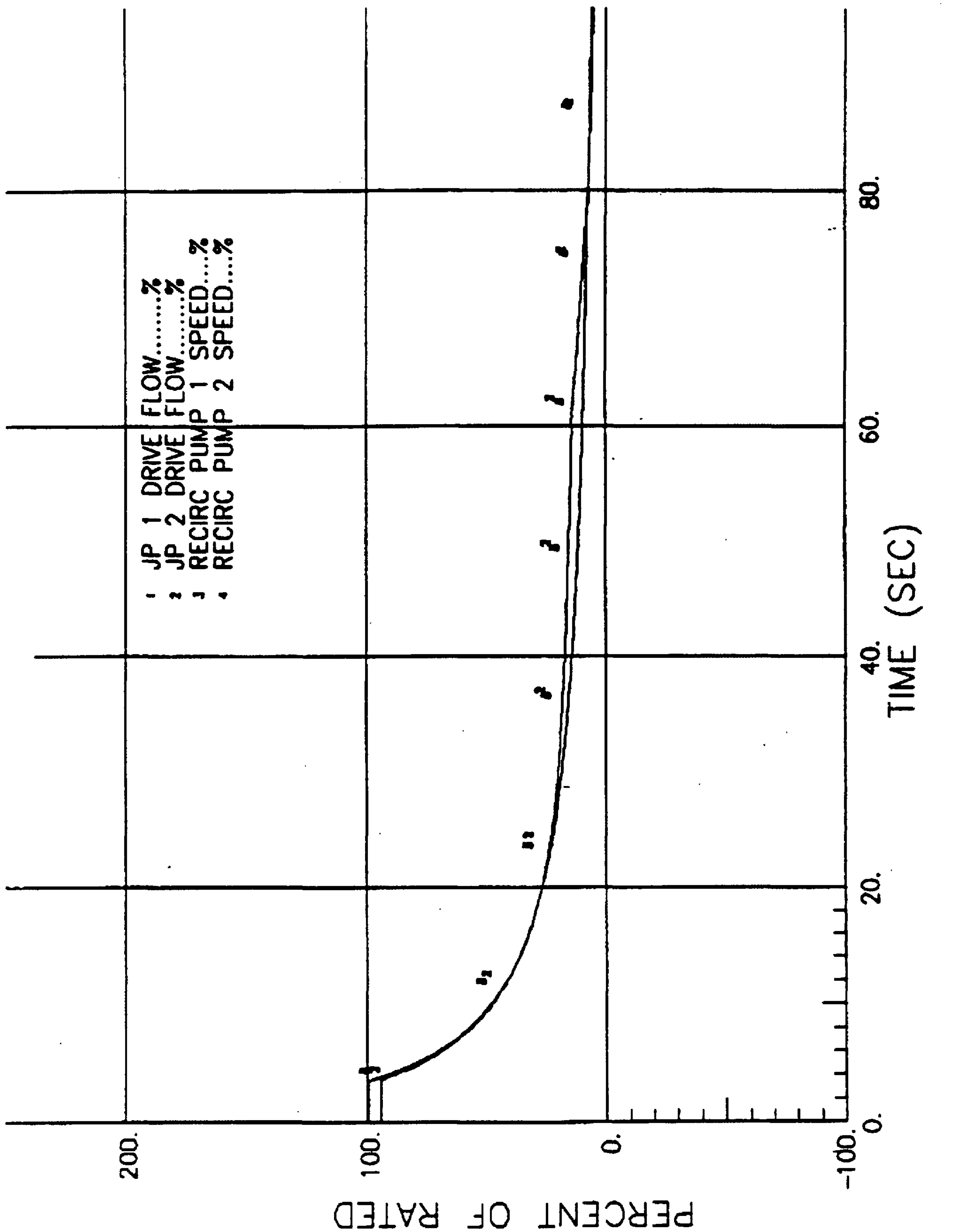


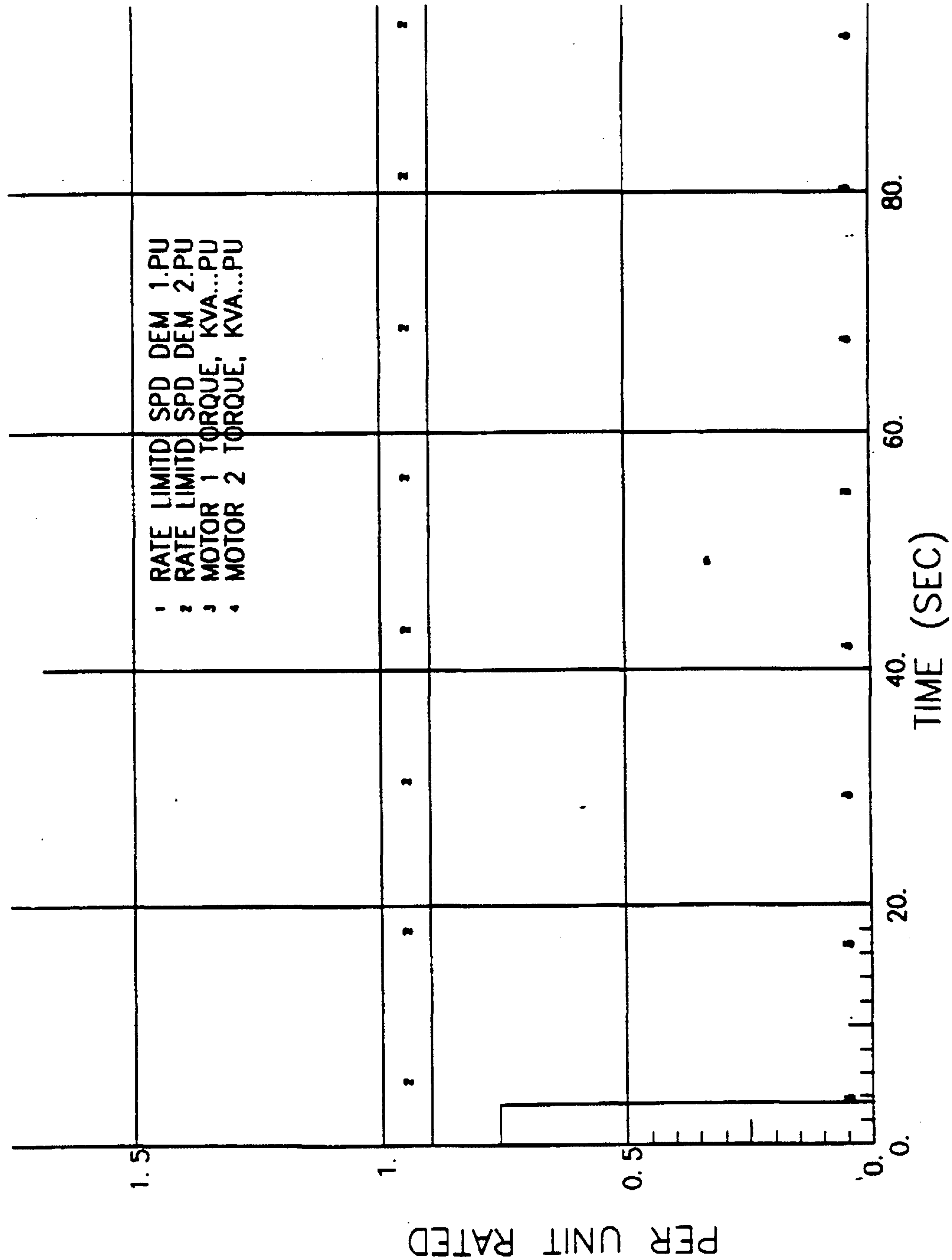
Columbia Generating Station
Final Safety Analysis Report

Pressure Regulator Failure - Open at 106.2% Uprated Power, 100% Flow

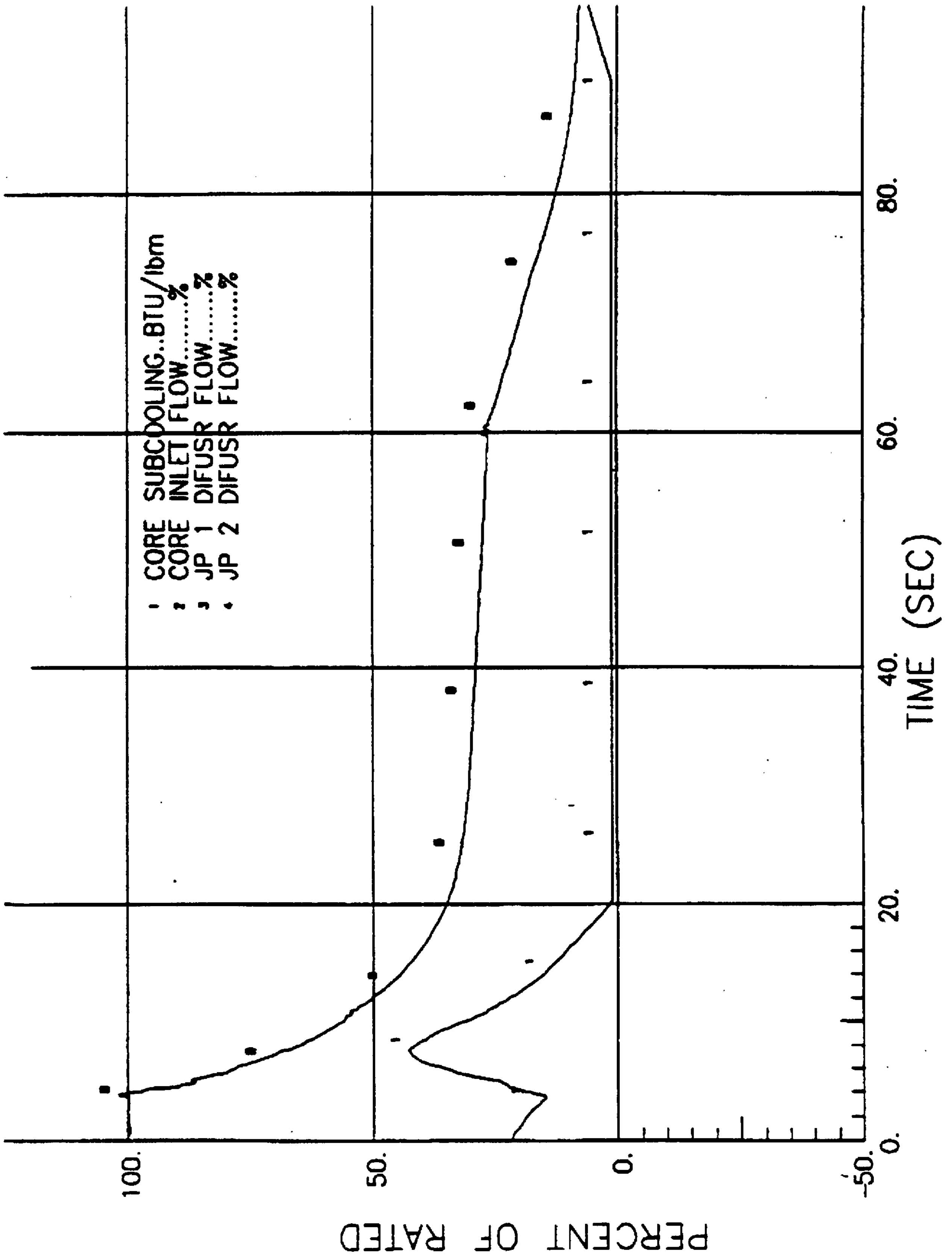
Draw. No. 020361.59	Rev.	Figure 15.1-2.3



Columbia Generating Station **Final Safety Analysis Report**	**Pressure Regulator Failure - Open at 106.2% Uprated Power, 100% Flow**	
Draw. No. 020361.60	Rev.	Figure 15.1-2.4



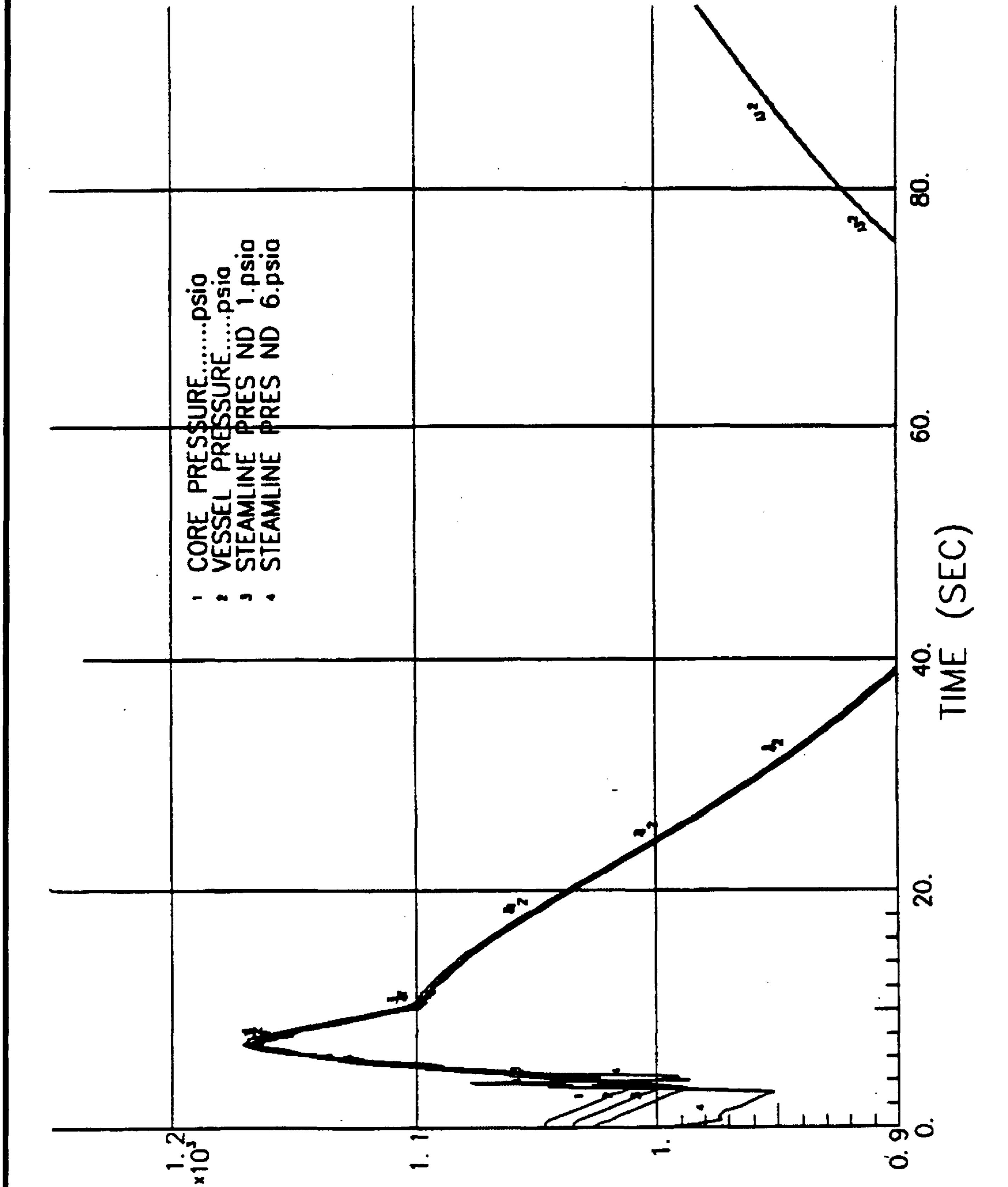
Columbia Generating Station Final Safety Analysis Report	Pressure Regulator Failure - Open at 106.2% Uprated Power, 100% Flow	
	Draw. No. 020361.61	Rev.
	Figure 15.1-2.5	



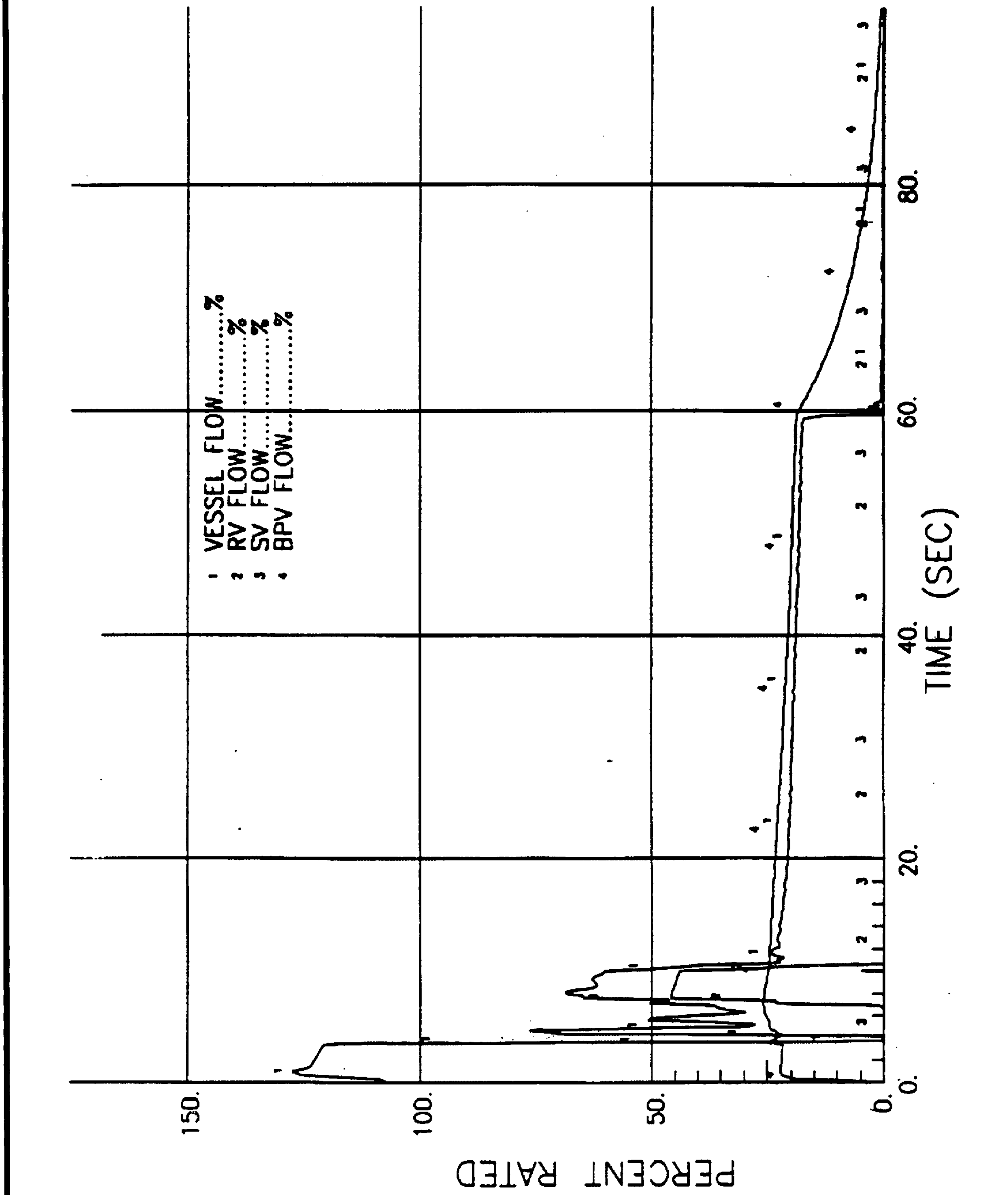
Columbia Generating Station
Final Safety Analysis Report

Pressure Regulator Failure - Open at 106.2% Uprated Power, 100% Flow

Draw. No. 020361.62	Rev.	Figure 15.1-2.6



Columbia Generating Station Final Safety Analysis Report	Pressure Regulator Failure - Open at 106.2% Uprated Power, 100% Flow		
	Draw. No. 020361.63	Rev.	Figure 15.1-2.7



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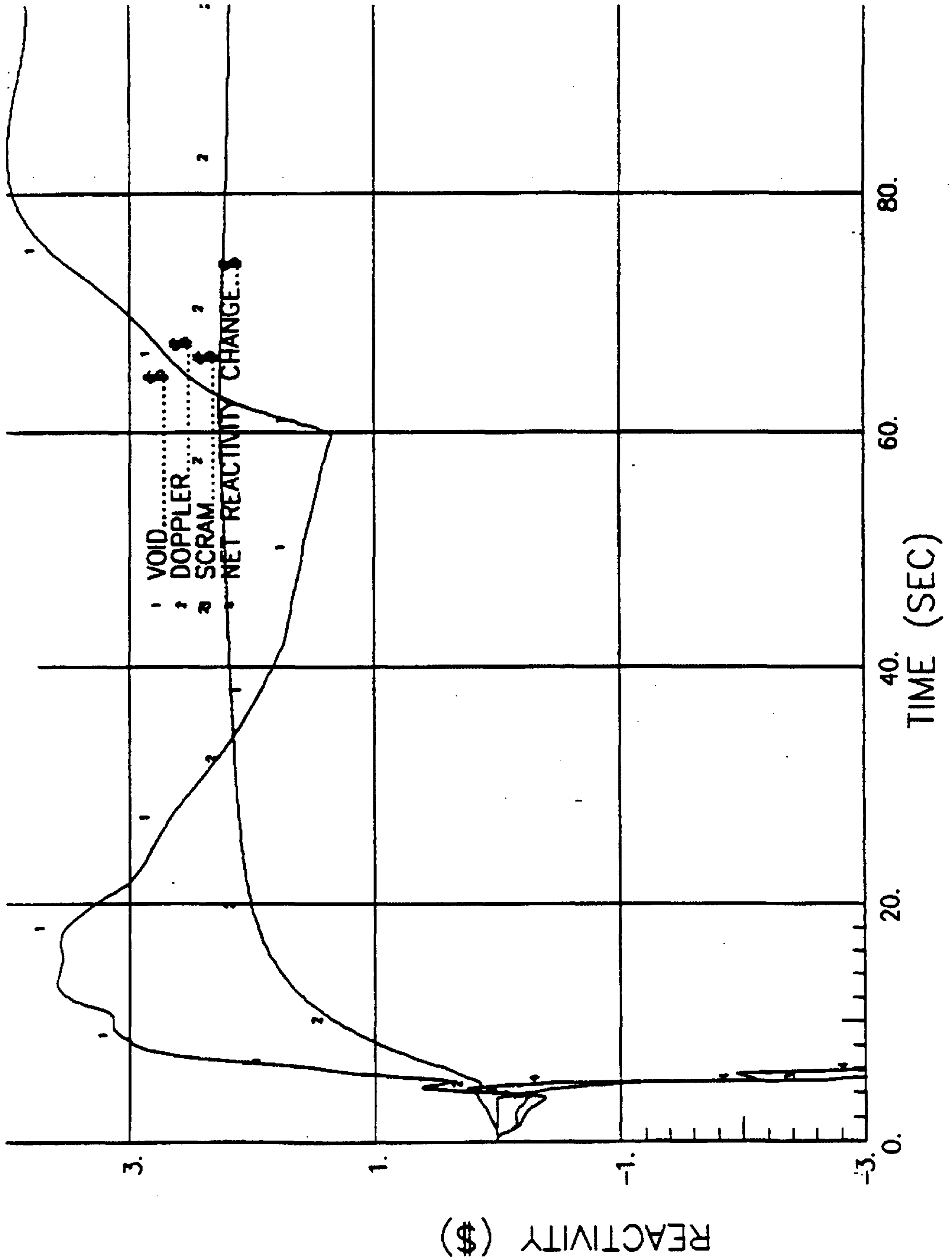
Pressure Regulator Failure - Open at 106.2% Uprated Power, 100% Flow

Draw. No. 020361.64	Rev.	Figure 15.1-2.8



Columbia Generating Station Final Safety Analysis Report	Pressure Regulator Failure - Open at 106.2% Uprated Power, 100% Flow
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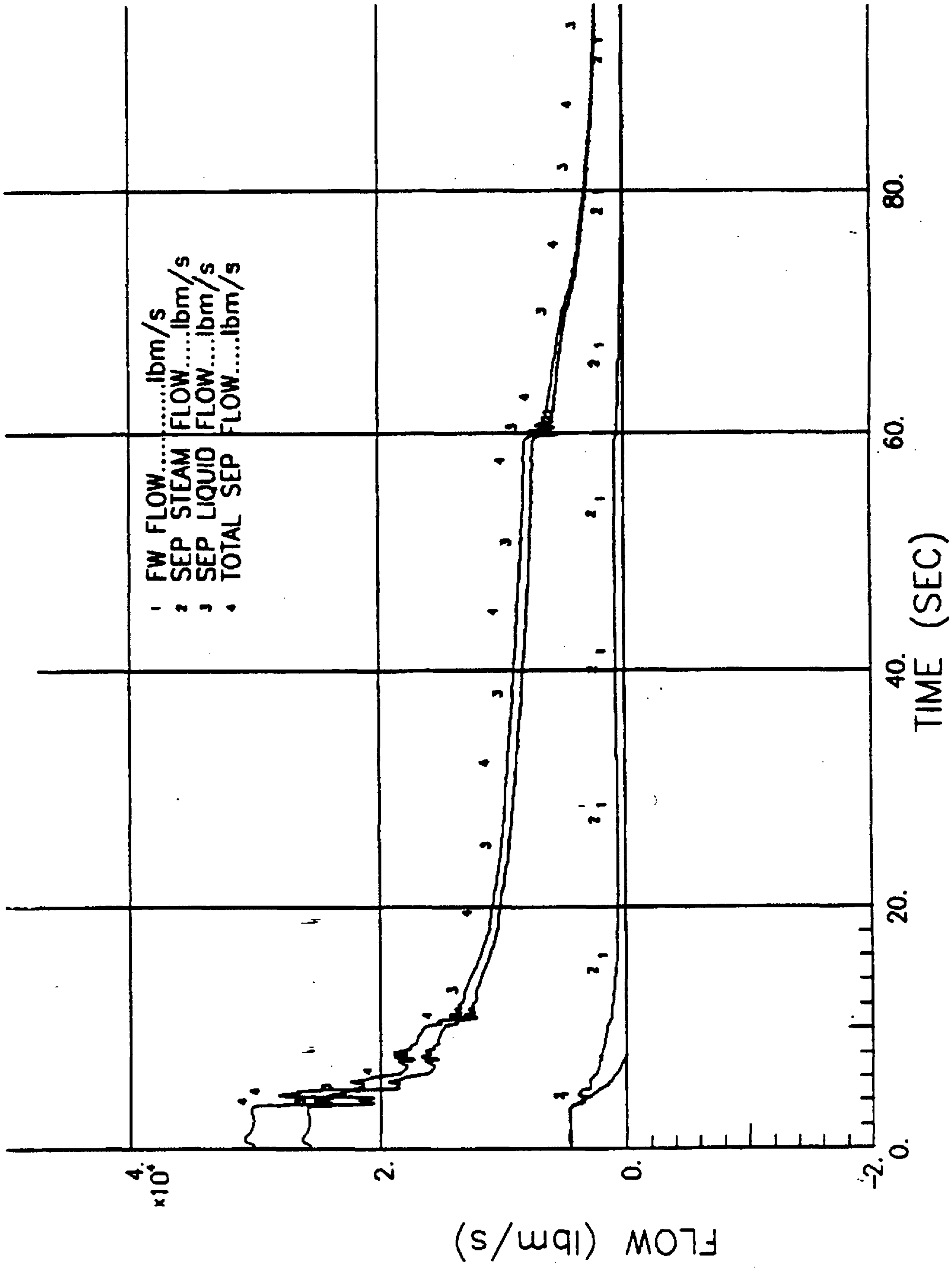
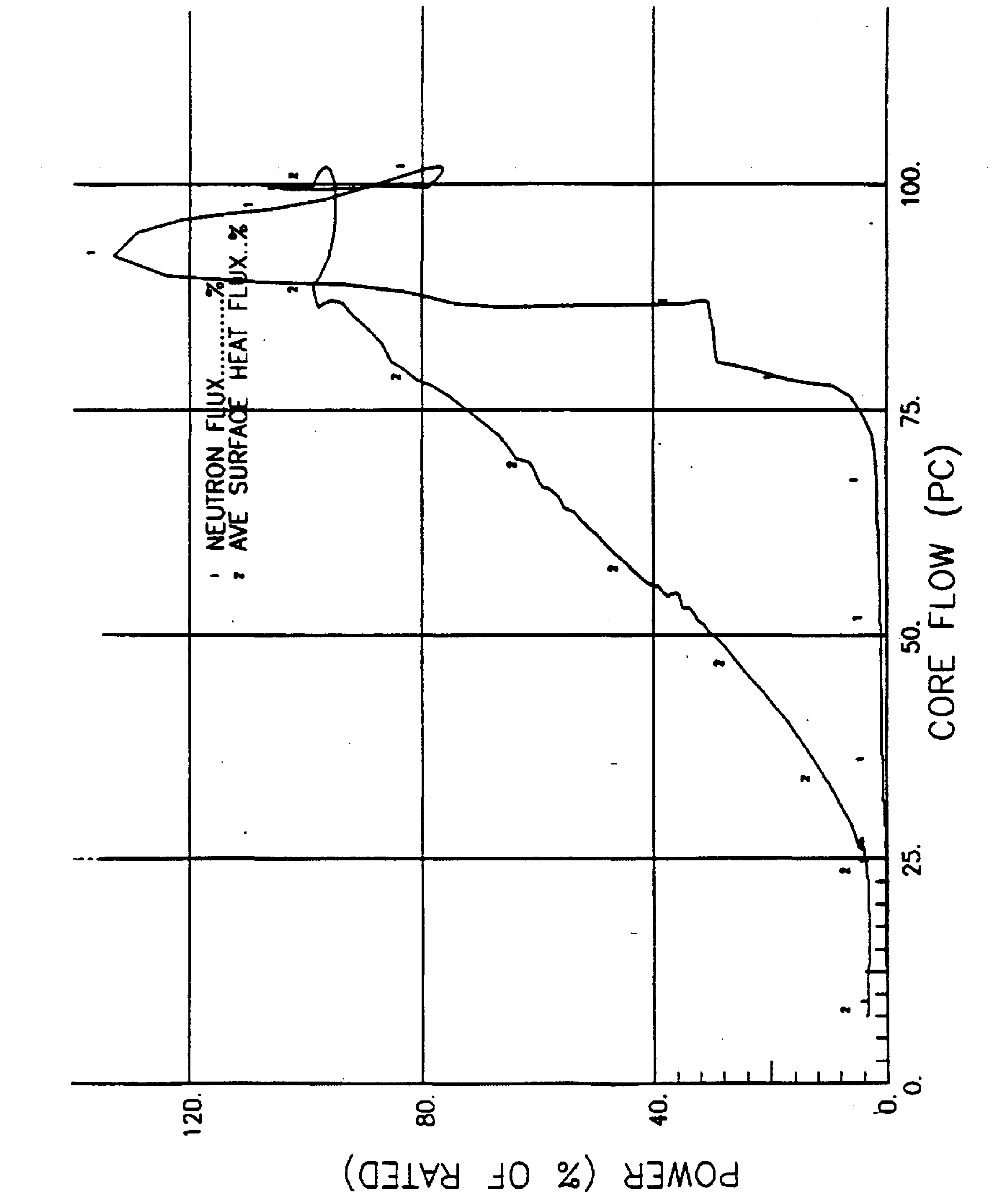


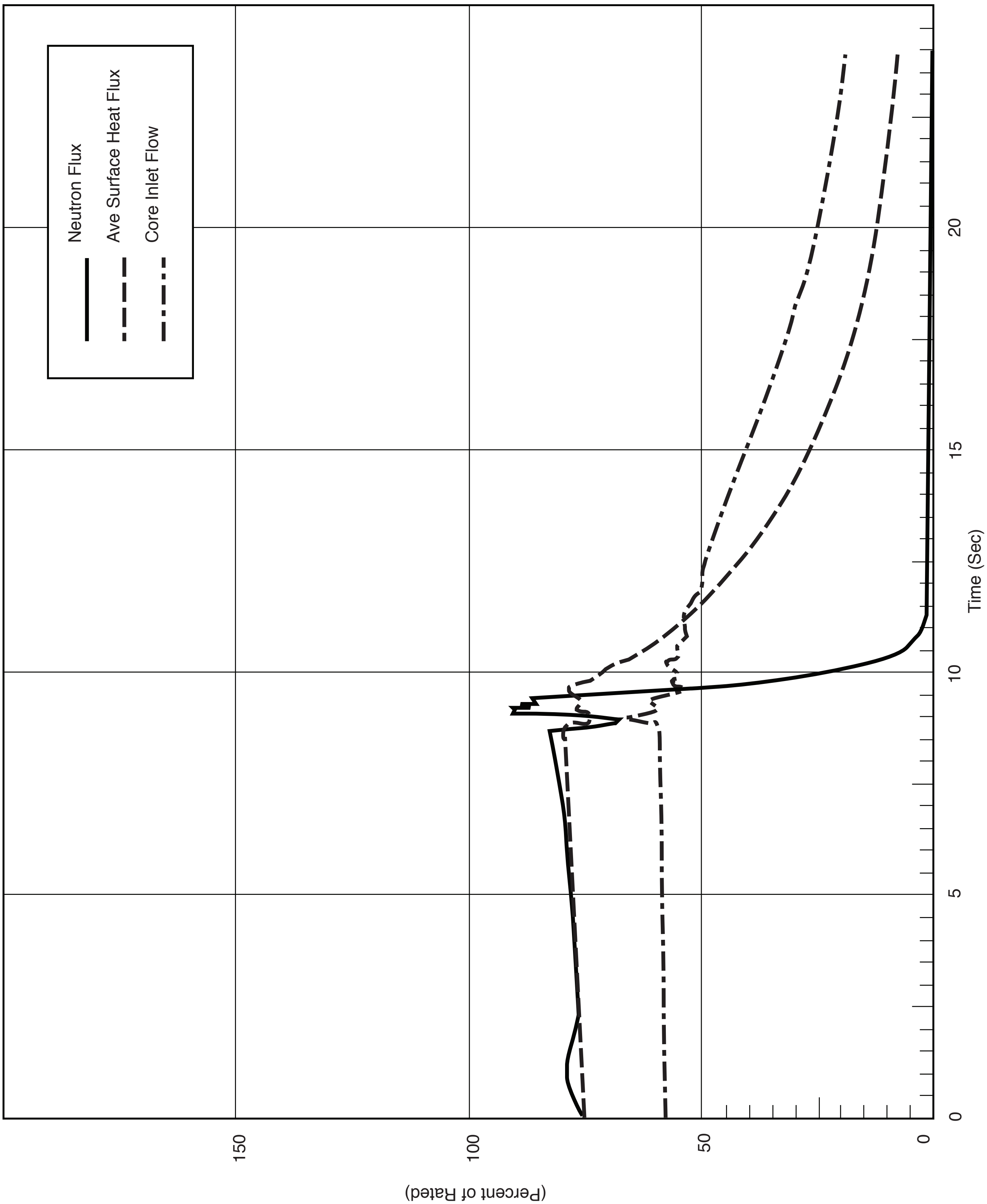
Columbia Generating Station
Final Safety Analysis Report

Pressure Regulator Failure - Open at 106.2% Uprated Power, 100% Flow

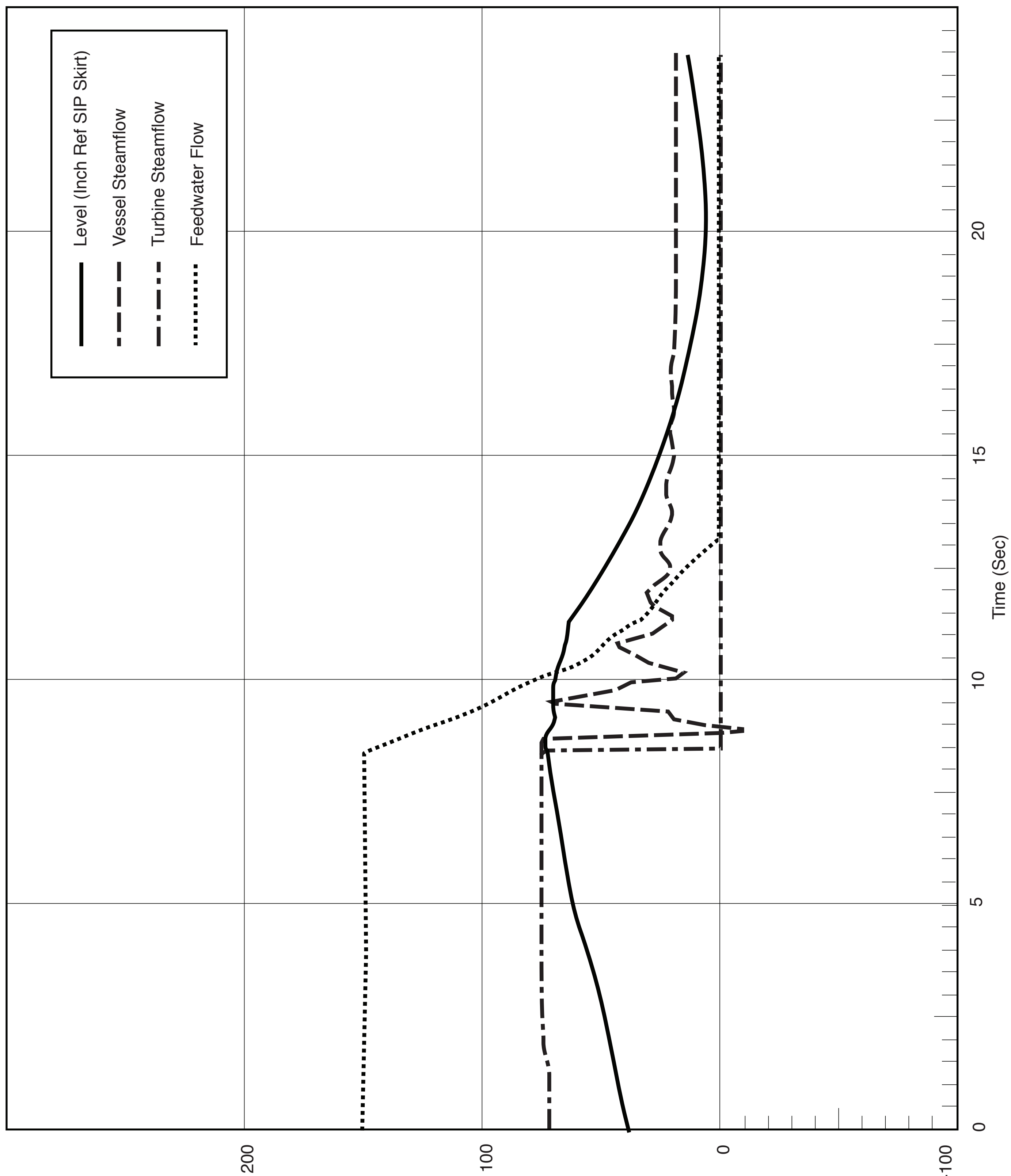
Draw. No. 020361.66	Rev.	Figure 15.1-2.10



Columbia Generating Station Final Safety Analysis Report	Pressure Regulator Failure - Open at 106.2% Uprated Power, 100% Flow		
	Draw. No. 020361.67	Rev.	Figure 15.1-2.11



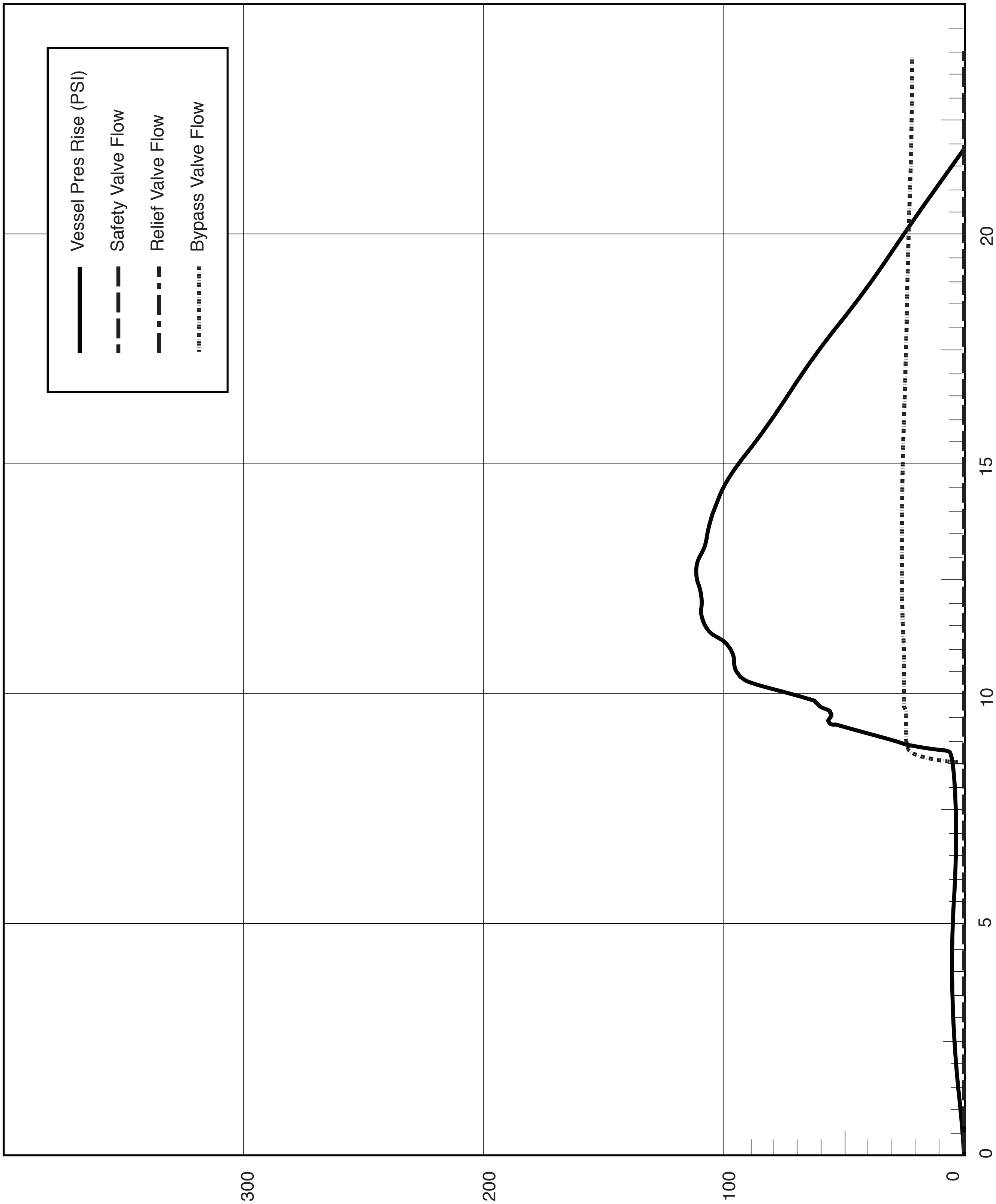
Columbia Generating Station **Final Safety Analysis Report**	**Feedwater Controller Failure, Maximum Demand, EOC RPT OOS, Single Loop Operation and 75% Uprated Power, 57% Flow**	
	Draw. No. **910402.41**	Rev.
	Figure **15.1-3.1**	



Columbia Generating Station
Final Safety Analysis Report

Feedwater Controller Failure, Maximum Demand, EOC RPT OOS, Single Loop Operation and 75% Uprated Power, 57% Flow

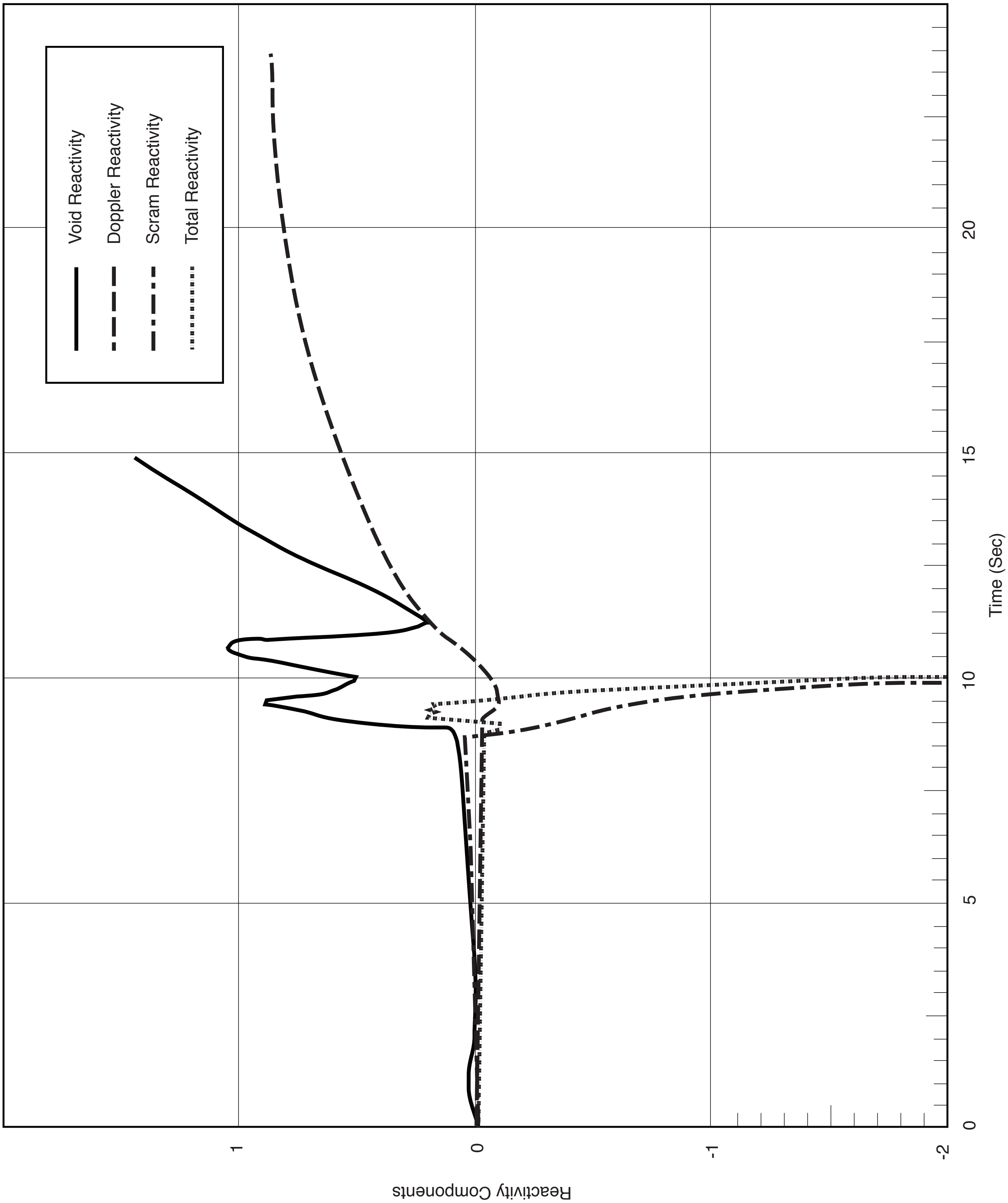
Draw. No. 910402.40 | Rev. | Figure 15.1-3.2



Columbia Generating Station
Final Safety Analysis Report

Feedwater Controller Failure, Maximum Demand, EOC RPT OOS, Single Loop Operation and 75% Uprated Power, 57% Flow

Draw. No. 910402.39	Rev.	Figure 15.1-3.3



Columbia Generating Station **Final Safety Analysis Report**	**Feedwater Controller Failure, Maximum Demand, EOC RPT OOS, Single Loop Operation and 75% Uprated Power, 57% Flow**		
	Draw. No. **910402.38**	Rev.	Figure **15.1-3.4**

15.2 INCREASE IN REACTOR PRESSURE

15.2.1 PRESSURE REGULATOR FAILURE - CLOSED

This transient is classified as a nonlimiting event for both original and uprated power conditions. Therefore, cycle specific analyses are not performed for this event. The analysis results presented in the section are based on uprated power conditions and a representative reload core (Cycle 8) as documented in Reference 15.2-5.

15.2.1.1 Identification of Causes and Frequency Classification

15.2.1.1.1 Identification of Causes

A triple redundant control system is provided to maintain primary system pressure control. The pressure upstream of the main turbine stop (throttle) valves is sensed by three redundant throttle pressure transmitters and the control system uses a median select logic to determine which pressure transmitter is used to control throttle pressure. The pressure control system compares the detected throttle pressure to a pressure setpoint to control the position of the main turbine control (governor) valves in order to control pressure.

It is assumed for purposes of this transient analysis that a single failure occurs on the controlling pressure transmitter which erroneously causes the DEH control system to close the turbine control (governor) valves and thereby increases reactor pressure. If this occurs, the self diagnostics ability and triple redundant control system is available.

15.2.1.1.2 Frequency Classification

This event is categorized as an incident of moderate frequency.

15.2.1.2 Sequence of Events and Systems Operation

15.2.1.2.1 Sequence of Events

A failure of a DEH control system component that causes the turbine control (governor) valves or turbine bypass valves to move towards the closed position will momentarily result in an initial pressure increase because the reactor is still generating the initial steam flow. The DEH control system is self diagnostic. It will detect the faulty component and disable it. The control system is redundant and will continue to perform its functions, and will restore steady state operation.

For a failure that causes the DEH turbine control pressure regulator to initiate a demand signal to close the turbine control (governor) valves (requires multiple component failures), there will be an increase in system pressure and reactor power. A scram will be initiated when the high

neutron flux scram setpoint is reached. The pressure rises to the pressure relief setpoint, part of the relief valves open, discharging steam to the suppression pool. The plant response is given in Table 15.2-1.

15.2.1.2.2 Systems Operation

Normal plant instrumentation and control is assumed to function except for the pressure regulator failure. The event is analyzed from 104.1% uprated power and 106% of rated core flow. The event results in a high flux trip initiated by the reactor protection system.

15.2.1.2.3 The Effect of Single Failures and Operator Errors

The first assumed failure produces a slight pressure increase in the reactor until the DEH control system adjusts to the single failure and gains control. No other action is significant in restoring normal operation. If subsequent failures occur such that the DEH control system further closes the turbine control (governor) valves the reactor pressure could rise to the point where a flux or pressure scram trip would be initiated to shutdown the reactor. This event is less severe than the turbine trip for the following reasons:

a. For the DEH control system failure-closure event the reactor scrams on high neutron flux or pressure but the recirculation pumps do not trip. As a result, core flow remains at 100% or greater throughout the critical portion of the transient with respect to the critical power ratio (CPR). This provides improved heat transfer capability in relation to the turbine trip transient; and

b. Since the turbine control (governor) valves close in response to a pressure error signal, their closure rate is not as fast as the turbine stop (throttle) or control (governor) valve response to a trip signal. This produces a slower pressurization rate for the DEH control system failure relative to the turbine trip event. This in turn results in a lower peak neutron flux and therefore a lower peak surface heat flux than the turbine trip event.

15.2.1.3 Core and System Performance

15.2.1.3.1 Mathematical Model

The one-dimensional ODYN model described in Section 15.0.3.3.1 is used to simulate this event.

15.2.1.3.2 Input Parameters and Initial Conditions

The analyses have been performed with plant conditions at 104.1% of uprated power and 106% of rated core flow. The input parameters are given in detail in Table 15.0-2 under the ODYN column.

15.2.1.3.3 Results

The closure of the turbine governor (control) valves results in a rise in reactor pressure, collapsing the coolant voids which in turn increases the neutron flux. One sec after the initiation of the event the neutron flux increases to the high flux setpoint signal and initiates a reactor scram. Two sec into the event the pressure in the reactor reaches the ATWS high pressure trip setpoint, initiating a recirculation pump trip signal. As the pressure in the reactor system continues to rise, the relief valves begin to open starting with Group 3. The maximum pressure is reached at 3.25 sec and is calculated to be 1220 psig at the bottom of the reactor vessel. Table 15.2-1 provides the sequence of events and Figure 15.2-1 depicts the plant parameters responses. Key transient peak values are presented in Table 15.0-1. This event is nonlimiting in that the pressurization event is less severe than the Generator Load Rejection with Bypass Failure and Turbine Trip with Bypass Failure events.

15.2.1.3.4 Consideration of Uncertainties

The uncertainties included in the initial power and flow considerations maximize the consequences of the plant response. The independent pressure regulators normally respond such that failure of one would be compensated by the other regulator with plant not experiencing a trip.

15.2.1.4 Barrier Performance

The consequences of this event do not result in any temperature or pressure transient (see Table 15.0-1) in excess of the criteria for which the fuel, pressure vessel, or containment are designed. Therefore, barrier integrity and function is maintained.

15.2.1.5 Radiological Consequences

Since this event does not result in any fuel failures or any release of primary coolant to either the secondary containment or to the environment, there are no radiological consequences associated with this event.

15.2.2 GENERATOR LOAD REJECTION

15.2.2.1 Identification of Causes and Frequency Classification

15.2.2.1.1 Identification of Causes

Fast closure of the turbine control (governor) valves (TCVs) is initiated whenever electrical grid disturbances occur which result 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 main TCVs will cause a sudden reduction in steam flow which results in an increase in system pressure, which may cause a reactor shutdown due to a high flux or high steam pressure condition.

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 a moderate frequency event.

15.2.2.2 Sequence of Events and System Operation

The generator load rejection with bypass failure event is the most limiting (with respect to thermal margin) of the class of transients characterized by rapid vessel pressurization, including load rejection with the bypass valves operating. The generator load rejection causes a TCV (governor valve) fast closure, which initiates a reactor scram and a recirculation pump trip (RPT). The compression wave produced by the TCV fast closure travels through the steam lines into the vessel and pressurizes the reactor vessel and core. Bypass flow to the condenser, which would mitigate the pressurization effect, is conservatively not allowed. The excursion of core power due to void collapse is primarily terminated by reactor scram and void growth due to RPT. The recirculation pump speed remains constant until tripped by the RPT system.

Events caused by low water level trips, including closure of main steam line isolation valves (MSIVs), and initiation of high-pressure core spray (HPCS) and reactor core isolation cooling (RCIC) are not included in the simulation. Should these events occur, they will follow after the primary concerns of fuel thermal margin and overpressure effects have occurred, and are expected to be less severe than those already experienced by the system.

15.2.2.2.1 Sequence of Events

15.2.2.2.1.1 Generator Load Rejection - Turbine Control Valve Fast Closure. This transient is classified as a nonlimiting event for both original and uprated power conditions. Therefore, the original rated power (3323 MWt) analysis has not been updated.

A loss of generator electrical load from high power conditions produces the sequence of events listed in Table 15.2-2.

15.2.2.2.1.2 Generator Load Rejection with Failure of Bypass. A loss of generator electrical load at 3486 MWt with bypass failure produces the sequence of events listed in Table 15.2-3.

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, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection, reactor pressure vessel (RPV) safety/relief valves (SRV), and reactor protection systems (RPS) unless stated otherwise. The bypass valve opening characteristics reflect the specified delay together with the specified opening characteristic required for bypass system operation.

Turbine control valve fast closure initiates a scram trip signal for power levels greater than 30% nuclear boiler rated (NBR). In addition, recirculation pump trip (RPT) is initiated. Both of these trip signals satisfy single failure criterion and credit is taken for these protection features.

The pressure relief system, which operates the SRVs independently when system pressure exceeds relief valve 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. Same as Section 15.2.2.2.2.1 except that failure of the main turbine bypass valves is assumed for the entire transient. In addition, the pressure relief system, which operates the SRVs independently when system pressure exceeds relief valve instrumentation setpoints, fails to operate. Pressure relief is provided by the safety function of the SRVs.

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 RPS functions. 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.

15.2.2.3 Core and System Performance

15.2.2.3.1 Mathematical Model

15.2.2.3.1.1 Generator Load Rejection with Bypass. The predicted dynamic behavior for the generator load reject with bypass valves operable has been determined using a computer simulated, analytical model of a generic direct-cycle BWR. This model is described in detail in Reference 15.2-4.

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

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

b. 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.

c. 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 safety/relief valve location), and turbine inlet pressure.

d. The active core void fraction is calculated from a relationship between core exit quality, inlet subcooling, and pressure. This relationship is generated from multimode 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 utilized.

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

f. The ability to simulate necessary reactor protection system functions is provided.

15.2.2.3.1.2 Generator Load Rejection with Bypass Failure. The predicted dynamic behavior for the load rejection with bypass inoperable has been determined using a computer simulated, analytical model of a direct-cycle BWR that is discussed in Section 15.1.2.3.1. This model is described in detail in Reference 15.2-2.

15.2.2.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Table 15.0-2 for the load rejection with bypass and in Table 15.0-2B for load rejection with bypass failure.

The turbine digital electrohydraulic control system power/load imbalance device detects load rejection before a measurable speed change takes place.

The closure characteristics of the TCVs are assumed such that the valves operate in the full arc (FA) mode and have a full stroke closure time, from fully open to fully closed, of 0.15 sec. In FA mode, at 100% power, the TCVs are not fully open, so the analysis assumes a closure time that is a fraction of the full stroke time proportional to the TCV initial position.

15.2.2.3.3 Results

Analyses were performed to analyze combinations of RPT operable/inoperable and Option A and Option B scram speeds (Reference 15.2-3). The excursion of core power due to void collapse is primarily terminated by reactor scram and void growth due to RPT.

15.2.2.3.3.1 Generator Load Rejection with Bypass. Figure 15.2-2.1 shows the results of the generator trip from original rated power. Peak neutron flux rises 156.8% above NBR conditions.

The average surface heat flux peaks at 102.9% of the initial value and minimum critical power ratio (MCPR) does not significantly decrease below its initial value.

15.2.2.3.3.2 Generator Load Rejection with Failure of Bypass. Figure 15.2-2.2 shows that, for the case of bypass failure, peak neutron flux reaches about 236% power, average surface heat flux reaches 111% of its initial value. Results reflect GE14 fuel introduction, some of which are dependent on fuel design and core loading pattern. Compliance with the event acceptance criteria is demonstrated by cycle-dependent analysis of potentially limiting events just prior to the operation of that cycle. The results are reported in the Supplemental Reload Licensing Report (Reference 15.2-3). As discussed in Section 15.0.2.1, when this event is initiated during single loop operation, the consequences are less severe than the consequences analyzed for the two loop operation.

15.2.2.3.4 Consideration of Uncertainties

The full stroke closure time of the TCV of 0.15 sec is conservative. Typically, the actual closure time is closer to 0.2 sec. The less time it takes to close, the more severe the pressurization effect.

All systems used for protection in this event were assumed to have the poorest allowable response. Expected plant behavior is, therefore, 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 peak steam line pressure reaches 1235 psig. The peak reactor coolant pressure boundary (RCPB) pressure reaches 1260 psig. The peak pressure remains well below the nuclear barrier transient pressure limit of 1375 psig.

15.2.2.5 Radiological Consequences

While the consequences of this event do not result in fuel failures, the result includes the discharge of normal coolant activity to the suppression pool by means of safety/relief valve (SRV) operation. Since this activity is contained in the primary containment, there will be no exposure to the public. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to hold the activity in containment or filter the discharge prior to release to the environment when conditions permit in accordance with established requirements.

15.2.3 TURBINE TRIP

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 will initiate a turbine trip. Some examples are moisture separator high levels, operator lockout, loss of control fluid pressure, low condenser vacuum, and reactor high water level.

15.2.3.1.2 Frequency Classification

15.2.3.1.2.1 Turbine Trip. This event is categorized as an incident of moderate frequency. In defining the frequency of this event, turbine trips which occur as a by-product of other transients such as loss of condenser vacuum or reactor high level trip events are not included. However, spurious low vacuum or high level trip signals which cause an unnecessary turbine trip are included in defining the frequency.

15.2.3.1.2.2 Turbine Trip with Failure of Bypass. This transient disturbance is categorized as a moderate frequency incident.

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. This transient is classified as a nonlimiting event for both original and uprated power conditions. Therefore, the original rated power (3323 MWt) analysis has not been updated. Turbine trip at high power produces the sequence of events listed in Table 15.2-4.

15.2.3.2.1.2 Turbine Trip with Failure of Bypass. Turbine trip at high power with bypass failure produces the sequence of events listed in Table 15.2-5.

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.

Turbine stop (throttle) valve closure initiates a reactor scram trip by means of valve position signals to the protection system.

Turbine stop valve closure initiates RPT thereby terminating the jet pump drive flow.

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

The severity of turbine trips from lower initial power levels decreases to the point where a scram can be avoided if auxiliary power is available from an external source and the power level is within the bypass capability.

15.2.3.2.2.2 Turbine Trip with Failure of Bypass. Same as 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. During the transient the SRVs open and close sequentially as the stored energy is dissipated until the pressure falls below the valve setpoints.

15.2.3.2.2.3 Turbine Trip at Low Power with Failure of Bypass. Same as Section 15.2.3.2.2.1 except that failure of the main turbine bypass system is assumed.

Below 30% NBR power level, a main stop valve scram trip inhibit signal derived from the first stage pressure of the turbine is activated. This is done to eliminate the stop valve scram trip 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 functional 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 30% Nuclear Boiler Rated. Mitigation of pressure increase, the basic nature of this transient, is accomplished by the RPS functions. Main stop valve closure scram trip and RPT are designed to satisfy single failure criterion.

15.2.3.2.3.2 Turbine Trips at Power Levels Less Than 30% Nuclear Boiler Rated. Same as Section 15.2.3.2.3.1 except RPT and stop valve closure scram trip is normally inoperative. Since protection is still provided by high flux, high pressure, etc., these will continue to function and scram the reactor should a single failure occur.

15.2.3.3 Core and System Performance

15.2.3.3.1 Mathematical Model

15.2.3.3.1.1 Turbine Trip with Bypass. The predicted dynamic behavior for the turbine trip has been determined using a computer simulated, analytical model of a generic direct-cycle BWR, as discussed in Section 15.2.2.3.1. This model is described in detail in Reference 15.2-4.

15.2.3.3.1.2 Turbine Trip with Bypass Failure. The one-dimensional ODYN model described in Section 15.0.3.3.1 is used to simulate this event.

15.2.3.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2 for the turbine trip with bypass and Table 15.0-2B for the turbine trip with bypass failure.

The turbine trip analysis was performed at the 105% of the original rated steam flow. The turbine trip with bypass failure was analyzed at an initial condition of 100% rated power (3486 MWt) and 106% rated core flow.

Turbine stop (throttle) valves full stroke closure time is 0.1 sec.

A reactor scram is initiated by position switches on the stop valves when the valves are 90% open or less. This stop valve scram trip signal is automatically bypassed when the reactor is below 30% NBR power level.

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

15.2.3.3.3 Results

15.2.3.3.3.1 Turbine Trip. The results of a turbine trip with the bypass system operating normally are shown in 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 147.5% of rated by the stop valve scram and the RPT system. Peak fuel surface heat flux does not exceed 101.7% of its initial value.

15.2.3.3.3.2 Turbine Trip with Failure of Bypass. The results of a turbine trip with failure of the bypass system are shown in Figure 15.2-4.

The peak neutron flux reaches 278% of its rated value, and peak surface heat flux reaches 111% of its initial value.

Results reflect GE14 fuel introduction, some of which are dependent on fuel design and core loading pattern. Compliance with the event acceptance criteria is demonstrated by cycle-dependent analysis of potentially limiting events just prior to the operation of that cycle. The results are reported in the Supplemental Reload Licensing Report (Reference 15.2-3).

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 30% of rated power, the turbine stop valve closure and TCV (governor valve) closure scrams are automatically bypassed. At these lower power levels, turbine first stage pressure is used to initiate the scram logic bypass. The scram which terminates the transient is initiated by high vessel pressure. The bypass valves are assumed to fail; therefore, system pressure will increase until the pressure relief setpoints are reached. At this time, because of the relatively low power of this transient event, relatively few relief valves will open to limit reactor pressure. Peak pressures are not expected to greatly exceed the pressure relief valve setpoints and will be significantly below the reactor coolant pressure boundary (RCPB) transient limit of 1375 psig. Peak surface heat flux and peak fuel center temperature remain at relatively low values and MCPR remains well above the GETAB safety limit.

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:

a. Slowest allowable control rod scram motion is assumed,
b. Scram worth shape for all-rod-out conditions is assumed,
c. Minimum specified valve capacities are utilized for overpressure protection, and
d. Setpoints of the SRVs include errors and uncertainties (high) for all valves.

15.2.3.4 <u>Barrier Performance</u>

15.2.3.4.1 Turbine Trip

Peak pressure in the bottom of the vessel reaches 1163 psig, which is below the American Society of Mechanical Engineers (ASME) Code limit of 1375 psig for the RCPB. Vessel dome pressure does not exceed 1136 psig.

15.2.3.4.2 Turbine Trip with Failure of Bypass

The peak steam line pressure reaches 1235 psig. The peak reactor coolant pressure boundary (RCPB) pressure reaches 1260 psig. The peak pressure remains well below the nuclear barrier transient pressure limit of 1375 psig.

15.2.3.4.2.1 <u>Turbine Trip with Failure of Bypass at Low Power</u>. Qualitative discussion is provided in Section 15.2.3.3.3.3.

15.2.3.5 <u>Radiological Consequences</u>

The consequence of this event does not result in fuel failure. It does result in the discharge of normal coolant activity to the suppression pool by means of SRV operation, which is contained in the primary containment. This event does not result in an uncontrolled release to the environment, so the plant operator can choose to hold the activity in containment or discharge it when conditions permit. If purging of the containment is chosen, the release would be in accordance with established requirements.

15.2.4 MAIN STEAM LINE ISOLATION VALVE CLOSURES

This transient is classified as a nonlimiting event for both original and uprated power conditions. Therefore, cycle specific analyses are not performed for this event. The analysis results presented in the section are based on uprated power conditions and a representative reload core (Cycle 8) as documented in Reference 15.2-5.

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 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 Main Steam Line Isolation Valves. This event is categorized as an incident of moderate frequency. To define the frequency of this event as an initiating event and not the byproduct of another transient, only the following contribute to the frequency: Manual action (purposely or inadvertent); spurious signals such as low pressure, low reactor water level, and low condenser vacuum; and equipment malfunctions such as faulty valves or operating mechanisms. A closure of one MSIV may cause an immediate closure of all the other MSIVs depending on reactor conditions. If this occurs, it is also included in this category. During the MSIV closure, position switches on the valves provide a reactor scram when the valves in three or more main steam lines are less than 90% open (except for interlocks which permit proper plant startup). Protection system logic permits the test closure of one valve without initiating scram from the position switches.

15.2.4.1.2.2 Closure of One Main Steam Line Isolation Valve. This event is categorized as an incident of moderate frequency. One MSIV at a time may be manually closed for testing purposes. Operator error or equipment malfunction may cause a single MSIV to be closed inadvertently. If reactor power is greater than about 75% when this occurs, a high flux or high steam line flow condition may result in a scram. If all MSIVs close as a result of the single event, the event is considered as a closure of all MSIVs. The results presented for this event assume all MSIVs close as a result of an unspecified initiating event.

15.2.4.2 Sequence of Events and Systems Operation

15.2.4.2.1 Sequence of Events

Table 15.2-6 lists the sequence of events for Figure 15.2-5. When the MSIV's reach their 85% open position, a reactor scram is initiated by the reactor protection system. The valve closure results in a system pressure increase which in turn results in a spike in reactor neutron flux. The reactor vessel pressure increase also results in an ATWS recirculation pump trip (RPT). As the pressure increases, the relief valves begin to open terminating the pressure increase.

15.2.4.2.2 Systems Operation

15.2.4.2.2.1 Closure of All Main Steam Line Isolation Valves. The MSIV closures initiate a reactor scram trip by means of position signals to the protection system. Credit is taken for successful operation of the protection system.

The pressure relief system which initiates opening of the relief valves when system pressure exceeds relief valve instrumentation setpoints is assumed to function normally during the time period analyzed.

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

15.2.4.2.2.2 Closure of One Main Steam Line Isolation Valve. A closure of a single MSIV will not initiate a reactor scram by means of the position signal to the protection system. This is because the valve position scram trip 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 maintain normal operation unless specifically designated to the contrary.

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 by means of MSIV position switches and the protection system. Relief valves also operate to limit system pressure. All of these aspects are designed to single failure criterion and additional single failures would not alter the results of this analysis.

Failure of a single relief valve to open is not expected to have any significant effect. Such a failure is expected to result in less than a 5 psi increase in the maximum vessel pressure rise. The peak pressure will still remain considerably below 1375 psig.

15.2.4.3 Core and System Performance

15.2.4.3.1 Mathematical Model

The point-kinetics REDY model described in Section 15.0.3.3.1 is used to simulate these transient events.

15.2.4.3.2 Input Parameters and Initial Conditions

It is assumed the closure of all MSIVs occurs with the plant operating at 106% of uprated power and 100% core flow. The input parameters are defined with the plant conditions tabulated in Table 15.0-2 for power uprate.

The MSIVs close in 3 to 5 sec. The worst case, the 3-sec closure time, is assumed in this analysis.

Position switches on the valves initiate a reactor scram when the valves are less than 90% open as described in Section 7.2 (85% is assumed in the analysis). Closure of these valves inhibits steam flow to the feedwater turbines terminating feedwater flow.

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

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 HPCS and RCIC systems.

15.2.4.3.3 Results

15.2.4.3.3.1 Closure of All Main Steam Line Isolation Valves. The reactor scram is initiated at 0.45 sec when the MSIVs reach 85% open position. The nuclear system relief valves begin to open at 3.08 sec after the start of isolation. The valves close sequentially as the stored heat is dissipated but continue to discharge the decay heat intermittently. Table 15.2-6 provides the sequence of events and Figure 15.2-5 depicts the plant parameters responses. Key transient peak values are presented in Table 15.0-1. This event is non-limiting in that the pressurization event and change in CPR margin is less severe than the Generator Load Rejection with Bypass Failure and Turbine Trip with Bypass Failure events.

15.2.4.3.3.2 Closure of One Main Steam Line Isolation Valve. Only one isolation valve is permitted to be closed at a time for testing purposes to prevent scram. Normal test procedure requires an initial power reduction to approximately 65% to 70% of design conditions to avoid high-flux scram, high-pressure scram, or full isolation from a high-steam flow condition in the open steam lines. With a 3-sec closure of one MSIV during 105% of original rated power conditions, the steam flow disturbance raises 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.

Inadvertent closure of one or all of the isolation valves while the reactor is shut down will produce no significant transient. Closures during plant heatup will be less severe than the maximum power cases (maximum stored and decay heat).

15.2.4.3.4 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:

a. Slowest allowable control rod scram motion is assumed,

b. Scram worth shape for all-rod-out conditions is assumed,

c. Minimum specified valve capacities are used for overpressure protection, and

d. Setpoints of the SRVs are assumed to be 15 psi higher than the valve's nominal setpoint.

15.2.4.4 Barrier Performance

15.2.4.4.1 Closure of All Main Steam Line Isolation Valves

The nuclear system relief valves begin to open at approximately 3.1 sec after the start of isolation. The valves close sequentially as the stored heat is dissipated but continue to discharge the decay heat intermittently. Peak pressure at the vessel bottom reaches 1234 psig, clearly below the pressure limits of the RCPB. Peak pressure in the main steam line is 1198 psig.

15.2.4.4.2 Closure of One Main Steam Line Isolation Valve

No significant effect is imposed on the RCPB, since if closure of the valve occurs at a high operating power level a flux or pressure scram will result. The main turbine bypass system will continue to regulate system pressure by means of the other three steam lines.

15.2.4.5 Radiological Consequences

While the consequence of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to the public. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to hold the activity in containment or discharge it to the environment when conditions permit. If purging of the containment is chosen, the release would be in accordance with established requirements.

15.2.5 LOSS-OF-CONDENSER VACUUM

This transient is classified as a nonlimiting event for both original and uprated power conditions. Therefore, cycle specific analyses are not performed for this event. The analysis results presented in the section are based on uprated power conditions and a representative reload core (Cycle 8) as documented in Reference 15.2-5.

15.2.5.1 Identification of Causes and Frequency Classification

15.2.5.1.1 Identification of Causes

Various malfunctions can cause a loss-of-condenser vacuum. The causes and estimated vacuum decay rates include failure or isolation of steam jet air ejectors (< 1 in. Hg/mm), loss of sealing steam shaft gland seals (1 to 2 in. Hg/minute), opening of vacuum breaker valves (2 to 12 in. Hg/minute), and loss of one or more circulating water pumps (4 to 24 in. Hg/minute).

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 Figure 15.2-6.

15.2.5.2.2 Systems Operation

It is conservatively assumed that condenser vacuum is lost at a rate of 2 inches of Hg per second. The bypass system is signaled to close approximately 10 inches of Hg less than the stop (throttle) valve closure vacuum setpoint level which means the bypass is available for approximately 5 sec before the turbine stop (throttle) valves close. The loss of vacuum initiates a main turbine trip and feedwater turbine trip. Upon reaching 90% close, the turbine throttle (stop) valves closure results in a reactor scram. As the reactor pressure increases, the relief valves will open. Subsequently, this results in the main steam line isolation valves to close. However, the effect of the MSIV closure is minimal since the turbine stop (throttle) valve and bypass valve closure have already terminated main steam line flow.

Tripping functions incurred by sensing main turbine condenser vacuum are designated in Table 15.2-8.

15.2.5.2.3 The Effect of Single Failures and Operator Errors

Single failure will not effect the vacuum monitoring and turbine trip devices which are redundant. The protective sequences of the anticipated operational transient are shown to be single failure proof.

15.2.5.3 Core and System Performance

15.2.5.3.1 Mathematical Model

The one-dimensional ODYN model described in Section 15.0.3.3.1 was used to simulate this transient event.

15.2.5.3.2 Input Parameters and Initial Conditions

This analysis was performed with plant conditions tabulated in Table 15.0-2 and at 104.1% of uprated power and 100% core flow. Turbine stop (throttle) valves full stroke closure time used in this analysis is 0.1 second and a reactor scram is initiated by position switches on the stop valves when the valves are less than 90% open. The 2 inches of Hg per second assumed in the analysis is conservative with respect to normal loss of vacuum and no operator actions are assumed.

Thus, the bypass system is available for several seconds since the bypass is signaled to close at a vacuum level of about 10 in. Hg less than the stop valve closure.

15.2.5.3.3 Results

The loss of condenser vacuum initiates a main turbine trip, which then initiates turbine bypass operation. The bypass is available for approximately 5 sec until both the turbine bypass valves and the main steam line isolation valves receive a signal to close on low condenser vacuum. The effect of MSIV closure tends to be minimal since the closure of main turbine stop valves and subsequently 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% of uprated NBR steam flow conditions. Peak neutron flux reaches 256% of NBR power while average fuel surface heat flux reaches 111% of rated value. The SRVs open to limit the pressure rise then sequentially reclose as the stored energy is dissipated.

15.2.5.3.4 Consideration of Uncertainties

The reduction or loss of vacuum in the main turbine condenser will sequentially trip the main and feedwater turbines and close the MSIVs and turbine bypass valves. While these are the major events occurring, other resultant actions will include scram (from stop valve closure)

and bypass opening with the main turbine trip. Because the protective actions are actuated at various levels of condenser vacuum, the severity of the resulting transient is dependent upon the rate at which the vacuum is lost. Normal loss of vacuum due to loss-of-cooling water pumps or steam jet air ejector problem produces a very slow rate of loss of vacuum (minutes, not seconds). If corrective actions by the reactor operators are not successful, then simultaneous trips of the main and feedwater turbines, and ultimately complete isolation by closing the bypass valves (opened with the main turbine trip) and the MSIVs, will occur.

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

a. Slowest allowable control rod scram motion is assumed,

b. Scram worth shape for all-rod-out conditions is assumed,

c. Minimum specified valve capacities are utilized for overpressure protection, and

d. Setpoints of the SRVs are assumed to be 15 psi higher than the valve's nominal setpoint.

15.2.5.4 Barrier Performance

The maximum calculated pressure for this event as presented in Table 15.0-1 is below the ASME Code limit of 1375 psig for the RCPB and the ASME Service Level C of 1500 psig. The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed. Therefore, barrier integrity and function is maintained.

15.2.5.5 Radiological Consequences

While the consequence of this event does not result in fuel failures, it does result in the discharge of normal coolant activity to the suppression pool by means of SRV operation. Since this activity is contained in the primary containment, there will be no exposure to the public. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to hold the activity in containment or discharge it to the environment when conditions permit. If purging of the containment is chosen, the release would be in accordance with established requirements.

15.2.6 LOSS OF ALTERNATING CURRENT POWER

This transient considers the loss of AC power to the plant from both an onsite cause (loss of auxiliary power transformer) and an offsite cause (loss of all grid connections).

This transient is classified as a nonlimiting event for both original and uprated power conditions. Therefore, cycle specific analyses are not performed for this event. The analysis results presented in the section are based on uprated power conditions and a representative reload core (Cycle 8) as documented in Reference 15.2-5.

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 Transformers. Causes for interruption or loss of the auxiliary power transformers can arise from normal operation or malfunctioning of transformer protection circuitry. These can include high transformer oil temperature, reverse of high current operation, and operator error which trips the transformer breakers.

15.2.6.1.1.2 Loss of All Grid Connections. 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 will 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

15.2.6.1.2.1 Loss of Auxiliary Power Transformers. This event is categorized as an incident of moderate frequency.

15.2.6.1.2.2 Loss of All Grid Connections. This event 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

15.2.6.2.1.1 Loss of Auxiliary Power Transformers. Table 15.2-9 lists the sequence of events for Figure 15.2-7.

15.2.6.2.1.2 Loss of All Grid Connections. Table 15.2-10 lists the sequence of events for Figure 15.2-8.

15.2.6.2.2 Systems Operation

15.2.6.2.2.1 Loss of Auxiliary Power Transformers. This event, unless otherwise stated, assumes and takes credit for normal functioning of plant instrumentation and controls, plant protection, and reactor protection systems.

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

a. Recirculation pumps and condenser circulatory water pumps trip off at time = 0. A 4 sec recirculation pump trip inertia time constant is assumed for this analysis;

b. Reactor scram and MSIV closure is initiated at 2 sec due to loss of power to the scram and MSIV relay solenoids; and

c. Feedwater turbines trip off at 4 sec due to MSIV closure at 2 sec.

Operation of the HPCS and RCIC are not simulated in this analysis. Their operation occurs at a time beyond the primary concerns of fuel thermal margin and overpressure effects of this analysis.

15.2.6.2.2.2 Loss of All Grid Connections. Same as Section 15.2.6.2.2.1 with the following additional concern.

The loss of all grid connections would add a generator load rejection to the above sequence at time, t=0. The load rejection immediately causes the TCVs (governor valves) to close, causes a scram, and initiates RPT [already tripped at reference time t = 0].

15.2.6.2.3 The Effect of Single Failures and Operator Errors

Loss of the auxiliary power transformers in general leads to a reduction in power level due to rapid pump coastdown with pressurization effects due to MSIV closure resulting from loss of power to the solenoids. Additional failures of the other systems assumed to protect the reactor would not result in an effect different from those reported. Failures of the protection systems have been considered and satisfy single failure criteria and, as such, no change in analyzed consequences is expected.

15.2.6.3 Core and System Performance

15.2.6.3.1 Mathematical Model

The point-kinetics REDY model described in Section 15.0.3.3.1 was used to simulate this event.

15.2.6.3.2 Input Parameters and Initial Conditions

15.2.6.3.2.1 Loss of Auxiliary Power Transformers. It is assumed the loss of the auxiliary power transformer occurs with the plant operating at 106% of uprated power and 100% core flow. The input parameters are defined with the plant conditions tabulated in Table 15.0-2 except as noted below.

a. The recirculation pump trip inertia time constant is 4 sec.

b. The relay-type Reactor Trip System (RTS) circuitry generates a reactor scram and Main Steam Isolation Valves (MSIV) closure signal due to loss of power to the scram and MSIV solenoids. This occurs 2 sec after the loss of offsite power.

c. The feedwater pumps trip due to MSIV closure 2 sec after the MSIV begin to close as a result of the loss of power to the MSIV solenoids.

15.2.6.3.2.2 Loss of All Grid Connections. It is assumed the loss of all grid connections occurs with the plant operating at 104% of uprated power and 100% core flow. The input parameters are defined with the plant conditions tabulated in Table 15.0-2 except as noted below.

a. The recirculation pump trip inertia time constant is 4 sec.

b. The relay-type Reactor Trip System (RTS) circuitry generates a Main Steam Isolation Valves (MSIV) closure signal due to loss of power to the MSIV solenoids. This occurs 2 sec after the loss of offsite power.

c. The feedwater pumps trip due to MSIV closure 2 sec after the MSIV begin to close as a result of the loss of power to the MSIV solenoids.

15.2.6.3.3 Results

15.2.6.3.3.1 Loss of Auxiliary Power Transformers. Initially the offsite power is cutoff causing both recirculation pumps to trip. The loss of power to the scram and MSIV solenoids causes a reactor scram, MSIV isolation and a feedwater pump trip 2 sec after the MSIV

isolation. Subsequently, the feedwater recirculation valves trip and the relief valves begin to open due to the rising pressure caused by the main steam line isolation. Table 15.2-9 provides the sequence of events and Figure 15.2-7 depicts the plant parameters responses. Key transient peak values are presented in Table 15.0-1. This event is non-limiting in that the pressurization event and change in MCPR margin are less severe than the Generator Load Rejection with Bypass Failure and Turbine Trip with Bypass Failure events.

15.2.6.3.3.2 Loss of All Grid Connections. Loss of all grid connections is a more general form of loss of auxiliary power. It essentially takes on the characteristic response of the standard full load rejection discussed in Section 15.2.2. Initially the offsite power is cutoff to the grid causing the turbine-generator to detect a loss of electrical load, and a power-load unbalance. The turbine generator overspeed protection control (OPC) initiates a control (governor) valve fast closure, turbine bypass valves opening and a reactor scram. At the same time both recirculation pump motors trip. Subsequently, MSIV isolation occurs and both feedwater pumps trip. The rising pressure due to the isolation of the steam line causes the relief valves to open. Table 15.2-10 provides the sequence of events and Figure 15.2-8 depicts the plant parameter responses. Key transient peak values are presented in Table 15.0-1. This event is non-limiting in that the pressurization event and change in MCPR margin are less severe than the Generator Load Rejection with Bypass Failure and Turbine Trip with Bypass Failure events.

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 HPCS and RCIC systems are not included in the simulation of the first 50 sec of this transient. Startup of the pumps occurs in the latter part of this time period but the system has no significant effect on the results of this transient.

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

15.2.6.4 Barrier Performance

15.2.6.4.1 Loss of Auxiliary Power Transformers

Safety/relief valves open in the pressure relief mode of operation as the pressure increases beyond their setpoints. The pressure in the dome is limited to a maximum value of 1169 psig well below the vessel pressure limit of 1375 psig.

15.2.6.4.2 Loss of All Grid Connections

Safety/relief valves open in the pressure relief mode of operation as the pressure increases beyond their setpoints. The pressure in the dome is limited to a maximum value of 1173 psig well below the vessel pressure limit of 1375 psig.

15.2.6.5 Radiological Consequences

The consequence of this event does not result in fuel failure. It does result in the discharge of normal coolant activity to the suppression pool by means of SRV operation, which is contained in the primary containment. This event does not result in an uncontrolled release to the environment, so the plant operator can choose to hold the activity in containment or discharge it when conditions permit. If purging of the containment is chosen, the release would be in accordance with established requirements.

15.2.7 LOSS-OF-FEEDWATER FLOW

This transient is classified as a nonlimiting event for both original and uprated power conditions. Therefore, cycle specific analyses are not performed for this event. The analysis results presented in the section are based on uprated power conditions and a representative reload core (Cycle 8) as documented in Reference 15.2-5. The analysis has not been updated for the change in MSIV isolation setpoint from Level 2 to Level 1 because the analysis is bounding and conclusions of the analysis are not affected (Reference 15.2-8).

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, or reactor system variables such as a high vessel water level (L8) trip signal.

15.2.7.1.2 Frequency Classification

This event 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

Feedwater flow terminates at approximately 5 sec. Subcooling decreases causing a reduction in core power level and pressure. As power level is lowered, the turbine steam flow starts to drop off. Water level continues to drop until the vessel level (L3) scram trip setpoint is reached, whereupon the reactor is shut down.

Main steam line isolation initiation occurs due to vessel water dropping to the L2 trip. Also at this time, the recirculation system is tripped and HPCS and RCIC operation is initiated. Operation of the HPCS and RCIC systems is not included in the simulation of the first 50 seconds of this transient since startup of the pumps occurs in the latter part of this time period. Therefore, the system has no significant effects on the results of this transient.

Table 15.2-11 lists the sequence of events for Figure 15.2-9.

15.2.7.2.2 Systems 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 trip actuation. Reactor protection system responds after this trip to scram the reactor. The low level (L3) scram trip function meets single failure criterion.

Vessel water level (L2) trip initiates main steam line isolation, recirculation pump trip and HPCS/RCIC system operation (not simulated). The recirculation pump motor circuit breakers then open causing decrease in core flow to natural circulation.

15.2.7.2.3 The Effect of Single Failures and Operator Errors

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

The potential exists for a single relief valve failing to close once it is opened. This would result in a complete depressurization of the reactor. Either the RCIC or the HPCS system is capable of maintaining adequate core coverage and will provide long-term inventory control. For the complete loss of feedwater flow event, operation of RCIC or HPCS is sufficient to avoid initiation of ADS on low vessel level (L1).

15.2.7.3 Core and System Performance

15.2.7.3.1 Mathematical Model

The point-kinetics REDY model described in Section 15.0.3.3.1 was used to simulate this event.

15.2.7.3.2 Input Parameters and Initial Conditions

The simultaneous trip of both feedwater pumps is assumed to occur while the plant is operating at 106% uprated power and 100% core flow. These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2.

15.2.7.3.3 Results

Table 15.2-11 provides the sequence of events and Figure 15.2-9 depicts the plant parameter responses. Key transient peak values are presented in Table 15.0-1. This event is non-limiting in that the pressurization event and change in MCPR are less severe than the Generator Load Rejection with Bypass Failure and Turbine Trip with Bypass Failure events.

15.2.7.3.4 Consideration of Uncertainties

End-of-cycle scram characteristics are assumed.

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

15.2.7.4 Barrier Performance

Peak pressure in the bottom of the vessel reaches 1152 psig, which is below the ASME Code limit of 1375 psig for the RCPB. Vessel dome pressure does not exceed 1142 psig. The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed. Therefore, barrier integrity and function are maintained.

15.2.7.5 Radiological Consequences

The consequence of this event does not result in fuel failure. It does result in the discharge of normal coolant activity to the suppression pool by means of SRV operation, which is contained in the primary containment. This event does not result in an uncontrolled release to the environment, so the plant operator can choose to hold the activity in containment or discharge it when conditions permit. If purging of the containment is chosen the release will be in accordance with established requirements.

15.2.8 FEEDWATER LINE BREAK

See Section 15.6.6.

15.2.9 FAILURE OF RESIDUAL HEAT REMOVAL SHUTDOWN COOLING

This transient is classified as a nonlimiting event for both original and uprated power conditions.

Normally, in evaluating component failures associated with the RHR shutdown cooling mode of operation, active pumps or instrumentation (all of which are redundant for the safety related portions of the RHR system) would be assumed to be the component failure. For purposes of a worst case analysis, a valve on the single recirculation suction line to the otherwise redundant RHR shutdown cooling loops is assumed to fail. Manual attempts to open the valve are assumed unsuccessful. Discovery is conservatively assumed to occur at 100 psig. This envelops discovery at normal RHR shutdown cooling operating limits (see Section 5.4.7). This failure disables the shutdown cooling mode but does not affect the remaining RHR modes of operation. Reference 15.2-1 establishes additional assumptions.

15.2.9.1 Identification of Causes and Frequency Classification

15.2.9.1.1 Identification of Causes

The plant is operating at 105% of original NBR steam flow when an event occurs, e.g., a long-term loss of offsite power, causing a plant shutdown. Reactor vessel depressurization is initiated to bring the reactor pressure to approximately 100 psig. Concurrent with the loss of offsite power a failure of a valve in the shutdown cooling suction line occurs which prevents the operator from establishing the normal shutdown cooling path through the RHR shutdown cooling lines. An additional failure is assumed which completely disables the RHR equipment in one division. The operator then establishes a shutdown cooling path for the vessel through the SRV valves.

15.2.9.1.2 Frequency Classification

This event is categorized as an incident of moderate frequency.

15.2.9.2 Sequence of Events and Systems Operation

15.2.9.2.1 Sequence of Events

The sequence of events for this event is shown in Table 15.2-12.

15.2.9.2.1.1 Identification of Operator Actions. For the early part of the transient, the operator actions are to restore and maintain reactor water level. The operator should reestablish reactor cooling by one or more of the following:

a. Maintain reactor water inventory with the RCIC (when single failure is not assumed to be a loss of Division 1 dc power) and HPCS systems,

b. At approximately 10 minutes into the transient, initiate suppression pool cooling, it is assumed that only one RHR heat exchanger is available,

c. Initiate RPV shutdown depressurization by manual actuation of the SRVs,

d. Attempts to open one of the two RHR shutdown cooling suction valves are assumed unsuccessful (reactor pressure is approximately 100 psig), and

e. Continue RPV depressurization by opening SRVs and establish a reactor cooling path as described in the notes for Figure 15.2-10.

Time required to initiate the necessary steps to maintain reactor pressure and level control is approximately 10 minutes.

15.2.9.2.2 Systems 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 ESF used.

15.2.9.2.3 The Effect of Single Failures and Operator Errors

The worst case single failure (loss of division power) has already been analyzed in this event. Therefore, no single failure or operator error can increase the consequences of this event.

15.2.9.3 Core and System Performance

The earliest time the shutdown cooling system can be actuated is 2 to 3 hr after shutdown is initiated. During this time MCPR remains high and nucleate boiling heat transfer is not exceeded at any time. Therefore, the core thermal safety margin remains essentially unchanged. The 10-minute time period approximated 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.

The transient behavior of the core during this event has been evaluated in Section 15.2.6.

15.2.9.4 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 the redundant shutdown cooling loop. In cases where the RHR shutdown cooling suction line valves cannot be opened, alternate paths are available to accomplish the shutdown cooling function (Figure 15.2-11). An evaluation has been performed assuming a failure that disables the RHR shutdown cooling suction line valves.

This 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 RCPB are not exceeded.

The alternate cooldown path chosen to accomplish the shutdown cooling function uses the RHR and ADS or normal relief valve systems (see Reference 15.2-1 and Figure 15.2-10). The alternate shutdown systems are capable of performing the function of transferring heat from the reactor to the environment using only safety systems. The systems are capable of bringing the reactor to a cold shutdown in approximately 36 hr or less after the transient occurs.

The systems have suitable redundancy in components such that even for onsite electrical power operation (offsite power is not available), the safety function of the systems can be accomplished assuming an additional single failure. The systems can be fully operated from the main control room.

The design evaluation is divided into two phases: (a) full power operation to approximately 100 psig vessel pressure, and (b) approximately 100 psig vessel pressure to cold shutdown (14.7 psia 200°F) conditions.

15.2.9.4.1 Full Power to Approximately 100 psig

Independent of the event that initiated plant shutdown (whether it be a normal plant shutdown or a forced plant shutdown), the reactor is normally brought to approximately 100 psig using either the main condenser or, in the case where the main condenser is unavailable, the HPCS and RCIC systems together with the nuclear boiler pressure relief system and the RHR heat exchanger in the suppression pool cooling mode.

For evaluation purposes, however, it is assumed that plant shutdown is initiated by a transient event (loss of offsite power) which results in relief valve actuation and subsequent suppression pool heatup. For this postulated condition, the reactor is shut down and the reactor vessel pressure is reduced to approximately 100 psig. Manual operation of the SRVs is used to depressurize the reactor vessel. Reactor vessel makeup water is automatically provided by means of the RCIC (until reduced vessel pressure is reached) and HPCS systems. While in

this condition, the RHR system (suppression pool cooling mode) is used to maintain the suppression pool temperature within shutdown limits.

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

15.2.9.4.2 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:

a. The vessel is at 100 psig and saturated conditions,

b. A worst-case single failure is assumed to occur (i.e., loss of a division of emergency power), and

c. There is no offsite power available.

In the event that the RHRs shutdown suction line is not available because of single failure, the first action to be taken will be to control reactor pressure. If a single electrical failure caused the suction line 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 restored to service, the capabilities described below will satisfy the normal shutdown cooling requirements and thus fully comply with GDC 34.

The RHR shutdown cooling line valves are in two divisions (Division 1 - the outboard valve, and Division 2 - the inboard valve) to satisfy containment isolation criteria. For evaluation purposes, the worst-case failure is assumed to be the loss of a division of emergency power, since this also prevents electrical actuation of one shutdown cooling line valve. Engineered safety features equipment and safe shutdown RCIC equipment (until reduced reactor pressure is reached) available for accomplishing the shutdown cooling function include (for the selected path):

ADS (dc Division 1 and dc Division 2)

RHR Loop A (Division 1)

HPCS (Division 3)

RCIC (dc Division 1)

LPCS (Division 1)

Since availability or failure of Division 3 equipment does not affect the normal shutdown mode, normal shutdown cooling is easily available through equipment powered from only Divisions 1 and 2. It should be noted that, HPCS is always available for coolant injection if either of the other two divisions fails. For failure of Division 1 or 2, the following systems are assumed functional:

a. Division 1 Fails, Division 2 and 3 Functional

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

Assuming the single failure is a failure of Division 1 emergency power, the safety function is accomplished by establishing one of the cooling loops described in Activity C1 of Figure 15.2-10.

b. Division 2 Fails, Division 1 and 3 Functional

Failed Systems	Functional Systems
RHR Loop B and C	HPCS ADS RHR Loop A LPCS RCIC (until reduced reactor pressure is reached)

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 15.2-10. Figures 15.2-12 through 15.2-15 show RHR loops A, B, and/or C (simplified).

15.2.9.4.3 Temperature Response – 3462 MWt

The reactor vessel temperature and pressure response versus time for the core conditions defined in Table 15.2-13 (105% of original rated steam flow, 3462 MWt rated power) are presented in Figures 15.2-16 and 15.2-17. Figure 5.2-16 presents the results for

Activity C1.b.1 or C2 described in Figure 15.2-10. Figure 5.2-17 presents the results for Activity C1.b.2. The bulk suppression pool temperature responses from the same analysis are presented in Figures 15.2-18 and 15.2-19. Figure 5.2-18 presents the results for Activity C1.b.1 or C2 and Figure 5.2-19 presents the results for Activity C1.b.2.

15.2.9.4.4 Temperature Response – 3702 MWt

Reference 15.2-7 analyzed the same two scenarios (Activities C2 and C1.b.2) for 106.2% of power uprate conditions (3702 MWt) to determine the peak bulk suppression pool temperature and the time required to cool the reactor vessel to cold shutdown (14.7 psia and 200°F). The analysis at power uprate conditions calculated a relative 4°F increase in the peak bulk pool temperature due to the power uprate. However, the peak temperature calculated was lower than the temperatures presented in this section due to the use of more realistic assumptions. These assumptions include a more realistic decay heat model, lower initial suppression pool temperature (90°F), and more realistic treatment of pump heat addition.

15.2.9.5 Barrier Performance

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

15.2.9.6 Radiological Consequences

The consequence of this event does not result in fuel failure. It does result in the discharge of normal coolant activity to the suppression pool by means of SRV operation, which is contained in the primary containment. This event does not result in an uncontrolled release to the environment, so the plant operator can choose to hold the activity in containment or discharge it when conditions permit. If purging of the containment is chosen, the release would be in accordance with established requirements.

15.2.10 REFERENCES

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15.2-2 NEDC-24154-P-A, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," Volumes 1, 2, 3 and 4, February 2000.

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15.2-4 R. B. Linford, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO 10802, April 1973.

15.2-5 For Power Uprate: GE Nuclear Energy, "WNP-2 Power Uprate Transient Analysis Task Report," GE-NE-208-08-0393, September 1993 (Proprietary).

15.2-6 Deleted.

15.2-7 GE Nuclear Energy, "WNP-2 Power Uprate Project NSSS Engineering Report," GE-NE-208-17-0993, Revision 1, December 1994 (Proprietary).

15.2-8 AREVA NP, Inc., "Columbia Generating Station MSIV Closure Level Setpoint Change - Loss of Feedwater Flow Transient Analysis," 51-9084418-000, July 2008.

15.2-9 GE Hitachi Nuclear Energy, "License Amendment Request for Proposed Changes to Columbia Technical Specifications: Changing Group 1 Isolation Valves' Low Reactor Water Level Isolation Signal from the Current Level 2 to Level 1," 0000-0081-6730-R1, July 2008.

Table 15.2-1

Sequence of Events for Figure 15.2-1

Pressure Regulator Failure - Closed
104.1% Uprated Power - 106% Flow

Time (sec)	Event
0	Failure of the pressure regulator causes closure of the turbine control valves.
1.0	Scram signal initiated at high neutron flux.
2.0	Recirculation pump motor circuit breakers open causing decrease in core flow to the natural circulation.
2.56	Group 3 relief valves actuated.
2.67	Group 4 relief valves actuated.
2.77	Group 5 relief valves actuated.
2.84	Turbine control valves close.

Table 15.2-2

Sequence of Events for Figure 15.2-2.1

Generator Load Rejection with Bypass On
Original Rated Power

Time (sec)	Event
(-)0.015[a]	Turbine generator detection of loss of electrical load.
0	Turbine generator overspeed protection control (OPC) devices trip to initiate turbine control (governor) valve fast closure.
0	Turbine generator OPC trip initiates main turbine bypass system operation.
0	Fast control valve closure initiates scram trip.
0	Fast control valve closure initiates an RPT.
0.07	Turbine control valves closed.
0.11	Turbine bypass valves start to open.
0.19	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation.
1.70	Group 1 relief valves actuated.
1.86	Group 2 relief valves actuated.
2.01	Group 3 relief valves actuated.
2.27	Group 4 relief valves actuated.

[a] Approximately.

Table 15.2-3

Sequence of Events for Figure 15.2-2.2

Generator Load Rejection with Bypass Failure
100% Power 106% Flow

Time (sec)	Event
(-)0.003[a]	Turbine generator detection of loss of electrical load.
0	Turbine generator OPC devices trip to initiate turbine control (governor) valve fast closure.
0	Turbine bypass valves fail to operate.
0.03	Time of scram trip.
0.15	Turbine control valves fully closed.
0.20	Time of RPT trip.
0.28	Start of control blade motion.
(b)	Group 1 MSRVs actuated (safety function).
(b)	Group 2 MSRVs actuated (safety function).
2.83	Group 3 MSRVs actuated (safety function).
3.17	Group 4 MSRVs actuated (safety function).

[a] Approximately.

[b] Not used - out of service for this analysis.

Table 15.2-4

Sequence of Events for Figure 15.2-3

Turbine Trip, Trip Scram - Bypass and RPT On
Original Rated Power

Time (sec)	Event
0	Turbine trip initiates closure of main stop (throttle) valves.
0	Turbine trip initiates bypass operation.
0.01	Main turbine stop valves reach 90% open position and initiate reactor scram trip.
0.01	Main turbine stop valves reach 90% open position and initiate an RPT.
0.10	Turbine stop valves closed.
0.10	Turbine bypass valves start to open to regulate pressure.
0.20	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation.
1.63	Group 1 relief valves actuated.
1.78	Group 2 relief valves actuated.
1.94	Group 3 relief valves actuated.
2.14	Group 4 relief valves actuated.
2.50	Group 5 relief valves actuated.
4.67	Feedwater turbines trip on L8 high water level.
5.1[a]	Group 5 relief valves start to close.
7.2[a]	All relief groups closed.
31.0	Turbine bypass starts to close.
32.3[a]	Turbine bypass closed.
39.7	Turbine bypass reopens on pressure increase at turbine inlet.
45.3	Main steam line isolation[b], HPCS system initiation, and RCIC system initiation on low level (L2) (not included in simulation).
50+	Group 1 relief valves cycle open and close on pressure.

[a] Estimated.

[b] The analysis has not been updated for the change in MSIV isolation setpoint from Level 2 to Level 1 because the analysis is bounding and conclusions of the analysis are not affected (Reference 15.2-9).

Table 15.2-5

Sequence of Events for Figure 15.2-4

Turbine Trip with Bypass Failure
at 100% Power/106% Core Flow

Time (sec)	Event
0	Turbine trip initiates closure of main stop (throttle) valves.
0	Turbine bypass valves fail to operate.
0.01	Main turbine stop valves reach 90% open position and initiate reactor scram trip.
0.10	Turbine stop valves closed.
0.20	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation.
(a)	Group 1 relief valves actuated.
(a)	Group 2 relief valves actuated.
2.83	Group 3 relief valves actuated.
3.18	Group 4 relief valves actuated.

[a] Not used - out of service for this analysis.

Table 15.2-6

Sequence of Events for Figure 15.2-5

Main Steam Line Isolation Valve Closure
106.2% Uprated Power, 100% Rated Flow

Time (sec)	Event
(-) 0.003 (approximately)	Turbine-Generator detection of loss of electrical load.
0	Initiate closure of all main steam line valves.
0.45	MSIVs reach 85% opening initiating a position valve scram.
2.0	Loss of feedwater begins as turbine loses steam supply.
2.53	Both recirculation pumps trip due to high pressure.
3.0	All MSIVs closed.
3.08	Groups 3 relief valves actuated.
3.16	Groups 4 relief valves actuated.
3.24	Groups 5 relief valves actuated.
11 (approx.)	All pressure relieve valves closed.
18.23	Groups 3 relief valves begin to cycle.

Table 15.2-7

Sequence of Events for Figure 15.2-6

Loss of Condenser Vacuum
104.1% Uprated Power, 100% Rated Flow

Time (sec)	Event
(-)5.0 (approximately)	Initiate simulated loss of condenser vacuum at 2 in. of Hg per second.
0.00	Low condenser vacuum main turbine trip and feedwater turbine trips initiated.
0.00	Main turbine trip initiates turbine bypass operation.
0.01	Main turbine stop valves reach 90% open position and initiate reactor scram.
0.19	Both recirculation pumps trip.
2.15	Group 3 relief valves actuated.
2.28	Group 4 relief valves actuated.
2.42	Group 5 relief valves actuated.
2.91	Feedwater recirculation valve is tripped.
5.00	Low condenser vacuum initiates turbine bypass valve closure and MSIV closure.
5.6 (approx.)	All relief valves closed.
6.0 (approx.)	Main steam isolation valves closed.
7.90	Group 3 relief valves reactuated.
8.35	Group 4 relief valves reactuated.
24.01	Group 5 relief valves reactuated.

Table 15.2-8

Trip Signals Associated with Loss-of-Condenser Vacuum

Vacuum[a]	Protective Action Initiated
27 to 30	Normal vacuum range.
20 to 23	Main turbine trip and feedwater turbine trip (stop valve closures).
7 to 10	Main steam line isolation valve closure and bypass valve closure.

[a] Inches of Hg.

Table 15.2-9

Sequence of Events for Figure 15.2-7

Loss of Auxiliary Power Transformers
106.2% Uprated Power, 100% Rated Flow

Time (sec)	Event
0.00	Loss of auxiliary power transformers occurs.
0.00	Recirculation system pump motors are tripped.
2.00	Reactor scram due to loss of power to the scram solenoid.
2.00	Main steam line isolation valves begin to close due to loss of power to MSIV solenoids.
4.00	Feedwater pumps are tripped due to MSIV closure.
5.98	Group 3 relief valves actuated.
6.15	Group 4 relief valves actuated.
6.34	Group 5 relief valves actuated.

Table 15.2-10

Sequence of Events for Figure 15.2-8

Loss of All Grid Connections
104.1% Uprated Power, 100% Rated Flow

Time (sec)	Event
0.00	Loss of grid causes turbine-generator to detect a loss of electrical load.
0.00	Turbine-generator PLU devices trip to initiate TCV fast closure and turbine bypass system operation.
0.00	Recirculation pumps trip.
0.00	Fast control valve closure initiates reactor scram.
2.00	Main steam line isolation is initiated due to loss of power to the solenoids.
2.12	Group 3 relief valves actuated.
2.25	Group 4 relief valves actuated.
2.37	Group 5 relief valves actuated.
4.00	Feedwater pump tripped due to MSIV closure.

Table 15.2-11

Sequence of Events for Figure 15.2-9

Loss of All Feedwater Flow
106.2% Uprated Power, 100% Rated Flow

Time (sec)	Event
0	Initiate trip of all feedwater pumps.
3.91	Recirculation runback initiated with narrow range sensed level less than L4 and feedwater pumps off.
7.38	Vessel water level (L3) trip initiates scram trip.
32.32	Vessel water level (L2) trip initiates main steam line isolation[a], recirculation pump trip and HPCS/RCIC system operation (not simulated).
32.51	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation.

[a] The analysis has not been updated for the change in MSIV isolation setpoint from Level 2 to Level 1 because the analysis is bounding and conclusions of the analysis are not affected (Reference 15.2-8).

Table 15.2-12

Sequence of Events for Failure of Residual Heat Removal Shutdown Cooling

Original Rated Power

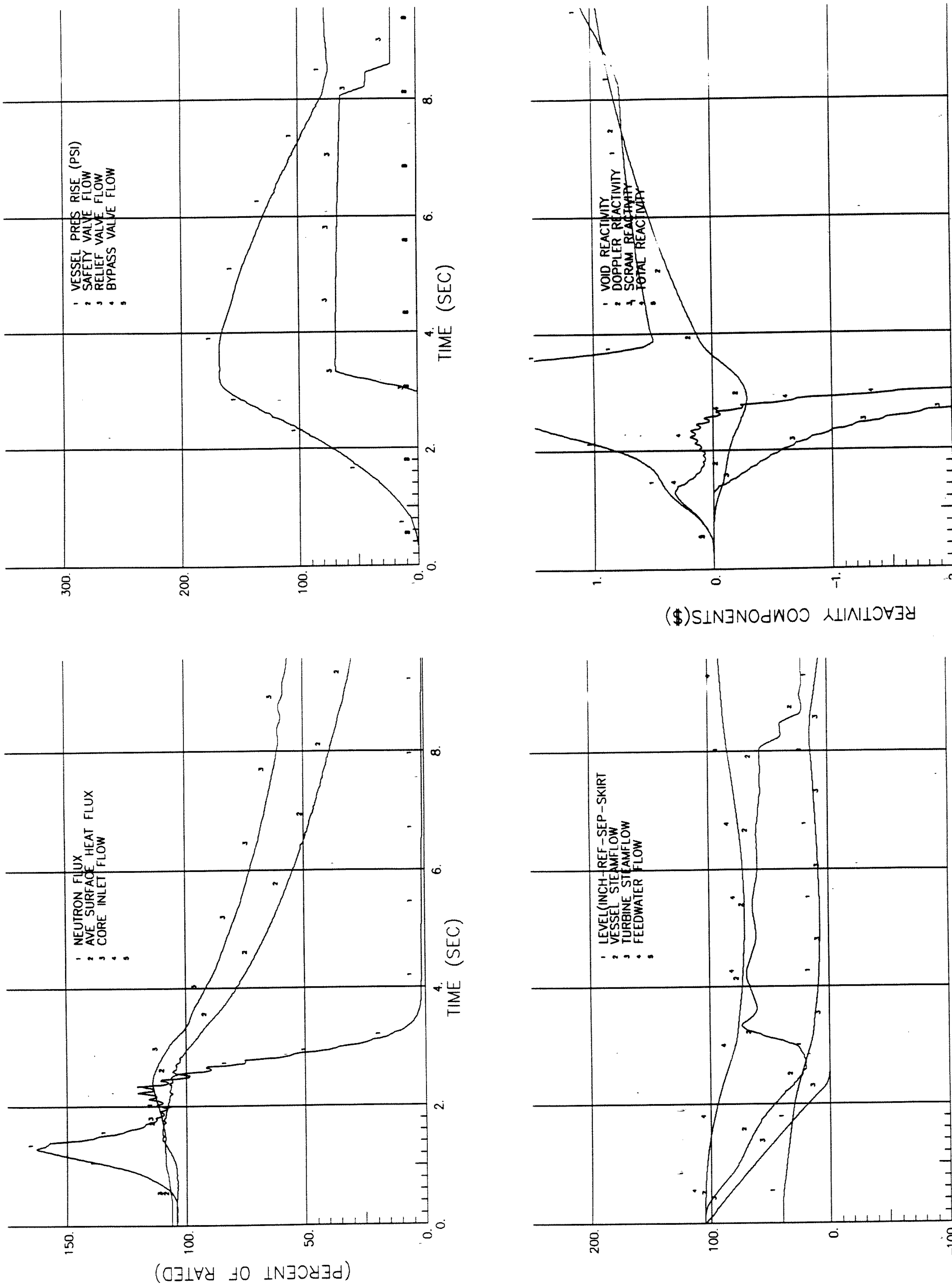
Time[a]	Event
0	Reactor is operating at 105% NBR steam flow when LOP transient occurs initiating plant shutdown.
0	Concurrently loss of division power occurs (i.e., loss of one diesel generator).
0	Initial suppression pool temperature at 95°F.
10 minutes	Suppression pool cooling initiated to prevent overheating from SRV actuation.
10 minutes	Controlled blowdown initiated.
2-3 hr	Blowdown to 100 psi completed.
2-3 hr	Personnel are sent in to open RHR shutdown cooling suction valve and fail.
2.5-3.5 hr	Complete blowdown to suppression pool by opening SRVs.
2.5-3.5 hr	Redirect RHR pump discharge from pool to vessel by means of the LPCI line. Alternate cooling path now established.
7 hr	Maximum suppression pool temperature attained.

[a] Approximately.

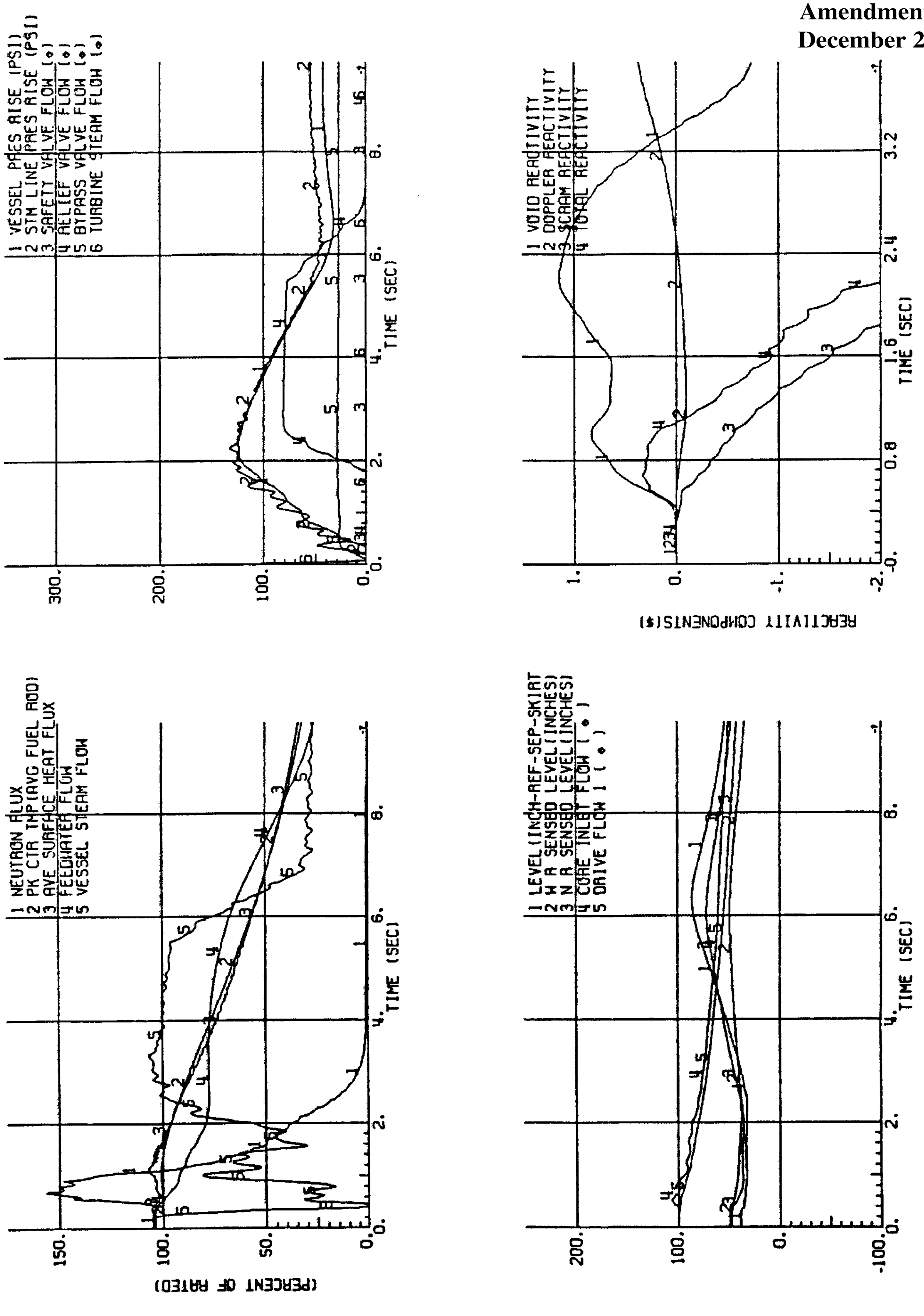
Table 15.2-13

Evaluation of Failure of Residual Heat Removal
Shutdown Cooling

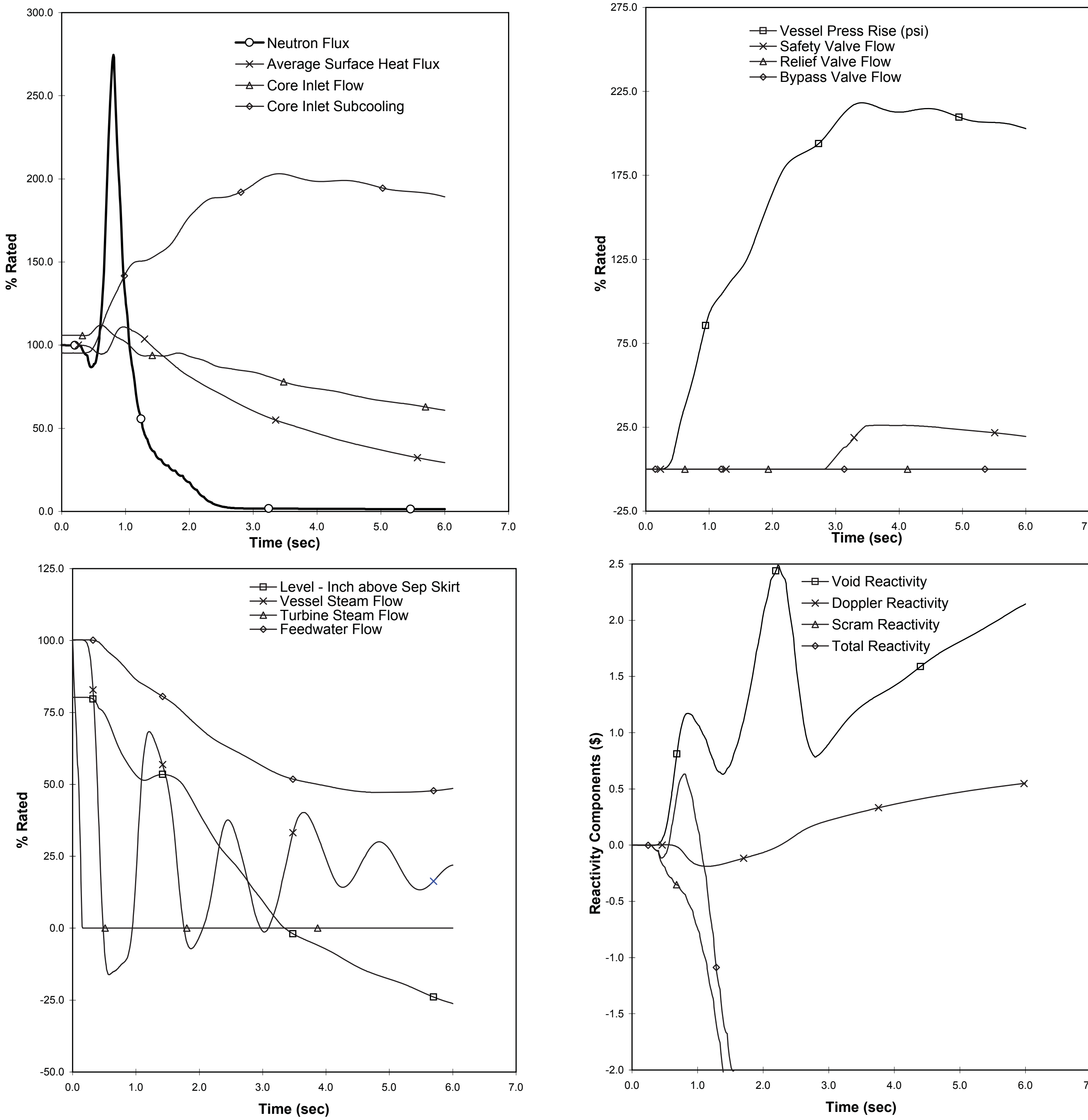
Parameter	Value
Initial power corresponding	105% original rated steam flow
To suppression pool mass (lbm)	8.52 E6
Residual heat removal (KHX value) (Btu/sec/°F)	289
Initial vessel condition	
Pressure (psia)	1055
Temperature (°F)	550.7
Initial primary fluid inventory (lbm)	7.016 E5
Initial pool temperature (°F)	95
Service water temperature (°F)	87
Vessel heat capacity (Btu/lbm/°F)	0.123
High-pressure core spray on-off water level (ft)	
HPCS ON	40.8
HPCS OFF	47
High-pressure core spray flow rate (lbm/sec)	868
Low-pressure coolant injection flow rate (lbm/sec)	982



Columbia Generating Station **Final Safety Analysis Report**	**Pressure Regulator Failure - Down Scale Failure at 104.1% Uprated Power, 106% Flow**
	Draw. No. 060108.06 | Rev. | Figure 15.2-1



Columbia Generating Station Final Safety Analysis Report	Generator Load Rejection with Bypass On - Original Rated Power
	Draw. No. 020361.68 · Rev. · Figure 15.2-2.1



Columbia Generating Station Final Safety Analysis Report	**Generator Load Rejection with BP Failure**
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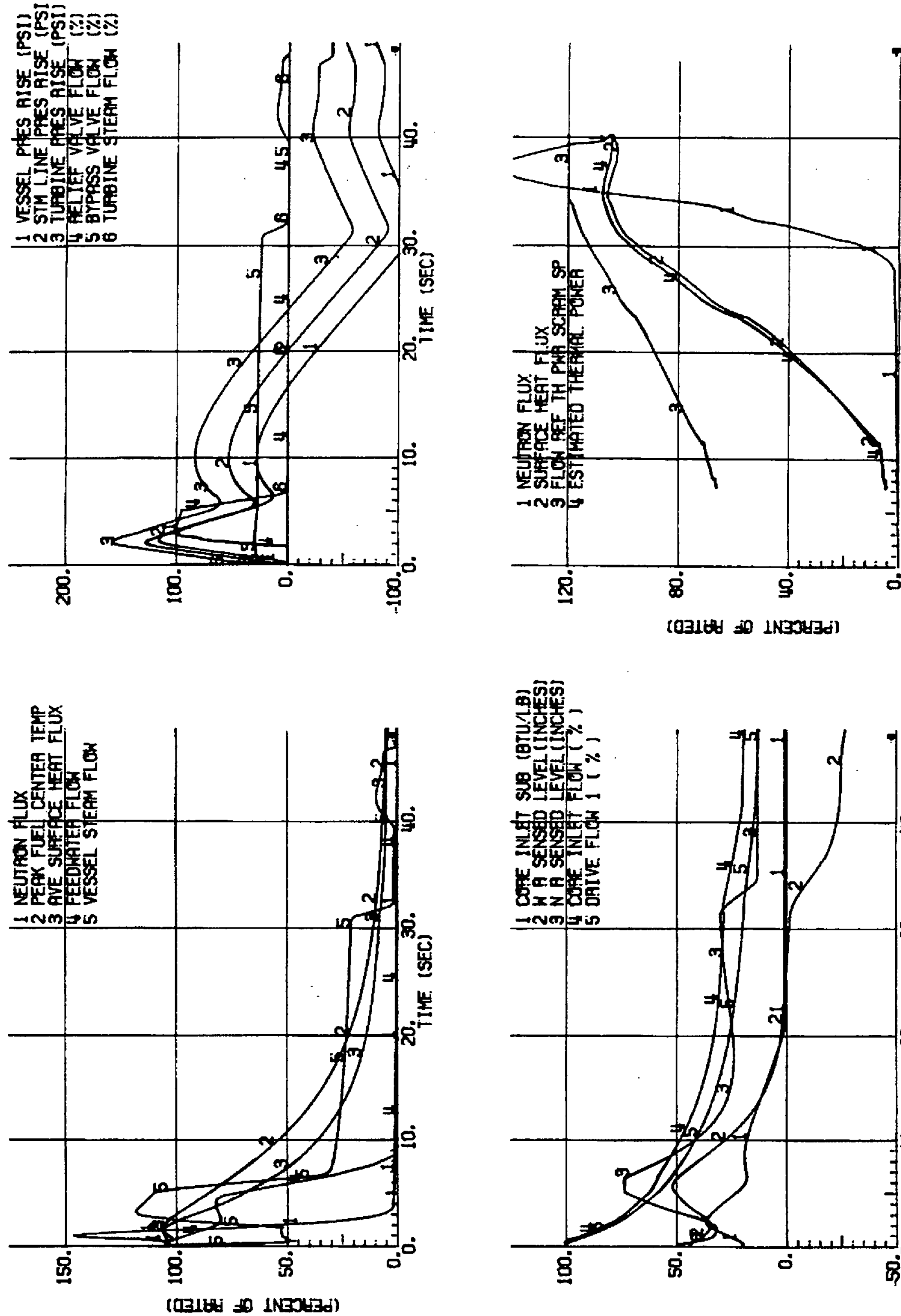
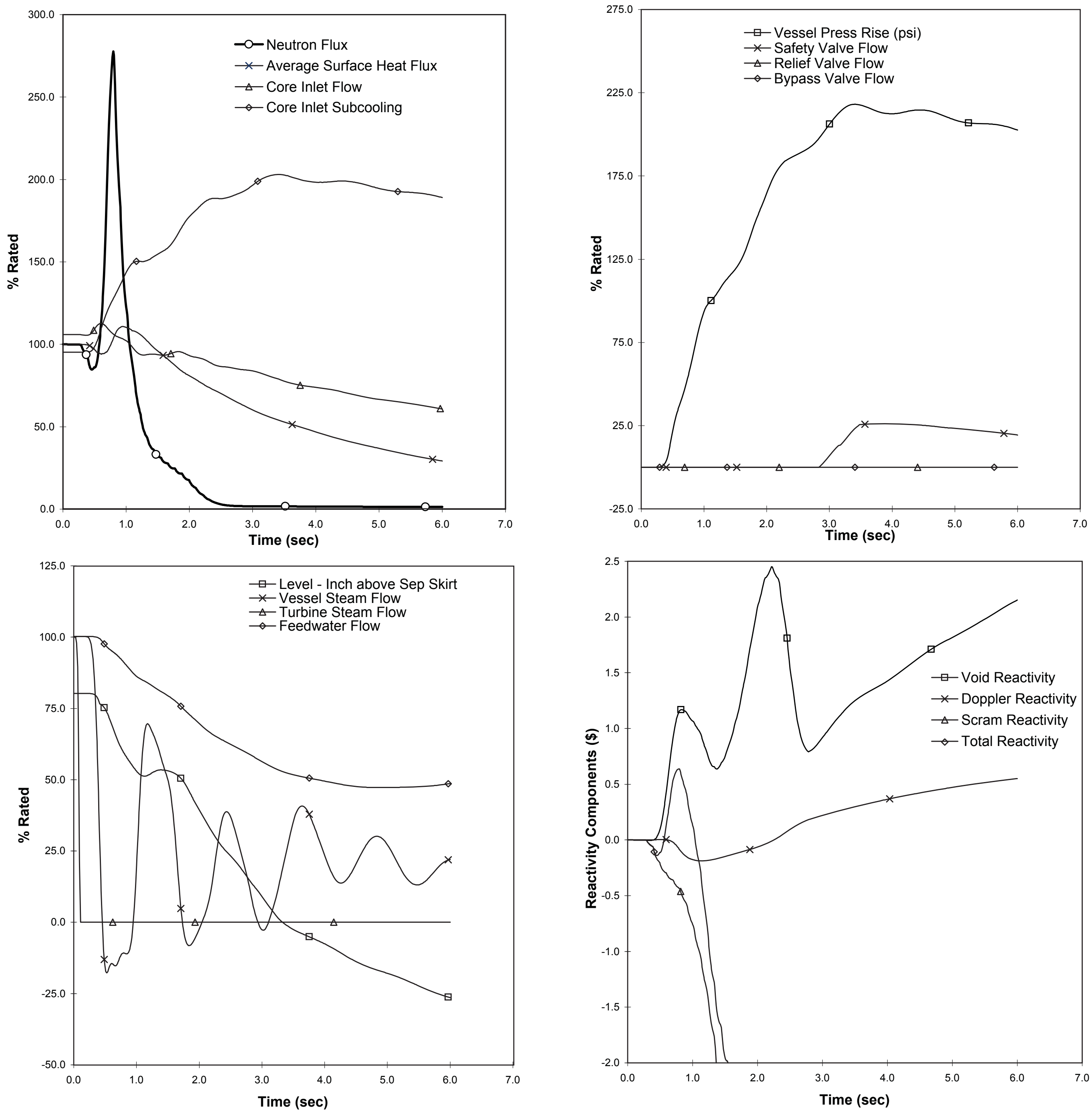
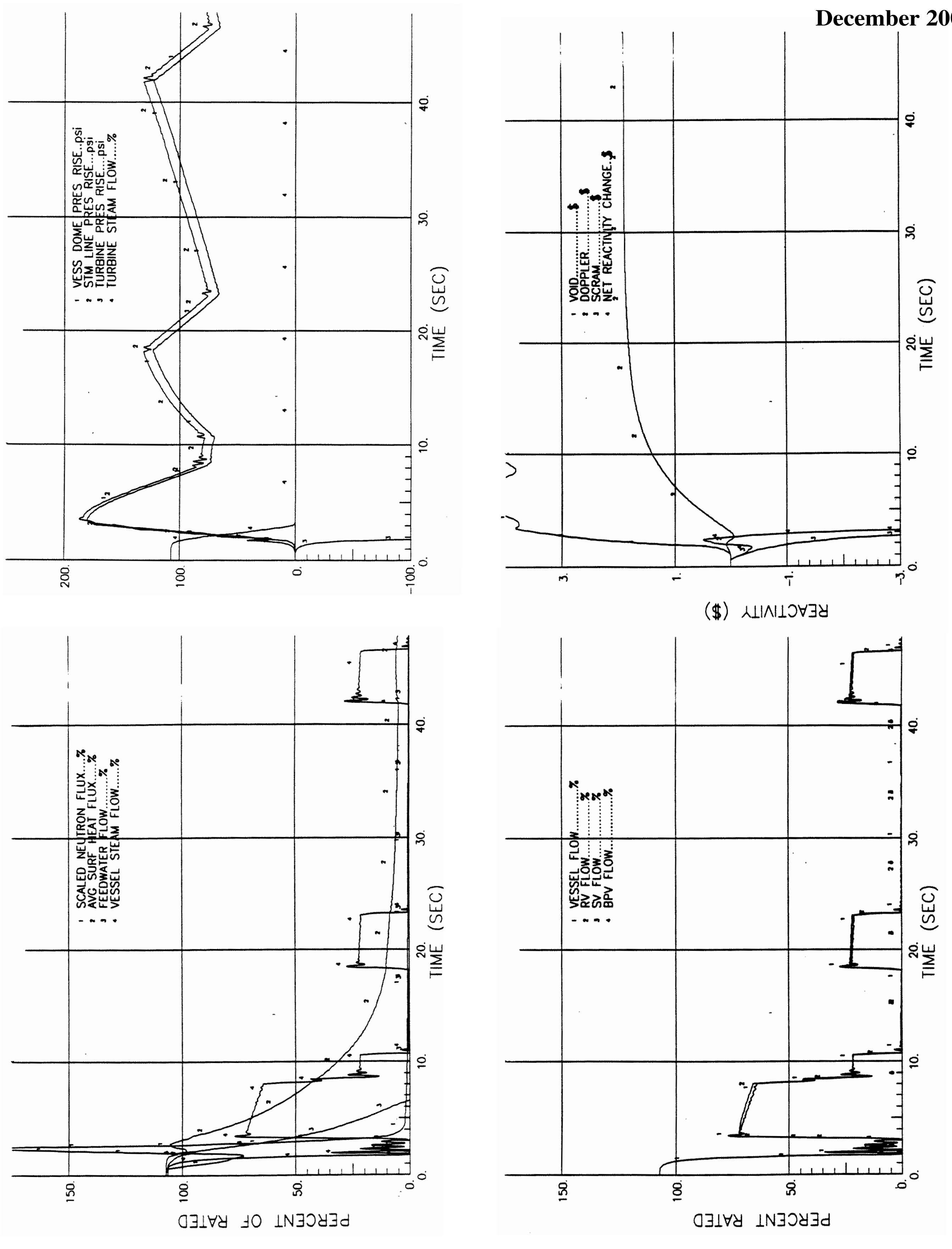


Columbia Generating Station Final Safety Analysis Report	Turbine Trip, Trip Scram, Bypass and RPT - On		
	Draw. No. 020361.70	Rev.	Figure 15.2-3



Columbia Generating Station Final Safety Analysis Report	Turbine Trip with Bypass Failure
	Draw. No. 020361.71 · Rev. · Figure 15.2-4



Columbia Generating Station
Final Safety Analysis Report

Main Steam Line Isolation Valve Closure at 106.2% Uprated Power, 100% Rated Flow

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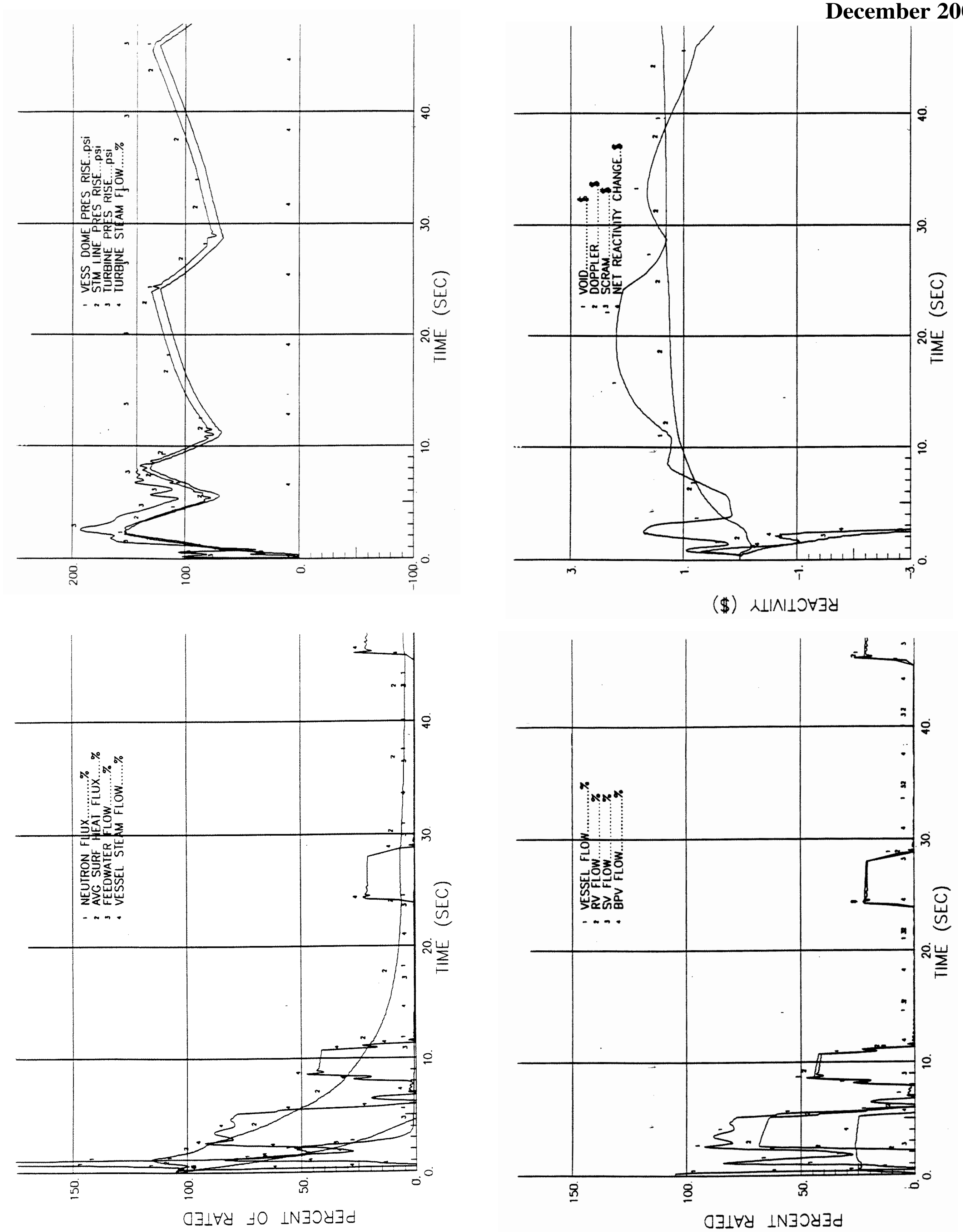
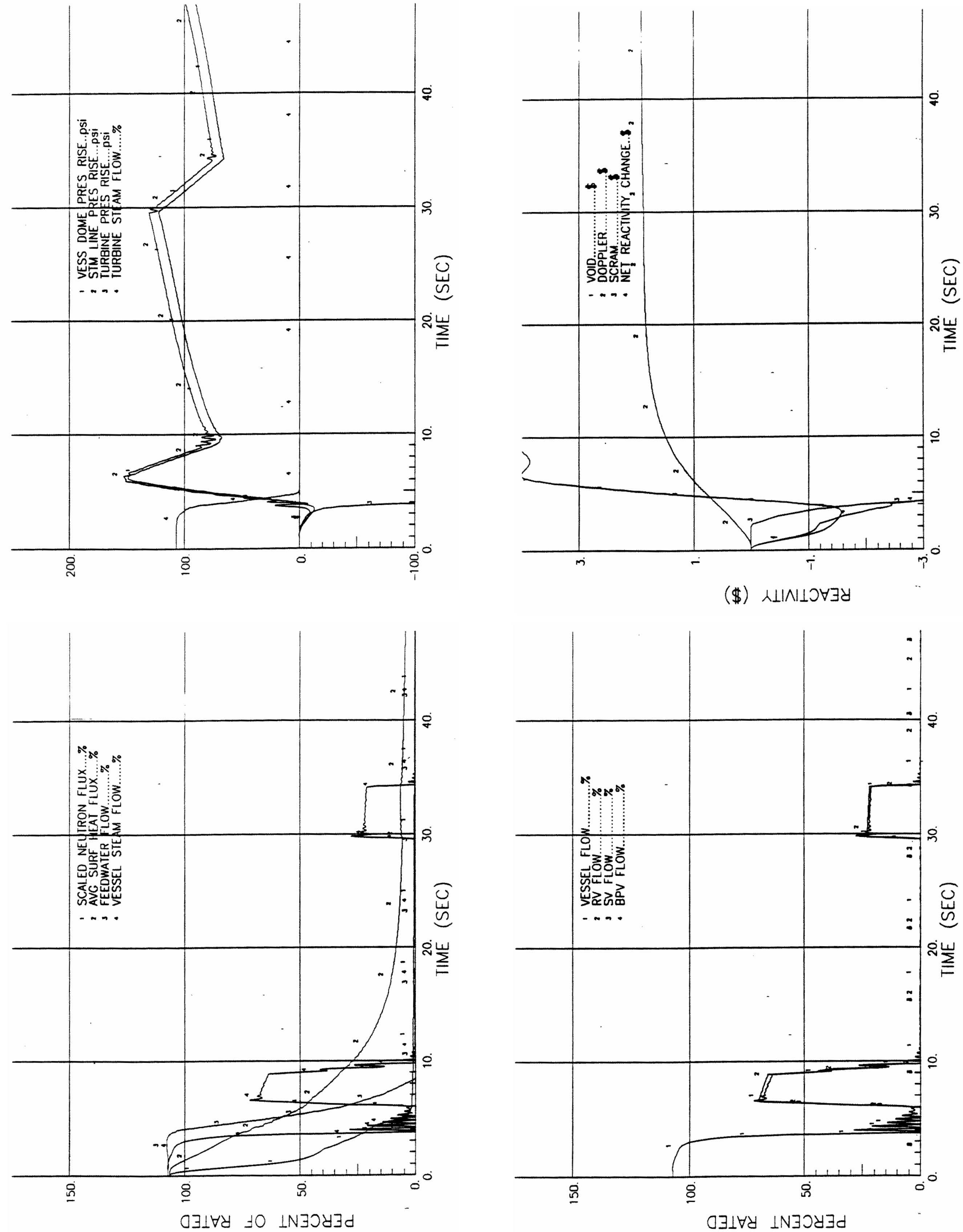


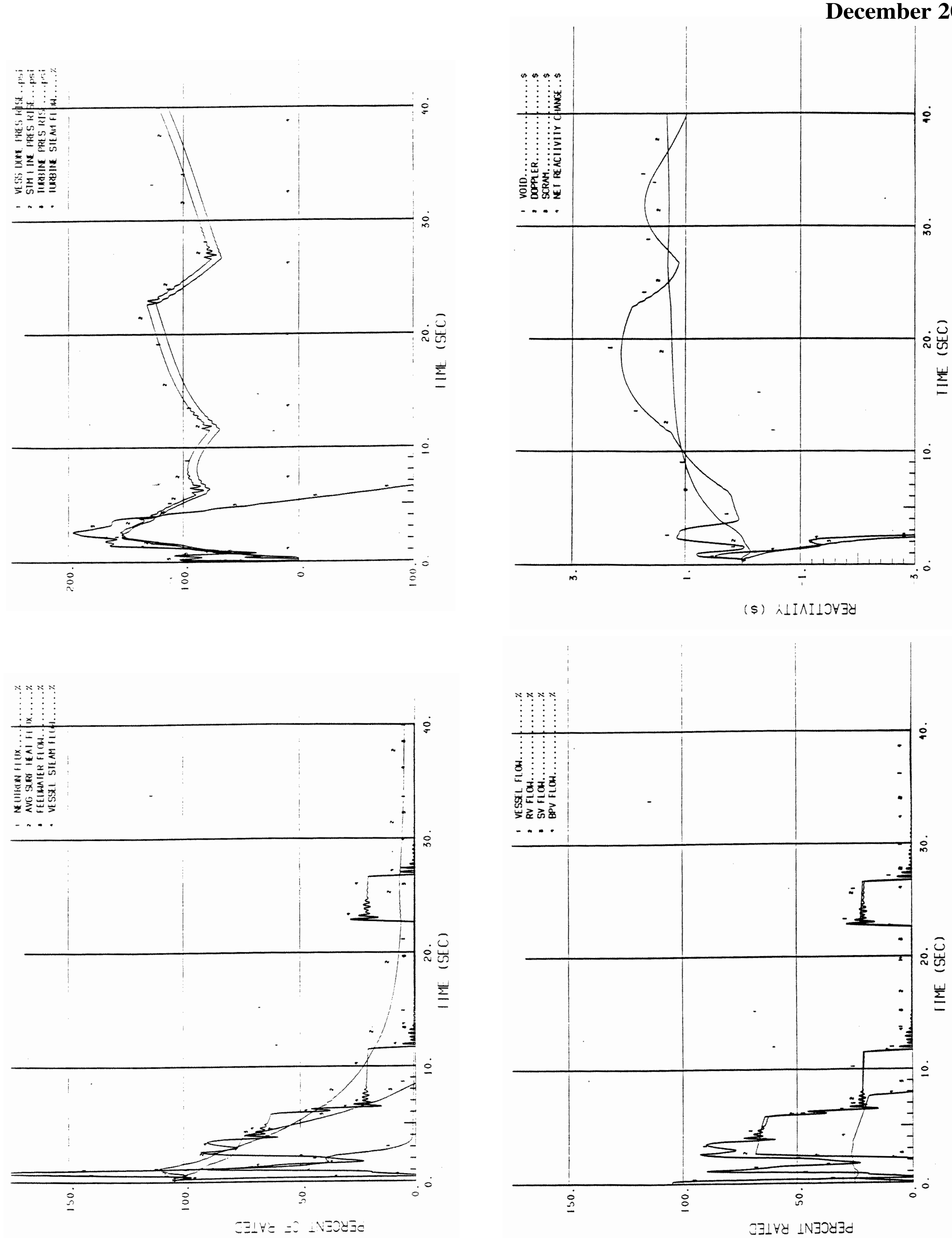
Columbia Generating Station Final Safety Analysis Report	Loss of Condenser Vacuum at 104.1% Uprated Power, 100% Rated Flow
	Draw. No. 020361.73 — Rev. — Figure 15.2-6



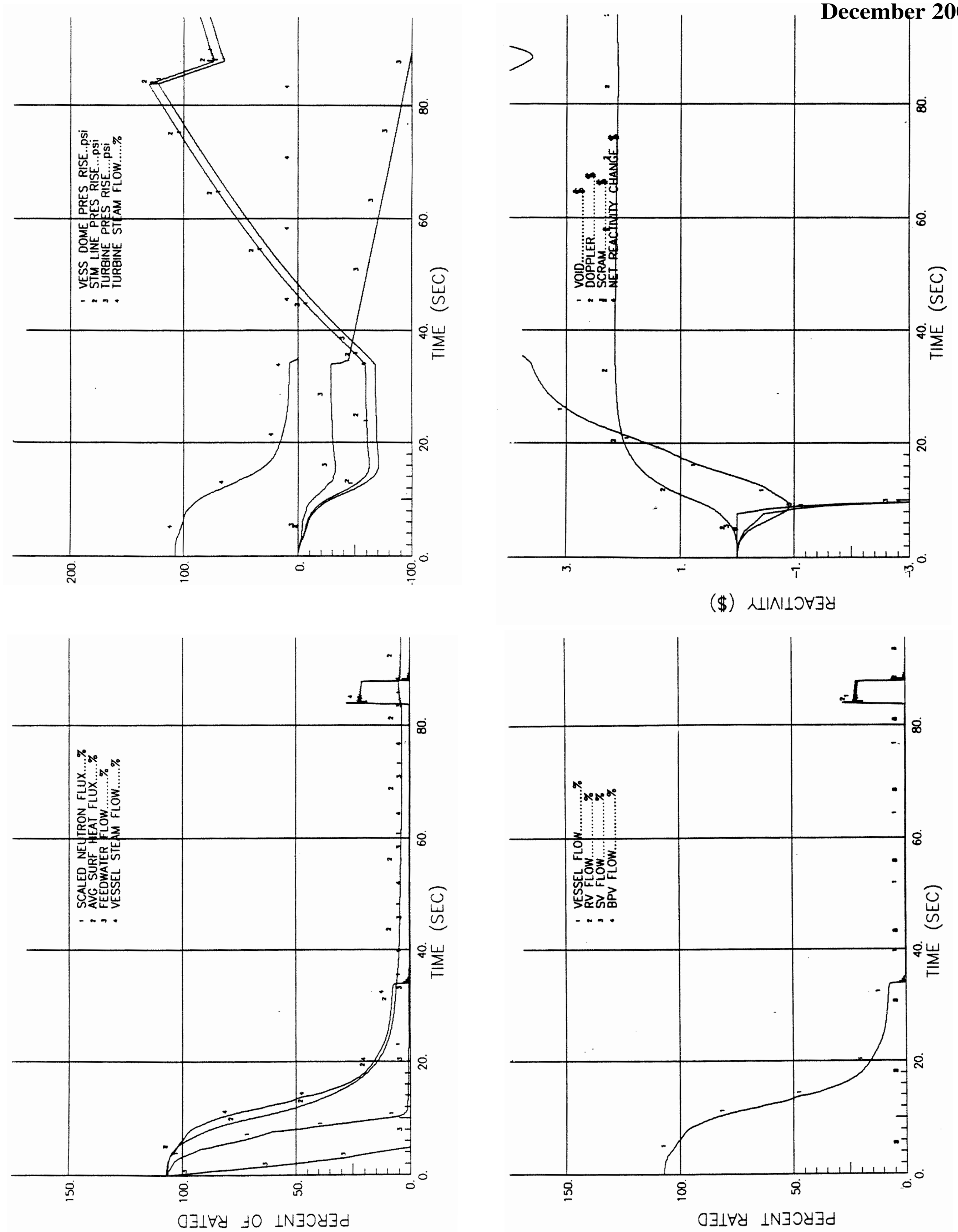
Columbia Generating Station
Final Safety Analysis Report

Loss of Auxiliary Power Transformers - at 106.2% Uprated Power, 100% Rated Flow

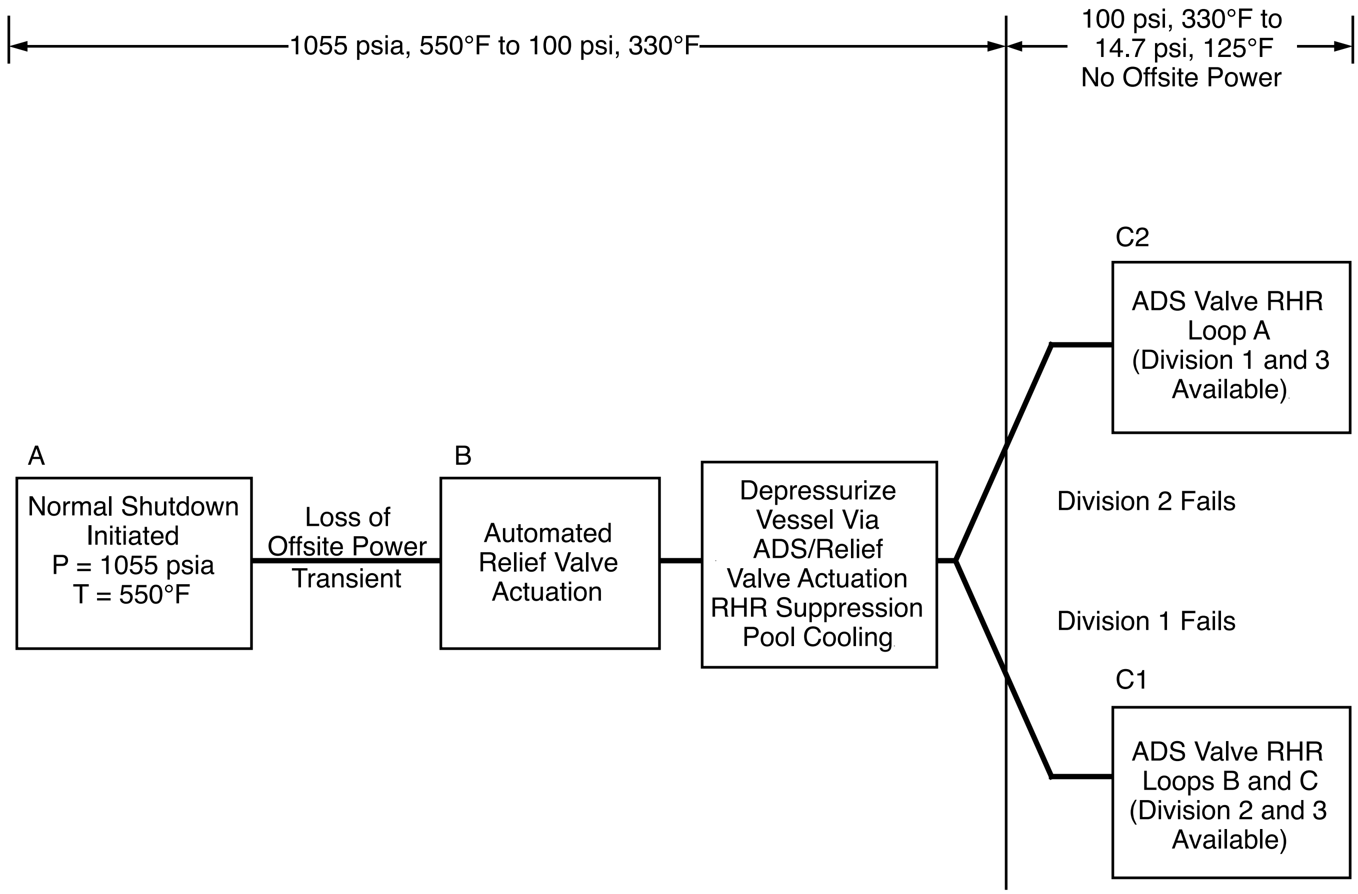
Draw. No. 020361.74	Rev.	Figure 15.2-7



Columbia Generating Station Final Safety Analysis Report	Loss of All Grid Connections - 104.1% Uprated Power, 100% Rated Flow
	Draw. No. 020361.75 / Rev. / Figure 15.2-8



Columbia Generating Station Final Safety Analysis Report	Loss of All Feedwater Flow - 106.2% Uprated Power, 100% Rated Flow		
	Draw. No. 020361.76	Rev.	Figure 15.2-9



Automatic Depressurization System/Residual Heat Removal Cooling Loops

NOTES

ACTIVITY A
Initial pressure =1055 psia
Initial temperature = 550°F
For purpose of this analysis, the following worst-case conditions are assumed to exist:

a. The reactor is assumed to be operating at 105% of original NBR steam flow,
b. A loss of power transient occurs,
c. A simultaneous loss of onsite power (Division 1 or Division 2), and
d. Operator unable to open one of the RHR shutdown cooling line suction valves.

ACTIVITY B
Initial system pressure =1055 psia
Initial system temperature = 550°F

Operator Actions
During approximately the first 30 minutes, reactor decay heat is passed to the suppression pool by the automatic operation of the reactor relief valves. Reactor water level will be returned to normal by the HPCS and RCIC systems automatic operation.

After approximately 10 minutes, the operator initiates depressurization of the reactor vessel to control vessel pressure. Controlled depressurization procedure consists of controlling vessel pressure and water level by using the SRV or HPCS and/or RCIC systems. After approximately 15 minutes, it is assumed one RHR heat exchanger is placed in the suppression pool cooling mode to remove decay heat. At this time, the suppression pool will be 121°F.

When the reactor pressure approaches 100 psig, the operator would normally prepare for operation of the RHR system in the shutdown cooling mode. At this time (121 minutes), the suppression pool will be 186°F.

ACTIVITY C1 (Division 1 fails, Division 2 available)
System pressure =100 psig
System temperature = 330°F

Operation Actions

The operator establishes a closed cooling path as follows:

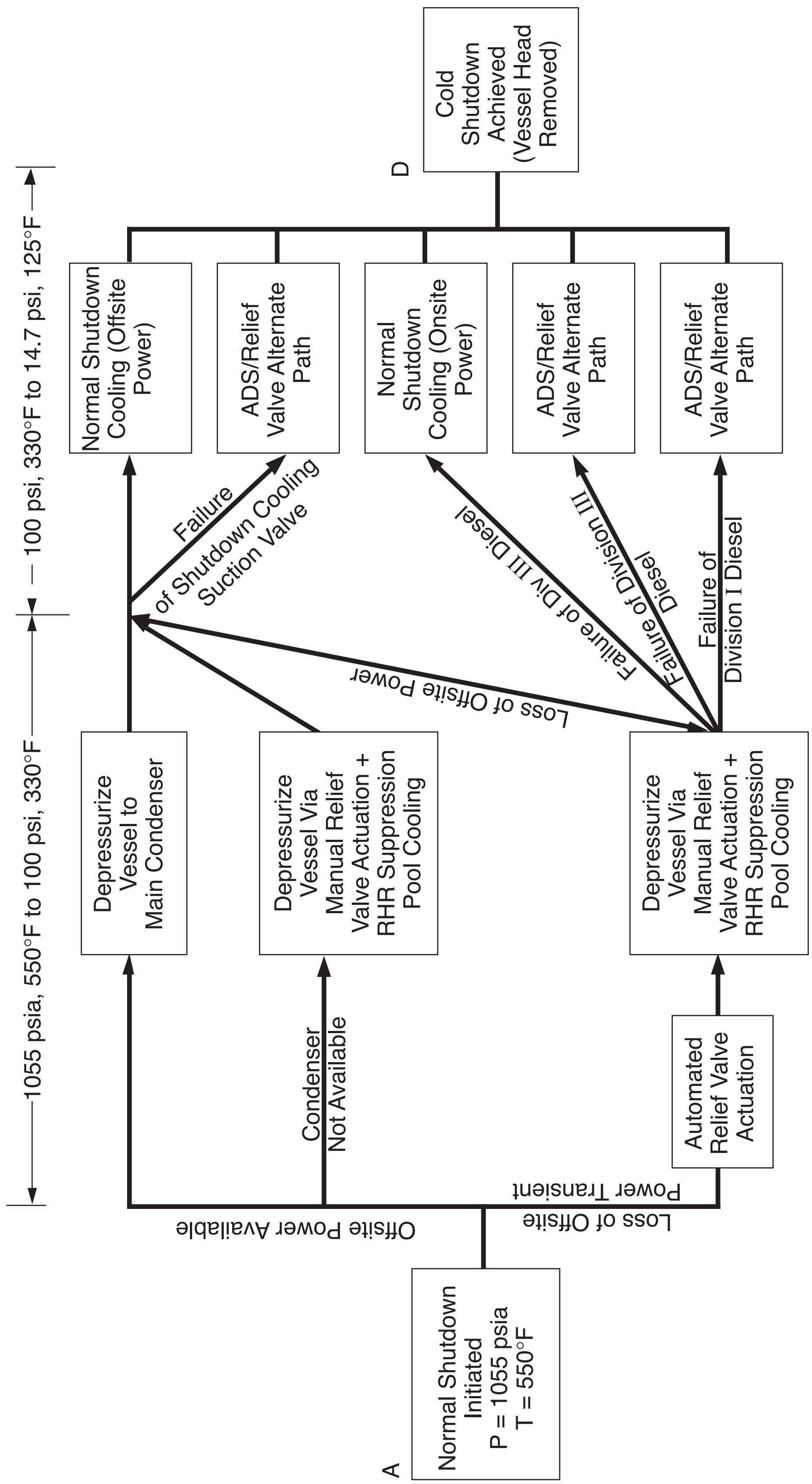
a. A minimum of two ADS valves (dc Division 2) are powered open.
b. Either of the following cooling paths are established:
    1. Using RHR loop B, water from the suppression pool is pumped through the RHR heat exchanger (where a portion of the decay heat is removed) into the reactor vessel. The cooled suppression pool water flows through the vessel (picking up a portion of the decay heat) out the ADS valves and back to the suppression pool. This alternate cooling path is shown in Figure 15.2-12.
    2. Using RHR loops B and C together, water is taken from the suppression pool and pumped directly into the reactor vessel. The water passes through the vessel (picking up decay heat) and out the ADS valves returning to the suppression pool as shown in Figure 15.2-13. Suppression pool water is then cooled by operation of RHR loop B in the pool cooling mode (see Figure 15.2-14). In this alternate cooling path, RHR loop C is used for injection and RHR loop B for cooling. Cold shutdown is achieved approximately 36 hr after the transient occurs.

ACTIVITY C2 (Division 2 fails, Division 1 available) (Figure 15.2-15)
System pressure =100 psig
System temperature = 330°F

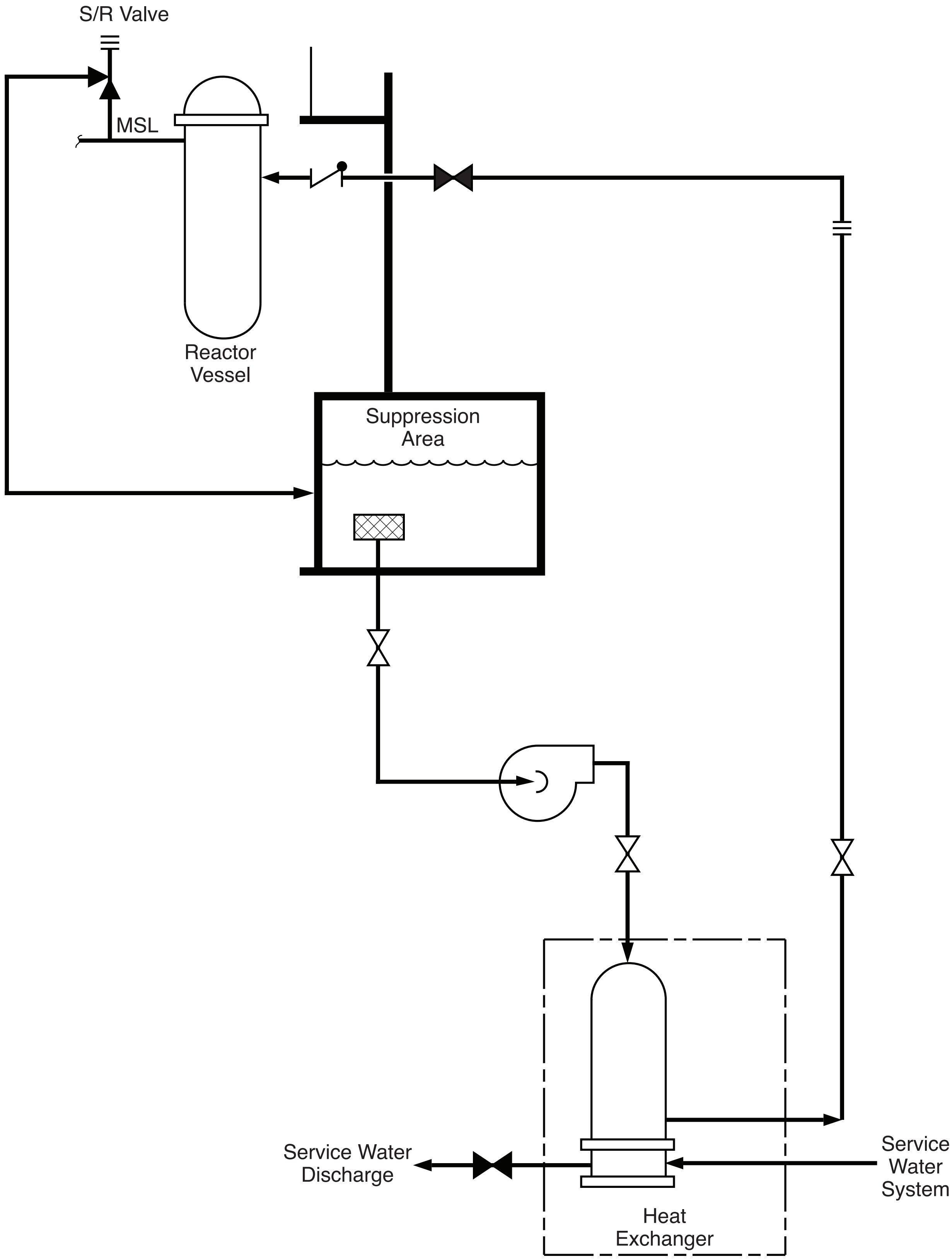
Operator Actions

The operator establishes a closed cooling path as follows:

a. A minimum of two ADS valves (dc Division 1) are powered open, and
b. Using RHR loop A instead of loop B, an alternate cooling path is established as shown in Activity C1. Cold shutdown is reached in approximately 15 hr.



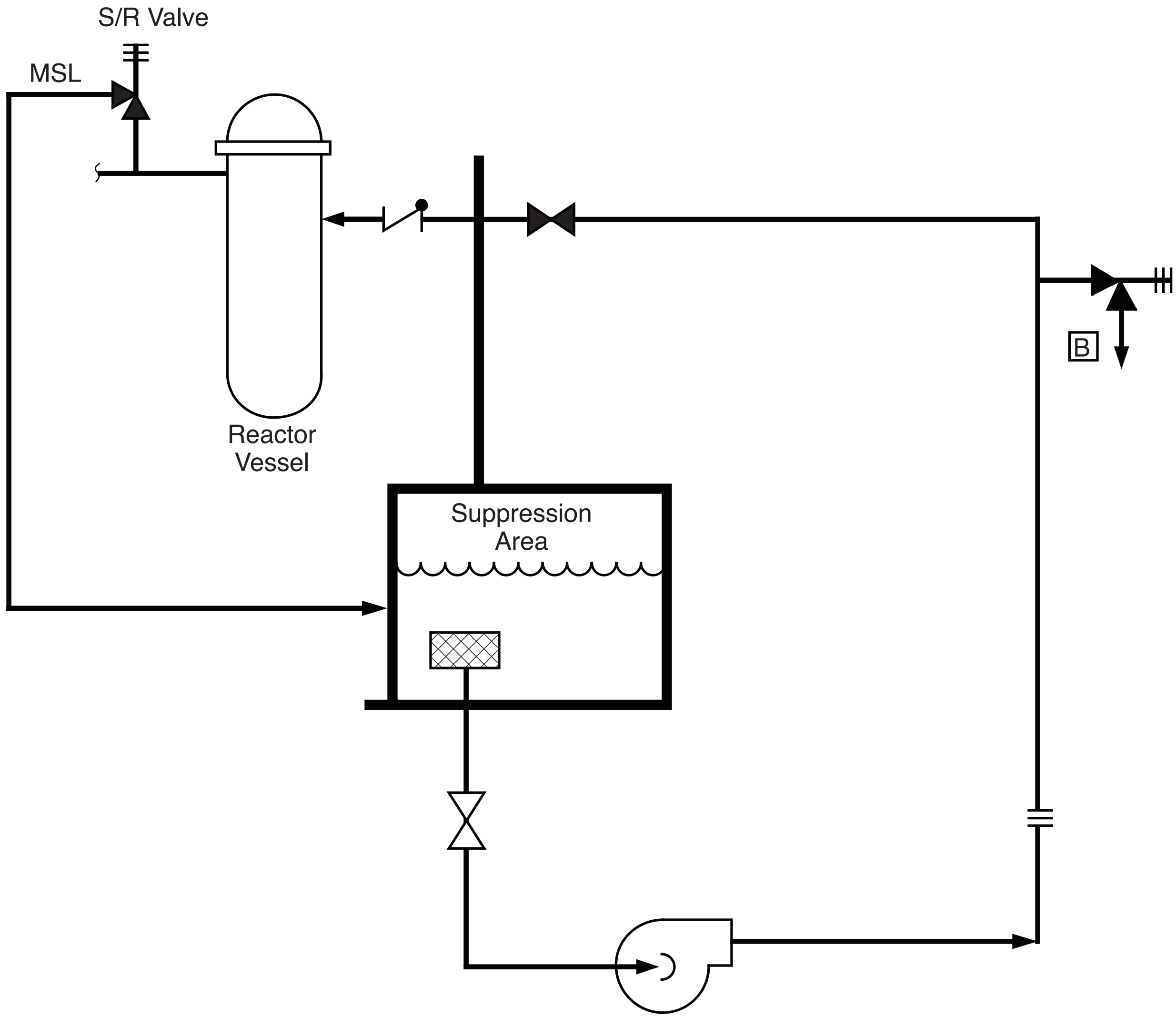
Columbia Generating Station Final Safety Analysis Report	**Summary of Paths Available to Achieve Cold Shutdown**
	Draw. No. **900547.64** Rev. Figure **15.2-11**



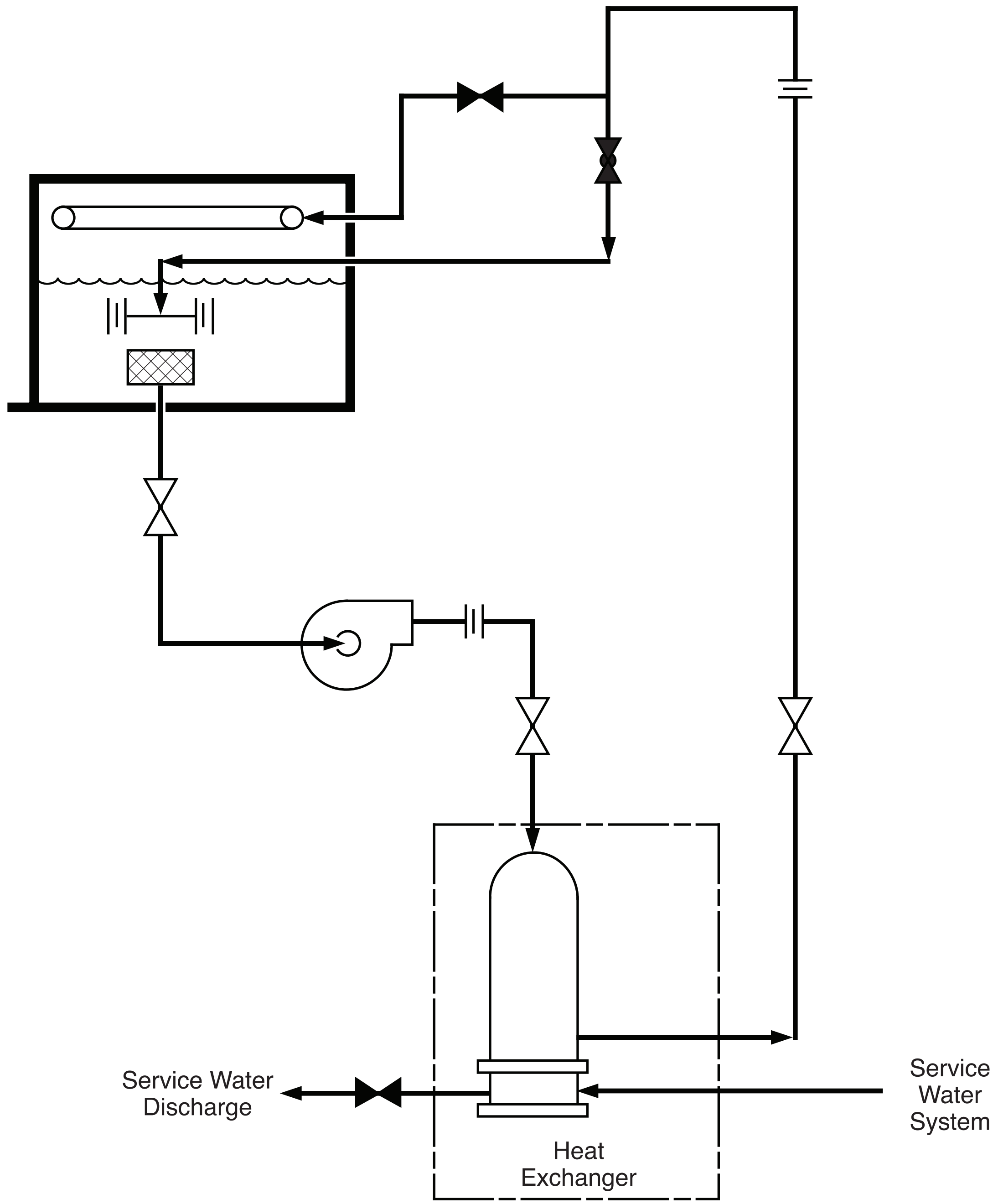
Columbia Generating Station
Final Safety Analysis Report

Activity C1 Alternate Shutdown Cooling Path Utilizing Residual Heat Removal Loop B

Draw. No. 900547.65 | Rev. | Figure 15.2-12



Residual Heat Removal Loop C



Columbia Generating Station
Final Safety Analysis Report

Residual Heat Removal Loop A(B) (Suppression Pool Cooling/ Rated Pump Flow Test Mode)

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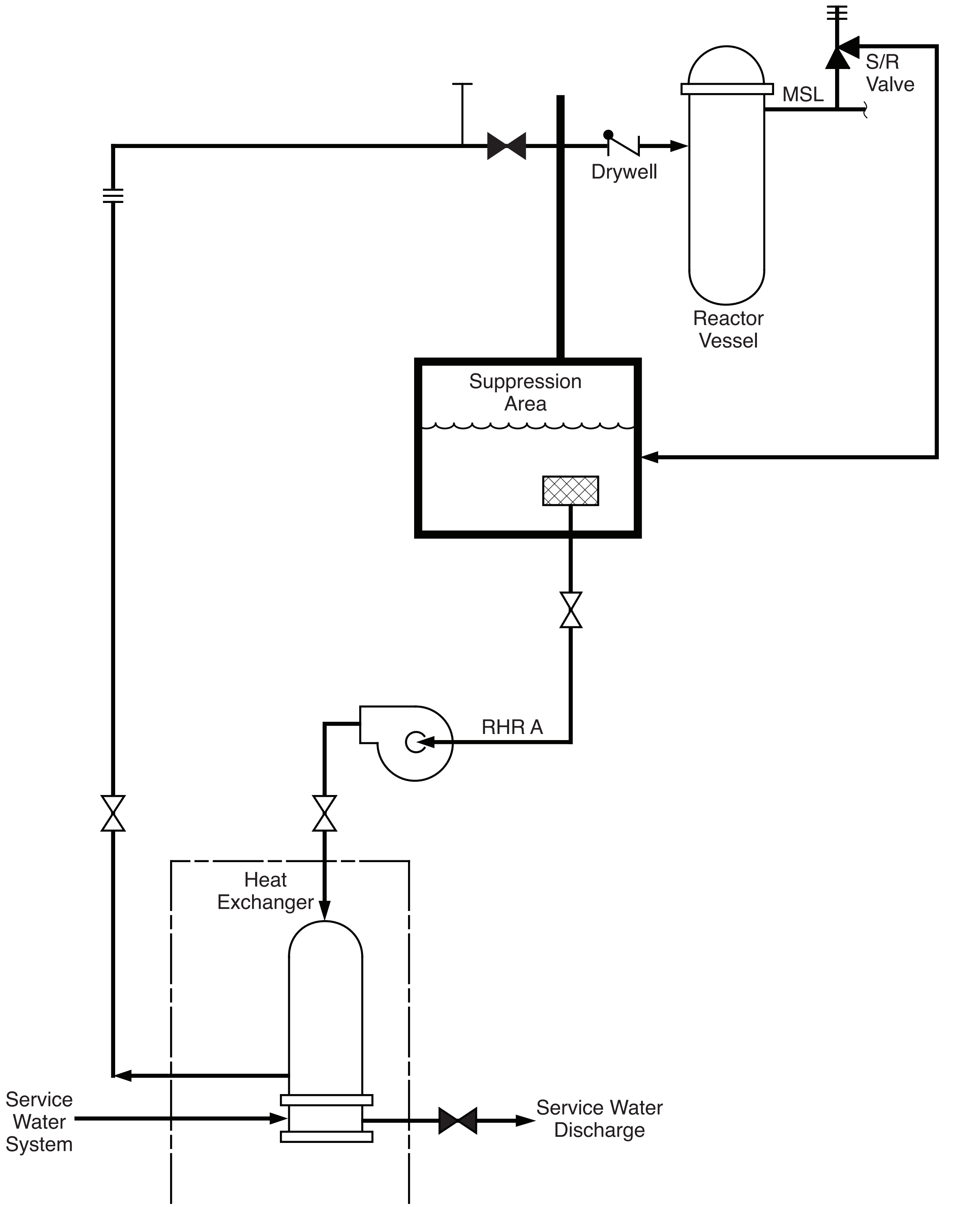
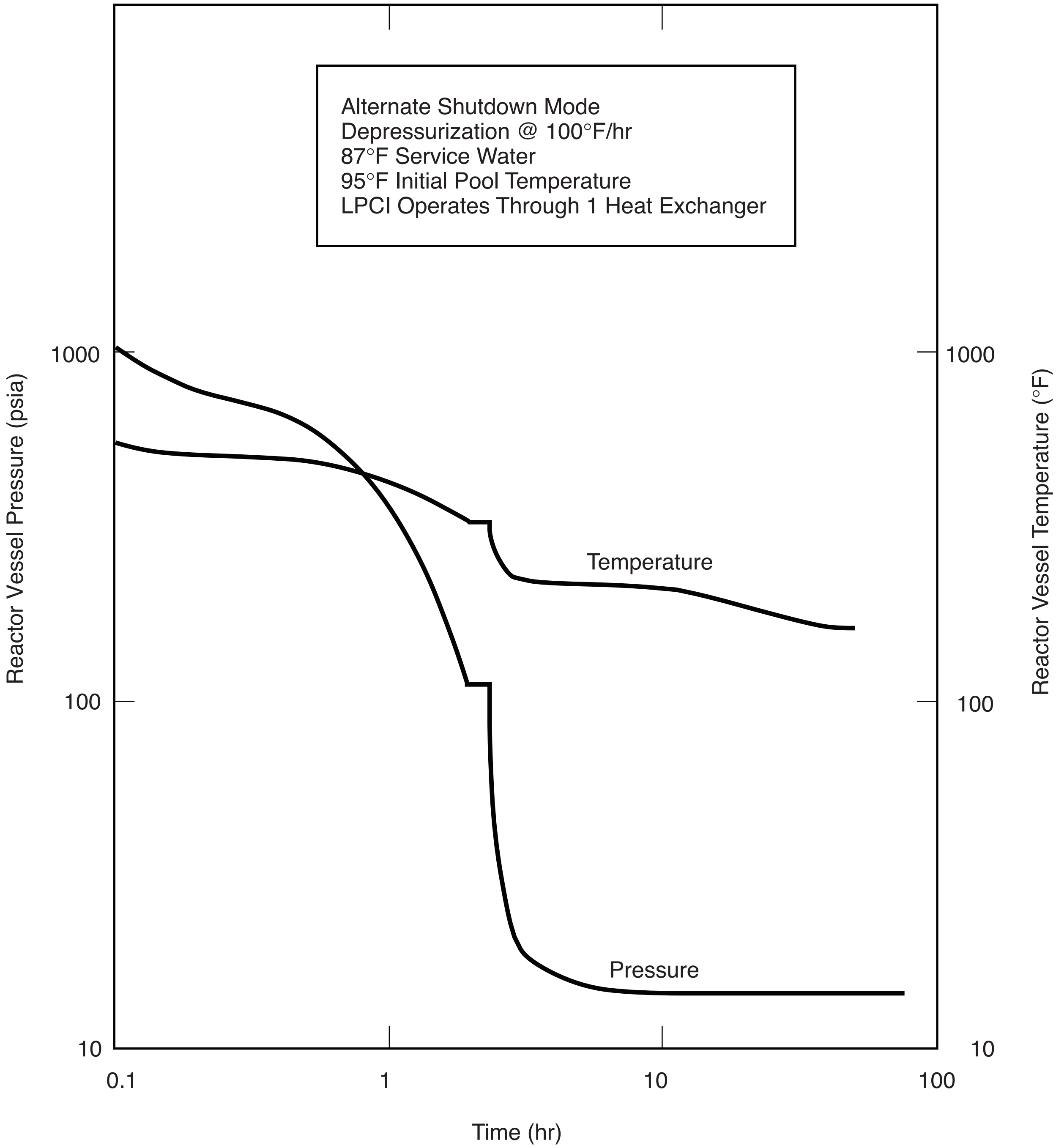
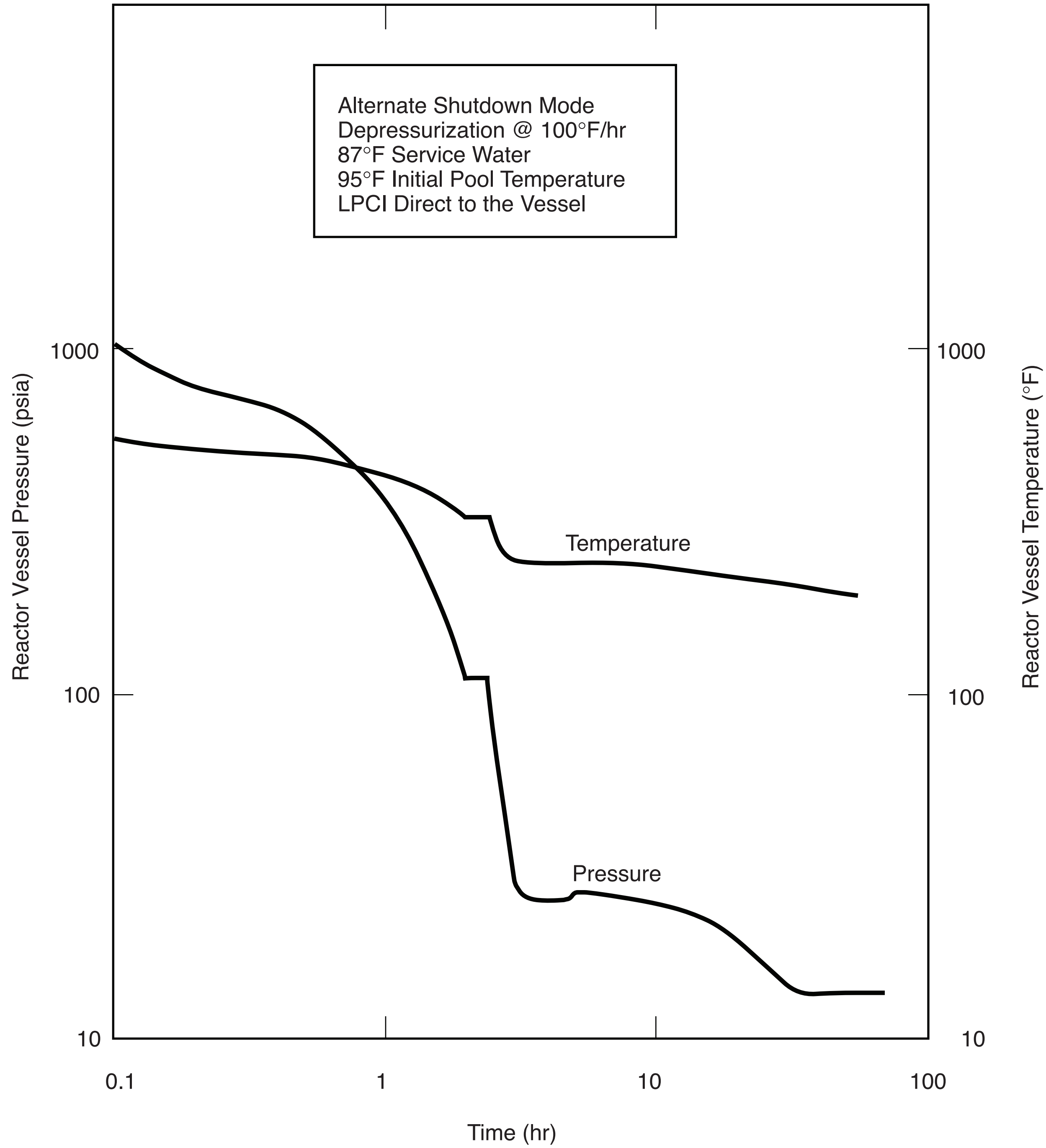


Columbia Generating Station Final Safety Analysis Report	Activity C2 Alternate Shutdown Cooling Path Utilizing Residual Heat Removal Loop A
	Draw. No. 900547.68 · Rev. · Figure 15.2-15



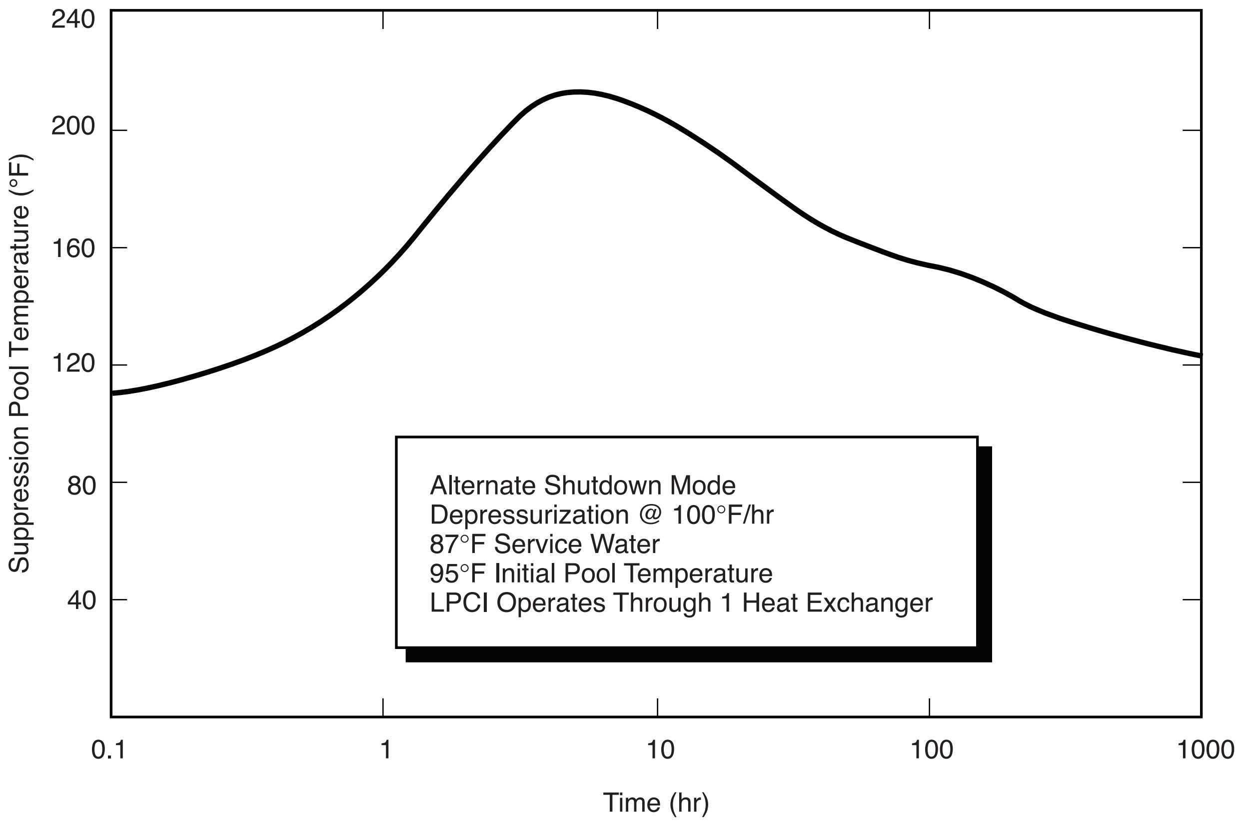
Vessel Temperature and Pressure Versus Time
(Activity C1.b.1 or C2)

Draw. No. 900547.69 | Rev. | Figure 15.2-16

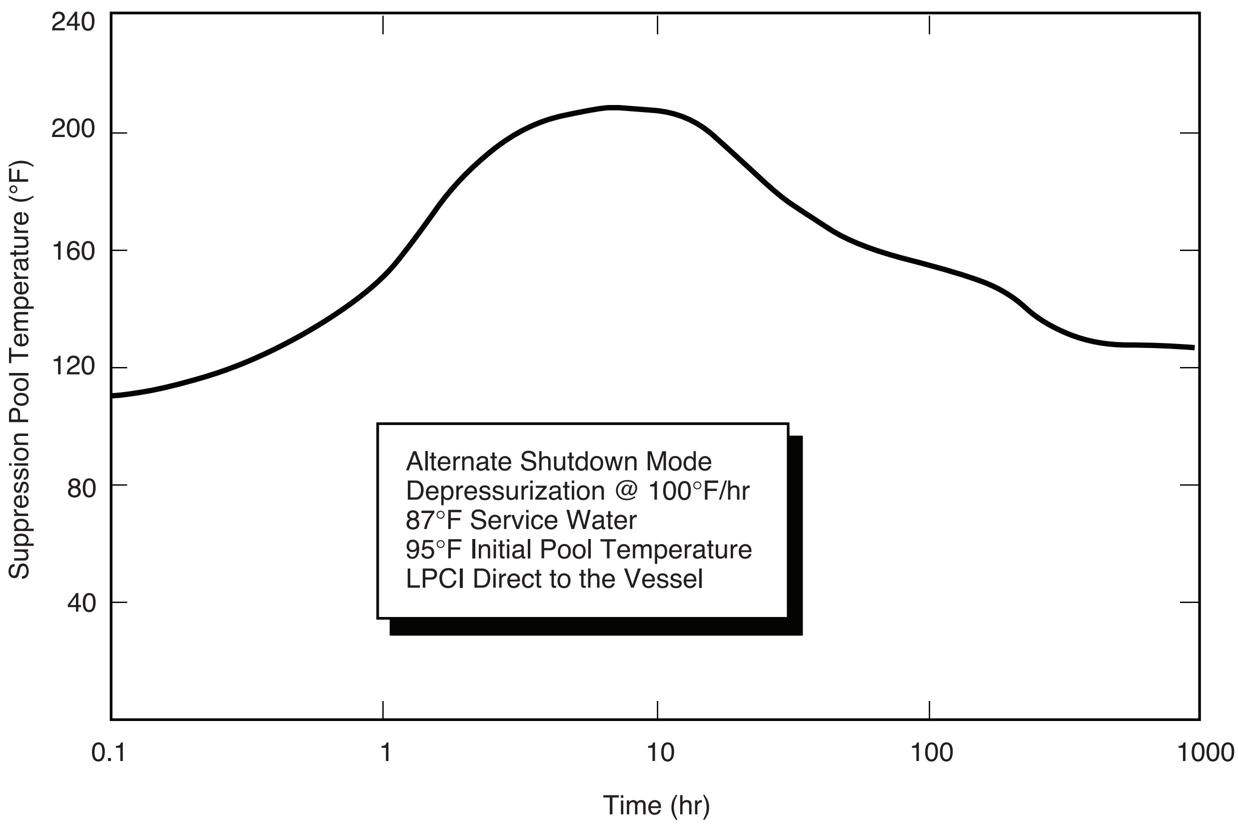


Vessel Temperature and Pressure Versus Time
(Activity C1.b.2)

Draw. No. 900547.70 | Rev. | Figure 15.2-17



Suppression Pool Temperature Versus Time
(with 87°F Service Water Temperature)
(Activity C1.b.1 or C2)



Suppression Pool Temperature Versus Time (with 87°F Service Water Temperature) (Activity C1.b.2)

Draw. No. 900547.72 Rev. Figure 15.2-19

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

15.3.1 RECIRCULATION PUMP TRIP

The events for two-recirculation pump operation are not limiting, therefore, the analyses have not been updated since the reactor power uprate analyses.

15.3.1.1 Identification of Causes and Frequency Classification

15.3.1.1.1 Identification of Causes

Recirculation pump motor operation can be tripped by design and by random operational failures. Design tripping will occur in response to:

a. Reactor vessel water level L2 setpoint trip,
b. Turbine control (governor) valve fast closure or stop (throttle) valve closure,
c. Failure to scram high pressure setpoint trip,
d. Motor branch circuit over-current protection,
e. Motor overload protection, and
f. Suction block valve not fully open.

Random tripping will occur in response to:

a. Operator error,

b. Loss of electrical power source to the pumps, and

c. Equipment or sensor failures and malfunctions which initiate the above intended trip response.

15.3.1.1.2 Frequency Classification

15.3.1.1.2.1 Trip of One Recirculation Pump. This event is categorized as an incident of moderate frequency.

15.3.1.1.2.2 Trip of Two Recirculation Pumps. This event is categorized as an incident of moderate frequency.

15.3.1.2 Sequence of Events and Systems Operation

15.3.1.2.1 Sequence of Events

15.3.1.2.1.1 Trip of One 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 Recirculation Pumps. Table 15.3-2 lists the sequence of events for Figure 15.3-2.

15.3.1.2.2 Systems Operation

15.3.1.2.2.1 Trip of One Recirculation Pump. Tripping a single recirculation pump requires no protection system or safeguard system operation. This analysis assumes normal functioning of plant instrumentation and controls.

15.3.1.2.2.2 Trip of Two Recirculation Pumps. Analysis of this event assumes normal functioning of plant instrumentation and controls and plant 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 stop (throttle) valves. High system pressure is limited by the pressure relief valve system operation.

15.3.1.2.3 The Effect of Single Failures and Operator Errors

15.3.1.2.3.1 Trip of One Recirculation Pump. None

15.3.1.2.3.2 Trip of Two Recirculation Pumps. Table 15.3-2 lists the vessel level (L8) trip event as the first response to initiate corrective action in this transient and it 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 transmission circuitry, however, is not built to single failure criterion. At this point the transient event is functionally over.

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

15.3.1.3 Core and System Performance

15.3.1.3.1 Mathematical Model

The point-kinetics REDY model described in Section 15.0.3.3.1 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 in Table 15.0-2.

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

15.3.1.3.3 Results

15.3.1.3.3.1 Trip of One Recirculation Pump. Figure 15.3-1 shows the response of the reactor system following the trip of one recirculation pump motor. Initially a recirculation pump is tripped in one loop, causing the core inlet flow to decrease, while the other recirculation loop flow increases. Subsequently jet pump diffuser flow reverses in the tripped recirculation loop. At approximately 45 sec the reactor reaches a new equilibrium operating point, at approximately 75% power and 57% core flow. During the transient, level swell is not sufficient to cause turbine trip.

15.3.1.3.3.2 Trip of Two Recirculation Pumps. Figure 15.3-2 shows the response of the reactor system following the trip of both recirculation pump motors. Initially both recirculation pumps are tripped, causing the core inlet flow to decrease, while vessel level rises until both main and feedwater turbines trip on high level (L8). A reactor scram is subsequently initiated at 90% turbine stop valve position. Shortly after the scram is initiated the stop valves close and the bypass valves open to regulate pressure. At this point the transient event is functionally over.

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 these maximize the flow reduction.

15.3.1.4 Barrier Performance

15.3.1.4.1 Trip of One Recirculation Pump

Figure 15.3-1 results indicate a basic reduction in system pressures from the initial conditions. Therefore, the reactor coolant pressure boundary (RCPB) barrier is not impacted.

15.3.1.4.2 Trip of Two Recirculation Pumps

The results shown in Figure 15.3-2 indicate peak pressures stay well below the limit allowed by the applicable American Society of Mechanical Engineers (ASME) code. Therefore, the RCPB barrier is not impacted.

15.3.1.5 Radiological Consequences

The consequence of this event does not result in fuel failure. It does result in the discharge of normal coolant activity to the suppression pool by means of safety/relief valve (SRV) operation, which is contained in the primary containment. This event does not result in an uncontrolled release to the environment, so the plant operator can choose to hold the activity in containment or discharge it when conditions permit. If purging of the containment is chosen, the release will be in accordance with established requirements.

15.3.2 RECIRCULATION FLOW CONTROL FAILURE - DECREASING FLOW

15.3.2.1 Identification of Causes and Frequency Classification

15.3.2.1.1 Identification of Causes

A postulated failure of the input demand signal, which is used in both loops, can decrease core flow at the maximum ramp demand rate established by the adjustable speed drive (ASD) control. Failure within either loop controller can result in a maximum ramp demand rate as limited by the ASD control.

15.3.2.1.2 Frequency Classification

This event is categorized as an incident of moderate frequency.

15.3.2.2 Sequence of Events and Systems Operation

15.3.2.2.1 Sequence of Events

15.3.2.2.1.1 Speed Decrease of One Recirculation Pump. Table 15.3-3 lists the sequence of events for Figure 15.3-3.

15.3.2.2.1.2 Speed Decrease of Two Recirculation Pumps. Table 15.3-4 lists the sequence of events for Figure 15.3-4.

15.3.2.2.2 Systems Operation

15.3.2.2.2.1 Speed Decrease of One Recirculation Pump. The most severe control system disturbance is a failure that causes the ASD internal controller to move at its maximum rate. Such transients may be obtained by instantaneous failure of a controller output into its upper or lower limits. Originally the recirculation flow was controlled by valve motion. For the current analysis the recirculation flow control valves have been locked at the full open position, and ASD units have been implemented to provide the necessary flow control.

15.3.2.2.2.2 Speed Decrease of Two Recirculation Pumps. The most severe control system disturbance is a failure that causes the ASD internal controller to move at its maximum rate. Such transients may be obtained by instantaneous failure of a controller output into its upper or lower limits. The independent and simultaneous failure of each individual loop controller would be highly improbable.

Thus, for the two loop controller failure event, the ASD internal controller is assumed to move at its maximum rate in both recirculation loops. Originally the recirculation flow was controlled by valve motion. For the current analysis the recirculation flow control valves have been locked at the full open position, and ASD units have been implemented to provide the necessary flow control.

15.3.2.2.3 The Effect of Single Failures and Operator Errors

The single failure and operator considerations for this event are essentially the same as in Section 15.3.1.2.3.2. The speed decrease of two instead of one recirculation pump would be the envelope case for the additional single component failure or operator error.

15.3.2.3 Core and System Performance

15.3.2.3.1 Mathematical Model

The point-kinetics REDY model described in Section 15.0.3.3.1 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 plant conditions listed in Table 15.0-2.

15.3.2.3.2.1 Speed Decrease of One Recirculation Pump. For the simulation of this event, a controller malfunction causes a zero demand signal to be sent to one of the recirculation ASD units, while the plant is operating at 106% uprated power and 100% core flow. A control demand error (low) signal causes the ASD to adjust the recirculation pump speed demand rate

limit downward at an assumed rate of 25%/sec for one loop failure. The ensuing transient is similar to a recirculation pump trip.

15.3.2.3.2.2 Speed Decrease of Two Recirculation Pumps. For the simulation of this event, a controller malfunction causes a zero demand signal to be sent to both of the recirculation ASD units, while the plant is operating at 106% uprated power and 100% core flow.
A control demand error (low) can cause the ASD units to adjust the recirculation pump speed downward in both loops at the 5%/sec pump speed rate limit.

15.3.2.3.3 Results

15.3.2.3.3.1 Speed Decrease of One Recirculation Pump. Figure 15.3-3 shows the response of the plant for this transient. Initially a negative recirculation pump speed demand is sent to the ASD due to a postulated controller failure. The negative pump speed demand causes the diffuser flow to decrease, and eventually reverse, in the failed loop. At the same time the active loop increases flow to compensate for the failed recirculation loop. At approximately 45 sec the reactor reaches a new equilibrium operating point, at approximately 74% power and 57% core flow. During the transient, level swell is not sufficient to cause turbine trip which would result in a reactor scram.

15.3.2.3.3.2 Speed Decrease of Two Recirculation Pumps. Figure 15.3-4 shows the response of the plant to this transient using the 5%/sec pump speed demand rate limit. Initially, a negative recirculation pump speed demand is sent to both ASD units due to a postulated controller failure. The negative pump speed demand causes the diffuser flows to decrease in the failed loops. During the transient, level swell is not sufficient to cause turbine trip which would result in a reactor scram.

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 expected under actual plant conditions.

These analyses are unaffected by deviations in pump/pump motor and driveline inertias since it is the ASD controller that causes rapid recirculation decreases.

15.3.2.4 Barrier Performance

15.3.2.4.1 Speed Decrease of One Recirculation Pump

The pressure in the vessel dome is well below the vessel pressure limit. The event does not result in a temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed. Therefore, barrier integrity and function is maintained.

15.3.2.4.2 Speed Decrease of Two Recirculation Pumps

The pressure in the vessel dome is well below the vessel pressure limit. The event does not result in a temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed and these barriers maintain their integrity and function as designed.

15.3.2.5 Radiological Consequences

Since this event does not result in any fuel failures or any release of primary coolant to either the secondary containment or to the environment there are no radiological consequences associated with this event.

15.3.3 RECIRCULATION PUMP SEIZURE

The pump seizure accident for single loop operation (SLO) is analyzed for the introduction of GE14 fuel into the Columbia reactor core.

15.3.3.1 Identification of Causes and Frequency Classification

The case of 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. The sudden decrease in core coolant flow while the reactor is at full power results in a degradation of core heat transfer which could result in fuel damage.

The event is categorized as an infrequent incident when operating with two recirculation pumps in service. For single loop operation, this event is considered to be a limiting fault, but is analyzed as an incident of moderate frequency for Global Nuclear Fuel reloads.

15.3.3.2 Sequence of Events and Systems Operation

15.3.3.2.1 Sequence of Events

Table 15.3-5 lists the sequence of events for Figure 15.3-5, for two loop operation. Table 15.3-6 lists the sequence of events for the recirculation pump seizure accident during SLO.

Identification of Operator Actions

The operator must verify that the reactor scrams with the turbine trip resulting from reactor water level swell. The operator should regain control of reactor water level through RCIC

operation or by restart of a feedwater pump, and must monitor reactor water level and pressure control after shutdown.

15.3.3.2.2 Systems Operation

In order to properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection, and reactor protection systems.

Operation of safe shutdown features, including operation of the HPCS and RCIC systems though not included in this simulation, may be used 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 by means of the high vessel level (L8) trip are similar to the considerations in Section 15.3.1.2.3.2.

15.3.3.3 Core and System Performance

15.3.3.3.1 Mathematical Model

The point-kinetics REDY model described in Section 15.0.3.3.1 is used to simulate this event for two loop operation. The computer model described in Reference 15.3-2 was used to simulate this event for SLO.

15.3.3.3.2 Input Parameters and Initial Conditions

This analysis has been performed for two loop operation, unless otherwise noted, with plant conditions tabulated in Table 15.0-2, column "REDY (ASD Events)". For the simulation of the event while in two loop operation, one recirculation pump was seized instantaneously (pump speed set to zero) while the plant is operating at 106% uprated power and 100% core flow.

For single loop operation, the analysis has been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2A, column "Original Rated Power". 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 the active recirculation pump shaft while the reactor is operating at 75% NBR power under SLO. Also, the reactor is assumed to be operating at thermally-limiting conditions. The void coefficient is adjusted to the most conservative value, that is, the least negative value in Table 15.0-2A.

15.3.3.3.3 Results

Figure 15.3-5 presents the results of the accident for two loop operation. Table 15.3-5 shows the sequence of events for this transient. Initially a recirculation pump is seized in one loop causing the flow in the seized loop to reverse and the flow in the active loop to increase. As the flow in the seized loop decreases, the vessel level rises until a turbine trip is initiated on high level, L8. Once L8 is reached, both feedwater pumps trip. A reactor scram is subsequently initiated due to 90% turbine stop (throttle) valve position. Shortly after the turbine trip is initiated the stop valves close and the bypass valves open to regulate pressure. Simultaneously the active recirculation loop trips due to the turbine trip. The MCPR does not decrease significantly before fuel surface heat flux begins dropping enough to restore greater thermal margins. After the time at which MCPR occurs, heat flux decreases more rapidly than the rate at which heat is removed by the coolant and the ΔCPR is less than 0.01.

Figure 15.3-6 presents the results of the event in SLO. Core coolant flow drops rapidly, reaching a minimum value of 25% rated at about 2.0 sec.

The RRC pump seizure while in SLO is more limiting than the RRC pump seizure in two loop operation. See Table 15.0-1A.

15.3.3.3.3.1 <u>Considerations of Uncertainties</u>. Considerations of uncertainties are included in the analysis.

15.3.3.4 <u>Barrier Performance</u>

The bypass valves open to limit the pressure well within the range allowed by the ASME vessel code. The RCPB is not impacted by overpressure. Therefore, barrier integrity and function is maintained.

15.3.3.5 <u>Radiological Consequences</u>

Since this event does not result in any fuel failures or any release of primary coolant to either the secondary containment or to the environment there are no radiological consequences associated with this event.

15.3.4 RECIRCULATION PUMP SHAFT BREAK

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

The breaking of the shaft of a recirculation pump is considered a design basis accident event. It has been evaluated as a mild accident in relation to other design basis accidents such as the loss-of-coolant accident. The analysis has been conducted with consideration to a single or

two loop operation. Two loop operation represents the worst case since single loop operation is limited to approximately 75% power.

This postulated event is bounded by the more limiting case of recirculation pump seizure.

15.3.4.1.1 Identification of Causes

The case of recirculation pump shaft breakage represents the unlikely event of rapid stoppage of the pump operation of one recirculation pump. This event produces a rapid decrease of core flow.

15.3.4.1.2 Frequency Classification

This event is categorized as an incident of infrequent frequency.

15.3.4.2 Sequence of Events and Systems Operation

15.3.4.2.1 Sequence of Events

A postulated instantaneous break of the pump motor shaft of one 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 (Level 8), a main turbine trip and feedwater pump trip will be initiated.

A reactor scram and the remaining recirculation pump trip will be initiated due to the turbine trip. Eventually the vessel water level will be controlled by HPCS and/or RCIC flow.

15.3.4.2.2 Systems Operation

Normal operation of plant instrumentation and control is assumed. This event takes credit for vessel water level (Level 8) instrumentation to scram the reactor and trip the main turbine and feedwater pumps. High system pressure is limited by the pressure relief system operation.

Operation of HPCS and/or RCIC 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 15.3.1.2.3.2.

Assumption of single component failure or operator error in other equipment has been examined and this has led to the conclusion that no other credible failure exists for this event. Therefore, the bounding case has been considered.

15.3.4.3 Core and System Performance

The pump shaft break event is bounded by the pump seizure event. Since this event is less limiting than that event, only qualitative evaluation is provided. Therefore, no discussion of mathematical model, input parameters, and consideration of uncertainties, etc., is necessary.

15.3.4.3.1 Qualitative Results

If this unlikely event occurs, core coolant flow will drop rapidly. The level swell produces a trip of the main and feedwater turbines. A scram is initiated due to turbine trip. Since heat flux decreases more rapidly than the rate at which heat is removed by the coolant, there is no impact on thermal limits. Additionally, the bypass valves and the potential for a momentary opening of some of the SRVs limit the pressure well within the range allowed by the ASME vessel code. Therefore, the RCPB is not impacted by overpressure.

The severity of this pump shaft break event is bounded by the pump seizure event. In either of these two events, the recirculation drive flow of the affected loop decreases rapidly.

In the case of the pump 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 seizure event. Therefore, the core flow decrease following a pump shaft break effect is slower than the pump seizure event. Thus, it can be concluded that the potential effects of the hypothetical pump shaft break accident are bounded by the effects of the pump seizure event.

15.3.4.4 Barrier Performance

The bypass valves and momentary opening of some of the SRVs limit the pressure well within the range allowed by the ASME vessel code. Therefore, the RCPB is not impacted by overpressure.

15.3.4.5 Radiological Consequences

Since this event does not result in any fuel failures or any release of primary coolant to either the secondary containment or to the environment there are no radiological consequences associated with this event.

15.3.5 REFERENCES

15.3-1 General Electric Company, "WNP-2 Power Uprate Transient Analysis Task Report," GE-NE-208-08-0393, September 1993.

15.3-2 NEDC-24154-P-A, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," Volumes 1, 2, 3 and 4, February 2000.

15.3-3 Advanced Nuclear Fuels Corporation, "WNP-2 Single Loop Operation Analysis," ANF-87-119, September 1987.

Table 15.3-1

Sequence of Events for Figure 15.3-1

Trip of One Recirculation Pump Motor
Uprated Power

Time (sec)	Event
0	Trip of one recirculation pump initiated.
9	Jet pump diffuser flow reverses in the tripped loop.
45[a]	Core flow and power level stabilize at new equilibrium conditions.

[a] Approximately.

Table 15.3-2

Sequence of Events for Figure 15.3-2

Trip of Both Recirculation Pump Motors
Uprated Power

Time (sec)	Event
0	Trip of both recirculation pumps initiated.
5.66	Vessel water level (L8) trip initiates turbine trip.
5.66	Feedwater pumps are tripped off.
5.67	Main turbine stop (throttle) valves reach 90% open position and initiate reactor scram trip.
5.76	Turbine bypass valves open.

Table 15.3-3

Sequence of Events for Figure 15.3-3

Recirculation Flow Control Failure
Decreasing Flow in One Loop
Uprated Power

Time (sec)	Event
0	Initiate fast down scale of recirculation pump speed in one loop.
4[a]	Jet pump diffuser flow reverses in the affected loop.
45[a]	Core flow and power level stabilize at new equilibrium conditions.

[a] Approximately.

Table 15.3-4

Sequence of Events for Figure 15.3-4

Recirculation Flow Control Failure
Decreasing Flow in Both Loops (5%/sec)
Uprated Power

Time (sec)	Event
0	Initiate 5%/sec down scale of recirculation pump speed in both loops.
85[a]	Core flow and power level stabilize at new equilibrium conditions.

Table 15.3-5

Sequence of Events for Figure 15.3-5

One Recirculation Pump Seizure
Uprated Power

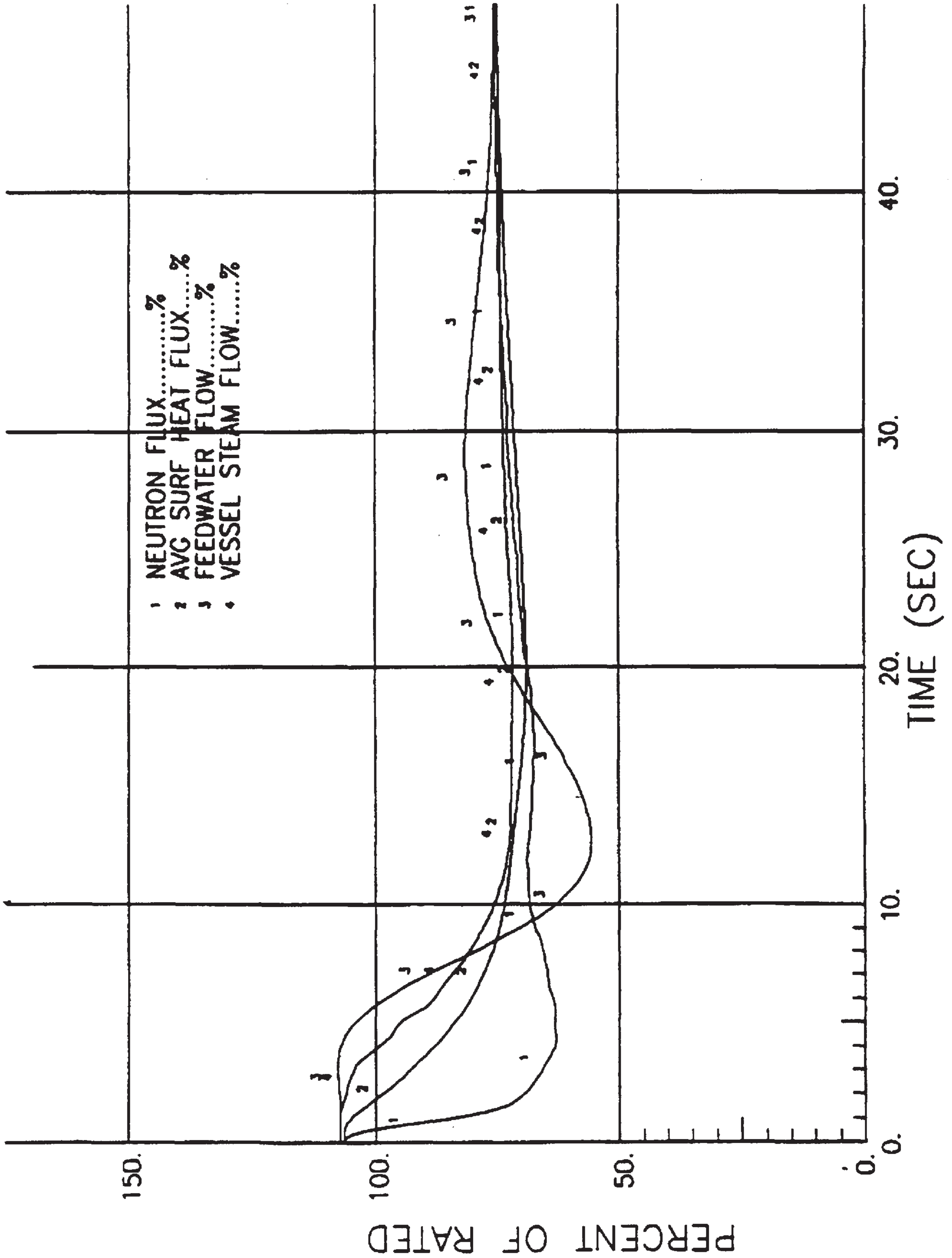
Time (sec)	Event
0	Seizure of one recirculation pump initiated.
1[a]	Jet pump diffuser flow reverses in the seized loop.
4.40	Vessel water high level (L8) trip initiates a turbine trip.
4.40	Feedwater pumps are tripped off.
4.41	Main turbine stop (throttle) valves reach 90% open position and initiate reactor scram.
4.59	Active recirculation loop trips due to previous turbine trip.

[a] Approximately.

Table 15.3-6

Sequence of Events for Pump Seizure
(for Single Loop Operation)

Time (sec)	Event
0.0	Recirculation pump motor trip off complete Single pump seizure was initiated; core flow decreases
~1.9	Reverse flow ceases in the idle loop
~6.0	Power and flow stabilize



Columbia Generating Station
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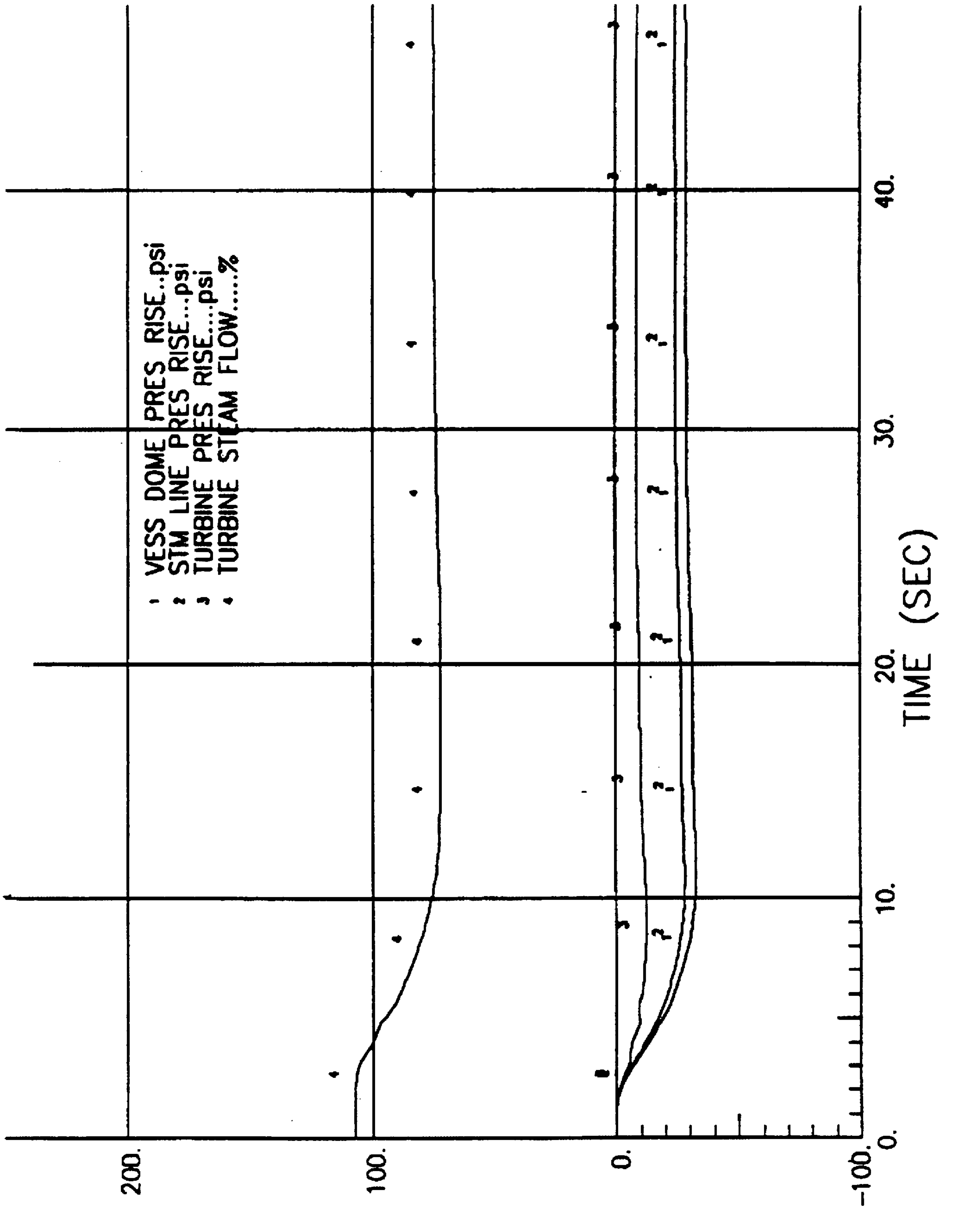
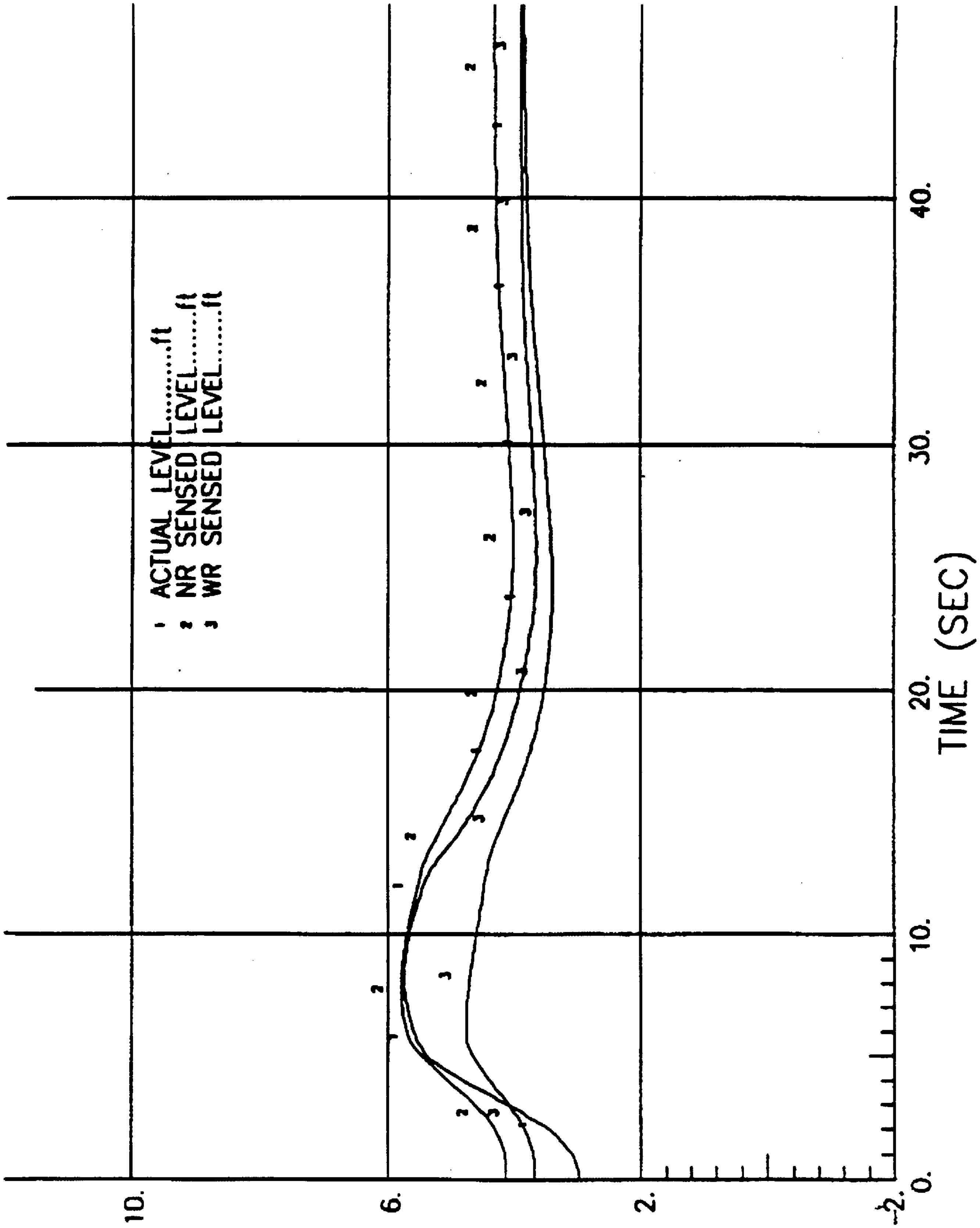
One Recirculation Pump Trip at 106.2% Uprated Power, 100% Flow

Draw. No. 020361.77	Rev.	Figure 15.3-1.1



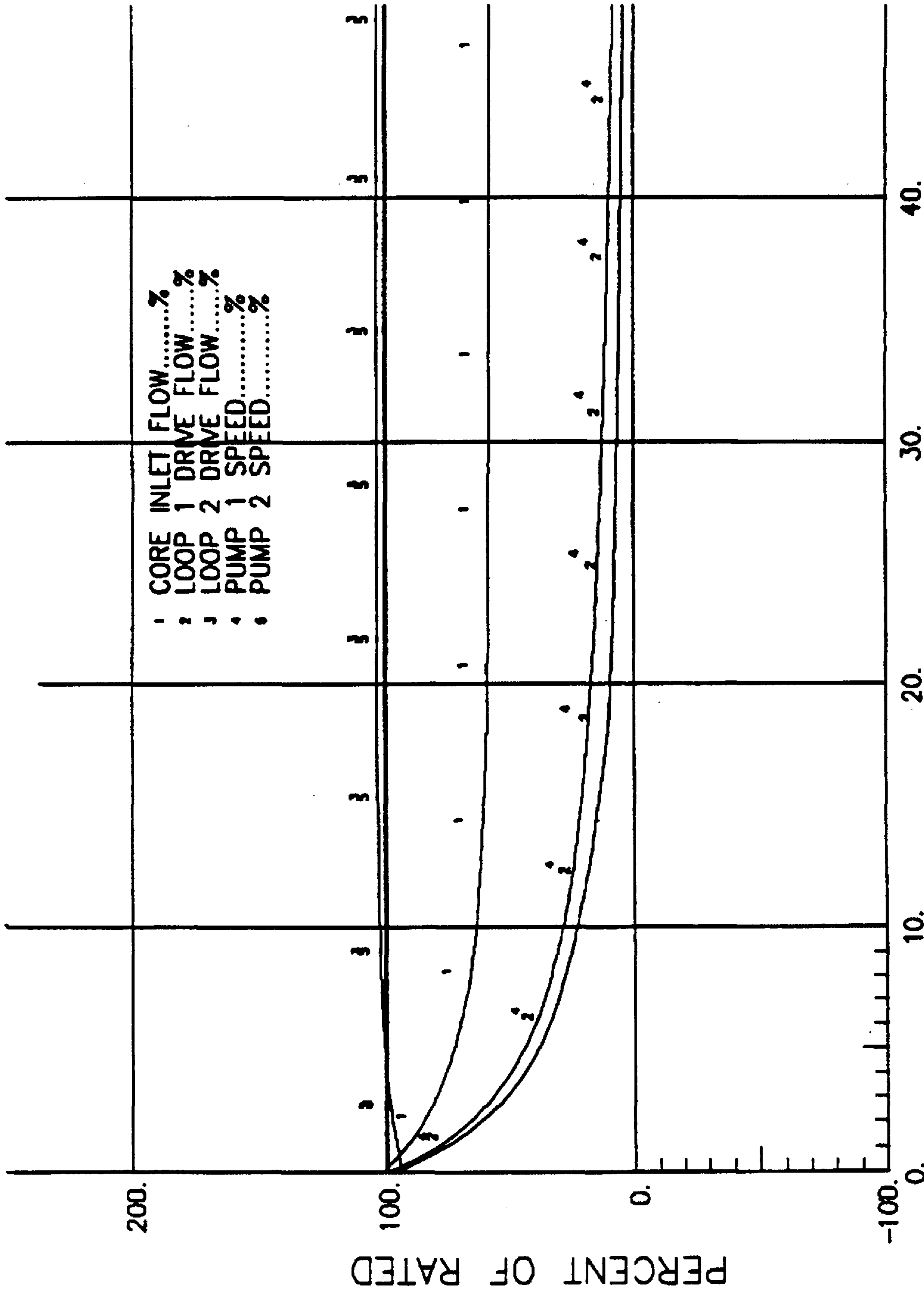
Columbia Generating Station Final Safety Analysis Report	One Recirculation Pump Trip at 106.2% Uprated Power, 100% Flow
	Draw. No. 020361.78 | Rev. | Figure 15.3-1.2



Columbia Generating Station
Final Safety Analysis Report

One Recirculation Pump Trip at 106.2%
Uprated Power, 100% Flow

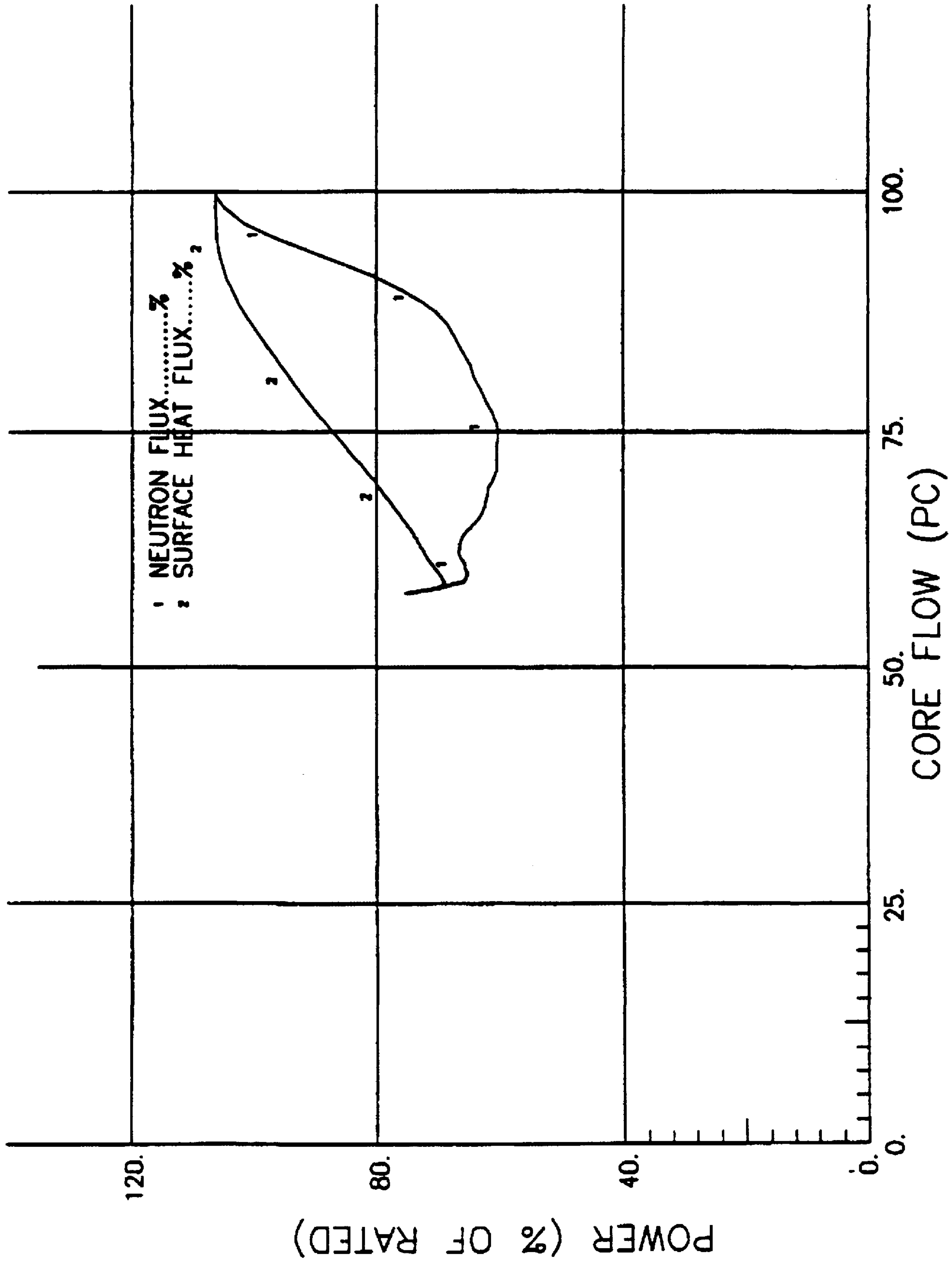
Draw. No. 020361.79	Rev.	Figure 15.3-1.3



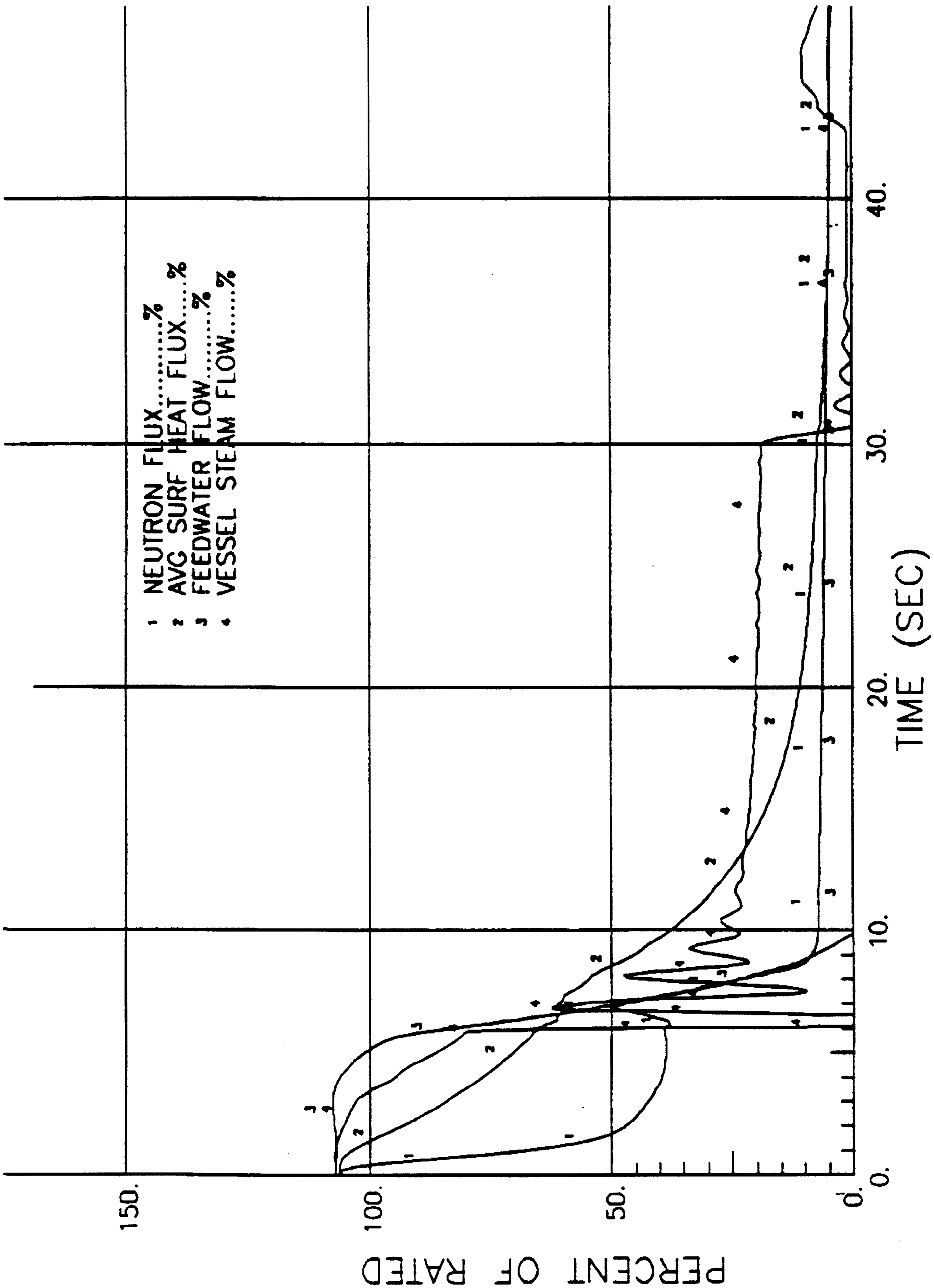
Columbia Generating Station
Final Safety Analysis Report

One Recirculation Pump Trip at 106.2% Uprated Power, 100% Flow

Draw. No. 020361.80	Rev.	Figure 15.3-1.4



Columbia Generating Station Final Safety Analysis Report	One Recirculation Pump Trip at 106.2% Uprated Power, 100% Flow		
	Draw. No. 020361.81	Rev.	Figure 15.3-1.5



Columbia Generating Station
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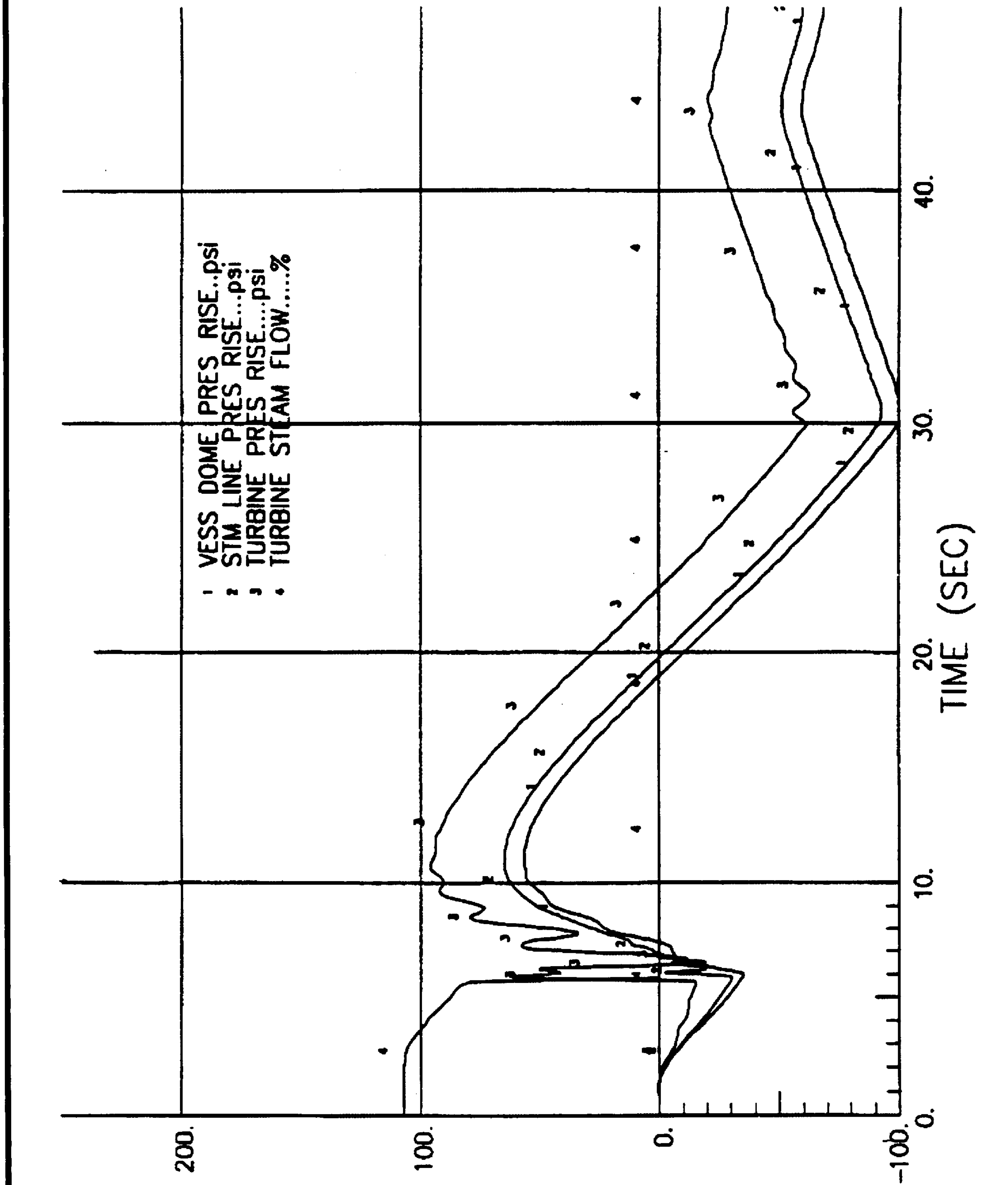
Two Recirculation Pump Trip at 106.2% Uprated Power, 100% Flow

Draw. No. 020361.82	Rev.	Figure 15.3-2.1



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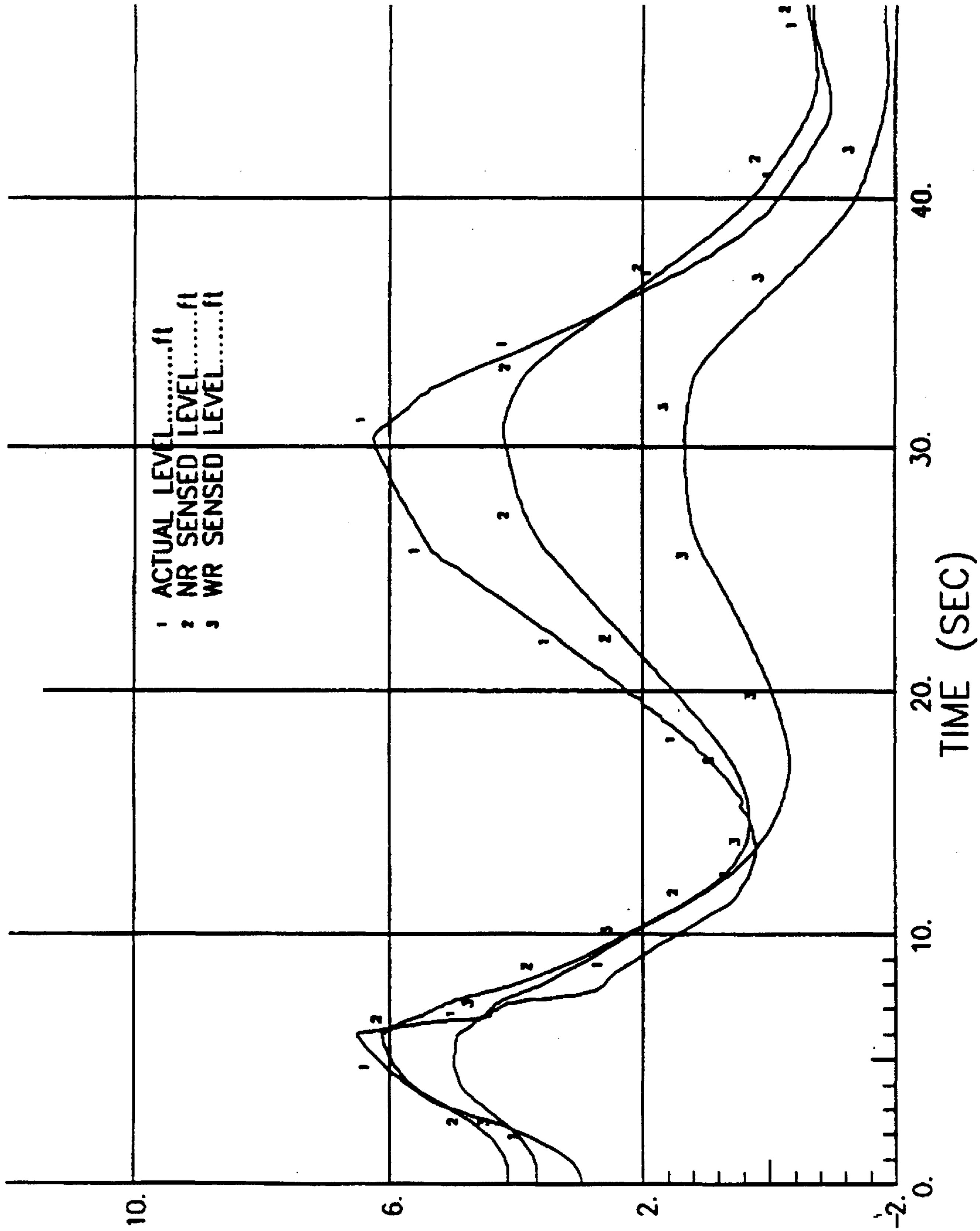
Two Recirculation Pump Trip at 106.2% Uprated Power, 100% Flow

Draw. No. 020361.83	Rev.	Figure 15.3-2.2



Columbia Generating Station Final Safety Analysis Report	Two Recirculation Pump Trip at 106.2% Uprated Power, 100% Flow		
	Draw. No. 020361.84	Rev.	Figure 15.3-2.3

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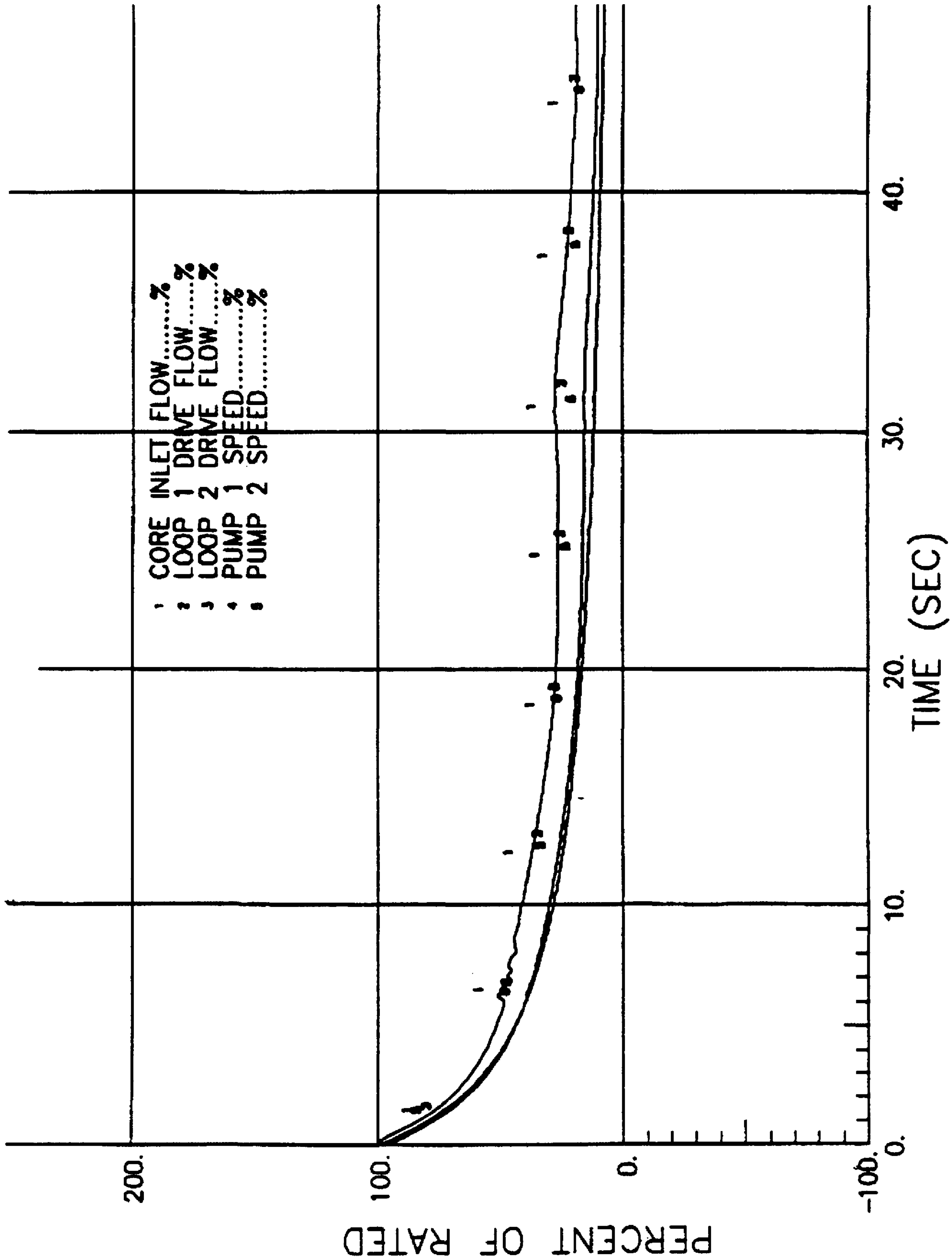
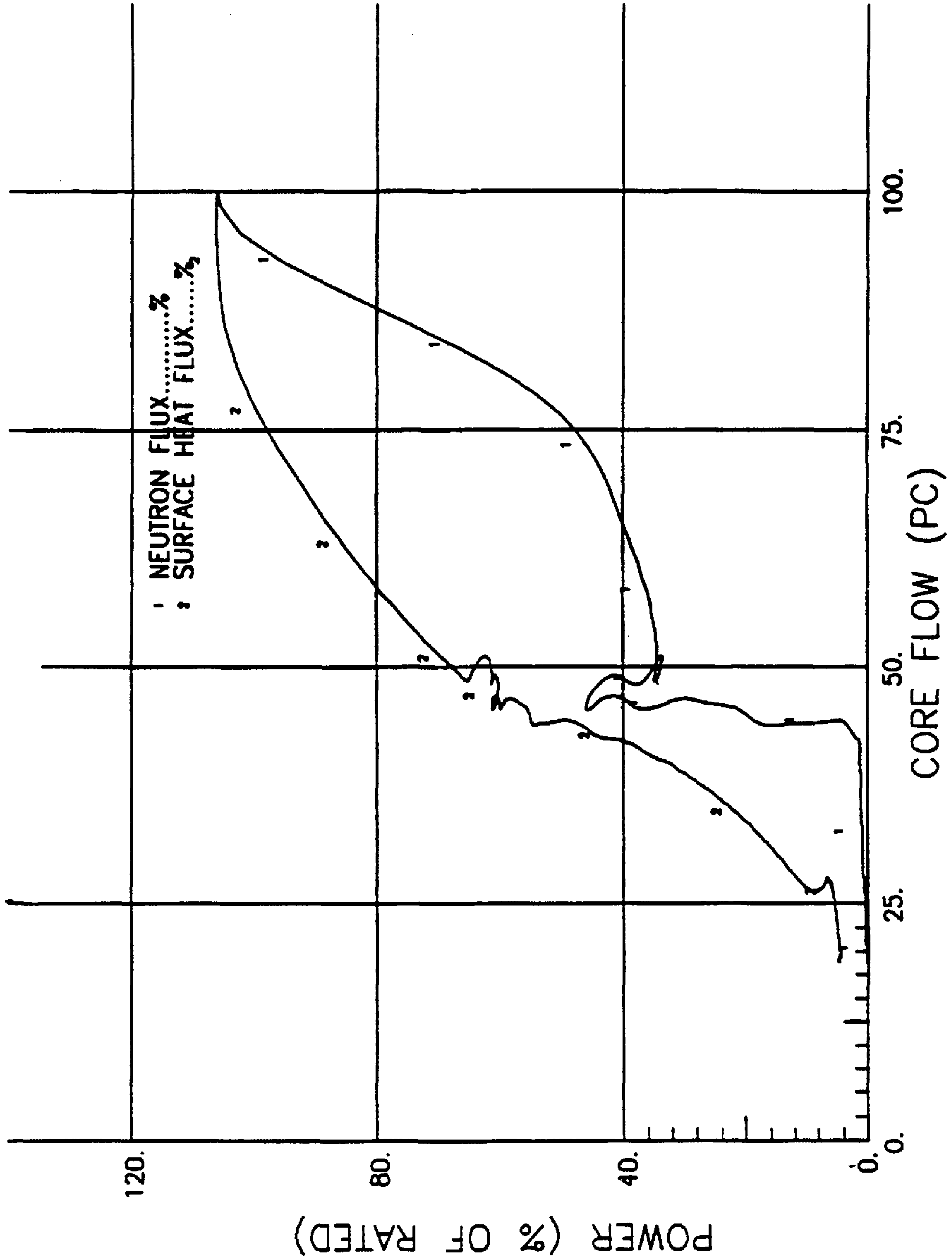


Columbia Generating Station Final Safety Analysis Report	Two Recirculation Pump Trip at 106.2% Uprated Power, 100% Flow		
	Draw. No. 020361.85	Rev.	Figure 15.3-2.4



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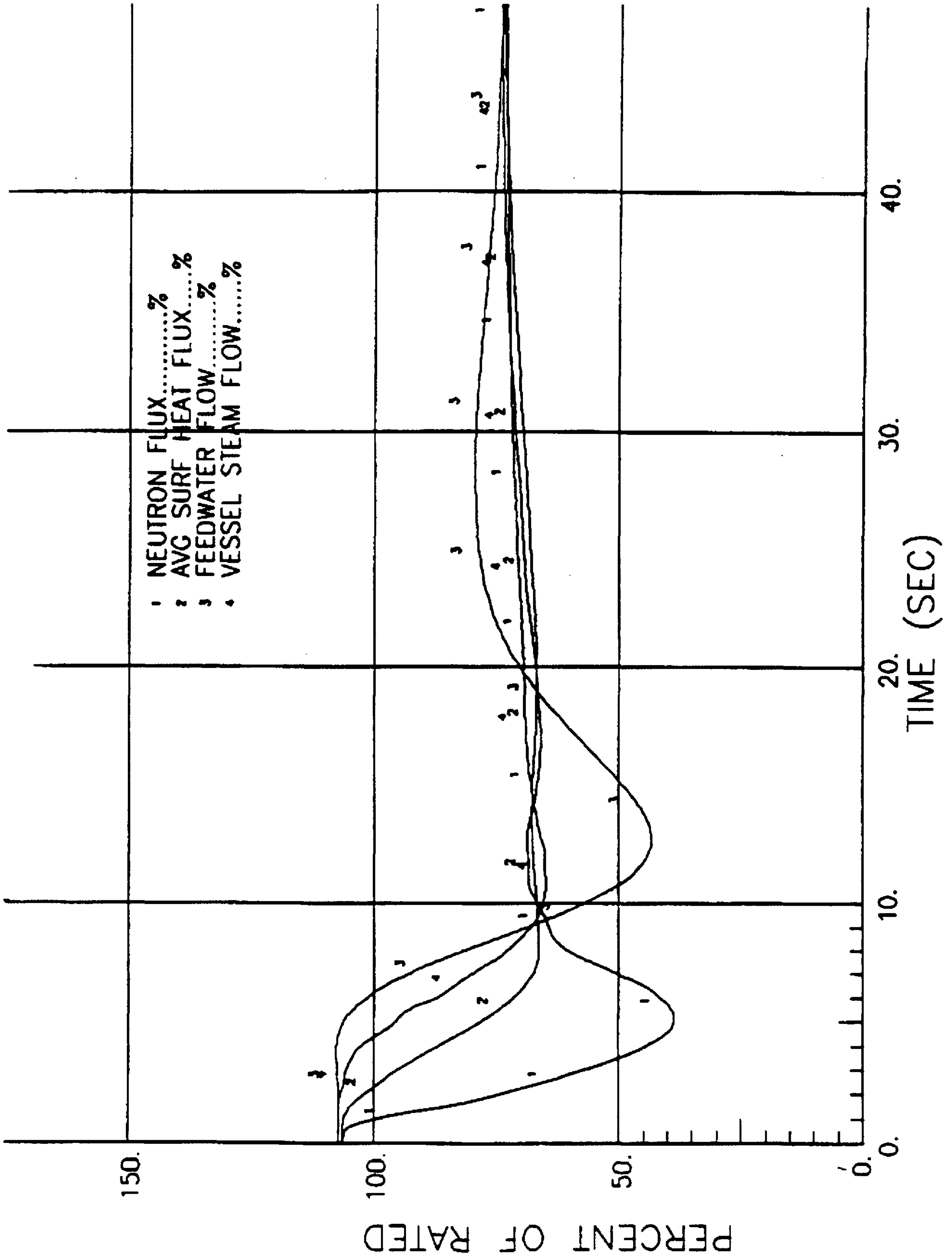
Two Recirculation Pump Trip at 106.2% Uprated Power, 100% Flow

Draw. No. 020361.86	Rev.	Figure 15.3-2.5



Columbia Generating Station Final Safety Analysis Report	Recirculation Flow Control Failure - Decreasing Flow in One Loop at 106.2% Uprated Power, 100% Flow		
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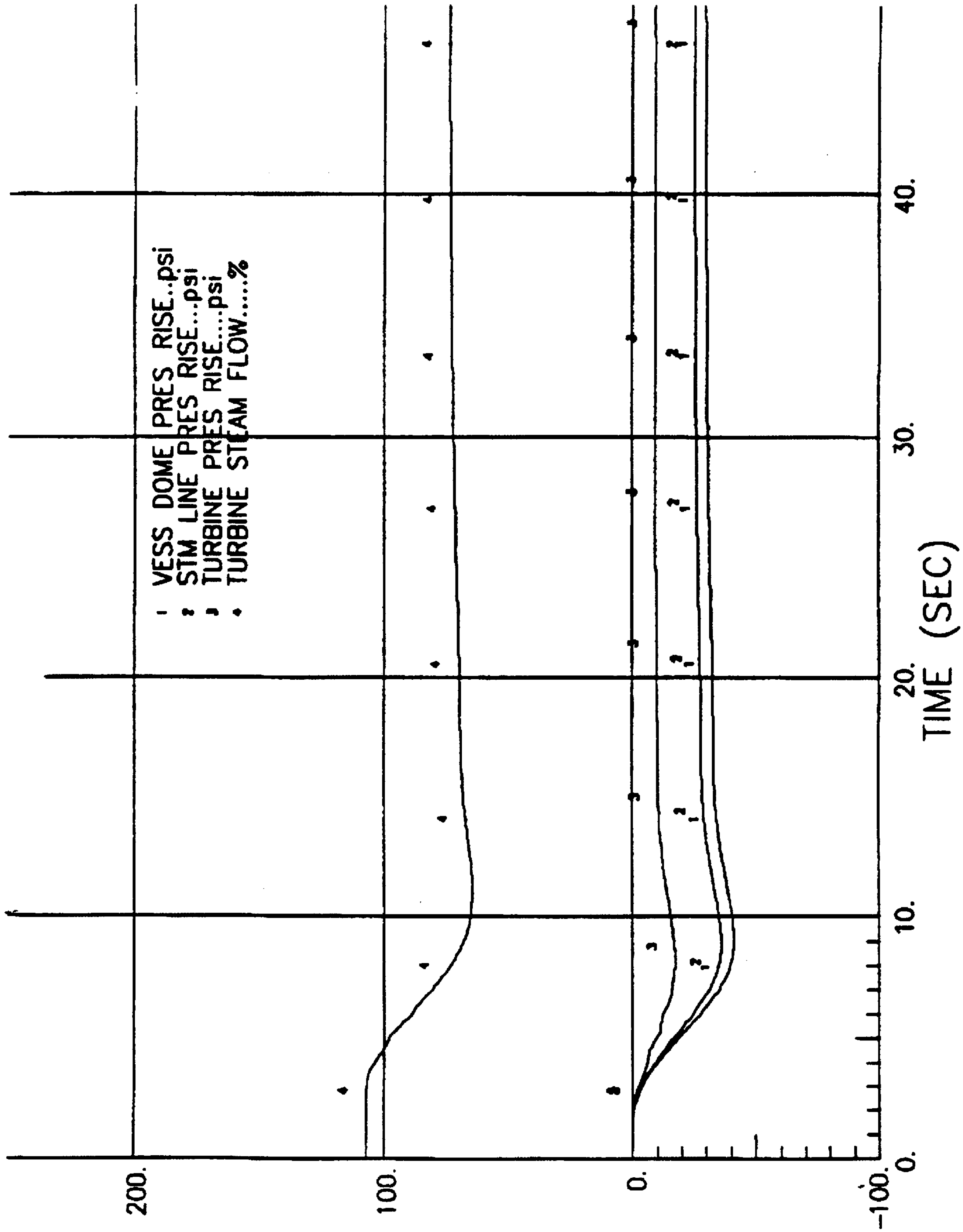
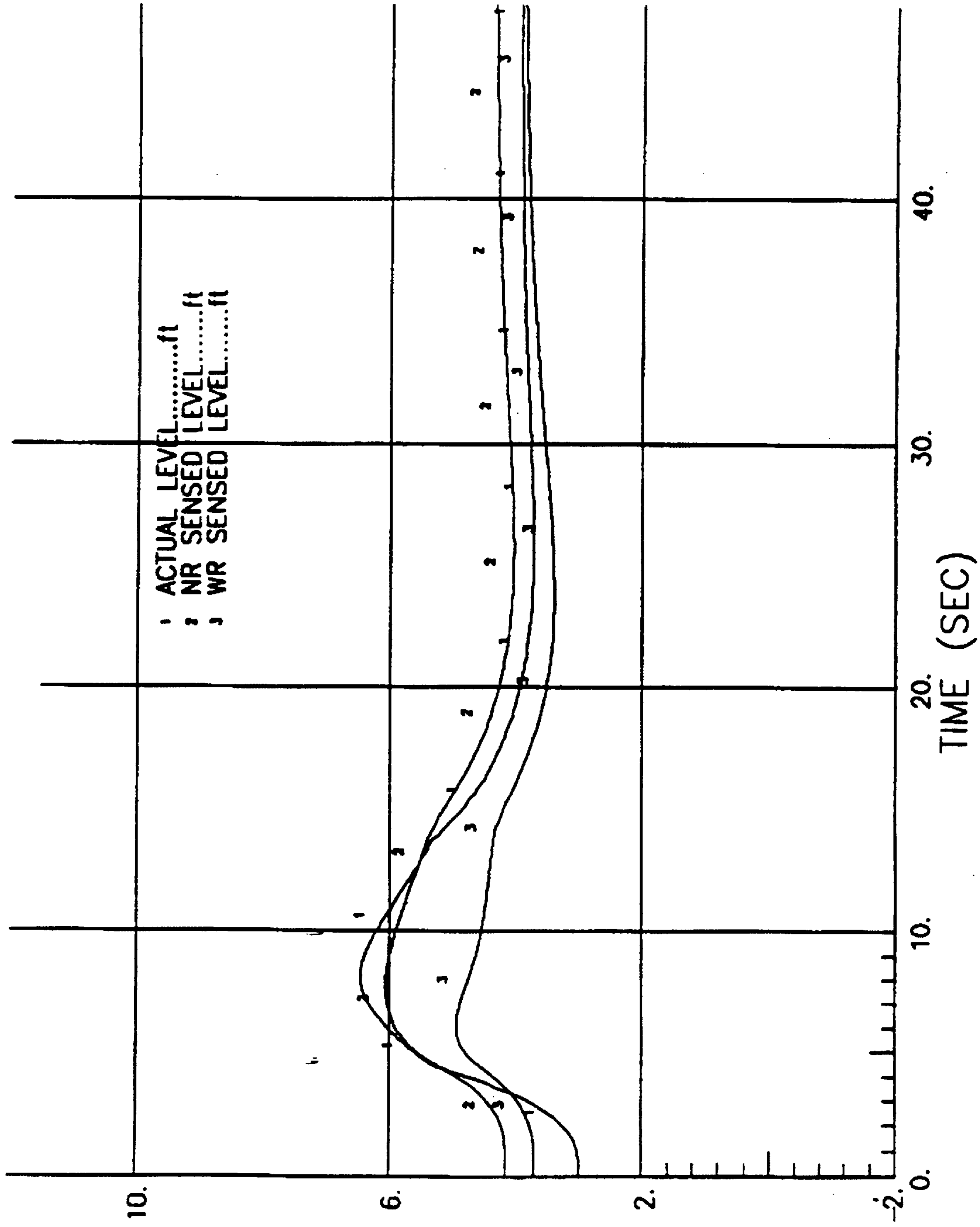


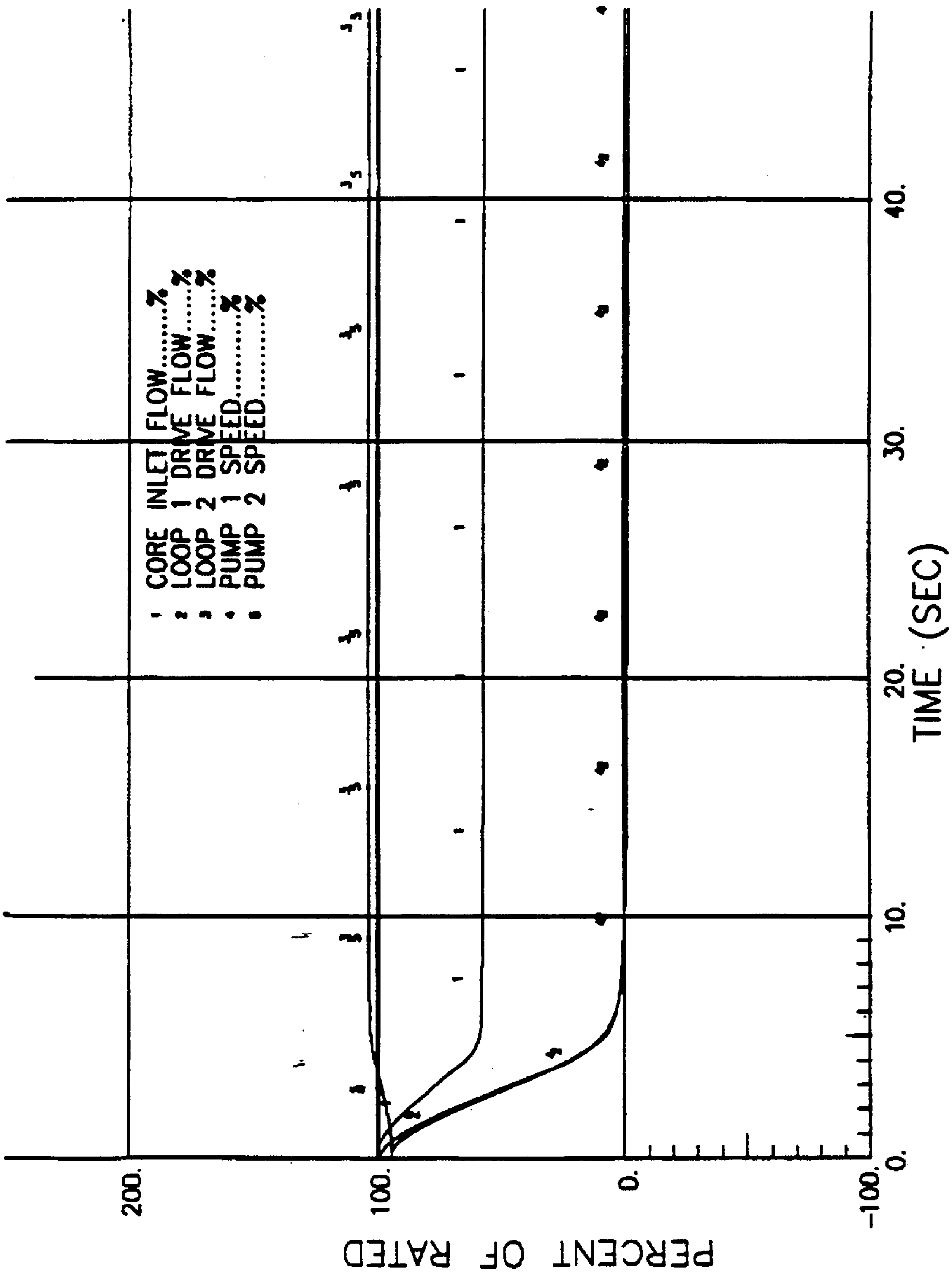
Columbia Generating Station Final Safety Analysis Report	Recirculation Flow Control Failure - Decreasing Flow in One Loop at 106.2% Uprated Power, 100% Flow		
	Draw. No. 020361.88	Rev.	Figure 15.3-3.2



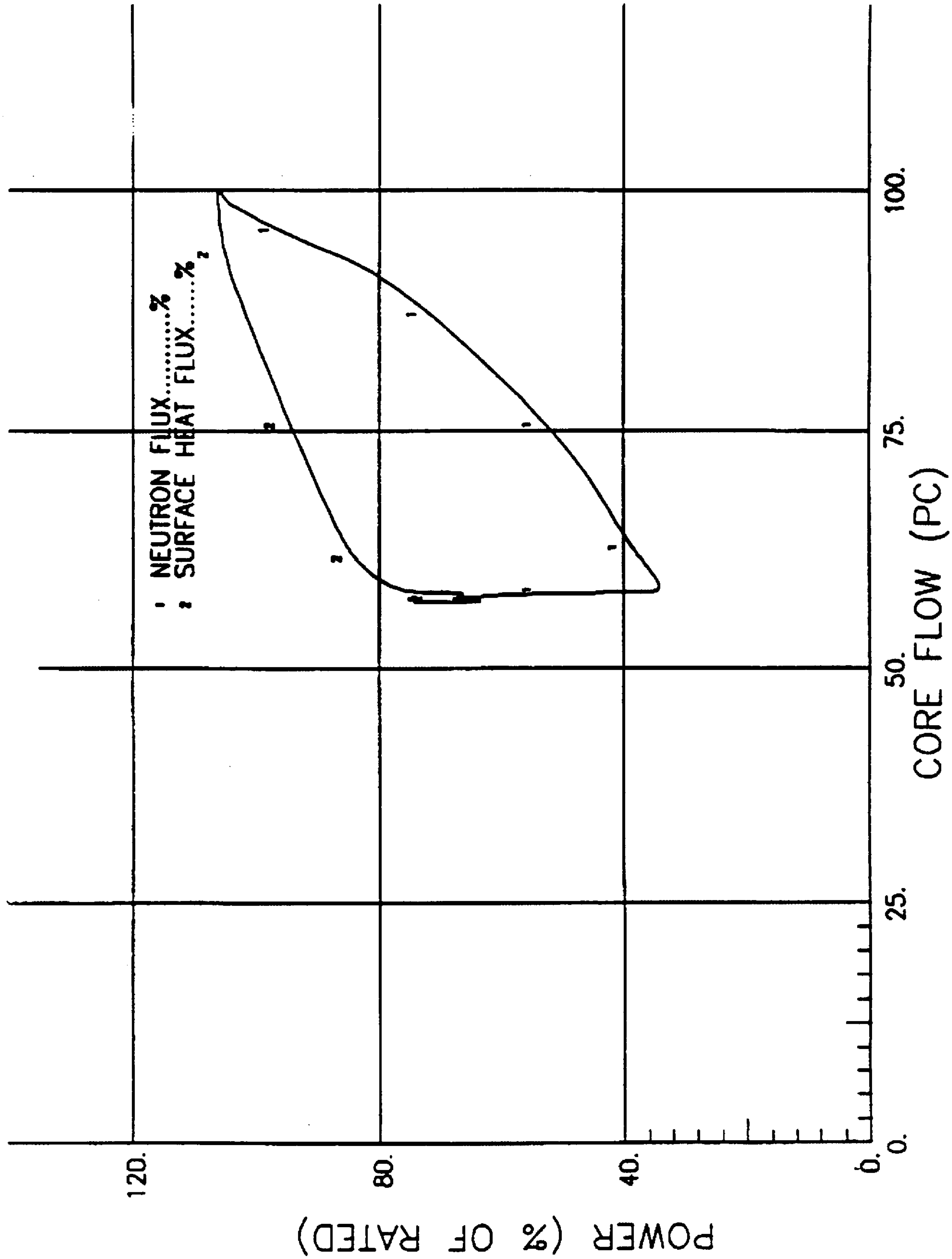
Columbia Generating Station
Final Safety Analysis Report

Recirculation Flow Control Failure - Decreasing Flow in One Loop at 106.2% Uprated Power, 100% Flow

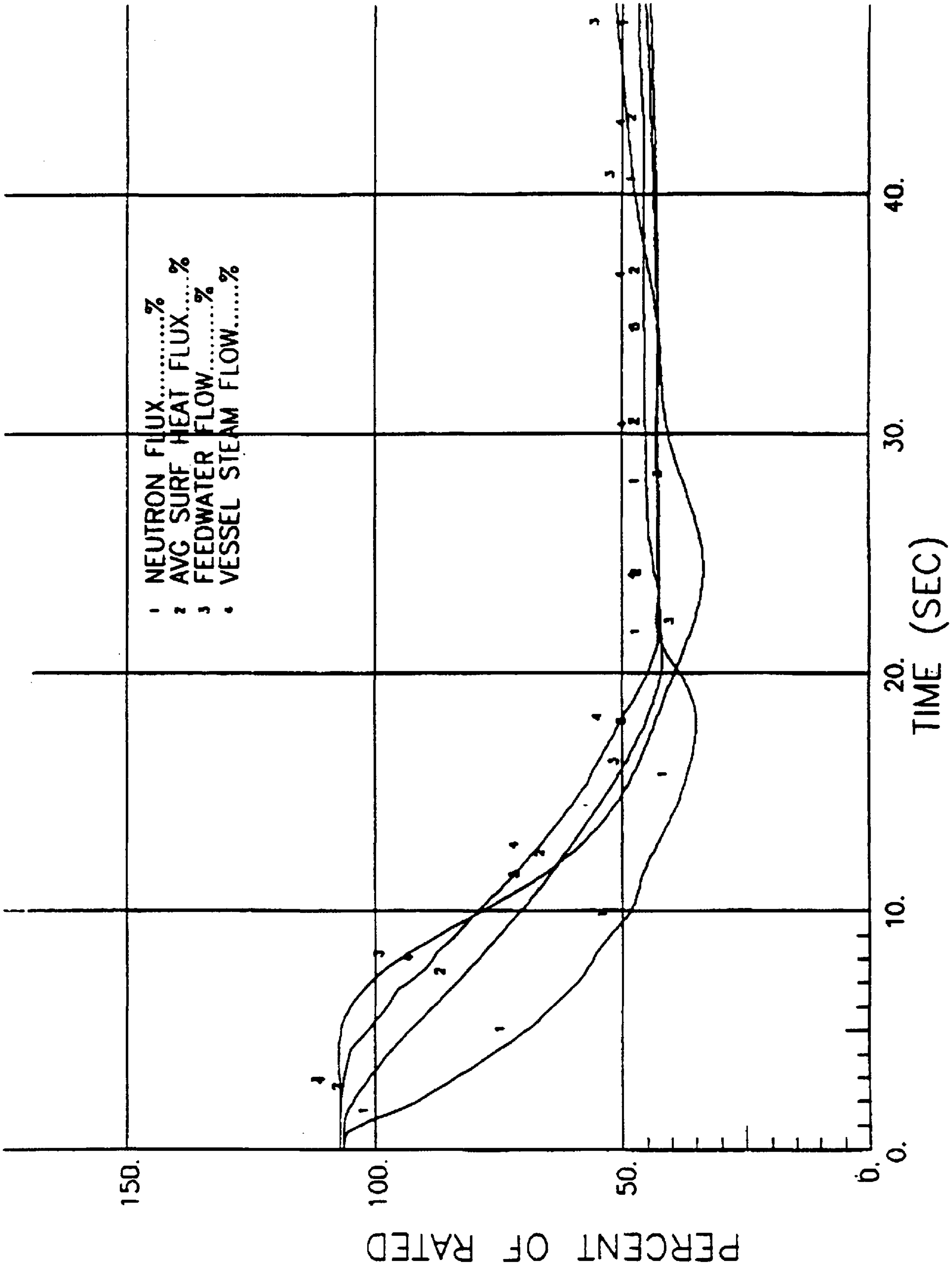
Draw. No. 020361.89	Rev.	Figure 15.3-3.3



Columbia Generating Station Final Safety Analysis Report	Recirculation Flow Control Failure - Decreasing Flow in One Loop at 106.2% Uprated Power, 100% Flow		
	Draw. No. 020361.90	Rev.	Figure 15.3-3.4



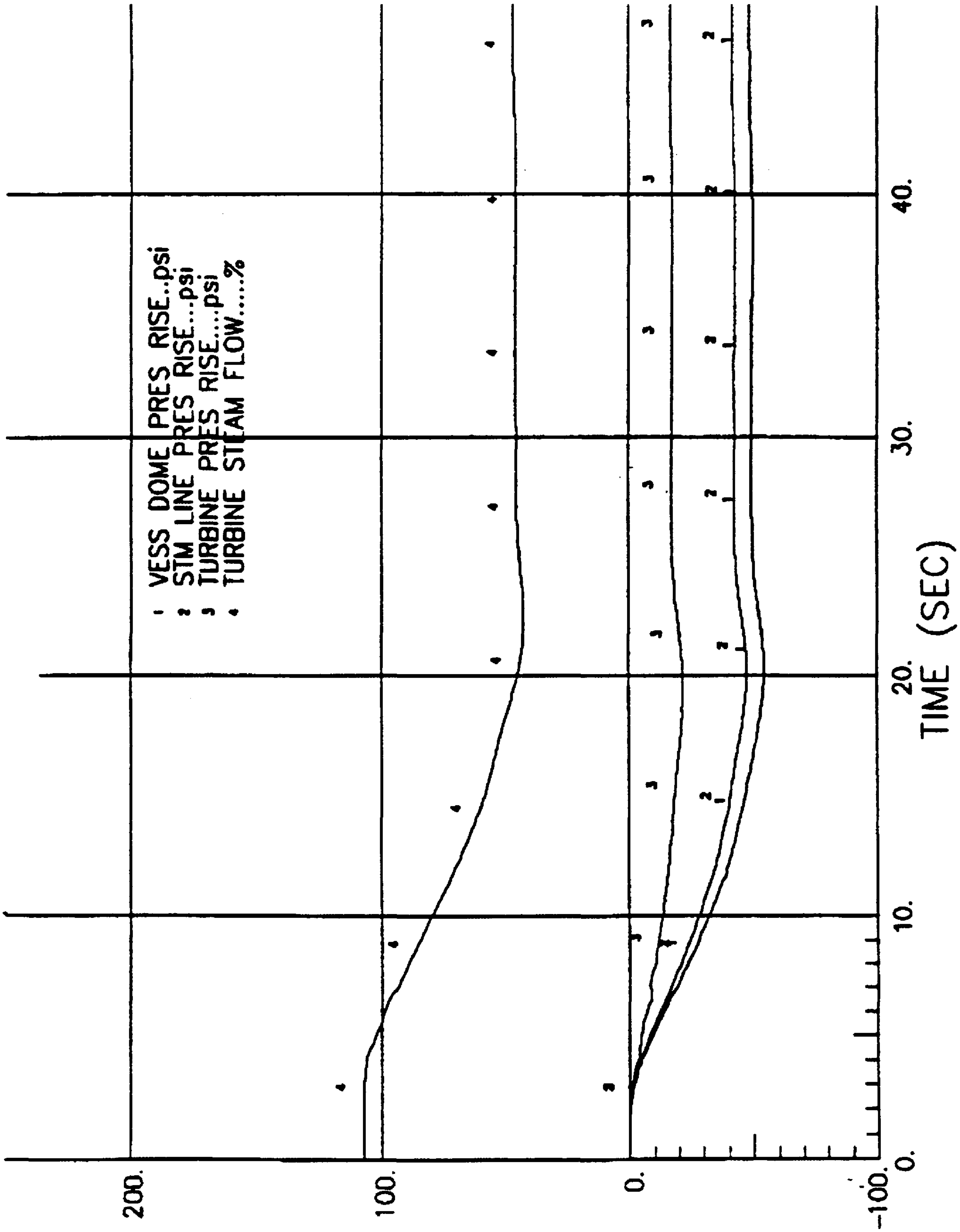
Columbia Generating Station **Final Safety Analysis Report**	**Recirculation Flow Control Failure - Decreasing Flow in One Loop at 106.2% Uprated Power, 100% Flow**
	Draw. No. 020361.91 | Rev. | Figure 15.3-3.5



Columbia Generating Station
Final Safety Analysis Report

Recirculation Flow Control Failure - Decreasing Flow in Two Loops, (5%/Sec Ramp) at 106.2% Uprated Power, 100% Flow

Draw. No. 020361.92	Rev.	Figure 15.3-4.1



Columbia Generating Station
Final Safety Analysis Report

Recirculation Flow Control Failure - Decreasing Flow in Two Loops, (5%/Sec Ramp) at 106.2% Uprated Power, 100% Flow

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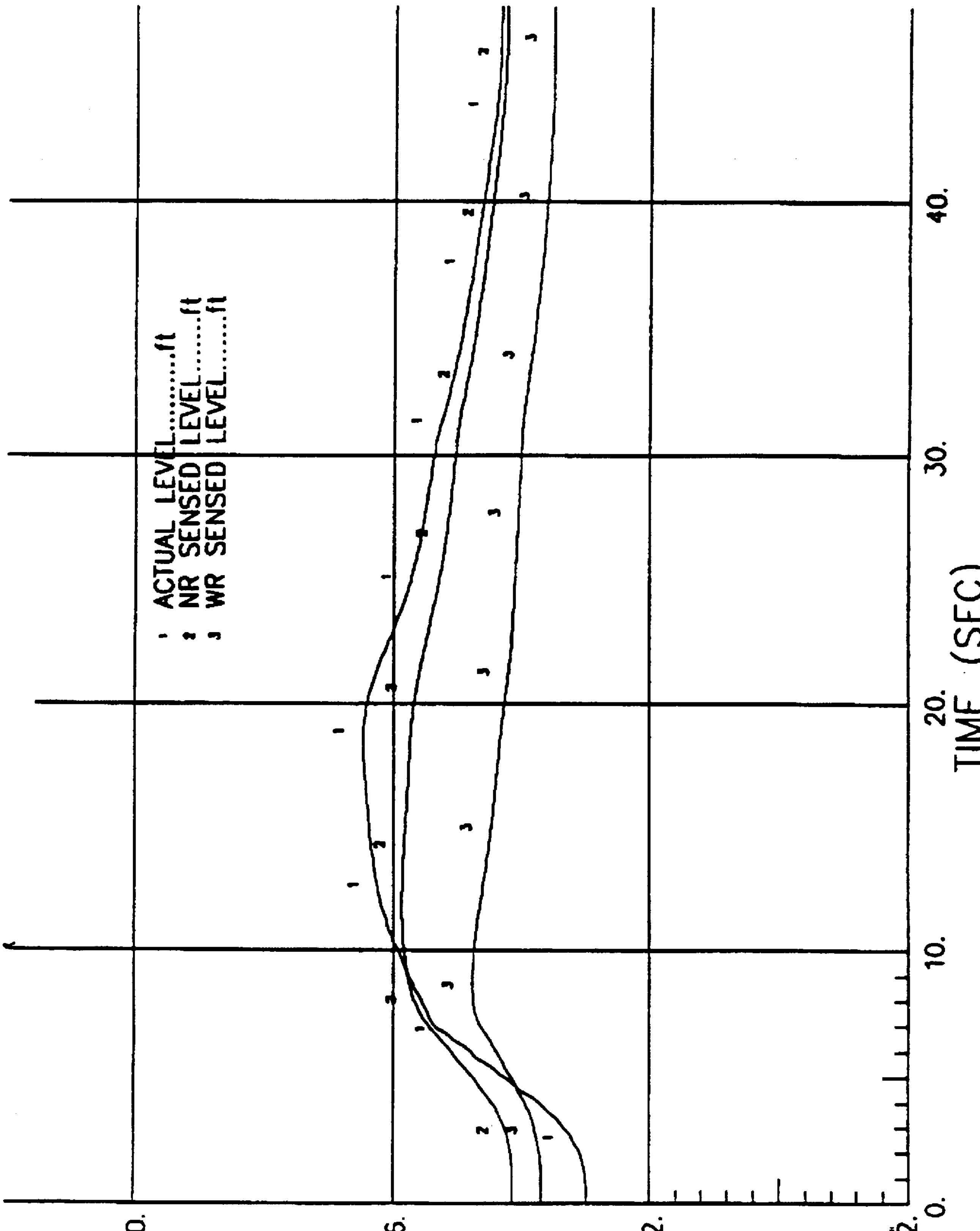
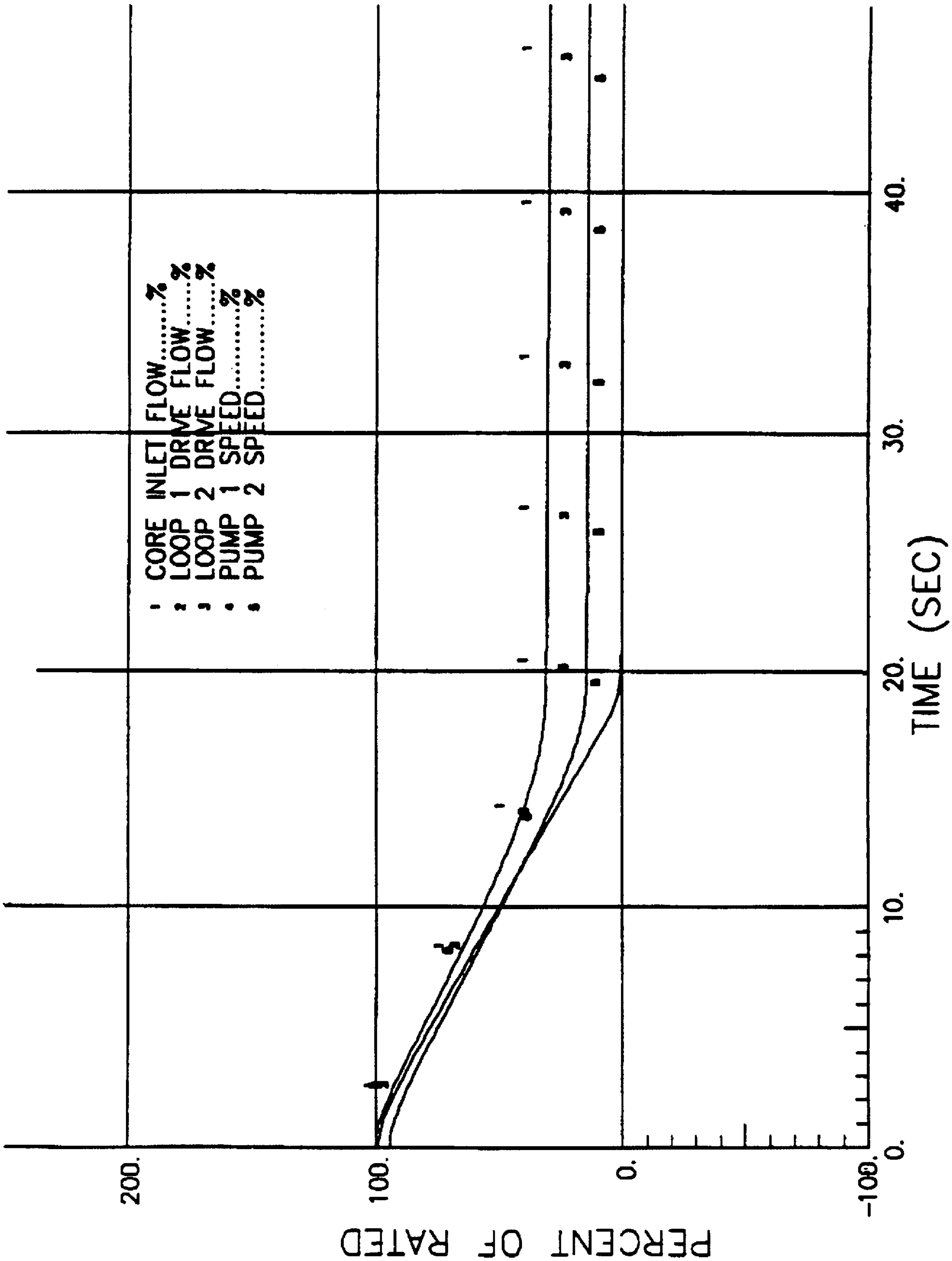
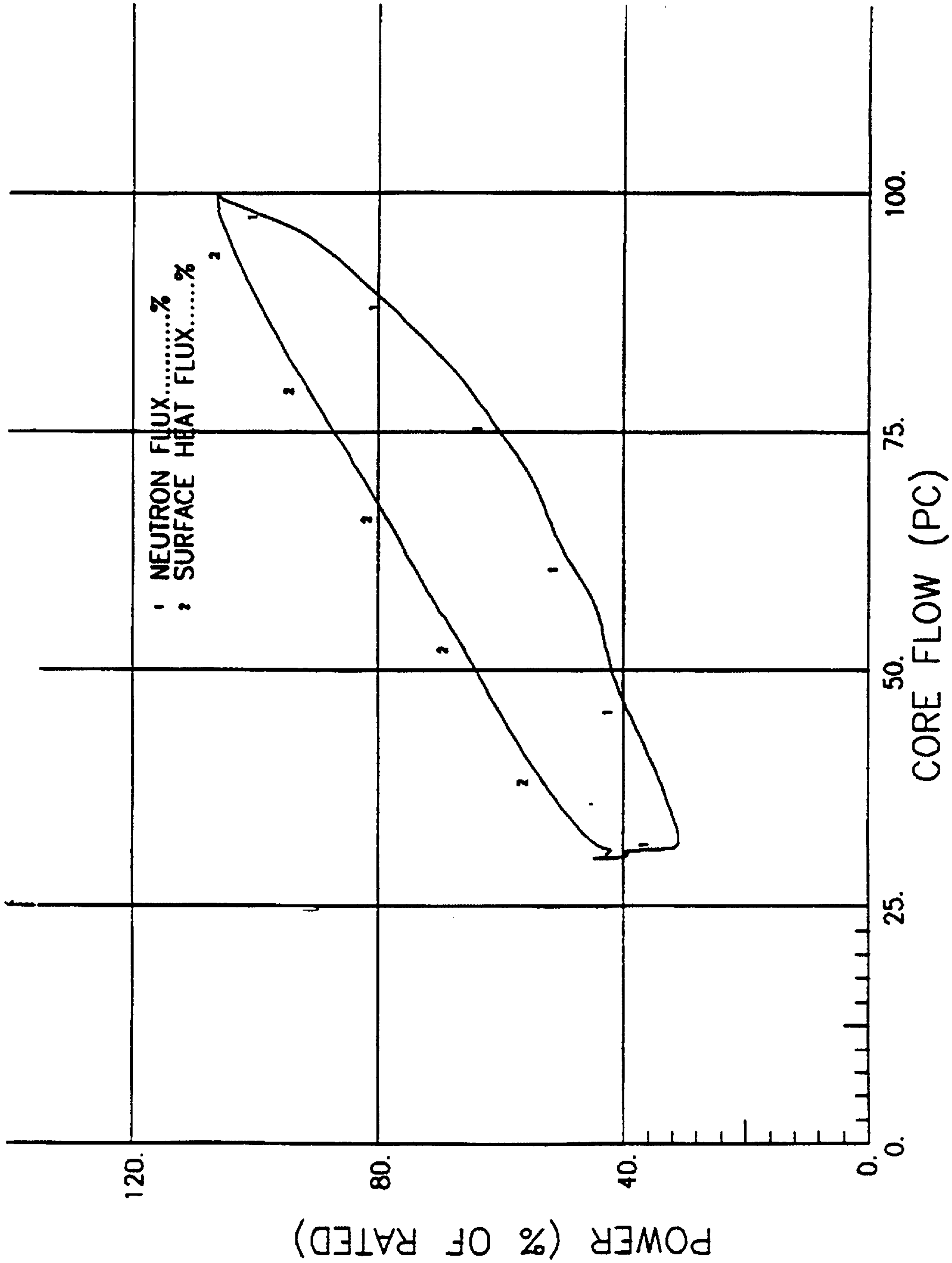


Columbia Generating Station Final Safety Analysis Report	Recirculation Flow Control Failure - Decreasing Flow in Two Loops, (5%/Sec Ramp) at 106.2% Uprated Power, 100% Flow		
	Draw. No. 020361.94	Rev.	Figure 15.3-4.3

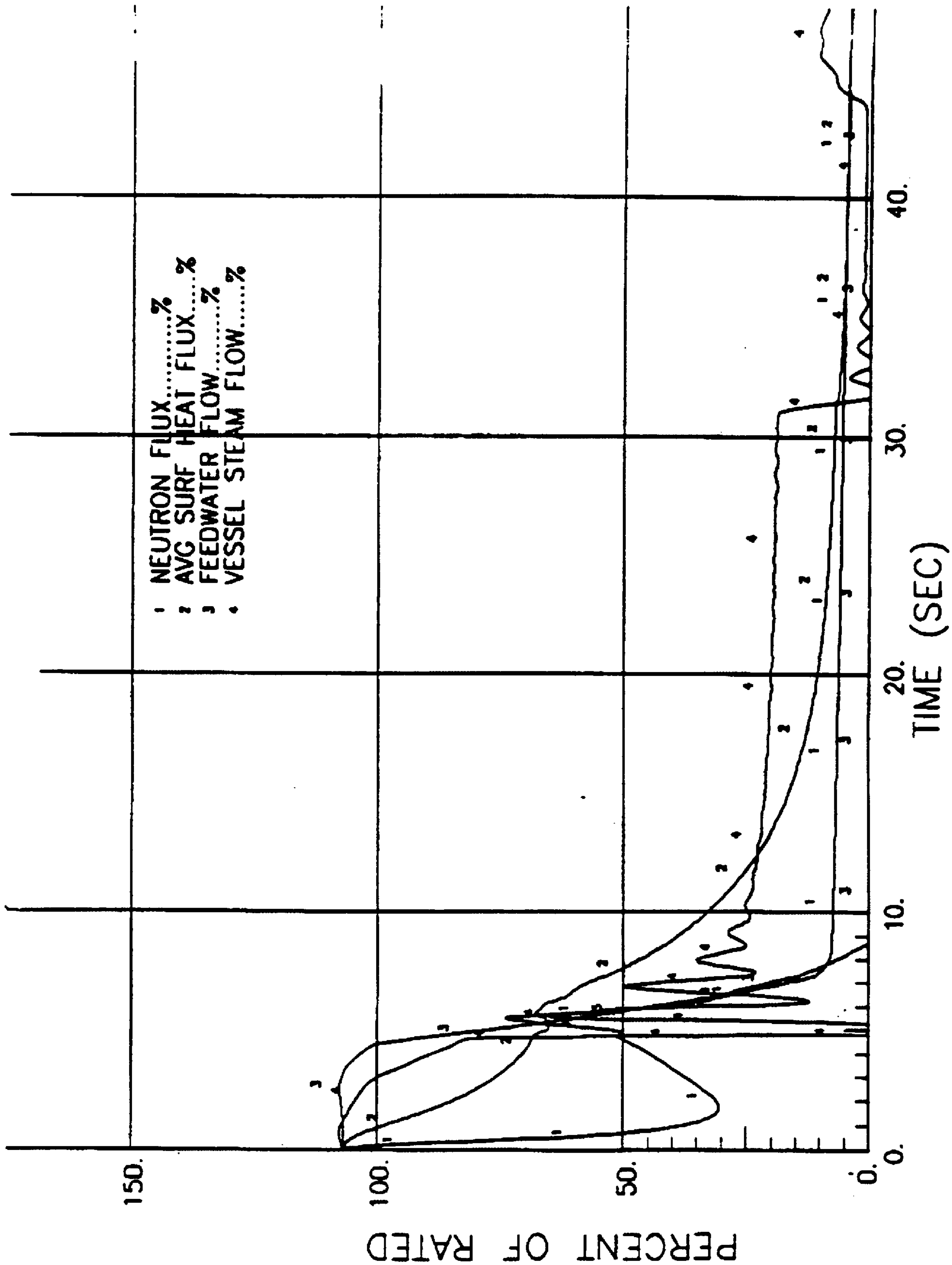


Recirculation Flow Control Failure - Decreasing Flow in Two Loops, (5%/Sec Ramp) at 106.2% Uprated Power, 100% Flow

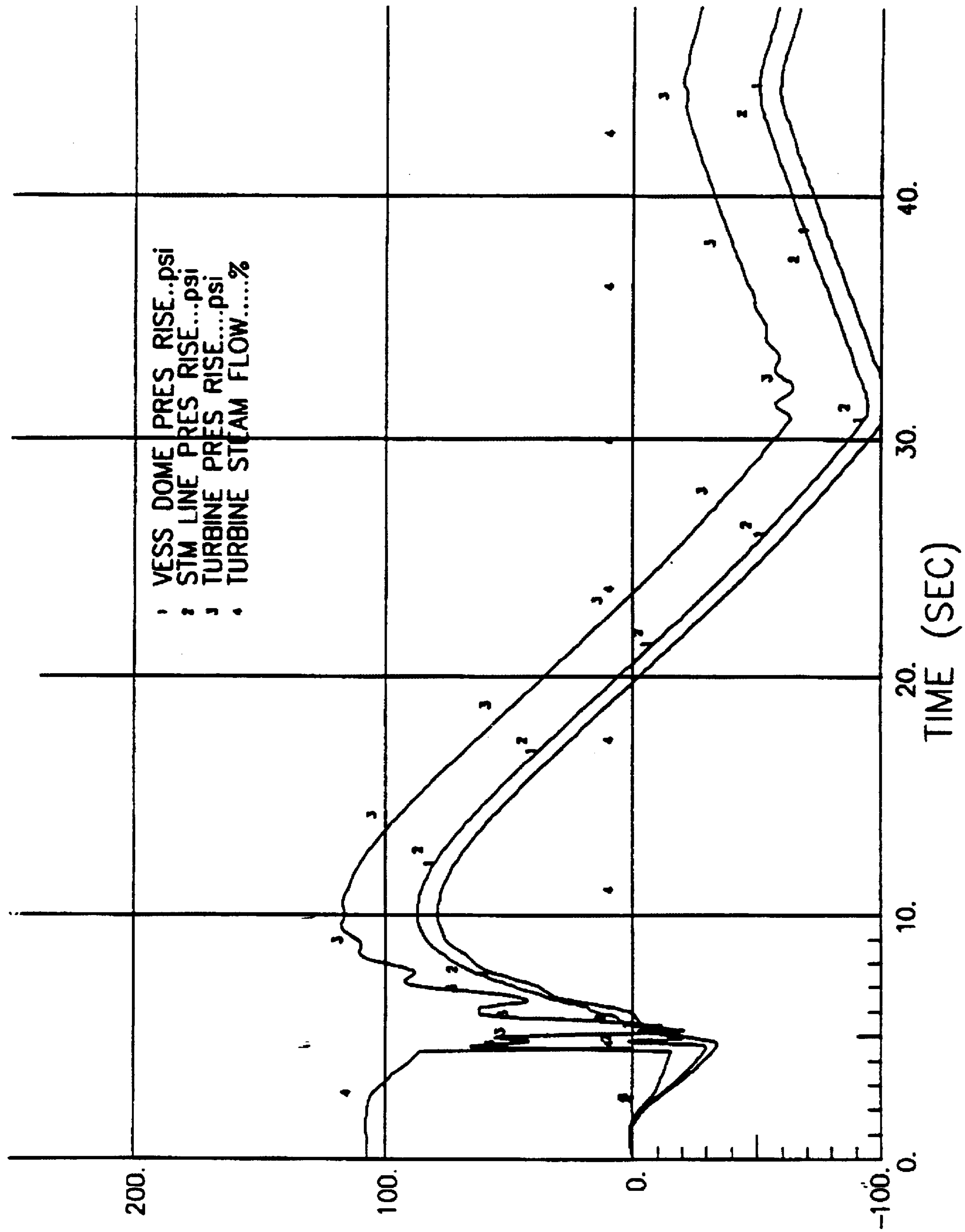
Draw. No. 020361.95	Rev.	Figure 15.3-4.4



Columbia Generating Station Final Safety Analysis Report	Recirculation Flow Control Failure - Decreasing Flow in Two Loops, (5%/Sec Ramp) at 106.2% Uprated Power, 100% Flow		
	Draw. No. 020361.96	Rev.	Figure 15.3-4.5

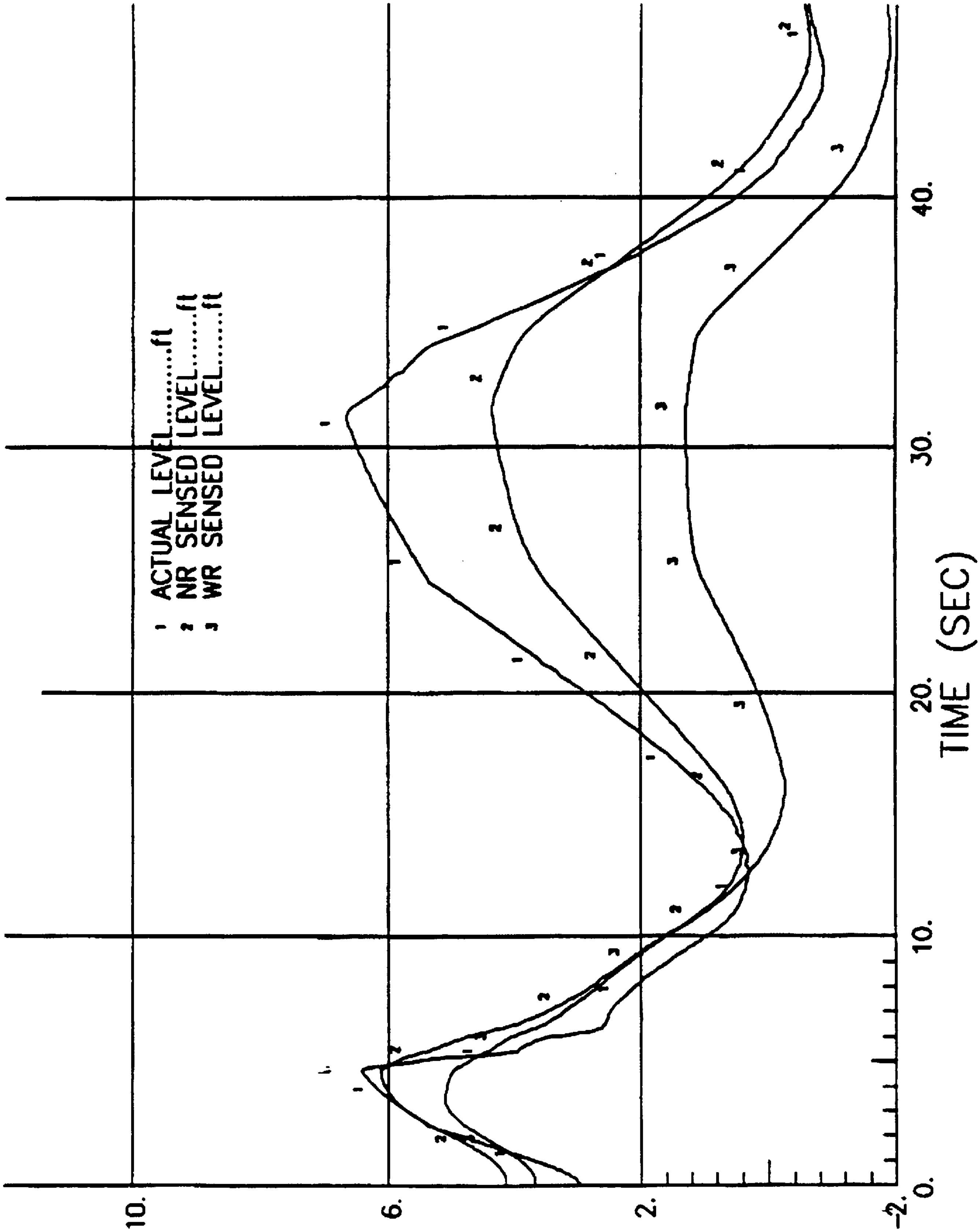


Columbia Generating Station **Final Safety Analysis Report**	**Recirculation Pump Seizure at 106% Uprated Power, 100% Flow**
	Draw. No. 020361.97 | Rev. | Figure 15.3-5.1



Columbia Generating Station Final Safety Analysis Report	Recirculation Pump Seizure at 106% Uprated Power, 100% Flow	
	Draw. No. 020361.98	Rev.
		Figure 15.3-5.2

Amendment 53
November 1998

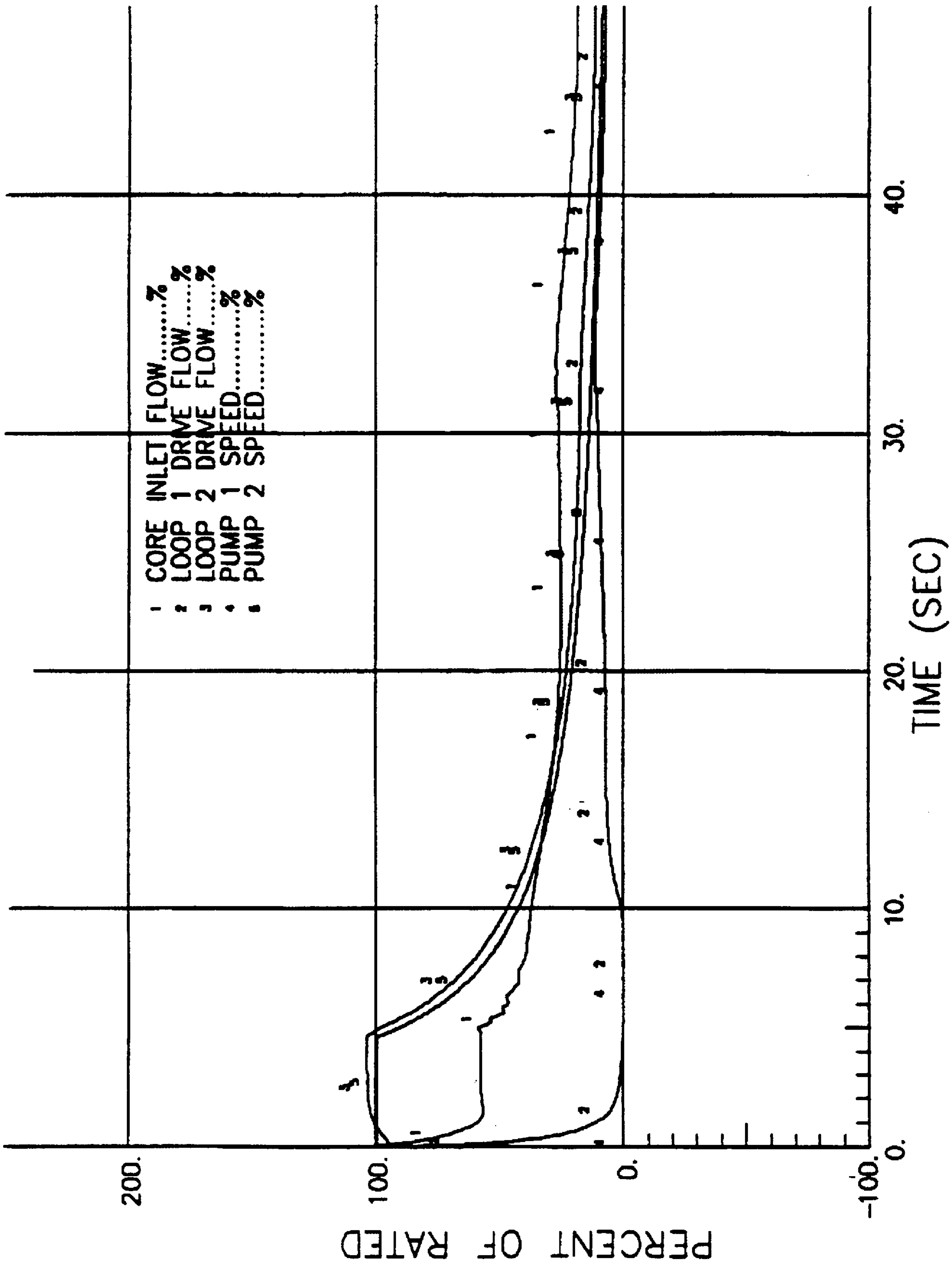


Columbia Generating Station
Final Safety Analysis Report

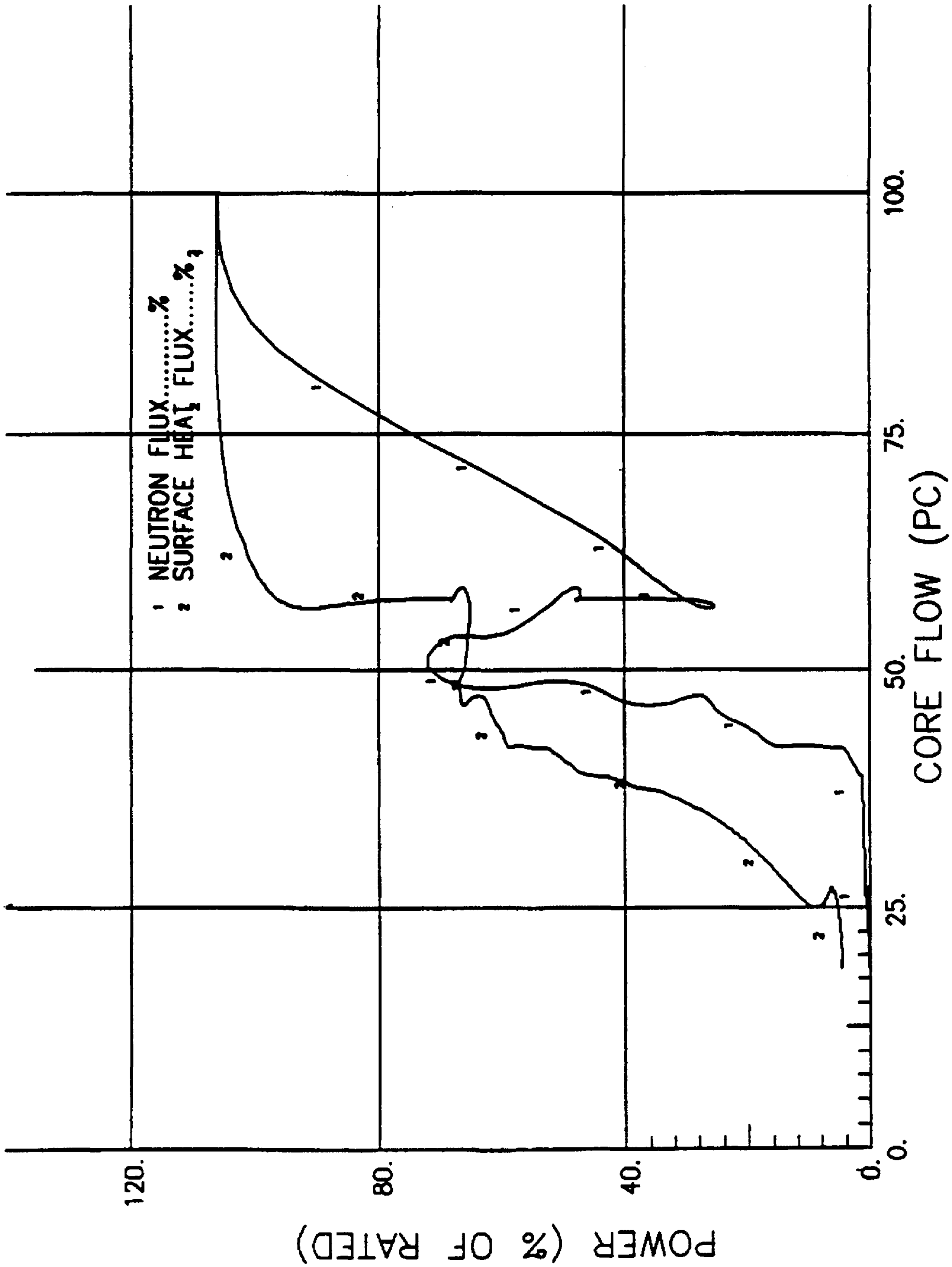
Recirculation Pump Seizure at 106% Uprated Power, 100% Flow

Draw. No. 020361.99	Rev.	Figure 15.3-5.3

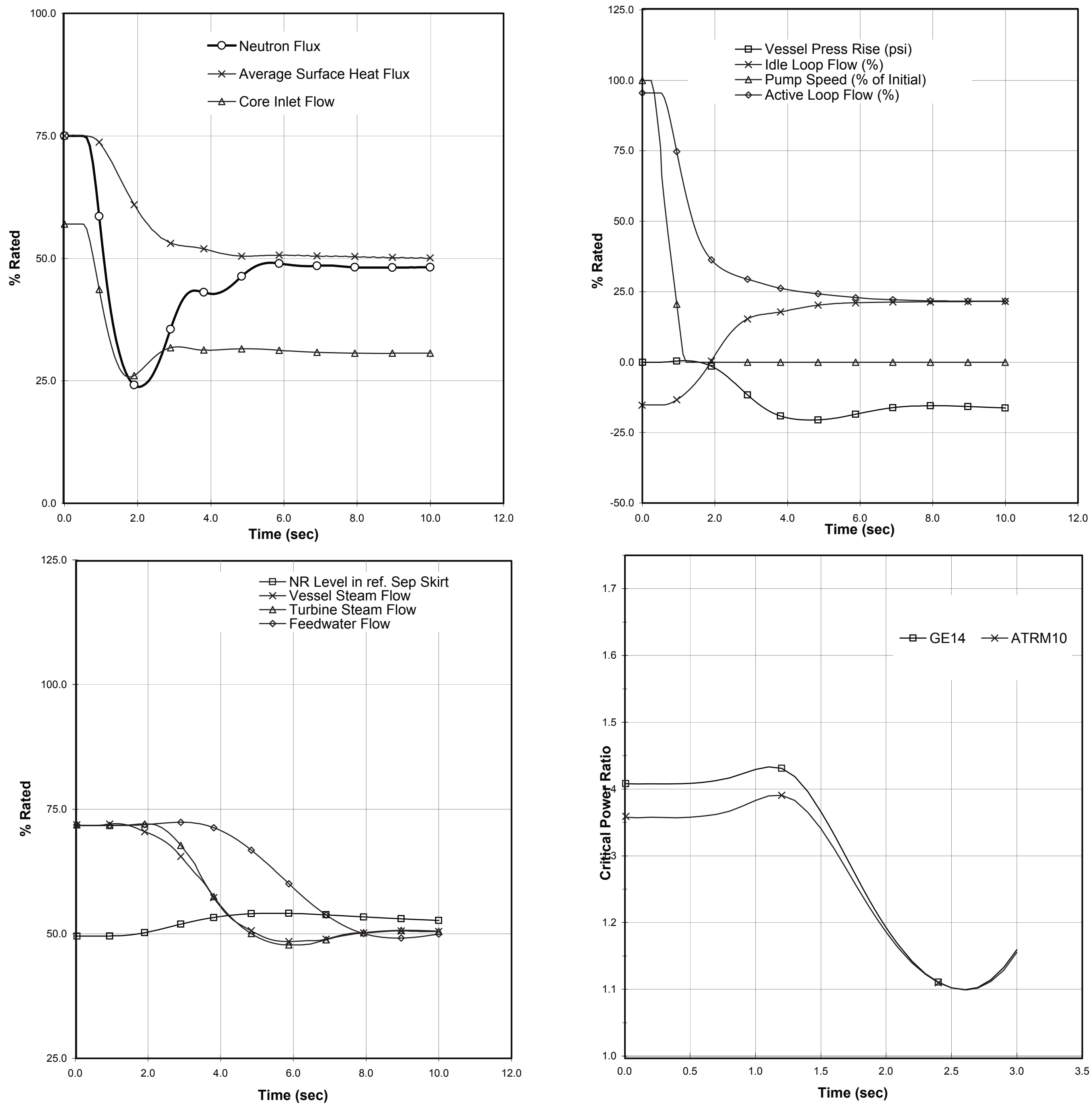
Form No. 960690



Columbia Generating Station Final Safety Analysis Report	Recirculation Pump Seizure at 106% Uprated Power, 100% Flow		
	Draw. No. 020002.01	Rev.	Figure 15.3-5.4



Columbia Generating Station Final Safety Analysis Report	**Recirculation Pump Seizure at 106% Uprated Power, 100% Flow**
	Draw. No. 020002.02 · Rev. · Figure 15.3-5.5


Neutron Flux
Average Surface Heat Flux
Core Inlet Flow
% Rated
Time (sec)
Vessel Press Rise (psi)
Idle Loop Flow (%)
Pump Speed (% of Initial)
Active Loop Flow (%)
% Rated
Time (sec)
NR Level in ref. Sep Skirt
Vessel Steam Flow
Turbine Steam Flow
Feedwater Flow
% Rated
Time (sec)
GE14
ATRM10
Critical Power Ratio
Time (sec)

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

15.4.1 ROD WITHDRAWAL ERROR - LOW POWER

This transient is classified as a nonlimiting event for both original and uprated power conditions. Furthermore, the low power Rod Withdrawal Error (RWE) is not affected by power uprate and therefore, the following qualitative analysis is valid for power uprate.

15.4.1.1 Control Rod Removal Error During Refueling

15.4.1.1.1 Identification of Causes and Frequency Classification

The event considered 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 its being categorized as an infrequent incident since there is no postulated set of circumstances which results in an inadvertent control rod withdrawal error (RWE) while in the refuel mode.

15.4.1.1.2 Sequence of Events and Systems Operation

15.4.1.1.2.1 Initial Control Rod Removal. During refueling operations, safety system interlocks provide assurance that inadvertent criticality does not occur because a control rod was removed or is withdrawn in coincidence with another control rod.

15.4.1.1.2.2 Fuel Insertion With Control Rod Removed. To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the core. This requirement is backed up by refueling interlocks on rod withdrawal and 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. 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 the 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. The design of the control rod, incorporating the velocity limiter, does not physically permit the upward removal of the control

rod without the simultaneous or prior removal of the four adjacent fuel bundles. This precludes any hazardous condition.

15.4.1.1.2.5 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 equipment failure or single operator error, the necessary safety actions are taken (e.g., rod block or scram) automatically prior to violation of any limits.

15.4.1.1.3 Core and System Performances

Since the probability of inadvertent criticality during refueling is precluded, the core and system performances were not analyzed. The withdrawal of the highest worth control rod during refueling will not result in criticality. This is verified experimentally by performing shutdown margin checks. Additional reactivity insertion is precluded by interlocks. As a result, no radioactive material is released from the fuel, making it unnecessary to assess any radiological consequences.

No mathematic 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.

15.4.1.1.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since it is a highly localized event and does not result in any change in the core pressure or temperature.

15.4.1.1.5 Radiological Consequences

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

15.4.1.2 Continuous Rod Withdrawal During Reactor Startup

15.4.1.2.1 Identification of Causes and Frequency Classification

This event is categorized as an infrequent incident. The probability of further development of this event is low because it is contingent upon the failure of the rod worth minimizer (RWM) system or failure of a second licensed operator (or technically qualified member of the technical staff) observing the out-of-sequence rod selection concurrent with a high worth rod, out-of-sequence rod selection contrary to procedures, and operator disregard of continuous alarm annunciations prior to safety system actuation.

15.4.1.2.2 Sequence of Events and Systems Operation

15.4.1.2.2.1 Sequence of Events. Control RWEs are not considered credible in the startup and low power ranges. The RWM or second licensed operator (or other technically qualified member of the technical staff) prevents the operator from selecting and withdrawing an out-of-sequence control rod.

Continuous control RWEs during reactor startup are precluded by the RWM or second qualified person. The RWM or second qualified person prevents the withdrawal of an out-of-sequence control rod from 100% control rod density to 10% of rated thermal power.

15.4.1.2.2.2 Effects of Single Failure and Operator Errors. If any one of the operations involved in the initial failure or error is followed by another single component failure or single operator error, the necessary safety actions are automatically taken to preclude violation of any limits.

15.4.1.2.3 Core and System Performance

The performance of the RWM or second licensed operator (or technically qualified member of the technical staff) prevents erroneous selection and withdrawal of an out-of-sequence control rod. Thus, core and system performance is not affected by such a single 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 applicable.

15.4.1.2.4 Barrier Performance

An evaluation of the barrier performance was not performed 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.

15.4.2 ROD WITHDRAWAL ERROR - AT POWER

15.4.2.1 Identification of Causes and Frequency Classifications

15.4.2.1.1 Identification of Causes

While operating in the power range in a normal mode of operation, the reactor operator makes a procedural error and withdraws the maximum worth control rod until the rod block monitor (RBM) system inhibits further withdrawal.

15.4.2.1.2 Frequency Classification

The probability of this event is considered low enough to warrant its being categorized as an infrequent incident. However, because of the lack of sufficient frequency database, this event is considered an incident of moderate frequency.

15.4.2.2 Sequence of Events and Systems Operation

15.4.2.2.1 Sequence of Events

The sequence of events for this transient is presented in Table 15.4-1.

15.4.2.2.2 Systems Operation

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

While operating in the power range in a normal operational mode, the reactor operator makes a procedural error and withdraws the maximum worth control rod until the RBM system inhibits further withdrawal.

Under normal operating conditions the nearest local power range monitor (LPRM) would detect the peak linear power exceeding design limits and alarm. The operator would acknowledge the alarm and take appropriate action.

If the RWE is severe, the RBM system would alarm, at which time the operator would acknowledge the alarm and take corrective action. Even for conditions such as highly abnormal control rod patterns, operator disregard of all alarms and warnings, and continuous control rod withdrawal, the RBM system will block further withdrawal of the control rod before the fuel reaches the point of boiling transition or the 1% plastic strain limit imposed on the clad.

15.4.2.2.3 Effect of Single Failure and Operator Errors

Operator errors do not impact the consequences of this event due to the single failure proof design of the RBM system.

15.4.2.3 Core and System Performance

15.4.2.3.1 Mathematical Model

The control RWE transient is classified as a "slow transient." A slow transient is a power increase transient that is sufficiently slow so that the assumption that steady-state conditions are achieved at each time step is either realistic or conservative. Using this assumption, this transient is calculated using a steady state, three dimensional, coupled nuclear thermal hydraulics computer program PANACEA. All spatial effects are included in the calculation. A detailed discussion of the code is presented in Reference 15.4-4.

The control RWE analysis has been performed to estimate the minimum critical power ratio (MCPR) and maximum linear heat generation rate (LHGR) in such a transient. A starting control rod pattern is established for the typical BWR reactor and a central control rod is withdrawn from the fully inserted position. Rod withdrawal results in an increase of the LHGR and decrease of the critical power ratio (CPR). The computed maximum LHGR and minimum CPR are compared to values of other transients to establish operating limits for the reactor. The analysis determines the transient MCPR as a function of the rod block monitor setpoint.

15.4.2.3.2 Input Parameters and Initial Conditions

The number of possible RWE transients is large due to the number of control rods and the wide range of exposures and power levels. In order to encompass all of the possible RWEs which could conceivably occur, a limiting analysis is defined such that a conservative assessment of the consequences is provided. These conditions bound the effects of the RWE at lower power or flow conditions, including operation with only one reactor recirculation pump.

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

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

c. The reactor is presumed to be in its most reactive state and devoid of all xenon. This ensures that the amount of reactivity is a maximum;

d. It is assumed that the operator has fully inserted the maximum worth rod prior to its removal and selected the remaining control rod pattern in such a way as to

approach thermal limits in the fuel bundles in the vicinity of the rod to be withdrawn (this control rod configuration would only be achieved by deliberate operator action or by numerous operator errors);

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

f. Of the four LPRM strings nearest to the control rod being withdrawn, the two highest reading LPRMs during the transient are assumed to have failed; and

g. One of the two instrument channels is assumed to be bypassed and out-of-service. The A and C LPRM chambers input to one channel, while the B and D chambers input to the other. The channel with the greatest response is assumed to be bypassed.

15.4.2.3.2.1 Rod Block Monitor System Operation. The RBM system minimizes the consequences of a RWE by blocking motion of the control rod before the safety limits are exceeded.

The RBM has three trip levels (rod withdrawal permissive removed). The trip levels may be adjusted and are nominally 8% of reactor power apart. The highest trip level is set so that the safety limit is not exceeded. The lower two trip levels are intended to provide a warning to the operator. Settings are 106%, 98%, and 90% of initial, steady-state, operating power at 100% flow. The trip levels are automatically varied with reactor coolant flow to protect against fuel damage at lower flows. The variation is set to ensure that no fuel damage will occur at any indicated coolant flow. The operator may encounter any number (up to three) of trip points depending on the starting power of a given control rod withdrawal. The lower two points may be passed up (reset) by manual operation of a push button. The reset permissive is actuated (and indicated by a light) when the RBM reaches 4% power less than the trip point. The operator would then assess his local power and either reset or select a new rod. The highest (power) trip point may not be reset. The reload licensing analysis was performed for a high trip level of 108% to accommodate the hysteresis effect of the trip setpoint at 106%. The corresponding low trip levels are 100% and 92% based on the assumed 8% trip level power difference stated above. All trip levels are in terms of percent of initial steady state operating power at 100% flow.

15.4.2.3.3 Results

At certain core exposures and power/flow conditions, this limiting transient may be a control RWE. Results reflect GE14 fuel introduction, some of which are dependent on fuel design and core loading pattern. Compliance with the event acceptance criteria is demonstrated by cycle-dependent analysis of potentially limiting events just prior to the operation of that cycle. The results are reported in the Supplemental Reload Licensing Report (Reference 15.4-16).

15.4.2.3.4 Considerations of Uncertainties

The conservative assumptions which ensure that this event has been conservatively analyzed have been previously discussed in Section 15.4.2.3.2.

15.4.2.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since this is a localized event with very little change in the gross core characteristics. Typically, an increase in total core power is less than 6% and the changes in pressure are negligible.

15.4.2.5 Radiological Consequences

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

15.4.3 CONTROL ROD MALOPERATION (SYSTEM MALFUNCTION OR OPERATOR ERROR)

This event is covered with the evaluation cited in Sections 15.4.1 and 15.4.2.

15.4.4 STARTUP OF IDLE RECIRCULATION PUMP

15.4.4.1 Identification of Causes and Frequency Classification

15.4.4.1.1 Identification of Causes

This action results directly from the operator's manual action to initiate 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 event is categorized as an incident of moderate frequency.

15.4.4.1.2.2 Abnormal Startup of Idle Recirculation Pump. This event 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 15.4-2 lists the sequence of events for Figure 15.4-1.

15.4.4.2.2 Systems Operation

This event assumes and takes credit for normal functioning of plant instrumentation and controls. No protection systems action is anticipated. No engineered safety feature (ESF) action occurs as a result of the event.

15.4.4.2.3 The Effect of Single Failures and Operator Errors

Attempts by the operator to start the pump at higher power levels will result in a reactor scram on flux.

15.4.4.3 Core and System Performance

15.4.4.3.1 Mathematical Model

The point-kinetics REDY model described in Section 15.0.3.3.1 is used to simulate this event.

15.4.4.3.2 Input Parameters and Initial Conditions

This analysis has been performed while the plant is operating with a single recirculation loop, at 58% uprated power and 34% core flow. Conservatively, the water in the idle loop is assumed to have a minimum temperature of 100°F. The average enthalpy is based on saturated water temperature at the suction inlet with a linear enthalpy gradient to the discharge outlet water temperature of 100°F.

The active recirculation loop is operating with a pump speed that produces about 45% of normal rated jet pump diffuser flow in the active jet pumps. The inactive recirculation loop jet pumps are forward flowing at about 2% of normal jet pump diffuser flow because of natural circulation affects. The core is receiving about 34% of its normal rated flow.

The idle recirculation pump suction and discharge block valves are open. Normal procedure requires leaving an idle loop in this condition to maintain the loop temperature within the required limits for restart.

15.4.4.3.3 Results

The transient response to the incorrect startup of a cold idle recirculation loop is shown in Figure 15.4-1. Shortly after the pump begins to move, the flow from the started jet pump diffusers causes the core inlet flow to increase. The pump startup demand is conservatively assumed to ramp at a rate of 3.3% until maximum pump speed is achieved. The diffuser flows on the started side of the reactor increase ultimately to about 144% of rated while the flow rate of the opposite loop diffusers decreases and eventually reverses to about -8% of

rated. As the inactive loop pump increases speed the cold fluid is pumped out of the recirculation loop piping and is mixed with hot downcomer fluid and the mixture flows to the core with a resulting increase of the core inlet subcooling.

A moderate-duration neutron flux peak to just above 124% of NB rated (NBR = 3486 MWt) is produced as the colder, increasing core flow reduces the void volume. Surface heat flux follows the slower response of the fuel and peaks at 110% of rated before decreasing after the cold water is washed out of the loop at about 30 sec. No damage occurs to the fuel barrier as the MCPR remains substantially above the safety limit.

15.4.4.3.4 Consideration of Uncertainties

This particular transient is analyzed for a maximum pump speed demand signal causing the ASD to adjust the recirculation pump speed upward at a nominal speed demand rate limit. A conservative idle loop temperature is assumed and no other uncertainties were included.

15.4.4.4 Barrier Performance

No evaluation of barrier performance is required for this event since no significant pressure increases are incurred during this transient. See Figure 15.4-1.

15.4.4.5 Radiological Consequences

Since this event does not result in any fuel failures or any release of primary coolant to either the secondary containment or to the environment, there are no radiological consequences associated with this event.

15.4.5 RECIRCULATION FLOW CONTROL FAILURE WITH INCREASING FLOW

15.4.5.1 Identification of Causes and Frequency Classification

15.4.5.1.1 Identification of Causes

An upscale failure of the master manual setpoint station can cause an increase in the core coolant flow rate. Upscale failure of an individual remote manual setpoint station or manual demand loop can also cause an increase in core coolant flow rate.

15.4.5.1.2 Frequency Classification

This event is an incident of moderate frequency.

15.4.5.2 Sequence of Events and Systems Operation

The increase in recirculation flow results in an increase in core flow. The increase in core flow causes an increase in core power level and shifts the power toward the top of the core by reducing the void fraction in the top of the core.

The rate and magnitude of the power increase are dependent on the rate and magnitude of the flow increase. The operator would be expected to control a slow or small increase through normal operating procedures. However, a rapid or significant increase in neutron flux could exceed the high flux scram setpoint and initiate a scram.

This analysis assumes a relatively gradual flow increase that challenges the thermal limits but does not initiate plant protective systems prior to operator action to terminate the transient. The turbine control (governor) valves and possibly the bypass valves open to control reactor pressure. Core power increases until a steady state power level is reached at the maximum recirculation flow. The operator then regains control of the flow control system and returns the plant to a normal operating condition.

The analysis of this event assumes and takes credit for normal functioning of plant instrumentation and controls, and the reactor protection system (RPS). Operation of ESF is not expected.

15.4.5.2.1 The Effect of Single Failures and Operator Errors

The greatest challenge to the thermal limits is the gradual flow increase without actuation of the RPS. The transient is terminated by operator action but not until the maximum core flow of 108.5% rated flow is reached. No actions, either automatic or manual, occur to mitigate the transient prior to event termination at the maximum core flow.

15.4.5.3 Core and System Performance

15.4.5.3.1 Mathematical Model

The core is assumed to be in a pseudo steady state condition in which all plant thermal hydraulics are in equilibrium. The feedwater inlet temperature is assumed to be at its equilibrium value at all power levels during the event. The flow control line used to define the power/flow points represents the steepest attainable during normal reactor operation. The core radial and axial peaking distributions are assumed not to change during the event. The MCPR hot channel analysis along the flow ascension path is calculated with ISCOR (References 15.4-4 and 15.4-14).

Only potentially MCPR limiting fuel is evaluated. Potentially limiting is defined as within 0.10 of the core MCPR for the nominal rated power rodded depletion (typically only fresh and

once burnt fuel). ISCOR is run at the power/flow level corresponding to the plant/cycle specific maximum flow and iterations are performed on the potentially MCPR limiting fuel channel radial peaking factor(s) such that the hot channel MCPR is equal to the MCPR safety limit (SLMCPR) ±0.005. ISCOR is then run along the specified flow control line at a range of core flows from 30% to 100% to obtain the potentially MCPR limiting fuel MCPR value(s) for each case. The results of this analysis determine the flow-dependent MCPR limits to assure that the MCPR will not fall below the SLMCPR for a flow increase event. These analyses bound final feedwater temperature reduction (FFWTR) as well as normal feedwater temperature conditions.

15.4.5.3.2 Input Parameters and Initial Conditions

The gradual increase in recirculation flow provides the greatest challenge to thermal limits. The final point in the transient is a power below the high flux scram setpoint at maximum flow. Maximum flow is the maximum flow that can be attained by a credible controller failure initiated from a given low flow starting point. A spectrum of initial, low flow starting points is analyzed at various points throughout the cycle. The analysis assumes that the event is quasi-steady state and that a flow biased scram does not occur. Table 15.4-3 contains a listing of the important input parameters and initial conditions.

15.4.5.3.3 Results

The reduced flow MCPR was calculated at discrete flow points. The reduced flow MCPR operating limit curve is shown in the cycle specific Core Operating Limits Report (COLR) for all cycle exposures, including FFWTR operation.

15.4.5.3.4 Considerations of Uncertainties

The conservative nature of the analysis approach bounds the uncertainties in void reactivity and power distribution characteristics expected for actual plant conditions.

15.4.5.4 Barrier Performance

The reduced flow MCPR is established so that the event does not challenge the safety limit MCPR. Therefore, no fuel damage is predicted as a result of this event.

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

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

15.4.7 MISPLACED BUNDLE ACCIDENT

The fuel loading error considers the consequences of either of two possible events: misorientation or mislocation of a fuel assembly. Further, the assumption is made that the error is not discovered during core verification. The purpose of the analysis is to determine the change in the minimum CPR and increase in LHGR between the correctly loaded core and the misloaded core. A combination of the misorientation and mislocation is not considered because of the very low probability of occurrence.

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. For the mislocation of a fuel bundle, three errors must occur during the initial core loading. First, a bundle must be misloaded into a wrong position in the core. Second, the bundle that was supposed to be loaded where the mislocation occurred would have to be placed in an incorrect location. Third, the misplaced bundles would have to be overlooked during the core verification performed following core loading. For the misorientation of a fuel bundle, the bundle is loaded 180° from the correct orientation and this error is overlooked during the core verification.

A fuel loading error would place a fuel assembly in an incorrect location in the core, potentially placing several highly reactive assemblies in close proximity. If a relatively high reactivity assembly is placed in a location not directly monitored by the LPRM/core monitoring system (i.e., unmonitored location), this incorrectly located assembly will operate at higher powers with reduced thermal margins relative to the symmetric monitored assembly. The incorrectly located assembly may violate operating limits if the symmetric, monitored assembly is operated close to limits. If the incorrect location is a directly monitored cell, the change in the local power readings may not be sufficient to either alert the operators of the loading error or to completely account for the reduction in thermal margin. In this situation, the incorrectly located assembly may violate operating limits while being treated by the monitoring system as if it were correctly loaded.

15.4.7.1.2 Frequency of Occurrence

This event is categorized as an infrequent incident but is analyzed for GNF reloads consistent with Section 15.3.3.1 as an incident of moderate frequency.

15.4.7.2 Sequence of Events and Systems Operation

A fuel bundle is misloaded (incorrect location or orientation) into the core and the error is not identified during the core verification process. The core is operated through the cycle at the conditions assumed for the reference core (the specified core load) with the control rod sequence developed for the reference core. At some point during the cycle, the control rod sequence places the limiting assembly (being monitored by the core monitoring system as a correctly loaded assembly) at the MCPR operating limit. Potentially, this causes the misloaded assembly to be operated below the MCPR operating limit curve and above the design LHGR limit curve. Because the operator may be unable to detect the error, the core operation continues throughout the cycle.

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 functioning occurs.

15.4.7.2.1 Effect of Single Failure and Operator Errors

This analysis already represents the worst case (i.e., operation with a misplaced bundle requires multiple equipment failures or operator errors).

15.4.7.3 Core and System Performance

15.4.7.3.1 Mathematical Model

A three-dimensional BWR simulator model is used to calculate the core performance resulting from this event. This model is described in detail in Reference 15.4-4 and Sections S.2.2.1.8 and S.2.2.1.9 of Reference 15.4-5.

15.4.7.3.2 Input Parameters and Initial Conditions

Initial input parameters and conditions are cycle specific. Sections S.2.2.1.8 and S.2.2.1.9 of Reference 15.4-5 describe the input parameters and initial conditions applied to the mislocated and misoriented bundle events. For both the mislocated and misoriented bundle events, the fuel loading error is undetected throughout the cycle by the core monitoring system.

15.4.7.3.3 Results

The results of the analyses for the fuel loading errors show that the resulting MCPR does not challenge the SLMCPR. No rods are expected to exceed the LHGR limits. Results reflect GE14 fuel introduction, some of which are dependent on fuel design and core loading pattern. Compliance with the event acceptance criteria is demonstrated by cycle-dependent analysis of

potentially limiting events just prior to the operation of that cycle. The results are reported in the Supplemental Reload Licensing Report (Reference 15.4-16).

15.4.7.3.4 Considerations of Uncertainties

A sufficient number of mislocated assembly cases are considered to assure that the limiting case is evaluated. The mislocated assemblies are loaded into positions that could produce limiting results. For the misoriented assembly case, the gap sizes resulting from the rotation are selected to assure a conservative estimate of the impact on MCPR.

15.4.7.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since it is a mild and highly localized event. No perceptible change in the core pressure would 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 from the fuel.

15.4.8 SPECTRUM OF ROD EJECTION ASSEMBLIES

This event is not applicable to BWR plants.

15.4.9 CONTROL ROD DROP ACCIDENT

15.4.9.1 Identification of Causes and Frequency Classification

15.4.9.1.1 Identification of Causes

The control rod drop accident (CRDA) is the result of a high worth control rod decoupled from the drive mechanism, dropping out of the core. The subsequent insertion of positive reactivity causes a localized power excursion. This is not an anticipated event because of the system failures and personnel errors that would have to occur in combination to present the reactivity required at the same time that the coupling failed. The control rod patterns are controlled in accordance with the banked position withdraw sequence (BPWS) to preclude situations in which rod drops would have sufficient reactivity to cause the damage postulated by the power excursion.

Detailed discussions of the rod drop analysis and BPWS are given in Reference 15.4-3 and 15.4-7.

15.4.9.1.2 Frequency Classification

The CRDA is categorized as a limiting fault because it is not expected to occur during the lifetime of the plant. However, postulated consequences include the potential for the release of radioactive material.

For reactivity anomalies, the CRDA is the limiting event.

15.4.9.2 Sequence of Events and Systems Operation

The CRDA assumptions include:

1. At some time, a fully inserted control rod becomes decoupled from its drive and sticks in the fully inserted position.

2. During the start up sequence, the rod patterns employed are permitted by the constraints on rod movements by technical specifications and the rod sequence control hardware, including the maximum allowable number of bypassed rods. At some time under critical reactor conditions, the rod pattern causes the decoupled rod to have the maximum worth from fully inserted to the position of its drive. The rod worth minimizer is not functioning. The rod drops at that time.

3. The reactor goes on a positive period, and the fuel temperature reactivity feedback terminates the initial power burst.

4. The reactor scrams on the APRM high flux scram signal.

5. All withdrawn rods, except for the decoupled rod, scram at the technical specification rate.

6. The scram terminates the accident.

7. If the mechanical vacuum pump (MVP) is maintaining condenser vacuum (e.g., the plant was operating at 5% power or less) and the main steam line radiation (MSLR) monitors detected radiation levels above the setpoint, the MSLR monitors would trip the MVP to reduce the fission product release from the condenser.

Although other normal plant instrumentation and controls are assumed to function, no credit for their operation is taken in the analysis of this event. No operator actions are required to terminate this event. Subsequent to reactor scram which terminates the event, normal vessel inventory makeup systems will be used as available.

15.4.9.2.1 Effect of Single Failures and Operator Errors

As discussed, the event is terminated, and therefore mitigated, by the APRM high flux scram signal to RPS. The RPS design meets the single failure criteria. The event is further mitigated by an initial control rod configuration that complies with the BPWS. The withdrawal (or insertion) sequence is implemented by the operator and enforced by the RWM. An operator error in control rod movement will be detected and stopped by the RWM. If the RWM system is not operable, rod movement can only continue with a backup for the operator verifying compliance with the BPWS sequence. Failure of the RWM concurrent with an operator error of moving an out-of-sequence rod, contrary to procedures would be required to result in a potentially more limiting event. Therefore, sufficient redundancy exists such that termination of this transient within the limiting criteria is assured.

At low power levels, the MVP trip maintains the condenser leak rates within the analytical assumptions. The MSLR monitor design meets the single failure criteria and no active failure would prevent the trip signal (Section 11.5.2.1).

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

15.4.9.3.1 Mathematical Model

The analytical methods, assumptions and conditions for evaluating the excursion aspects of the CRDA are described in detail in References 15.4-1, 15.4-3, 15.4-13. To limit the worth of the postulated dropped rod, the rod pattern control systems are programmed to follow the BPWS, which is generically defined in Reference 15.4-7.

15.4.9.3.2 Input Parameters and Initial Conditions

The data presented in Reference 15.4-7 shows that BPWS reduces the control rod worths to the degree that the detailed analyses presented in References 15.4-1, 15.4-3, 15.4-13 or the bounding analyses presented in Reference 15.4-11 do not need to be performed each cycle.

15.4.9.3.3 Results

Control rod drop accident results from BPWS plants have been statistically analyzed and documented in Reference 15.4-12. The results show that, in all cases, the peak fuel enthalpy in a CRDA would be much less than the 280 cal/gm design limit even with a maximum incremental rod worth corresponding to 95% probability at the 95% confidence level. Based on these results, it was proposed to the NRC, and subsequently found acceptable, to delete the CRDA from the standard GE BWR reload package for the BPWS plants.

The US Supplement to Reference 15.4-5 reports the results of radiological analyses for initial cores that are orders of magnitude below those identified in 10 CFR 100. The radiological consequences of the CRDA, assuming a full core of more recent GE/GNF fuel designs, are discussed in Reference 15.4-5. With implementation of Alternative Source Term (AST), the radiological release acceptance criterion becomes 10 CFR 50.67. An evaluation of fuel damage was performed because the maximum deposited fuel rod enthalpy exceeded 170 cal/gm, which is the enthalpy limit assumed for eventual cladding perforation. The number of fuel rods predicted to fail (Reference 15.4-15) are bounded by the number assumed in the radiological consequences analysis for this event.

The total energy deposited and the associated increase in reactor system pressure during a rod drop event is not high relative to other events such as turbine trip without bypass or main steam line isolation valve closure, both of which are quantitatively analyzed. As such, the increase in reactor system pressure is not anticipated to result in penetration of the stress limits defined in Section III of the ASME boiler and pressure code.

15.4.9.4 Barrier Performance

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

15.4.9.5 Radiological Consequences

The radiological analysis is based on the AST described in Reference 15.4-8. Specific models, assumptions, and the program used for computer evaluation are described in Reference 15.4-6. Specific parametric values used in the evaluation are presented in Table 15.4-4. The radiological consequences remain bounding because cycle specific analyses have confirmed that the number of fuel rods with an enthalpy greater than the threshold for fuel failure are well below the number assumed in the analysis.

15.4.9.5.1 Fission Product Release from Fuel

The failure of 1.8% of the core was assumed for this analysis. The mass fraction of the fuel in the damaged rods which reaches or exceeds the initiation temperature of fuel melting (taken as 2804°C) is assumed to be 0.0077.

Fuel reaching melt condition is assumed to release 100% of the noble gas inventory and 50% of the iodine inventory. The remaining fuel rods with clad damage only (no melting), will undergo a gap release which is assumed to release 10% of noble gases and 10% of iodine inventories.

The core inventory of fission products is based on a plant-specific ORIGEN 2 run for pre-power uprate basis of 3489 MW with 1000 days of exposure, adjusted as follows:

- A scale factor of 1.0528 to bound the dose impact of power level to 3556 MWt,
- A correction to increase by 25 percent short-lived krypton values (based on comparisons to other core inventory tables), and
- An increase by 60% in the activity of longer lived isotopes to bound longer plant operation at a higher burnup rate.

The assumed core power of 3556 MWt is the licensed power increased by 2% to account for power measurement uncertainties. These adjustments resulted in a conservative source term (in terms of activity available). A peaking factor of 1.7 is assumed and no delay time is considered between departure from that power condition and the initiation of the accident.

15.4.9.5.2 Fission Product Transport to the Environment

The transport pathway consists of a release from the core to the coolant, carryover with steam to the turbine condenser and leakage from the condenser to the environment. The release fractions are given in Reference 15.4-6 and are consistent with Reference 15.4-8. No credit is taken for mixing in the turbine building or filtration by the CREF.

Of the activity reaching the condenser, 100% of the noble gases, and 10% of the iodine (due to partitioning and plate-out) and 1% of the particulates remain airborne and are available for release to the environment. The activity airborne in the condenser leaks to the environment as a ground level release at a rate of 1% of condenser volume per day. Release from the condenser is assumed to terminate 24 hours following the onset of the accident. Radioactive decay is accounted for during residence in the condenser; however, it is neglected after release to the environment. If the condenser is not isolated from the offgas system, the activity is processed through the offgas system. In this case, radioactive decay is accounted for during the residence in the offgas system. Response of the offgas system to elevated radiation levels is described in Section 11.3.2.4.5.

The activity airborne in the condenser is presented in Table 15.4-5. The cumulative release of activity to the environment is presented in Table 15.4-6.

15.4.9.5.3 Results

The calculated exposures from the design basis analysis are presented in Table 15.4-7 and are within the limits of 10 CFR 50.67.

15.4.10 REFERENCES

15.4-1 Stirn, R. C., "Rod Drop Accident Analysis for Large Boiling Water Reactors," NEDO-10527, March 1972.

15.4-2 GE Nuclear Energy, "WNP-2 Power Uprate Transient Analysis Task Report," GE-NE-208-08-0393, September 1993 (Proprietary).

15.4-3 Stirn, R. C., "Rod Drop Accident Analysis for Large BWRs," Supplement 1, NEDO-10527, July 1972.

15.4-4 "Steady -State Nuclear Methods," NEDE-30130-P-A, April 1985.

15.4-5 "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A and "Supplement for United States," NEDE-24011-P-A-US (most recent approved version referenced in COLR).

15.4-6 Energy Northwest, "Columbia Generating Station Alternative Source Term," CGS-FTS-0168, Revision 0, August 2007.

15.4-7 Paone, C. J., "Bank Position Withdrawal Sequence," NEDO-21231.

15.4-8 NRC Regulatory Guide 1.183, "Alternative Source Term for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

15.4-9 Deleted.

15.4-10 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, Class I, October 1992.

15.4-11 "GE BWR Generic Reload Application for 8x8 Fuel," Supplement 3 to Revision 1, NEDO-20360.

15.4-12 Letter from R. E. Engel (GE) to D. M. Vassallo (NRC), "Elimination of Control Rod Drop Accident Analysis for Banked Position Withdrawal Sequence Plants," February 24, 1982.

15.4-13 Stirn, R. C., "Rod Drop Accident Analysis for Large Boiling Water Reactors Addendum No. 2 Exposed Cores," Supplement 2, NEDO-10527, January 1973.

15.4-14 Letter from D. G. Eisenhut (NRC) to R. L. Gridley (GE), "Safety Evaluation for the General Electric Topical Report, Generic Reload Fuel Application (NEDE-24011-P)," May 12, 1978, MFN-212-78.

15.4-15 "GE14 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR II)," NEDC-32868P, Revision 4, January 2012.

15.4-16 Supplemental Reload Licensing Report for Columbia (most recent version referenced in COLR).

Table 15.4-1

Sequence of Events - Rod Withdrawal Error in Power Range

Time (sec)[a]	Event
0	Core is assumed to be operating at rated conditions.
0	Operator selects and withdraws the maximum worth control rod.
1	The total core power and the local power in the vicinity of the control rod increase.
5	The LPRM system indicates excessive localized peaking.
5	The operator ignores warning and continues withdrawal.
15	The RBM system indicates excessive localized peaking.
15	The operator ignores warning and continues withdrawal.
20	The RBM system initiates a rod block inhibiting further withdrawal.
40	Reactor core stabilizes at higher core power level.
60	Operator re-inserts control rod to reduce core power level.
80	Core stabilizes at rated conditions.

[a] Approximately.

Table 15.4-2

Sequence of Events for an Abnormal Startup
of an Idle Recirculation Loop

Time (sec)	Event
0.00	Plant operating with one recirculation loop only.
5.00	Start idle recirculation loop pump motor.
29.6	Peak value of core inlet subcooling.
29.6	Peak thermal power. Estimated APRM thermal power approximately 1% below APRM thermal power setpoint.
45.7	Pump motor at full speed.
80+	Reactor reaches new equilibrium condition.

Table 15.4-3

Reactor Recirculation Pump Flow Increase
Input Parameters and Initial Conditions

Parameter	Value
Reactor Power/Core Flow	Flow increase is initiated from several power/flow points and terminates at 111% rated power, 108.5%rated flow
Power Distribution	The MCPR equals the safety limit at the final power/flow condition
Reactivity	The results are applicable from Beginning of Cycle to End of Cycle for nominal and reduced feedwater temperatures
Control Rod Configuration	The control rod pattern is the same at the initial and final points.

Table 15.4-4

Control Rod Drop Accident Evaluation Parameters

I.	Data and assumptions used to estimate radioactive source from postulated accidents.		
	A.	Power level	Section 15.4.9.5.1
	B.	Burnup	Section 15.4.9.5.1
	C.	Fuel damaged	1.8% of core
	D.	Release of activity by nuclide	Table 15.4-6
	E.	Iodine fractions	
		(1) Organic	0.0015
		(2) Elemental	0.0485
		(3) Particulate	0.95
	F.	Reactor coolant activity before the accident.	N/A
II.	Data and assumptions used to estimate activity released.		
	A.	Condenser leak rate (%/day)	1.0
	B.	Turbine building leak rate (%/day)	N/A
	C.	Valve closure time (sec)	N/A
	D.	Adsorption and filtration efficiencies	
		(1) Organic iodine	N/A
		(2) Elemental iodine	N/A
		(3) Particulate iodine	N/A
		(4) Particulate fission products	N/A
	E.	Recirculation system parameters	
		(1) Flow rate	N/A
		(2) Mixing efficiency	N/A
		(3) Filter efficiency	N/A
	F.	Containment spray parameters (flow rate, drop size, etc.)	N/A

Table 15.4-4

Control Rod Drop Accident Evaluation Parameters (Continued)

	G.	Containment volumes	N/A
	H.	All other pertinent data and assumptions	None
III.	Dispersion data		Table 15.0-4
IV.	Dose data		
	A.	Method of dose calculation	Reference 15.4-6
	B.	Dose conversion assumptions	Reference 15.4-6
	C.	Peak activity concentrations in condenser	Table 15.4-5
	D.	Doses	Table 15.4-7

Table 15.4-5

Control Rod Drop Accident
Activity Airborne in the Condenser (Curies)

3556 MWth

	1 m	30 m	2 hrs	8 hrs	12 hrs	24 hrs
Kr83m	4.12E+04	3.43E+04	1.96E+04	2.06E+03	4.61E+02	5.13E+00
Kr85m	8.50E+04	7.88E+04	6.21E+04	2.40E+04	1.27E+04	1.90E+03
Kr85	4.77E+03	4.77E+03	4.76E+03	4.75E+03	4.74E+03	4.72E+03
Kr87	1.54E+05	1.18E+05	5.20E+04	1.95E+03	2.18E+02	3.05E-01
Kr88	2.19E+05	1.95E+05	1.34E+05	3.03E+04	1.12E+04	5.72E+02
Kr89	2.05E+05	3.71E+02	1.14E-06	6.96E-18	1.01E-16	7.70E-18
Xe131m	3.24E+03	3.23E+03	3.22E+03	3.16E+03	3.13E+03	3.02E+03
Xe133m	1.93E+04	1.91E+04	1.88E+04	1.74E+04	1.65E+04	1.41E+04
Xe133	6.30E+05	6.28E+05	6.23E+05	6.01E+05	5.87E+05	5.47E+05
Xe135m	1.23E+05	3.40E+04	6.24E+02	7.12E-05	1.68E-09	2.52E-16
Xe135	1.52E+05	1.46E+05	1.31E+05	8.30E+04	6.13E+04	2.47E+04
Xe137	4.52E+05	2.62E+03	2.99E-04	4.13E-18	3.26E-17	2.41E-17
Xe138	4.00E+05	1.22E+05	3.11E+03	1.30E-03	7.24E-08	2.47E-18
Total noble gases	2.49E+06	1.39E+06	1.05E+06	7.68E+05	6.97E+05	5.96E+05
I-131*	3.06E+05	2.64E+05	1.83E+05	8.50E+04	6.15E+04	2.47E+04
I-132	6.71E+05	1.98E+05	1.34E+05	3.03E+04	1.12E+04	5.72E+02
I-133	6.05E+05	1.22E+05	3.11E+03	1.30E-03	7.24E-08	1.02E-17
I-134	3.24E+03	3.23E+03	3.22E+03	3.16E+03	3.13E+03	3.02E+03
I-135	2.51E+06	1.41E+06	1.07E+06	7.85E+05	7.14E+05	6.10E+05
Total Iodine	4.09E+06	1.99E+06	1.39E+06	9.03E+05	7.90E+05	6.38E+05

* The isotopic iodine activity is the sum of the elemental and organic iodines with 97% elemental and 3% organic.

The particulate iodine comprise 0%.

Table 15.4-5

Control Rod Drop Accident
Activity Airborne in the Condenser (Curies) (Continued)

3556 MWth

	1 m	30 m	2 hrs	8 hrs	12 hrs	24 hrs
Rb86	5.87E-02	5.86E-02	5.84E-02	5.77E-02	5.73E-02	5.60E-02
Cs134	8.23E+00	8.23E+00	8.22E+00	8.20E+00	8.18E+00	8.14E+00
Cs136	1.82E+00	1.82E+00	1.81E+00	1.79E+00	1.77E+00	1.71E+00
Cs137	6.63E+00	6.63E+00	6.62E+00	6.61E+00	6.59E+00	6.56E+00
Sb127	1.38E-02	1.38E-02	1.36E-02	1.30E-02	1.26E-02	1.14E-02
Sb129	3.95E-02	3.66E-02	2.88E-02	1.10E-02	5.84E-03	8.61E-04
Te127m	1.95E-03	1.95E-03	1.94E-03	1.94E-03	1.93E-03	1.91E-03
Te127	1.38E-02	1.38E-02	1.38E-02	1.36E-02	1.34E-02	1.25E-02
Te129m	5.81E-03	5.80E-03	5.79E-03	5.75E-03	5.72E-03	5.63E-03
Te129	3.72E-02	3.74E-02	3.43E-02	1.52E-02	8.10E-03	1.20E-03
Te131m	1.75E-02	1.73E-02	1.67E-02	1.45E-02	1.32E-02	9.97E-03
Te132	1.67E-01	1.66E-01	1.64E-01	1.55E-01	1.49E-01	1.33E-01
Ba137m	1.54E+00	6.62E+00	6.62E+00	6.61E+00	6.59E+00	6.56E+00
Ba139	7.82E-02	6.14E-02	2.90E-02	1.43E-03	1.94E-04	4.75E-07
Ba140	7.65E-02	7.64E-02	7.61E-02	7.49E-02	7.41E-02	7.17E-02
Mo99	1.02E-02	1.02E-02	1.00E-02	9.39E-03	8.99E-03	7.90E-03
Tc99m	9.06E-03	9.12E-03	9.27E-03	9.45E-03	9.33E-03	8.55E-03
Ru103	9.81E-03	9.81E-03	9.79E-03	9.72E-03	9.68E-03	9.55E-03
Ru105	7.21E-03	6.69E-03	5.33E-03	2.14E-03	1.16E-03	1.87E-04
Ru106	4.26E-03	4.26E-03	4.26E-03	4.24E-03	4.23E-03	4.21E-03
Rh105	6.83E-03	6.83E-03	6.80E-03	6.42E-03	6.04E-03	4.87E-03
Y90	3.50E-05	6.38E-05	1.52E-04	4.90E-04	7.02E-04	1.28E-03
Y91	4.56E-04	4.66E-04	4.95E-04	5.82E-04	6.22E-04	6.87E-04
Y92	6.39E-04	4.76E-03	1.26E-02	1.28E-02	7.81E-03	1.10E-03
Y93	5.94E-04	5.74E-04	5.18E-04	3.42E-04	2.59E-04	1.13E-04
Zr95	7.13E-04	7.13E-04	7.12E-04	7.08E-04	7.06E-04	6.98E-04
Zr97	7.23E-04	7.09E-04	6.66E-04	5.21E-04	4.42E-04	2.70E-04

Table 15.4-5

Control Rod Drop Accident
Activity Airborne in the Condenser (Curies) (Continued)

3556 MWth

	1 m	30 m	2 hrs	8 hrs	12 hrs	24 hrs
Nb95	7.13E-04	7.13E-04	7.13E-04	7.11E-04	7.10E-04	7.06E-04
La140	8.08E-04	1.43E-03	3.33E-03	1.04E-02	1.46E-02	2.54E-02
La141	7.26E-04	6.66E-04	5.10E-04	1.75E-04	8.58E-05	1.01E-05
La142	6.91E-04	5.55E-04	2.81E-04	1.84E-05	3.00E-06	1.29E-08
Pr143	6.31E-04	6.32E-04	6.35E-04	6.44E-04	6.49E-04	6.58E-04
Nd147	2.86E-04	2.85E-04	2.84E-04	2.79E-04	2.76E-04	2.66E-04
Am241	1.28E-07	1.28E-07	1.28E-07	1.28E-07	1.27E-07	1.27E-07
Cm242	2.91E-05	2.91E-05	2.90E-05	2.89E-05	2.89E-05	2.87E-05
Cm244	2.36E-06	2.35E-06	2.35E-06	2.35E-06	2.34E-06	2.33E-06
Ce141	1.85E-03	1.85E-03	1.85E-03	1.83E-03	1.82E-03	1.80E-03
Ce143	1.67E-03	1.66E-03	1.60E-03	1.40E-03	1.28E-03	9.85E-04
Ce144	1.36E-03	1.36E-03	1.36E-03	1.35E-03	1.35E-03	1.34E-03
Np239	2.93E-02	2.91E-02	2.85E-02	2.64E-02	2.51E-02	2.15E-02
Pu238	3.99E-06	3.99E-06	3.99E-06	3.98E-06	3.97E-06	3.95E-06
Pu239	7.89E-07	7.89E-07	7.89E-07	7.87E-07	7.85E-07	7.82E-07
Pu240	1.30E-06	1.30E-06	1.30E-06	1.29E-06	1.29E-06	1.29E-06
Pu241	3.70E-04	3.70E-04	3.69E-04	3.68E-04	3.68E-04	3.66E-04
Sr89	3.37E-02	3.37E-02	3.37E-02	3.35E-02	3.33E-02	3.29E-02
Sr90	5.58E-03	5.58E-03	5.57E-03	5.56E-03	5.55E-03	5.52E-03
Sr91	4.32E-02	4.17E-02	3.74E-02	2.42E-02	1.81E-02	7.54E-03
Sr92	5.01E-02	4.41E-02	2.97E-02	6.14E-03	2.15E-03	9.15E-05

Table 15.4-6

Control Rod Drop Accident
Activity Airborne to the Environment (Curies)

3556 MWth

	1 m	0.5 hr	2 hrs	8 hrs	12 hrs	24 hrs
Kr83m	2.78E-01	7.86E+00	2.43E+01	4.38E+01	4.56E+01	4.61E+01
Kr85m	5.74E-01	1.71E+01	6.09E+01	1.61E+02	1.91E+02	2.19E+02
Kr85	3.21E-02	9.92E-01	3.97E+00	1.59E+01	2.38E+01	4.75E+01
Kr87	1.04E+00	2.83E+01	7.87E+01	1.17E+02	1.18E+02	1.18E+02
Kr88	1.48E+00	4.31E+01	1.45E+02	3.20E+02	3.52E+02	3.69E+02
Kr89	1.54E+00	8.07E+00	8.08E+00	8.08E+00	8.08E+00	8.08E+00
Xe131m	2.18E-02	6.73E-01	2.69E+00	1.07E+01	1.59E+01	3.13E+01
Xe133m	1.30E-01	4.00E+00	1.58E+01	6.10E+01	8.92E+01	1.66E+02
Xe133	4.24E+00	1.31E+02	5.22E+02	2.05E+03	3.04E+03	5.88E+03
Xe135m	8.48E-01	1.48E+01	2.00E+01	2.01E+01	2.01E+01	2.01E+01
Xe135	1.02E+00	3.10E+01	1.18E+02	3.80E+02	5.00E+02	7.01E+02
Xe137	3.32E+00	2.09E+01	2.10E+01	2.10E+01	2.10E+01	2.10E+01
Xe138	2.75E+00	5.00E+01	7.03E+01	7.08E+01	7.08E+01	7.08E+01
Total Noble Gases	1.73E+01	3.58E+02	1.09E+03	3.28E+03	4.50E+03	7.70E+03
I-131*	4.08E-01	1.19E+01	4.01E+01	1.05E+02	1.35E+02	2.12E+02
I-132	4.81E+00	1.48E+02	5.83E+02	2.21E+03	3.23E+03	6.10E+03
I-133	8.80E-01	1.58E+01	2.40E+01	3.60E+01	4.39E+01	6.76E+01
I-134	2.06E+00	5.93E+01	1.97E+02	4.97E+02	6.18E+02	8.19E+02
I-135	4.80E+00	6.40E+01	1.66E+02	3.41E+02	3.73E+02	3.90E+02
Total Iodine	1.30E+01	2.99E+02	1.01E+03	3.19E+03	4.40E+03	7.59E+03

* The isotopic iodine activity is the sum of the elemental and organic iodines with 97% elemental and 3% organic.
The particulate iodine comprise 0%.

Table 15.4-6

Control Rod Drop Accident
Activity Airborne to the Environment (Curies) (Continued)

3556 MWth

	1 m	0.5 hr	2 hrs	8 hrs	12 hrs	24 hrs
Rb86	3.95E-07	1.22E-05	4.88E-05	1.94E-04	2.90E-04	5.73E-04
Cs134	5.54E-05	1.71E-03	6.86E-03	2.74E-02	4.11E-02	8.19E-02
Cs136	1.23E-05	3.79E-04	1.52E-03	6.02E-03	8.99E-03	1.77E-02
Cs137	4.47E-05	1.38E-03	5.52E-03	2.21E-02	3.31E-02	6.60E-02
Sb127	9.31E-08	2.87E-06	1.14E-05	4.47E-05	6.60E-05	1.26E-04
Sb129	2.66E-07	7.92E-06	2.83E-05	7.46E-05	8.82E-05	1.01E-04
Te127m	1.31E-08	4.05E-07	1.62E-06	6.48E-06	9.70E-06	1.93E-05
Te127	9.31E-08	2.88E-06	1.15E-05	4.58E-05	6.83E-05	1.33E-04
Te129m	3.91E-08	1.21E-06	4.83E-06	1.93E-05	2.88E-05	5.72E-05
Te129	2.50E-07	7.77E-06	3.04E-05	9.08E-05	1.10E-04	1.28E-04
Te131m	1.18E-07	3.63E-06	1.43E-05	5.33E-05	7.65E-05	1.34E-04
Te132	1.12E-06	3.46E-05	1.38E-04	5.35E-04	7.88E-04	1.49E-03
Ba137m	5.43E-06	1.21E-03	5.35E-03	2.19E-02	3.29E-02	6.58E-02
Ba139	5.29E-07	1.45E-05	4.15E-05	6.44E-05	6.55E-05	6.56E-05
Ba140	5.15E-07	1.59E-05	6.36E-05	2.53E-04	3.77E-04	7.41E-04
Mo99	6.89E-08	2.12E-06	8.44E-06	3.27E-05	4.80E-05	9.02E-05
Tc99m	6.11E-08	1.89E-06	7.65E-06	3.12E-05	4.69E-05	9.18E-05
Ru103	6.61E-08	2.04E-06	8.17E-06	3.26E-05	4.88E-05	9.69E-05
Ru105	4.86E-08	1.45E-06	5.19E-06	1.39E-05	1.66E-05	1.93E-05
Ru106	2.87E-08	8.87E-07	3.55E-06	1.42E-05	2.13E-05	4.24E-05
Rh105	4.60E-08	1.42E-06	5.68E-06	2.23E-05	3.27E-05	6.00E-05
Y90	2.33E-10	1.02E-08	7.78E-08	8.86E-07	1.88E-06	6.88E-06
Y91	3.07E-09	9.60E-08	3.97E-07	1.75E-06	2.76E-06	6.06E-06
Y92	3.78E-09	5.63E-07	6.33E-06	4.33E-05	6.04E-05	7.81E-05
Y93	4.00E-09	1.22E-07	4.63E-07	1.52E-06	2.02E-06	2.90E-06

Table 15.4-6

Control Rod Drop Accident
Activity Airborne to the Environment (Curies) (Continued)

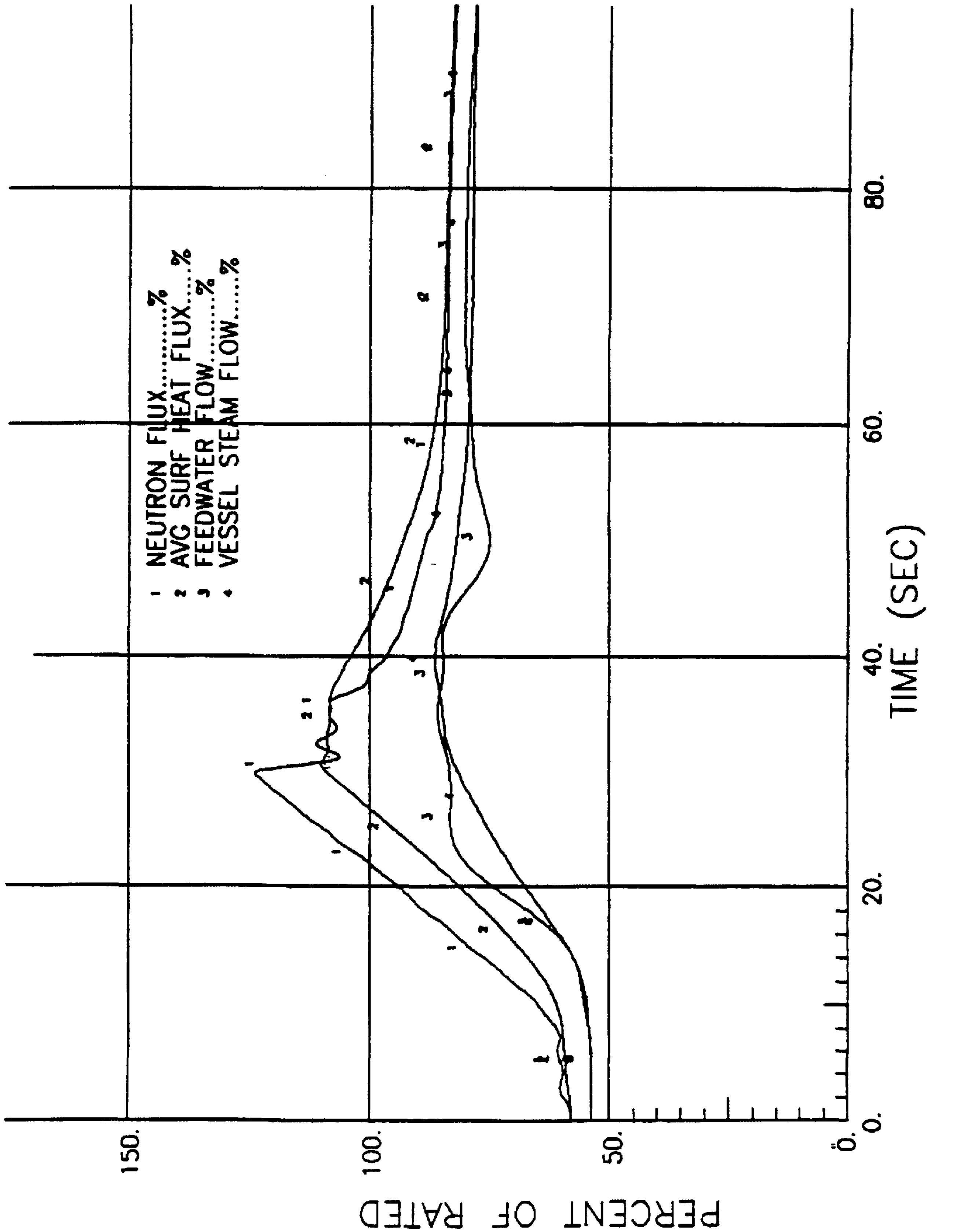
3556 MWth

	1 m	0.5 hr	2 hrs	8 hrs	12 hrs	24 hrs
Zr95	4.81E-09	1.48E-07	5.94E-07	2.37E-06	3.55E-06	7.06E-06
Zr97	4.87E-09	1.49E-07	5.79E-07	2.06E-06	2.86E-06	4.60E-06
Nb95	4.81E-09	1.48E-07	5.94E-07	2.38E-06	3.56E-06	7.10E-06
La140	5.37E-09	2.31E-07	1.72E-06	1.91E-05	4.00E-05	1.41E-04
La141	4.90E-09	1.45E-07	5.11E-07	1.29E-06	1.50E-06	1.68E-06
La142	4.67E-09	1.30E-07	3.81E-07	6.23E-07	6.37E-07	6.40E-07
Pr143	4.25E-09	1.32E-07	5.28E-07	2.13E-06	3.21E-06	6.48E-06
Nd147	1.92E-09	5.94E-08	2.37E-07	9.42E-07	1.40E-06	2.76E-06
Am241	8.63E-13	2.67E-11	1.07E-10	4.27E-10	6.39E-10	1.28E-09
Cm242	1.96E-10	6.05E-09	2.42E-08	9.67E-08	1.45E-07	2.89E-07
Cm244	1.59E-11	4.90E-10	1.96E-09	7.84E-09	1.18E-08	2.34E-08
Ce141	1.25E-08	3.85E-07	1.54E-06	6.15E-06	9.20E-06	1.83E-05
Ce143	1.13E-08	3.47E-07	1.37E-06	5.12E-06	7.36E-06	1.30E-05
Ce144	9.15E-09	2.83E-07	1.13E-06	4.52E-06	6.77E-06	1.35E-05
Np239	1.97E-07	6.07E-06	2.41E-05	9.28E-05	1.36E-04	2.52E-04
Pu238	2.69E-11	8.31E-10	3.33E-09	1.33E-08	1.99E-08	3.98E-08
Pu239	5.32E-12	1.64E-10	6.58E-10	2.63E-09	3.94E-09	7.86E-09
Pu240	8.75E-12	2.70E-10	1.08E-09	4.33E-09	6.48E-09	1.29E-08
Pu241	2.49E-09	7.69E-08	3.08E-07	1.23E-06	1.85E-06	3.68E-06
Sr89	2.27E-07	7.02E-06	2.81E-05	1.12E-04	1.68E-04	3.34E-04
Sr90	3.76E-08	1.16E-06	4.65E-06	1.86E-05	2.78E-05	5.55E-05
Sr91	2.91E-07	8.84E-06	3.36E-05	1.09E-04	1.44E-04	2.05E-04
Sr92	3.38E-07	9.81E-06	3.26E-05	7.00E-05	7.63E-05	7.96E-05

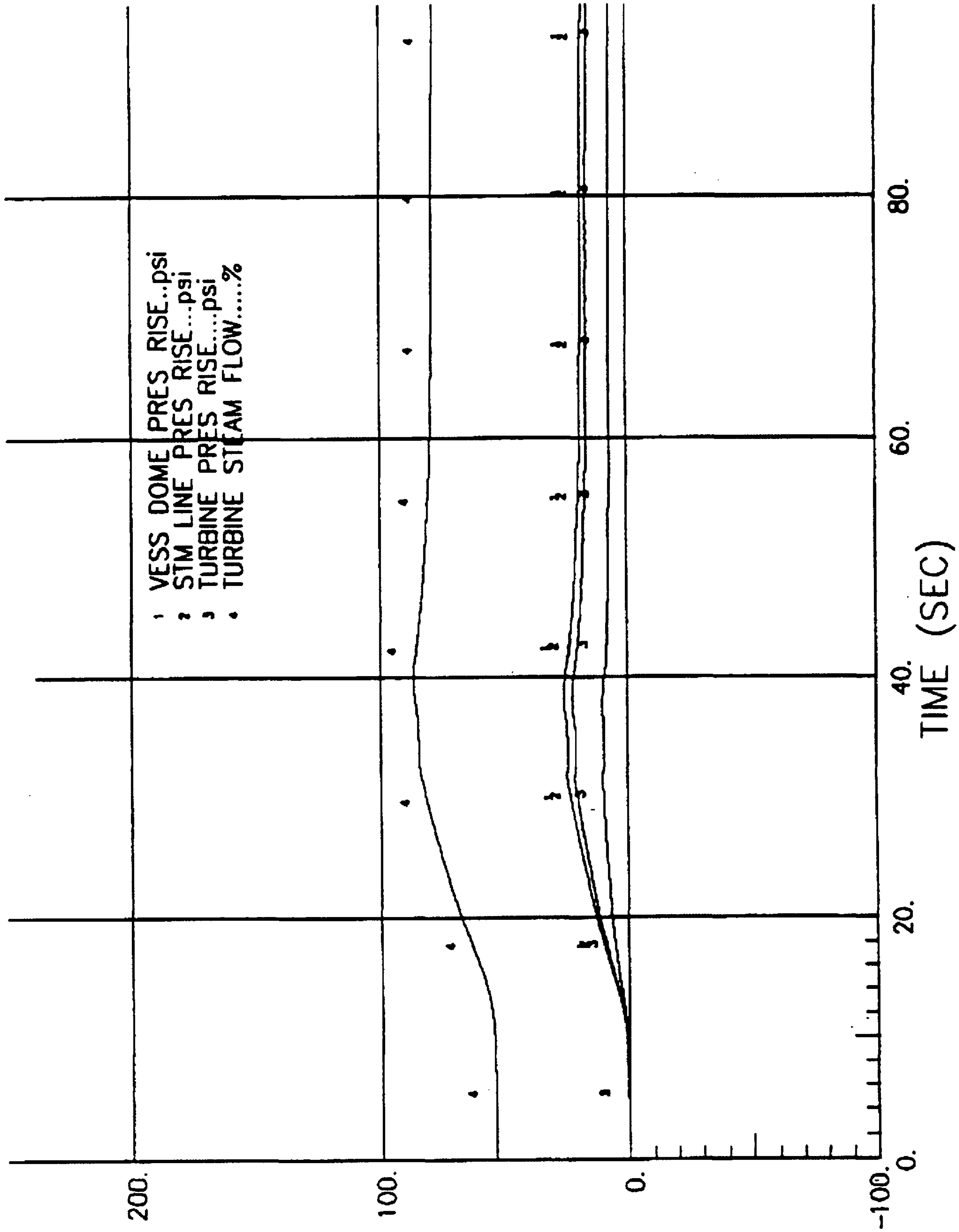
Table 15.4-7

Control Rod Drop Accident
Radiological Effects (rem)

Area	Time	TEDE Dose
Exclusion area (1950 m)	2 hr	0.03
Low population zone (4827 m)	30 days	0.03
Control Room	30 days	0.7



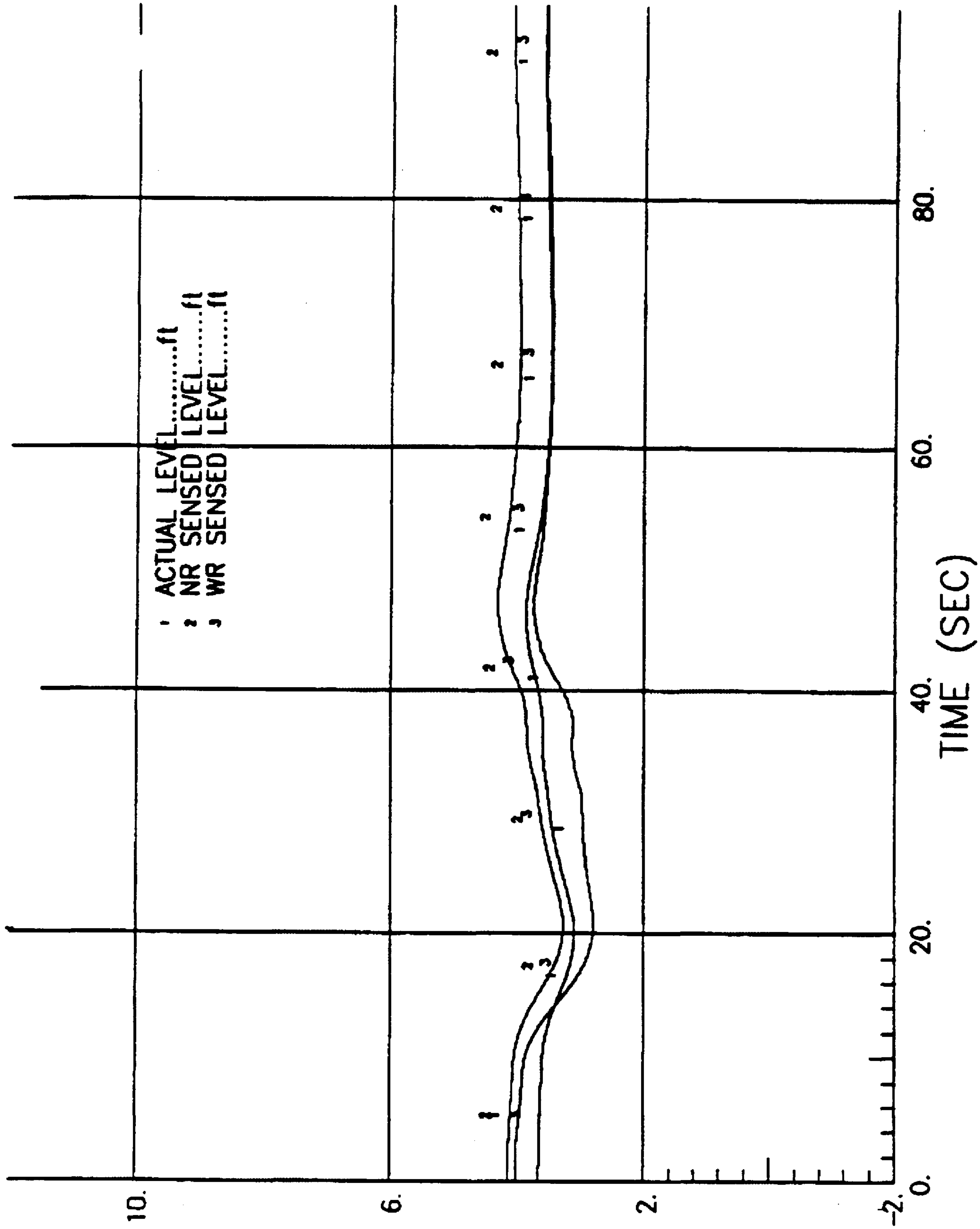
Columbia Generating Station Final Safety Analysis Report	Abnormal Startup of an Idle Recirculation Loop at 57.9% Uprated Power, 34.1% Flow		
	Draw. No. 020002.03	Rev.	Figure 15.4-1.1



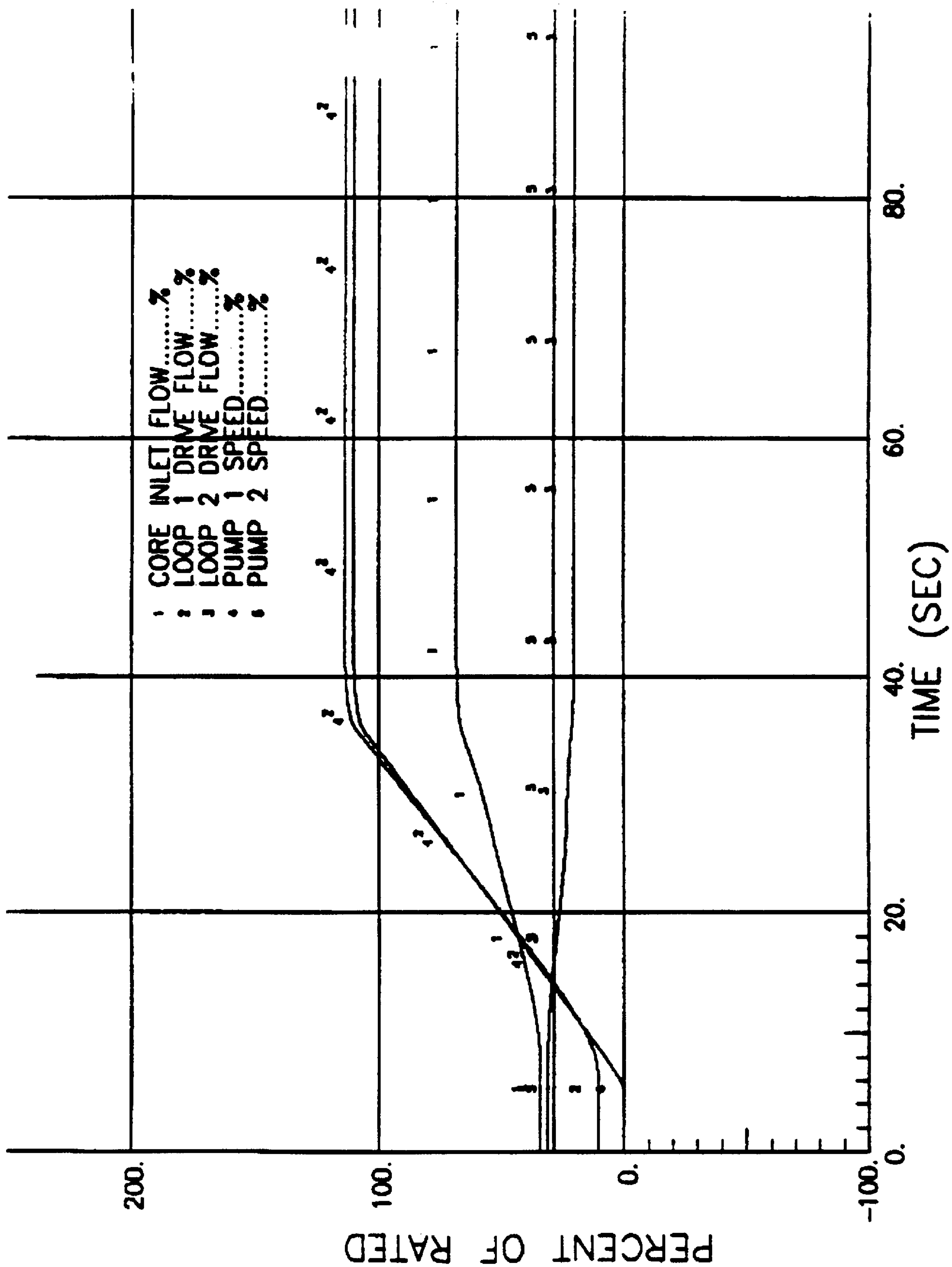
Columbia Generating Station
Final Safety Analysis Report

Abnormal Startup of an Idle Recirculation Loop at 57.9% Uprated Power, 34.1% Flow

Draw. No. 020002.04	Rev.	Figure 15.4-1.2



Columbia Generating Station Final Safety Analysis Report	Abnormal Startup of an Idle Recirculation Loop at 57.9% Uprated Power, 34.1% Flow		
	Draw. No. 020002.05	Rev.	Figure 15.4-1.3



Columbia Generating Station Final Safety Analysis Report	**Abnormal Startup of an Idle Recirculation Loop at 57.9% Uprated Power, 34.1% Flow**
	Draw. No. 020002.06 | Rev. | Figure 15.4-1.4

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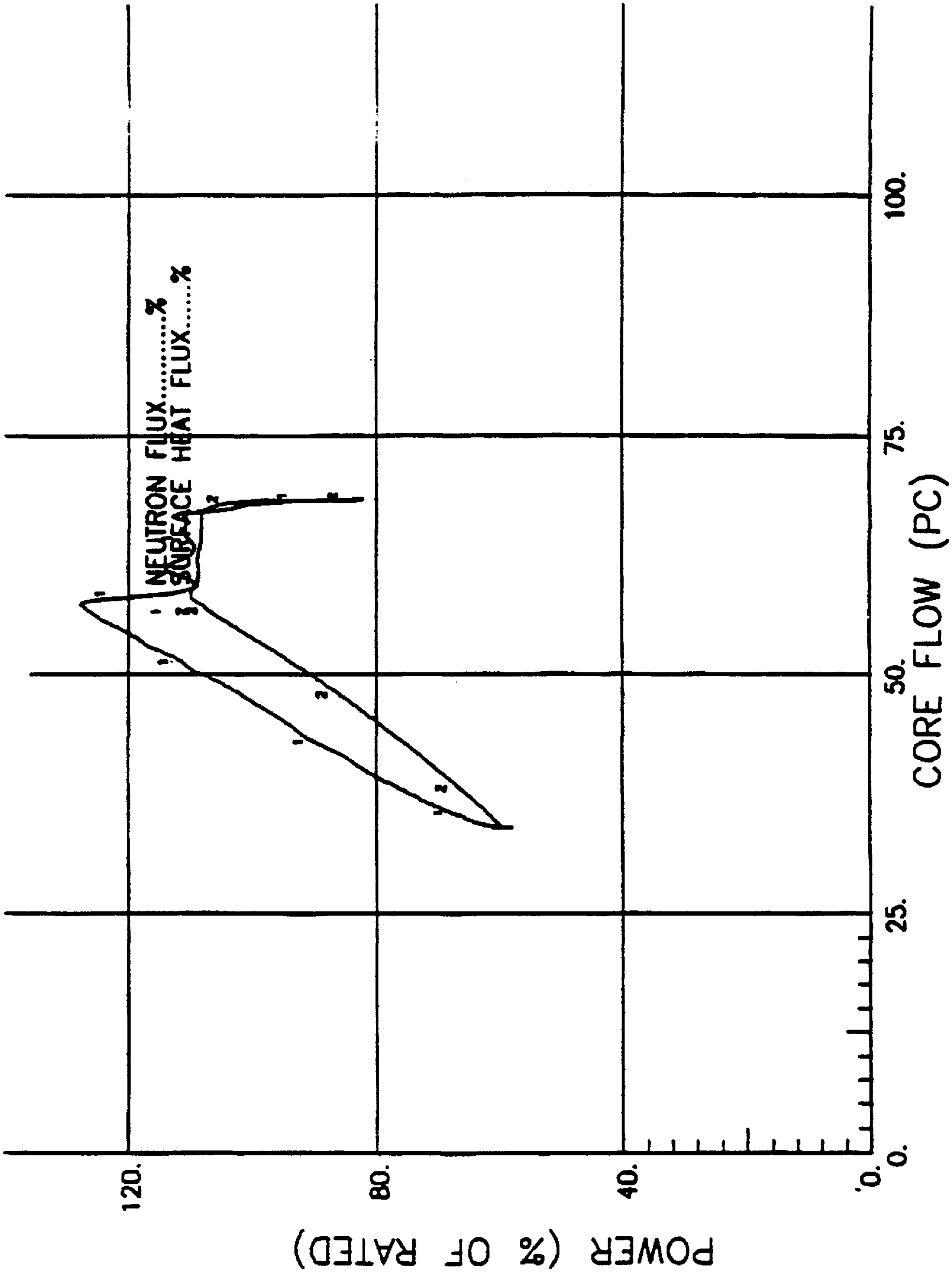
November 1998



Columbia Generating Station Final Safety Analysis Report	Abnormal Startup of an Idle Recirculation Loop at 57.9% Uprated Power, 34.1% Flow		
	Draw. No. 020002.07	Rev.	Figure 15.4-1.5

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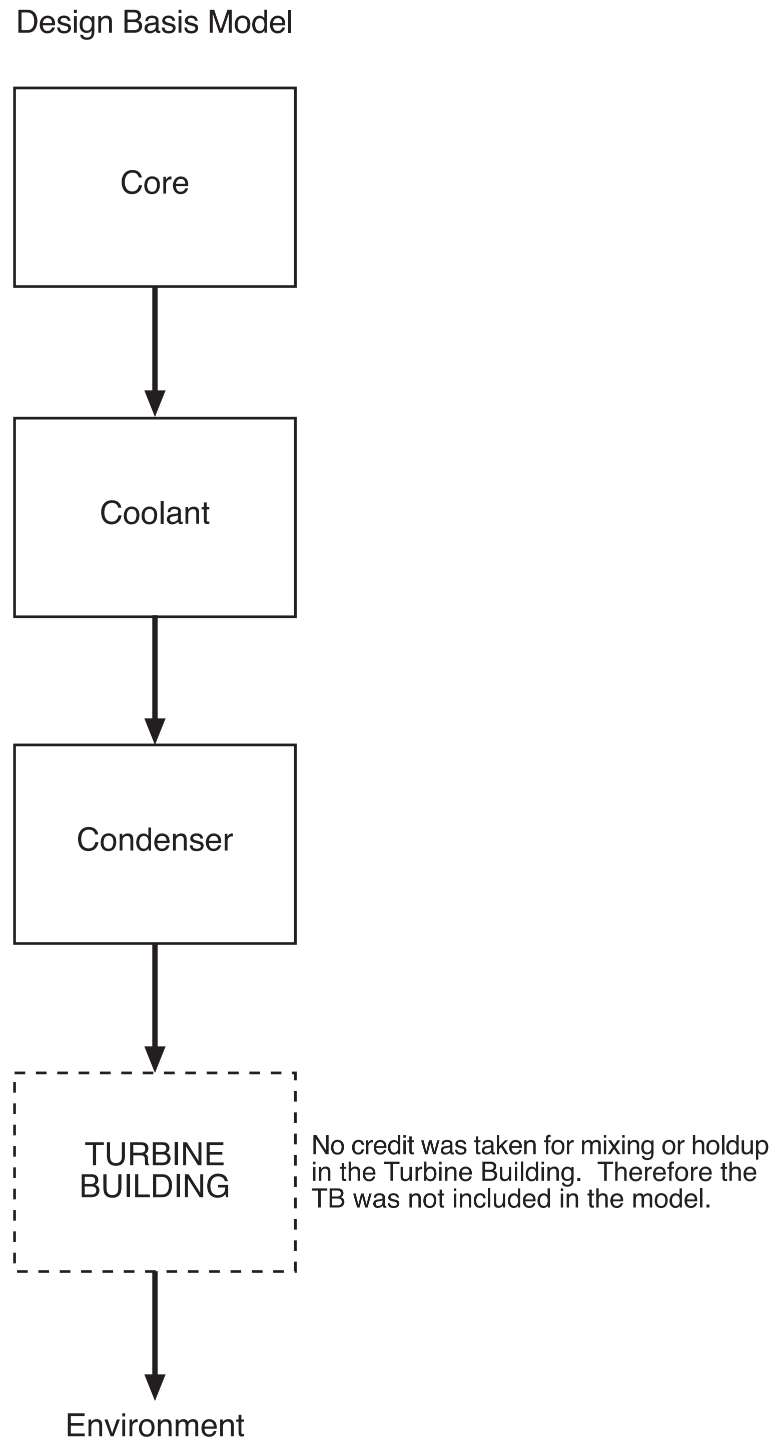


Columbia Generating Station
Final Safety Analysis Report

Leakage Path Model for Rod Drop Accident

Draw. No. 960222.71 | Rev. | Figure 15.4-2

15.5 INCREASE IN REACTOR COOLANT INVENTORY

15.5.1 INADVERTENT HIGH-PRESSURE CORE SPRAY STARTUP

This transient is classified as a nonlimiting event for both original and uprated power conditions. The transient is not analyzed for each reload, but was analyzed for the GE14 New Fuel Introduction. Inadvertent startup of the high-pressure core spray (HPCS) system was analyzed because it provides the greatest auxiliary source of cold water into the vessel.

15.5.1.1 Identification of Causes and Frequency Classification

15.5.1.1.1 Identification of Causes

Manual startup (i.e., operator error) and continued injection of the HPCS 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

The HPCS system is manually initiated and injects into the reactor vessel, reaching full flow in approximately one second. The addition of the cooler water to the upper plenum causes a reduction in steam flow. This causes some reactor pressure decrease as the turbine control system responds to the event. As the steam flow decreases, the feedwater system responds by decreasing flow. In less than a minute, the reactor and the auxiliary steam systems stabilize at a new, lower power level. The analysis assumes normal functioning of plant instrumentation and controls. No engineered safety feature (ESF) function is expected in response to this transient. Plant parameter responses are shown in Figure 15.5-1.

15.5.1.2.1 The Effect of Single Failures and Operator Errors

Inadvertent operation of the HPCS system results in a mild depressurization. Level control and pressure regulator actuation are expected to establish a new stable operating state. The effect of a single failure in the DEH control system will have no effect on the transient because of its redundant design.

The effect of a single failure in the level control system has rather straightforward consequences including level rise or fall by improper control of the feedwater system. Increasing level will trip the turbine and automatically stop injection by the HPCS system. Decreasing level will automatically initiate a scram at the L3 level trip.

15.5.1.3 Core and System Performance

15.5.1.3.1 Mathematical Model

The one-dimensional ODYN transient analysis model described in Reference 15.5-1 was used to simulate this transient.

15.5.1.3.2 Input Parameter and Initial Conditions

The important parameters are shown in Table 15.5-1.

15.5.1.3.3 Results

The calculated uncorrected ΔCPR for the simulated bundles is less than 0.01 (Reference 15.5-2). A summary of transient key peak values is found in Table 15.0-1.

15.5.1.3.3.1 Consideration of Uncertainties. Important analytical factors including reactivity coefficient and feedwater temperature change have been assumed to be at the worst conditions so that any deviations in the actual plant parameters will produce a less severe transient.

15.5.1.4 Barrier Performance

Figure 15.5-1 indicates only a slight pressure reduction from initial conditions; therefore, reactor coolant pressure boundary pressure margins are not impacted.

15.5.1.5 Radiological Consequences

Since this event does not result in any fuel failures or any release of primary coolant to either the secondary containment or to the environment, there are no radiological consequences associated with this event.

15.5.2 CHEMICAL VOLUME CONTROL SYSTEM MALFUNCTION (OR OPERATOR ERROR)

This event is not applicable to boiling water reactor plants.

15.5.3 BOILING WATER REACTOR TRANSIENTS WHICH INCREASE REACTOR COOLANT INVENTORY

These events are discussed in Sections 15.1 and 15.2.

15.5.4 REFERENCES

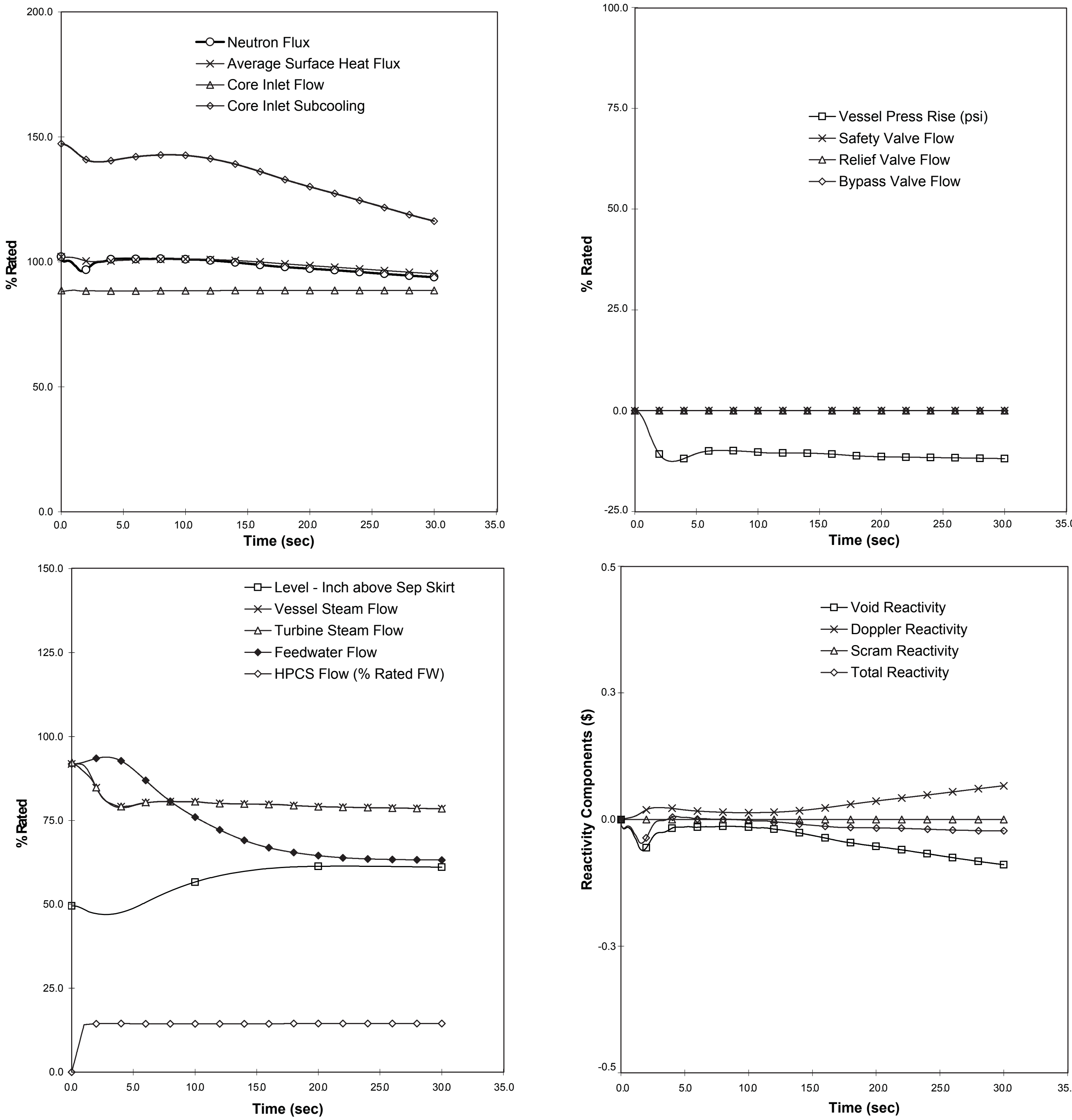
15.5-1 "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," Volumes 1, 2, 3 and 4, NEDC-24154-P-A, February 2000.

15.5-2 GE Hitachi Nucear Energy, "Inadvertent High Pressure Core Spray Startup Analysis," 0000-0098-5369-R0, March 2009.

Table 15.5-1

Input Parameters and Initial Conditions
HPCS Injection

Parameter	Value
Reactor power	102% (3556 MWth)
Core flow	106% ICF and 88% ELLLA
HPCS water source	Suppression pool
HPCS source pressure	14.7 psia
HPCS source temperature	40° F
HPCS source enthalpy	11.0 Btu/lbm
HPCS pump flow	12.6 % of rated feedwater flow (3800 gal/min)
Vessel-to-suppression pool differential pressure	1020 psid



Columbia Generating Station Final Safety Analysis Report	Inadvertent Start of High Pressure Core Spray Pump at 102% Uprated Power, 88% Flow		
	Draw. No. 020002.18	Rev.	Figure 15.5-1

15.6 DECREASE IN REACTOR COOLANT INVENTORY

15.6.1 INADVERTENT SAFETY/RELIEF VALVE OPENING

This event is discussed in Section 15.1.4.

15.6.2 INSTRUMENT LINE PIPE BREAK

This faulted condition is not a limiting event for either original or uprated power conditions. Therefore, the uprated power analysis for the accident has not been updated.

This event involves the postulated pipe break in a small steam or liquid line that is connected to the reactor coolant pressure boundary (RCPB) and is located in the reactor building. If the break were inside primary containment, the event would be bounded by the steam line break inside containment (LOCA) (see Section 15.6.5). That event is bounding because the instrument line sizes are bounded by the spectrum of breaks considered in the LOCA analysis.

15.6.2.1 Identification of Causes and Frequency Classification

15.6.2.1.1 Identification of Causes

There is no specific event or circumstance identified which results in the failure of an instrument line. These lines are designed to specific engineering specifications and standards, and seismic and environmental requirements. However, for the purpose of evaluating the consequences of this event, the failure of an instrument line is assumed to occur.

15.6.2.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.6.2.2 Sequence of Events and Systems Operation

The instrument line ruptures (complete circumferential break) and releases reactor coolant into the secondary containment. The analysis assumes that the reactor coolant activity is at the Technical Specification limit corresponding to an iodine spike of 4 μCi/g dose equivalent ^{131}I and that the break cannot be isolated. The operators have a variety of methods to detect the leak such as monitoring plant area temperatures and radiation levels, system pressures, or sump inventories, or during operator rounds. It is assumed that the reactor operators identify the break after 20 minutes and initiate a reactor scram.

Using available plant systems, the operators maintain reactor water level and cool down and depressurize the reactor within 5 hr, at a rate less than or equal to the 100°F/hr limit in the Technical Specifications. Examples of plant systems or components the operators can use for

inventory and temperature controls include HPCS, RCIC, SRVs, RHR, or the condensate feed system. No credit is taken for the automatic initiation of the RPS or ESF.

15.6.2.2.1 The Effect of Single Failures and Operator Errors

The event is handled by operator actions. Assuming additional single equipment failure or single operator error occurrences, adequate equipment would be available to respond to the loss of reactor coolant.

15.6.2.3 Core and System Performance

The inventory loss is within the capacity of the make up systems available and the shutdown and the cool down are controlled evolutions. Therefore, no fuel damage will occur and no other barriers are challenged.

15.6.2.3.1 Qualitative Summary - Results

Since instrument line breaks result in a slower rate of coolant loss and are bounded, the results are qualitative rather than quantitative. Since the rate of coolant loss is slow, an orderly reactor system depressurization follows reactor scram and the primary system is cooled down and maintained without ECCS actuation. No fuel damage or core uncovery occurs as a result of this event.

15.6.2.4 Barrier Performance

15.6.2.4.1 General

The release of primary coolant through the orificed instrument line would not result in an increase in secondary containment pressure.

15.6.2.5 Radiological Consequences

The radiological consequences are based on the following assumptions and methods:

a. The broken instrumentation line contains a 0.5-in. diameter flow restricting orifice inside the drywell.

b. Flow is critical at the orifice and is determined using the GOTHIC computer program (Reference 15.6-1) that employs the Henry model for subcooled liquid and the Moody model for saturated and superheated vapors (Reference 15.6-4).

c. The total integrated mass of fluid released by means of the break during the blowdown is 121,000 lb. Of this total, 29,800 lb flash to steam.

d. The specific models, assumptions and the program used for the radiological analysis is in Reference 15.6-3. Specific values of parameters used in the evaluation are presented in Table 15.6-1. The leakage path used in these calculations is shown in Figure 15.6-1.

e. The activity released from the instrument line break is based on the iodine spike concentration of 4μCi/g dose equivalent ^{131}I and is assumed to not mix or be held up within the secondary containment and is released to the environment at a flow rate of 80,000 cfm without SGT filtration.

f. The activity released to the secondary containment and the environment is presented in Table 15.6-2.

15.6.2.5.1 Results

The calculated exposures are presented in Table 15.6-3.

15.6.3 STEAM GENERATOR TUBE FAILURE

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

15.6.4 STEAM SYSTEM PIPING BREAK OUTSIDE CONTAINMENT

This event involves the postulation of a large steam line pipe break outside the primary containment. The analysis assumes that a main steam line instantaneously and circumferentially breaks at a location downstream of the outboard isolation valve. The plant is designed to immediately detect such an occurrence, initiate isolation of the broken line, and actuate the necessary protective features. The main steam line was selected for analysis because the postulated event envelops evaluation of steam line failures outside containment.

15.6.4.1 Identification of Causes and Frequency Classification

15.6.4.1.1 Identification of Causes

A main steam line break is postulated without the cause being identified. These lines are designed to specific engineering codes and standards, and seismic and environmental requirements. However, for the purpose of evaluating the consequences of a postulated large steam line rupture, the failure of a main steam line is assumed to occur.

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

When the steam line breaks, steam flow through the failed line will rapidly increase, initiating the high main steam line flow trip that initiates the signal to close the main steam isolation valves (MSIVs). The closing MSIVs initiate a reactor protection (RPS) signal, scramming the reactor. The analysis assumes that the MSIVs are fully closed 6 seconds after the break. Performance of the engineered safety feature (ESF) systems in response to the loss of coolant is discussed in Chapter 6.

The sequence of events and approximate time required to reach the event is given in Table 15.6-4.

15.6.4.2.2 Systems Operation

A postulated guillotine 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. Initially, only steam will issue from the broken end of the steam line. The flow in each line is limited by critical flow at the limiter to a maximum of 200% of rated flow for each line. 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. Mass loss (steam and water) is reduced and finally terminated (except for leakage) as the MSIVs close.

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. All of the protective sequences for this event are capable of single equipment failure or single operator error accommodation and yet still complete the necessary safety action.

15.6.4.3 Core and System Performance

The temperature and pressure transients resulting as a consequence of this accident are insufficient to cause fuel damage.

15.6.4.3.1 Input Parameters and Initial Conditions

See Section 6.3 for initial conditions.

15.6.4.3.2 Results

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

See Section 6.3 for ECCS analysis.

15.6.4.3.3 Considerations of Uncertainties

Discussions of the uncertainties associated with the ECCS performance and the containment isolation systems are discussed in Sections 6.3 and 7.3, respectively.

15.6.4.4 Barrier Performance

Since this break occurs outside the primary containment, barrier performance within the containment envelope is not applicable. There are sufficient vent openings in the steam tunnel to ensure that the secondary containment structure will not be damaged.

15.6.4.5 Radiological Consequences

The radiological analysis is based on NRC Regulatory Guide 1.183 (Reference 15.6-5). The dispersion of the plume is based on the puff model given in Regulatory Guide 1.194 (Reference 15.6-6).

The specific models, assumptions, and the program used for computer evaluation are described in Reference 15.6-3. Specific values of parameters used in the evaluation are presented in Table 15.6-5. There is no fuel damage as a result of this accident. The only activity available for release from the break is that which is present in the reactor coolant and steam lines prior to the break. The iodine inventories and the subsequent exposures are based on the equilibrium conditions and maximum reactor coolant activity for an iodine spiking event as allowed by the Technical Specifications. The analysis assumes all the activity in this discharge becomes airborne and released directly and unfiltered to the environment. The release of activity to the environment is presented in Table 15.6-6.

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

a. The reactor is operating at the power level associated with maximum mass release,

b. Nuclear system pressure is 1060 psia and remains constant during closure,

c. An instantaneous circumferential break of the main steam line occurs,

d. Isolation valves start to close at 0.5 sec on high flow signal and are fully closed at 6 sec,

e. The Moody critical flow model (Reference 15.6-4) is applicable, and

f. The flow limiters allow up to 200% of rated flow through the MSIVs, and

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

The total integrated mass leaving the RPV through the steam line break is 130,000 lb of which 105,000 lb is liquid and 25,000 lb is steam. Only the liquid portion of the discharged coolant is assumed to carry the iodine activity of 4 μCi/gm dose-equivalent of I-131. The entire amount of activity in the liquid is assumed to be released to the environment.

The transport pathway is a direct unfiltered release to the environment and an unfiltered entrance to the control room as presented in Figure 15.6-2.

15.6.4.5.1 Results

The calculated doses for the design basis analysis are presented in Table 15.6-7. The doses are within the limits of 10 CFR 50.67.

15.6.5 LOSS-OF-COOLANT ACCIDENTS (RESULTING FROM SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY) - INSIDE CONTAINMENT

Accidents that could result in the release of radioactive fission products directly into the containment are the results of postulated breaks in the reactor coolant system pressure boundary (RCPB). The analysis postulates that the most severe pressurization transient to the primary containment is caused by a complete circumferential break of the suction line of one of the two recirculation loops. Flow through the break transports the reactor vessel contents to the suppression pool.

The loss-of-coolant accident (LOCA) postulates the break of any of the spectrum of piping systems that form the RCPB. The plant and operator responses to the spectrum of breaks are presented in Chapter 6. Chapter 6 demonstrates that fuel, core, and barrier performance requirements are met for the spectrum of breaks. The bounding radiological analysis for the LOCA event detailed in this section reflects an inadequate core cooling accident that degrades to complete core damage. The event assumed for the analysis is the break inside containment of one of the main steam lines. The radiological analysis assumptions presented in

Section 15.6.5 are separate and distinct and are not mechanistically tied to the pipe break analyzed in Chapter 6.

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 containment of the magnitude required to cause the fuel damage sufficient to release the source terms assumed in this section. The piping is designed to specific engineering codes and standards and for severe seismic and environmental conditions. However, since such an accident provides an upper limit estimate of the dose consequences for the limiting faulted condition, 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

The sequence of events and system operations are discussed in Chapter 6. The effect of single failures and operator errors is discussed in Chapter 6.

15.6.5.3 Core and System Performance

For the plant response to the LOCA and the evaluation of the system and core performance, see Chapter 6.

15.6.5.4 Radiological Consequences

The radiological consequences are based on the guidance provided in Regulatory Guide 1.183 (Reference 15.6-5) for the purpose of determining adequacy of the plant design to meet 10 CFR 50.67 limits.

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

15.6.5.4.1 Design Basis Analysis

The specific models, assumptions, and computer code used to evaluate the radiological consequences of the bounding LOCA based on the above criteria are presented in Reference 15.6-3. Specific values of parameters used in this evaluation are presented in Table 15.6-8.

15.6.5.4.1.1 Fission Product Release from Fuel. It is assumed that 100% of the noble gases and 30% of the iodine are released from an equilibrium core operating at a power level of 3556 MWt for 1000 days prior to the accident. Of this release, 100% of the noble gases become airborne. Some of the iodine is removed by plate-out and filtration; therefore, it is not available for airborne release to the environment. The activity airborne in the containment is presented in Table 15.6-9.

15.6.5.4.1.2 Fission Product Transport to the Environment. The fission product transport to the environment consists of two basic pathways. One transport pathway consists of leakage from the containment to the secondary containment by several different mechanisms and is discharged to the environment through the SGT system at an elevated location. Of the secondary containment flow, 50 cfm bypasses the SGT filters. The second transport pathway consists of leakage from the containment directly to the environment through piping systems that originate in containment and terminate outside the reactor building. The individual mechanisms for leakage from the primary containment are:

a. Containment leakage - Leakage from primary containment to the secondary containment. Prior to the completion of secondary containment drawdown this leakage is assumed to be a direct release to the environment; however, when drawdown is complete, 20 minutes post accident, this leakage is treated by the SGT system before it is released to the environment. No credit is taken for forced mixing and holdup within the secondary containment.

b. ESF leakage - Leakage from engineered safety feature (ESF) components outside the primary containment (all ESF equipment which circulates primary coolant or suppression pool water during the course of the postulated accident) to the secondary containment. This leakage is treated in a manner similar to the containment leakage described above.

c. Hydrogen purge - No hydrogen purge is required or assumed throughout the postaccident period.

d. MSIV leakage - Leakage from the primary containment (or the RPV) through the main steam isolation valves (MSIVs) to the turbine building. Iodine is assumed to plate-out in three of the four main steam lines. Plate-out in the broken fourth main steam line is not credited. In that line it is assumed one MSIV fails open (single failure), whereas all MSIVs in the other three steam lines are assumed to close.

e. Bypass leakage - Leakage from the primary containment that bypasses the secondary containment and is released, untreated, directly to the environment.

Fission product activities in secondary containment and those released to the environment based on the above assumptions are given in Tables 15.6-10 and 15.6-11, respectively.

15.6.5.4.1.3 Suppression Pool pH Control. The suppression pool pH is maintained above 7.0 for the duration of the accident as a result of the standby liquid control system injection of sodium pentaborate solution. The solution is assumed to be injected and fully mixed with the suppression pool water within 8 hours post-LOCA.

15.6.5.4.1.4 Results. The calculated doses for the design basis analysis are presented in Table 15.6-12 and are within the limts of 10 CFR 50.67.

15.6.6 FEEDWATER LINE BREAK - OUTSIDE CONTAINMENT

In order to evaluate large liquid process line pipe breaks outside containment, the failure of a feedwater line is assumed to evaluate the response of the plant design to this postulated event. The postulated break of the feedwater line, representing the largest liquid line outside the containment, provides the envelope evaluation relative to this type of occurrence. The break is assumed to be instantaneous, circumferential, and outboard of the outermost isolation valve. The analysis has not been updated for the change in MSIV isolation setpoint from Level 2 to Level 1 because the analysis remains bounded by the recirculation line break LOCA (Reference 15.6-9).

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 specific engineering codes and standards.

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 15.6-13.

15.6.6.2.2 Systems Operation

It is assumed that the normally operating plant instrument and controls are functioning. Credit is taken for the actuation of the reactor isolation system and ECCS. The RPS (SRVs, ECCS,

and control rod drives) and plant protection system (RHR heat exchanger) are assumed to function. The ESF system is assumed to operate normally. Although not an ECCS and not credited nor required for mitigation of this event, RCIC will also be used if available for maintaining vessel level as it initiates at approximately the same low reactor vessel level as HPCS.

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 general LOCA break spectrum considered in Section 6.3. The general single-failure analysis for LOCAs is discussed in Section 6.3.3.3. For the feedwater line break outside the containment, since the break is isolable, the HPCS can provide adequate flow to the vessel to maintain core cooling and prevent fuel rod cladding failure. A single failure of the HPCS would require actuation of ADS and the low-pressure core cooling systems to keep the core covered with water.

15.6.6.3 Core and System Performance

15.6.6.3.1 Qualitative Summary

The accident evaluation qualitatively considered in this section is considered to be conservative and to envelope assessment of the consequences of the postulated failure of one of the feedwater piping lines external to the containment. The accident is postulated to occur at the input parameters and initial conditions given in Table 6.3-2.

15.6.6.3.2 Qualitative Results

The feedwater line break outside the containment is less limiting than the steam line break outside the containment or the LOCA inside the containment.

The reactor vessel is isolated on low-low water level and the HPCS would restore the reactor water level to the normal elevation. The fuel is covered throughout the event and there are no pressure or temperature transients sufficient to cause fuel damage.

15.6.6.3.3 Consideration of Uncertainties

This event was conservatively analyzed and uncertainties were adequately considered (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 containment is a complete severance of one of the

main steam lines. The feedwater system piping break is less severe than the main steam line break.

15.6.6.5 Radiological Consequences

The specific models, assumptions, and the program used for computer evaluation are described in Reference 15.6-2. Specific values of parameters used in the evaluation are presented in Table 15.6-14. A diagram of the leakage path for this accident is shown in Figure 15.6-4.

15.6.6.5.1 Fission Product Release

Fission product release is assumed to occur from two pathways: activity being pumped from the condenser hotwell and activity returning to the feedwater system from the reactor water cleanup (RWCU) system. The activity in both of these sources is based on the Technical Specification coolant limit.

Noble gas activity in the condensate is negligible since the air ejectors remove most of the noble gas from the condenser.

15.6.6.5.2 Fission Product Transport to the Environment

The transport pathway consists of liquid release from the break, carryover to the turbine building atmosphere due to flashing and partitioning and unfiltered release to the environment through the turbine building ventilation system.

Of the 860,000 lb of condensate released from the break, 86,000 lb flashes to steam with assumed iodine carryover of 100%. Of the activity remaining in the unflashed liquid, 5% is assumed to become airborne. Normally, all feedwater reaching the break location will have passed through condensate demineralizers which have a 90% iodine removal efficiency. However, as a result of the increased feedwater flow caused by the break, differential pressure across the demineralizers is assumed to initiate flow through the demineralizer bypass line. This bypass line then carries 15% of the total flow resulting in an effective iodine removal efficiency for all flow of 76.5%. In addition, it is also assumed that 2771 lb of liquid returning from the RWCU are released prior to isolation of the RWCU. The activity concentration in this return steam is 1% of the RPV coolant concentration.

Taking no credit for holdup, decay, or plate-out during transport through the turbine building, the release of activity to the environment is presented in Table 15.6-15. The release is assumed to take place within 2 hr of the occurrence of the break.

15.6.6.5.3 Results

The calculated exposures for the realistic analysis are presented in Table 15.6-16 and are a small fraction of 10 CFR 100 guidelines.

15.6.7 REFERENCES

15.6-1 GOTHIC Containment Analysis Package, Technical Manual, Version 4.0, Numerical Applications, Inc., NA18907-06, Revision 3.

15.6-2 Nguyen, D., "Realistic Accident Analysis for General Electric Boiling Water Reactor - The RELAC Code and User's Guide," (NEDO-21142).

15.6-3 Energy Northwest, "Columbia Generating Station Alternative Source Term," CGS-FTS-0168, Revision 0, August 2007.

15.6-4 Moody, F. J, "Maximum Two-Phase Vessel Blowdown from Pipes," ASME Paper Number 65-WA/HT-1, March 15, 1965.

15.6-5 Regulatory Guide 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July, 2000.

15.6-6 Regulatory Guide 1.194, Revision 0, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," July, 2003.

15.6-7 Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," Oak Ridge National Laboratory, 1988.

15.6-8 Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water and Soil," Oak Ridge National Laboratory, 1993.

15.6-9 GE Hitachi Nuclear Energy, "License Amendment Request for Proposed Changes to Columbia Technical Specifications: Changing Group 1 Isolation Valves' Low Reactor Water Level Isolation Signal from the Current Level 2 to Level 1," 0000-0081-6730-R1, July 2008.

Table 15.6-1

Instrument Line Break Accident - Parameters
Tabulated for Postulated Accident Analyses

Parameters	Design Basis Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accidents	
A. Power level	N/A
B. Burnup	N/A
C. Fuel damaged	None
D. Airborne activity by nuclide	Table 15.6-2
E. Iodine fractions	
(1) Organic	0.15%
(2) Elemental	4.85%
(3) Particulate	95%
F. Initial reactor coolant activity with iodine spike	4 µci/g
^{131}I	1.6 µci/g
^{132}I	14.8 µci/g
^{133}I	11.0 µci/g
^{134}I	30.0 µci/g
^{135}I	16.0 µci/g
II. Data and assumptions used to estimate activity released	
A. Primary containment leak rate (%/day)	N/A
B. Secondary containment effluent rate (cfm)	80,000[a]
C. Valve movement times	N/A
D. Adsorption and filtration efficiencies	
(1) Organic iodine	N/A
(2) Elemental iodine	N/A
(3) Particulate iodine	N/A
(4) Particulate fission products	N/A
E. Recirculation system parameters	
(1) Flow rate	N/A
(2) Mixing efficiency	N/A
(3) Filter efficiency	N/A
F. Containment spray parameters (flow rate, drop size, etc.)	N/A
G. Containment volumes	N/A
H. All other pertinent data and assumptions	None

Table 15.6-1

Instrument Line Break Accident - Parameters
Tabulated for Postulated Accident Analyses (Continued)

Parameters			Design Basis Assumptions
III.	Dispersion data		
	A.	Offsite	See Table 15.0-4
	B.	Control Room	See Table 15.0-5
IV.	Dose data		
	A.	Method of dose calculation	Reference 15.6-5
	B.	Dose conversion assumptions	Reference 15.6-7
	C.	Peak activity released from secondary containment	Table 15.6-2
	D.	Doses	Table 15.6-3

[a] No forced mixing in secondary containment is considered.

Table 15.6-2

Instrument Line Failure

Activity Airborne in Secondary Containment (Ci)					
	2 hr	5 hr	8 hr	1 day	30 days
^{133}Xe	7.02E-07	2.60E-07	0.00E+00	0.00E+00	0.00E+00
^{135}Xe	1.22E-05	3.67E-06	0.00E+00	0.00E+00	0.00E+00
Total	1.29E-05	3.93E-06	0.00E+00	0.00E+00	0.00E+00
^{131}I	2.00E-02	8.09E-03	0.00E+00	0.00E+00	0.00E+00
^{132}I	1.03E-01	1.72E-02	0.00E+00	0.00E+00	0.00E+00
^{133}I	1.29E-01	4.80E-02	0.00E+00	0.00E+00	0.00E+00
^{134}I	7.60E-02	2.80E-03	0.00E+00	0.00E+00	0.00E+00
^{135}I	1.64E-01	4.93E-02	0.00E+00	0.00E+00	0.00E+00
Total	4.92E-01	1.25E-01	0.00E+00	0.00E+00	0.00E+00

Activity Airborne in the Environment (Ci)					
	2 hr	5 hr	8 hr	1 day	30 days
^{133}Xe	1.90E-03	3.93E-03	4.91E-03	6.06E-03	6.08E-03
^{135}Xe	3.56E-02	6.76E-02	8.02E-02	9.06E-02	9.06E-02
Total	3.75E-02	7.15E-02	8.51E-02	9.67E-02	9.67E-02
^{131}I	4.87E+01	8.46E+01	8.72E+01	8.72E+01	8.72E+01
^{132}I	3.51E+02	4.81E+02	4.86E+02	4.86E+02	4.86E+02
^{133}I	3.26E+02	5.51E+02	5.65E+02	5.65E+02	5.65E+02
^{134}I	4.91E+02	5.52E+02	5.53E+02	5.53E+02	5.53E+02
^{135}I	4.46E+02	7.06E+02	7.22E+02	7.22E+02	7.22E+02
Total	1.66E+03	2.37E+03	2.41E+03	2.41E+03	2.41E+03

Table 15.6-3

Instrument Line Failure
Radiological Effects

Area	Total Effective Dose Equivalent (rem)	Limit (rem TEDE)
Control Room	1.58	5
Exclusion area (1950 m) (2 hr)	0.36	2.5
Low population zone (4827 m) (30 days)	0.16	2.5

Table 15.6-4

Sequence of Events for Steam Line Break
Outside Containment

Time	Event
0	Guillotine break of one main steam line outside primary containment.
0.5[a]	High steam line flow signal initiates closure of MSIV.
<1.0	Reactor begins to scram.
≥ 6.0	Main steam line isolation valves fully closed.
10	Safety/relief valves open on high vessel pressure. The valves open and close to maintain vessel pressure at approximately 1100 psi.
600	Operator initiates ADS or manually controls relief valves. Vessel depressurizes rapidly.
750	High-pressure core spray initiates on low water level.
1270	Core effectively reflooded. No fuel rod failure.

[a] Approximately.

Table 15.6-5

Steam Line Break Accident - Parameters
Tabulated for Postulated Accident Analyses

Parameters	Design Basis Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accidents.	
A. Power level	N/A
B. Burnup	N/A
C. Fuel damaged	None
D. Release of activity by nuclide	Table 15.6-6
E. Iodine fractions[1]	
(1) Organic	N/A
(2) Elemental	N/A
(3) Particulate	N/A
F. Reactor coolant activity before the accident corresponds to the iodine spike of 4 µci/gm dose-equivalent I-131	4 µci/gm
II. Data and assumptions used to estimate activity released.	
A. Primary containment leak rate (%/day)	N/A
B. Secondary containment leak rate (%/day)	N/A
C. Isolation valve closure time (sec)	6
D. Adsorption and filtration efficiencies	
(1) Organic iodine[1]	N/A
(2) Elemental iodine	N/A
(3) Particulate iodine	N/A
(4) Particulate fission products	N/A

[1] Since no filtration is credited, speciation of iodines is not applicable.

Table 15.6-5

Steam Line Break Accident - Parameters
Tabulated for Postulated Accident Analyses (Continued)

				Design Basis Assumptions
	E.	Recirculation system parameters		N/A
		(1)	Flow rate	N/A
		(2)	Mixing efficiency	N/A
		(3)	Filter efficiency	N/A
	F.	Containment spray parameters (flow rate, drop size, etc.)		N/A
	G.	Containment volumes		N/A
	H.	All other pertinent data and assumptions		None
III.	Dispersion data			
		(1)	Offsite	Table 15.0-4
		(2)	Control Room	8.19 E-4 sec/m^3
IV.	Dose data			
	A.	Method of dose calculation		Reference 15.6-3
	B.	Dose conversion assumptions		Reference 15.6-7
	C.	Peak activity concentrations in containment		N/A
	D.	Doses		Table 15.6-7

Table 15.6-6

Steam Line Break Accident
Activity Release to Environment (Curies)

Isotope	Activity Released
^{131}I	1.91E 02

Table 15.6-7

Steam Line Break Accident
Radiological Effects of a Puff Release

Area	TEDE (rem)
Exclusion area (1950 m)	0.40
Low population zone (4827 m)	0.11
Control Room	1.8

Table 15.6-8

Loss-of-Coolant Accident - Parameters
Tabulated for Postulated Accident Analysis

Parameters			Design Basis Assumptions	
I.	Data and assumptions used to estimate radioactive source from postulated accidents			
	A.	Power level	3556	
	B.	Burnup	N/A	
	C.	Fuel damaged	100%	
	D.	Airborne activity by nuclide	Table 15.6-10 and 15.6-11	
	E.	Iodine fractions		
		(1) Organic	0.0015	
		(2) Elemental	0.0485	
		(3) Particulate	0.95	
	F.	Reactor coolant activity before the accident	N/A	
II.	Data and assumptions used to estimate activity released			
	A.	Primary containment leak rate includes MSIV leakage (% volume/day)	0 – 24 hrs 24 – 720 hrs	0.5 0.25
	B.	Secondary containment leak rate (%/day)	N/A	
	C.	Drawdown period (minutes)	20	
	D.	Adsorption and filtration efficiencies (%)		
		(1) Organic iodine	98%	
		(2) Elemental iodine	98%	
		(3) Particulate iodine	98%	
		(4) Particulate fission products	98%	

Table 15.6-8

Loss-of-Coolant Accident - Parameters
Tabulated for Postulated Accident Analysis (Continued)

		Parameters		Design Basis Assumptions	
	E.	Secondary containment volumetric flow rate bypassing SGT filters[1], cfm		50	
	F.	Secondary containment bypass leakage		0 – 24 hrs 24 – 720 hrs	0.04% volume per day 0.02% volume per day
	G.	Recirculation system parameters			
		(1)	Flow rate (cfm)	N/A	
		(2)	Mixing efficiency	N/A	
		(3)	Filter efficiency	N/A	
	H.	Containment spray removal rates		Time (hr)	Removal Rate (1/hr)
				0	0.0
				0.25	6.20
				2.44	0.62
				24.0	0.0
	I.	Containment volumes		Table 6.2-1	
	J.	MSIV leak rate per steam line		0 – 24 hrs 24 – 720 hrs	16 scfh 8 scfh
	K.	ESF leakage into secondary containment		2 gpm	
	L.	CREF bypass leakage		50 cfm	
III.	Dispersion data				
		(1)	Offsite	Table 15.0-4	
		(2)	Control room	Table 15.0-5	

[1] SGT filter bypass will reduce SGT filter efficiency from 99% to 98%.

Table 15.6-8

Loss-of-Coolant Accident - Parameters
Tabulated for Postulated Accident Analysis (Continued)

Parameters			Design Basis Assumptions
IV.	Dose data		
	A.	Method of dose calculation	Reference 15.6-5
	B.	Dose conversion assumptions	Reference 15.6-7, 15.6-8
	C.	Peak activity concentrations in containment	Table 15.6-9
	D.	Doses	Table 15.6-12

Table 15.6-9

Loss-of-Coolant Accident
Primary Containment[1] Activity (Curies)

Isotope	0.25 hr	0.5 hr	0.8 hr	1.0 hr	2.0 hr	4.0 hr	8.0 hr	24.0 hr	30 day
^{131}I	2.15E+06	1.72E+06	2.42E+06	2.63E+06	2.69E+06	3.93E+05	1.25E+05	4.09E+04	3.03E+03
^{132}I	2.82E+06	2.10E+06	2.75E+06	2.79E+06	2.15E+06	2.17E+05	3.57E+04	1.12E+02	1.03E-01
^{133}I	4.16E+06	3.30E+06	4.62E+06	4.97E+06	4.95E+06	6.82E+05	1.92E+05	3.92E+04	3.25E-06
^{134}I	3.81E+06	2.49E+06	2.87E+06	2.55E+06	1.18E+06	3.47E+04	4.52E+02	4.13E-04	0
^{135}I	3.78E+06	2.94E+06	4.05E+06	4.29E+06	3.97E+06	4.77E+05	1.02E+05	6.80E+03	0
Total iodines	1.67E+07	1.26E+07	1.67E+07	1.72E+07	1.49E+07	1.80E+06	4.55E+05	8.71E+04	3.03E+03
^{83m}Kr	2.51E+05	4.92E+05	1.80E+06	3.02E+06	5.87E+06	2.84E+06	6.34E+05	1.58E+03	0
^{85m}Kr	5.45E+05	1.13E+06	4.35E+06	7.71E+06	1.86E+07	1.39E+07	7.36E+06	5.84E+05	0
^{85}Kr	3.17E+04	6.82E+04	2.74E+05	5.05E+05	1.43E+06	1.46E+06	1.46E+06	1.45E+06	1.29E+06
^{87}Kr	9.02E+05	1.69E+06	5.92E+06	9.53E+06	1.56E+07	5.33E+06	5.97E+05	9.35E+01	0
^{88}Kr	1.38E+06	2.79E+06	1.05E+07	1.82E+07	4.03E+07	2.51E+07	9.29E+06	1.76E+05	0
^{89}Kr	6.47E+04	5.31E+03	8.12E+02	5.71E+01	3.42E-04	1.60E-15	3.36E-38	5.95E-129	0
^{131m}Xe	2.15E+04	4.63E+04	1.86E+05	3.42E+05	9.66E+05	9.81E+05	9.71E+05	9.30E+05	1.56E+05
^{133m}Xe	1.28E+05	2.74E+05	1.10E+06	2.01E+06	5.63E+06	5.60E+06	5.33E+06	4.33E+06	6.17E+02
^{133}Xe	4.19E+06	9.00E+06	3.61E+07	6.64E+07	1.87E+08	1.89E+08	1.86E+08	1.72E+08	3.52E+06
^{135m}Xe	4.40E+05	4.87E+05	1.00E+06	9.51E+05	1.88E+05	9.29E+02	2.18E-02	0	0
^{135}Xe	1.02E+06	2.23E+06	8.57E+06	1.56E+07	4.20E+07	4.21E+07	3.82E+07	1.79E+07	0
^{137}Xe	2.50E+05	3.75E+04	1.05E+04	1.35E+03	8.99E-02	5.09E-11	1.59E-29	1.52E-103	0
^{138}Xe	1.50E+06	1.75E+06	3.81E+06	3.82E+06	9.34E+05	7.13E+03	3.98E-01	3.87E-18	0
Total noble gases	1.07E+07	2.00E+07	7.36E+07	1.28E+08	3.19E+08	2.86E+08	2.50E+08	1.97E+08	4.97E+06
Alkali metals	9.84E+05	7.85E+05	9.21E+05	9.62E+05	9.72E+05	1.29E+05	3.05E+04	9.57E+01	7.73E+01
Te-group	2.83E+05	2.31E+05	2.04E+06	2.49E+06	2.41E+06	3.09E+05	6.56E+04	1.72E+02	5.34E+01
Noble metals	0.00E+00	0.00E+00	1.55E+05	1.97E+05	1.99E+05	2.57E+04	5.82E+03	1.60E+01	3.99E+00
La-group	0.00E+00	0.00E+00	2.40E+04	3.19E+04	3.17E+04	9.64E+03	3.11E+03	1.33E+01	1.00E+01
Ce-group	0.00E+00	0.00E+00	1.15E+05	1.46E+05	1.52E+05	1.98E+04	4.49E+03	1.18E+01	1.05E+00

[1]Primary Containment includes the Dry Well & the Wet Well Air-Space

Table 15.6-10

Loss-of-Coolant Accident
Secondary Containment Activity (Curies) - 20 Minute Drawdown

Isotope	0.25 hr	0.5 hr	0.75 hr	1 hr	2 hr	4 hr	8 hr	24 hr	30 day
^{131}I	0	6.89E+00	9.90E+00	1.15E+01	1.50E+01	7.11E+00	6.08E+00	5.47E+00	4.11E-01
^{132}I	0	8.38E+00	1.13E+01	1.22E+01	1.19E+01	3.26E+00	8.84E-01	6.86E-03	1.79E-07
^{133}I	0	1.32E+01	1.89E+01	2.17E+01	2.75E+01	1.23E+01	9.36E+00	5.24E+00	0
^{134}I	0	9.98E+00	1.18E+01	1.12E+01	6.54E+00	6.29E-01	2.20E-02	5.53E-08	0
^{135}I	0	1.18E+01	1.65E+01	1.88E+01	2.21E+01	8.61E+00	4.95E+00	9.08E-01	0
Total iodines	0	5.02E+01	6.84E+01	7.54E+01	8.29E+01	3.19E+01	2.13E+01	1.16E+01	4.11E-01
^{83m}Kr	0	1.65E+00	5.88E+00	1.02E+01	2.02E+01	9.84E+00	2.20E+00	5.47E-03	0
^{85m}Kr	0	3.77E+00	1.42E+01	2.59E+01	6.40E+01	4.81E+01	2.55E+01	2.03E+00	0
^{85}Kr	0	2.28E-01	8.97E-01	1.70E+00	4.91E+00	5.06E+00	5.06E+00	5.03E+00	2.24E+00
^{87}Kr	0	5.66E+00	1.94E+01	3.21E+01	5.36E+01	1.85E+01	2.07E+00	3.25E-04	0
^{88}Kr	0	9.33E+00	3.44E+01	6.13E+01	1.38E+02	8.69E+01	3.22E+01	6.10E-01	0
^{89}Kr	0	1.78E-02	2.66E-03	1.92E-04	1.17E-09	5.56E-21	1.17E-43	0	0
^{131m}Xe	0	1.55E-01	6.08E-01	1.15E+00	3.32E+00	3.40E+00	3.37E+00	3.22E+00	2.70E-01
^{133m}Xe	0	9.17E-01	3.59E+00	6.78E+00	1.93E+01	1.94E+01	1.85E+01	1.50E+01	1.07E-03
^{133}Xe	0	3.01E+01	1.18E+02	2.24E+02	6.43E+02	6.59E+02	6.49E+02	6.06E+02	6.47E+00
^{135m}Xe	0	1.63E+00	3.29E+00	3.20E+00	6.44E-01	3.22E-03	7.58E-08	0	0
^{135}Xe	0	7.70E+00	2.87E+01	5.35E+01	1.50E+02	1.61E+02	1.57E+02	8.24E+01	0
^{137}Xe	0	1.25E-01	3.43E-02	4.53E-03	3.09E-07	1.77E-16	5.52E-35	0	0
^{138}Xe	0	5.87E+00	1.25E+01	1.28E+01	3.21E+00	2.47E-02	1.38E-06	0	0
Total noble gases	0	6.72E+01	2.42E+02	4.33E+02	1.10E+03	1.01E+03	8.95E+02	7.14E+02	8.98E+00
Alkali metals	0	2.75E+00	3.18E+00	3.33E+00	3.38E+00	4.51E-01	1.07E-01	3.34E-04	1.34E-04
Te-group	0	8.69E-01	6.92E+00	8.69E+00	8.45E+00	1.08E+00	2.29E-01	5.99E-04	9.28E-05
Noble metals	0	0.00E+00	5.18E-01	6.76E-01	6.92E-01	8.96E-02	2.03E-02	5.60E-05	6.92E-06
La-group	0	0.00E+00	8.11E-02	1.12E-01	1.12E-01	3.36E-02	1.09E-02	4.65E-05	1.74E-05
Ce-group	0	0.00E+00	3.82E-01	5.04E-01	5.30E-01	6.92E-02	1.57E-02	4.11E-05	1.81E-06

Table 15.6-11

Loss-of-Coolant Accident
Activity Released to the Environment (Curies) - 20 Minute Drawdown

Isotope	0.25 hrs	0.5 hrs	0.75 hrs	1 hrs	2 hrs	4 hrs	8 hrs	24 hrs	30 d
^{131}I	6.59E+01	1.39E+02	1.80E+02	2.31E+02	4.48E+02	5.49E+02	6.37E+02	8.35E+02	3.03E+03
^{132}I	8.83E+01	1.82E+02	2.31E+02	2.89E+02	5.27E+02	7.45E+02	1.13E+03	2.52E+03	1.15E+04
^{133}I	1.28E+02	2.69E+02	3.47E+02	4.44E+02	8.49E+02	1.03E+03	1.17E+03	1.42E+03	1.67E+03
^{134}I	1.23E+02	2.45E+02	2.99E+02	3.53E+02	4.98E+02	5.25E+02	5.28E+02	5.28E+02	5.28E+02
^{135}I	1.17E+02	2.44E+02	3.13E+02	3.97E+02	7.35E+02	8.74E+02	9.63E+02	1.05E+03	1.06E+03
Total iodines	5.22E+02	1.08E+03	1.37E+03	1.71E+03	3.06E+03	3.72E+03	4.43E+03	6.35E+03	1.78E+04
^{83m}Kr	1.09E+01	4.59E+01	1.47E+02	3.80E+02	2.19E+03	4.92E+03	6.81E+03	7.35E+03	7.35E+03
^{85m}Kr	2.34E+01	1.02E+02	3.43E+02	9.21E+02	6.15E+03	1.67E+04	2.99E+04	4.36E+04	4.42E+04
^{85}Kr	1.34E+00	6.00E+00	2.09E+01	5.82E+01	4.33E+02	1.37E+03	3.25E+03	1.07E+04	1.64E+05
^{87}Kr	3.98E+01	1.62E+02	5.03E+02	1.25E+03	6.46E+03	1.27E+04	1.55E+04	1.59E+04	1.59E+04
^{88}Kr	5.95E+01	2.55E+02	8.42E+02	2.22E+03	1.40E+04	3.49E+04	5.53E+04	6.71E+04	6.73E+04
^{89}Kr	9.01E+00	1.13E+01	1.16E+01	1.16E+01	1.16E+01	1.16E+01	1.16E+01	1.16E+01	1.16E+01
^{131m}Xe	9.12E-01	4.07E+00	1.42E+01	3.94E+01	2.93E+02	9.28E+02	2.18E+03	7.06E+03	5.55E+04
^{133m}Xe	5.42E+00	2.41E+01	8.40E+01	2.33E+02	1.72E+03	5.38E+03	1.24E+04	3.71E+04	9.18E+04
^{133}Xe	1.77E+02	7.92E+02	2.76E+03	7.66E+03	5.68E+04	1.80E+05	4.22E+05	1.35E+06	6.45E+06
^{135m}Xe	2.28E+01	6.80E+01	1.40E+02	2.36E+02	4.35E+02	4.58E+02	4.59E+02	4.59E+02	4.59E+02
^{135}Xe	4.33E+01	1.98E+02	6.85E+02	1.86E+03	1.34E+04	4.24E+04	9.96E+04	2.67E+05	3.33E+05
^{137}Xe	2.69E+01	3.79E+01	3.99E+01	4.04E+01	4.04E+01	4.04E+01	4.04E+01	4.04E+01	4.04E+01
^{138}Xe	7.66E+01	2.35E+02	5.02E+02	8.78E+02	1.74E+03	1.87E+03	1.87E+03	1.87E+03	1.87E+03
Total noble gases	4.97E+02	1.94E+03	6.09E+03	1.58E+04	1.04E+05	3.02E+05	6.49E+05	1.81E+06	7.23E+06
Alkali metals	2.99E+01	6.16E+01	7.76E+01	9.54E+01	1.68E+02	1.96E+02	2.10E+02	2.14E+02	2.16E+02
Te-group	6.96E+00	1.64E+01	3.87E+01	8.25E+01	2.70E+02	3.38E+02	3.71E+02	3.81E+02	3.82E+02
Noble metals	0	0	1.54E+00	4.94E+00	2.01E+01	2.56E+01	2.84E+01	2.93E+01	2.94E+01
La-group	0	0	2.30E-01	7.72E-01	3.23E+00	4.56E+00	5.82E+00	6.30E+00	6.65E+00
Ce-group	0	0	1.13E+00	3.66E+00	1.51E+01	1.94E+01	2.15E+01	2.22E+01	2.22E+01

Table 15.6-12

Loss-of-Coolant Accident
(Design Basis Analysis)
Radiological Effects

Total Effect	TEDE (rem)
Exclusion area (1950 m) (2 hr)	4.1
Low population zone (4827 m) (30 days)	4.0
Control Room (30 days)	3.5

Table 15.6-13

Sequence of Events for Feedwater Line Break
Outside Containment

Time	Event
0	One feedwater line breaks.
0+	Feedwater line check valves isolate the reactor from the break.
<30 sec	At low reactor water level, reactor scram would initiate and, at low-low reactor water level, HPCS and MSIV closure[b] would initiate and recirculation pumps would trip.
2 minutes[a]	The SRVs open and close and maintain the reactor vessel pressure at approximately 1100 psig.
1 to 2 hr	Normal reactor cooldown established.

[a] Approximately.

[b] The analysis has not been updated for the change in MSIV isolation setpoint from Level 2 to Level 1 because it remains bounded by the recirculation line break LOCA (Reference 15.6-9).

Table 15.6-14

Feedwater Line Break Accident - Parameters
Tabulated for Postulated Accident Analysis

Parameter			Value
I.	Data and assumptions used to estimate radioactive source from postulated accidents		
	A.	Power level	N/A
	B.	Burnup	N/A
	C.	Fuel damaged	None
	D.	Release of activity by nuclide	Table 15.6-15
	E.	Iodine fractions	
		(1) Organic	0
		(2) Elemental	1%
		(3) Particulate	0
		(4) Reactor coolant activity before the accident	Section 15.6.6.5.1
II.	Data and assumptions used to estimate activity released		
	A.	Primary containment leak rate (%/day)	N/A
	B.	Secondary containment leak rate (%/day)	N/A
	C.	RWCU total isolation valve closure time (sec)	75
	D.	Adsorption and filtration efficiencies	
		(1) Organic iodine	N/A
		(2) Elemental iodine	N/A
		(3) Particulate iodine	N/A
		(4) Particulate fission products	N/A
	E.	Recirculation system parameters	N/A
		(1) Flow rate	N/A
		(2) Mixing efficiency	N/A
		(3) Filter efficiency	N/A
	F.	Containment spray parameters (flow rate, drop size, etc.)	N/A
	G.	Containment volumes	N/A
	H.	All other pertinent data and assumptions	None

Table 15.6-14

Feedwater Line Break Accident - Parameters
Tabulated for Postulated Accident Analysis (Continued)

Parameter			Value
III.	Dispersion data		
	A.	Boundary and LPZ distance (m)	1950/4827
	B.	χ/Qs for time intervals of 0-2 hr - EAB/LPZ	2.62×10^{-4}/1.06×10^{-4}
IV.	Dose data		
	A.	Method of dose calculation	Reference 15.6-2
	B.	Dose conversion assumptions	Reference 15.6-2
	C.	Peak activity concentrations in containment	N/A
	D.	Doses	Table 15.6-16

Table 15.6-15

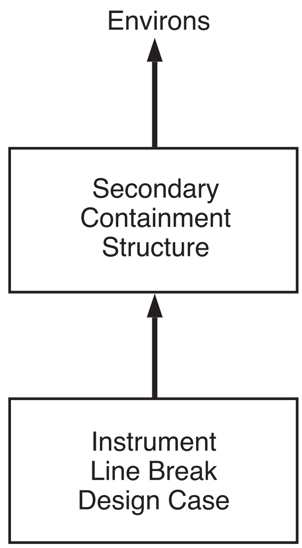
Feedwater Line Break Accident
Activity Release to Environment (Curies)

Isotope	Activity
^{131}I	2.22×10^{-2}
^{132}I	2.05×10^{-1}
^{133}I	1.52×10^{-1}
^{134}I	4.45×10^{-1}
^{135}I	2.22×10^{-1}
Total	1.04 x 10

Table 15.6-16

Feedwater Line Break Accident
Biological Effects of a Puff Release

Area	Whole Body Dose (rem)	Thyroid Dose (rem)
Exclusion area (1950 m)	1.37×10^{-4}	5.47×10^{-3}
Low population zone (4827 m)	5.53×10^{-5}	2.21×10^{-3}



Columbia Generating Station
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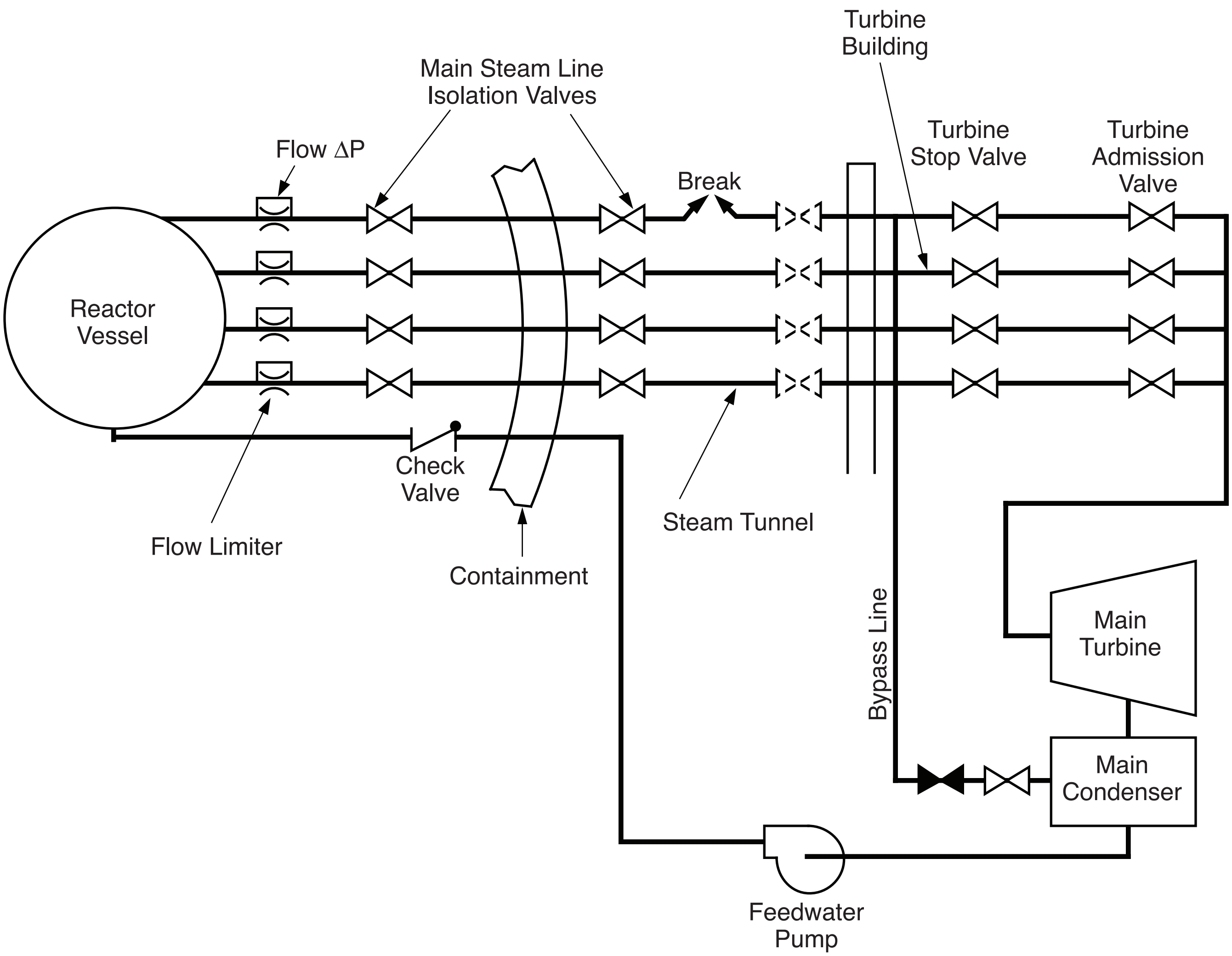
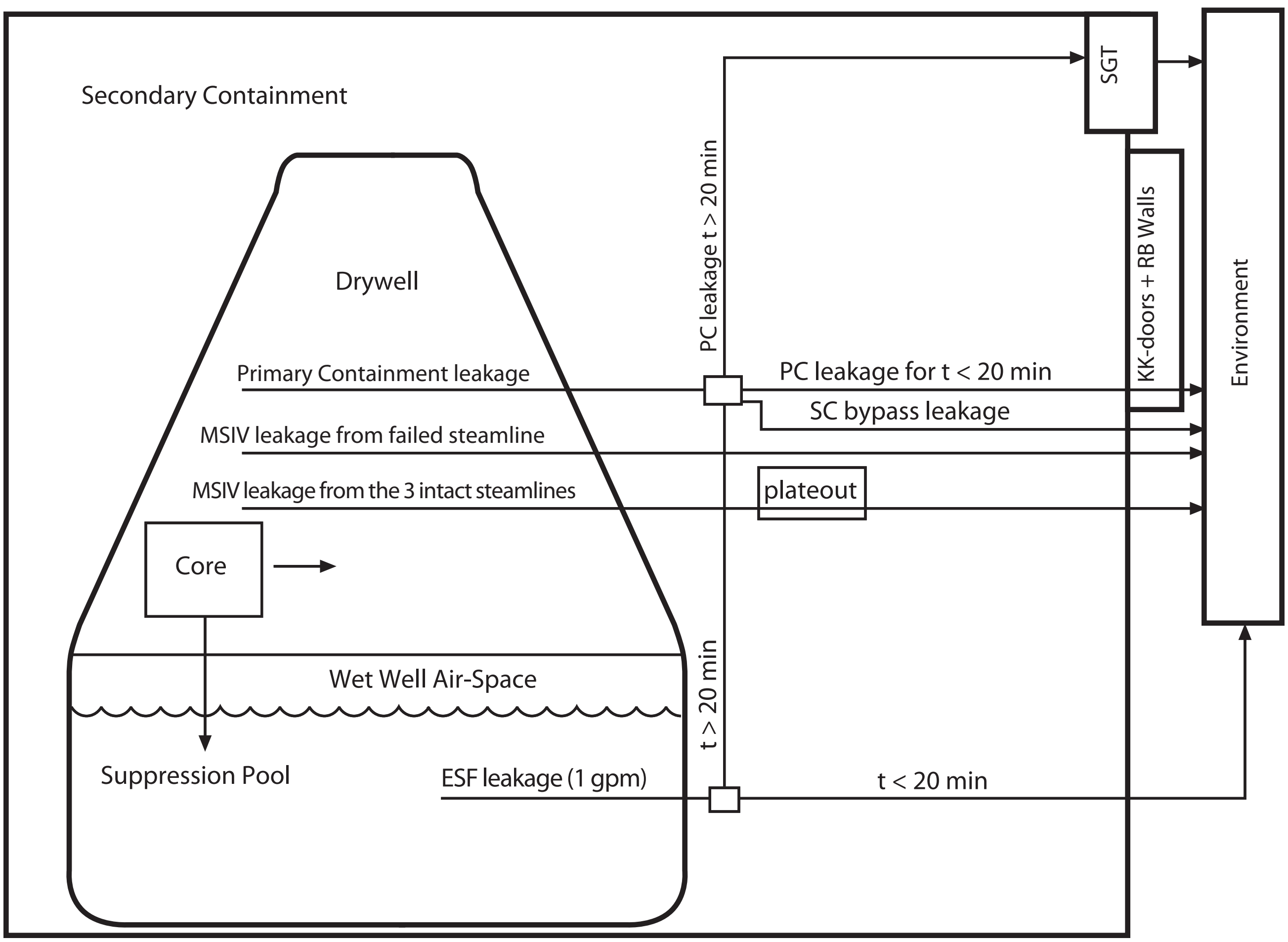
Leakage Path for Instrument Line Break

Draw. No. 900547.73 | Rev. | Figure 15.6-1



Steam Flow Schematic for Steam Break Outside Containment

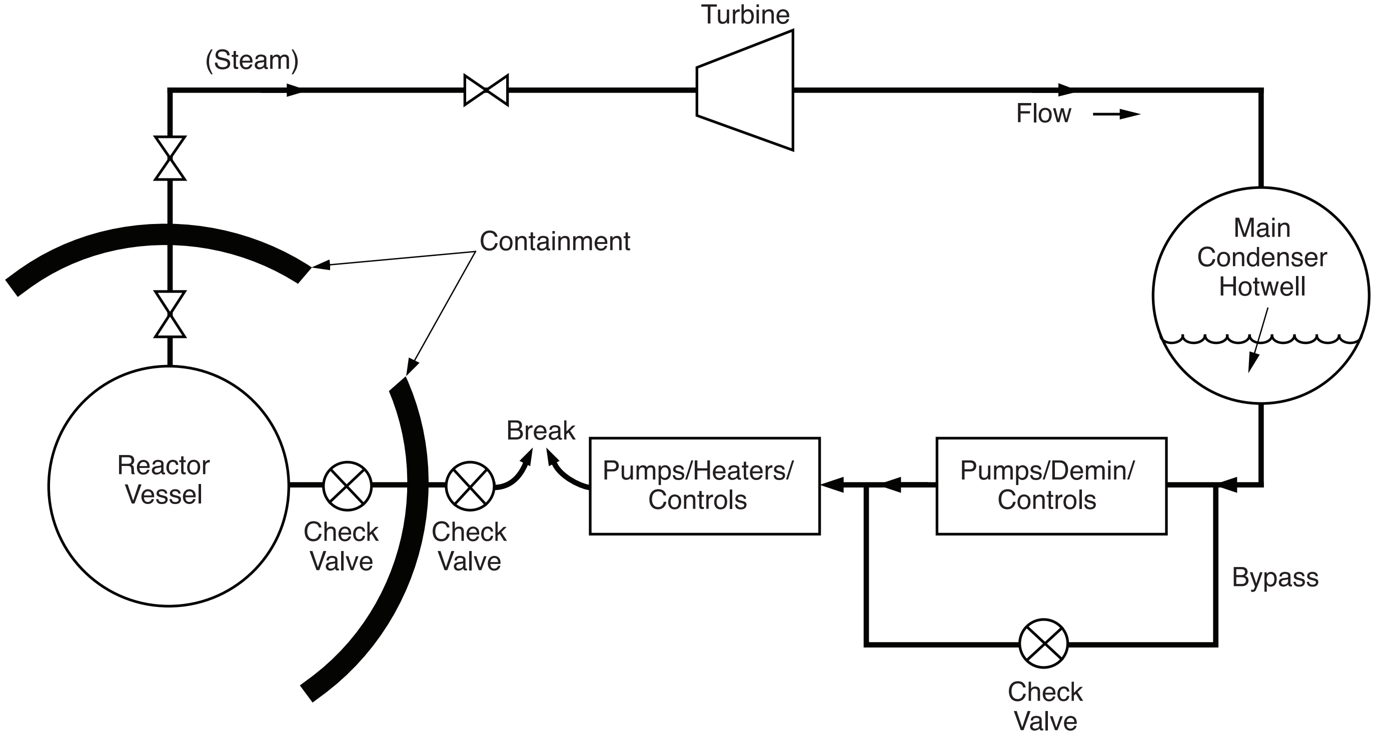
Draw. No. 900547.74 | Rev. | Figure 15.6-2



Columbia Generating Station
Final Safety Analysis Report

Leakage Path for LOCA

Draw. No. 900547.75 | Rev. | Figure 15.6-3



Leakage Path for Feedwater Line Break
Outside Containment

Draw. No. 900547.76 | Rev. | Figure 15.6-4

15.7 RADIOACTIVE RELEASE FROM SUBSYSTEMS AND COMPONENTS

These events are classified as nonlimiting events for both original and uprated power conditions. Therefore, no further analysis has been performed.

15.7.1 RADIOACTIVE GAS WASTE SYSTEM LEAK OR FAILURE

Not applicable.

15.7.2 LIQUID RADIOACTIVE SYSTEM FAILURE

Not applicable.

15.7.3 POSTULATED RADIOACTIVE RELEASES DUE TO LIQUID RADWASTE TANK FAILURE

15.7.3.1 Identification of Causes and Frequency Classification

15.7.3.1.1 Identification of Causes

The liquid radwaste tanks are constructed to specific engineering codes and standards and to the uniform building code seismic requirements. These tanks operate at atmosphere pressure and low temperatures. A positive action interlock system is provided to prevent inadvertent opening of a drain valve because of operator error. Accordingly, the possibility of a complete tank failure or drainage is considered small.

An unspecified event is postulated to cause the complete release of the average radioactivity inventory in the tank containing the largest quantities of significant radionuclides in the liquid radwaste system. The tank postulated to rupture is one of the two decontamination solution concentrated waste tanks (see Figure 11.2-1).

15.7.3.1.2 Frequency Classification

This accident is categorized as a limiting fault.

15.7.3.2 Sequence of Events and Systems Operation

15.7.3.2.1 Sequence of Events

The sequence of events expected to occur is as follows:

Sequence of Events - Liquid Radwaste Tank Failure

Events	Time
Event begins-failure occurs	0
Area radiation alarms alert plant personnel	~ 1 minute
Operator action begins	~ 10 minute

15.7.3.2.2 Systems Operation

Failure of a concentrated waste tank does not require a shutdown nor does it impair a safe shutdown. It will lead to limited operation of the concentrated waste system using the remaining tank.

The liquid contents of this tank will also be contained by the building walls and an unlined, 18-in. high concrete dike around the radwaste tank area. Floor drain sump pumps would receive a high water level alarm, activate automatically, and remove the spilled liquid.

15.7.3.2.3 The Effects of Single Failures and Operator Errors

This event has been analyzed without taking credit for any expected operator action or system operation; therefore, a discussion of single equipment failure or single operator error is not applicable.

15.7.3.3 Core and System Performance

The failure of this liquid radwaste system component does not directly affect the nuclear steam supply system (NSSS). It will lead to decoupling of NSSS with the subject system.

This failure has no applicable effect on the reactor core or the NSSS safety performance. Specific assumptions and parameters are presented in Table 15.7-1.

15.7.3.4 Barrier Performance

This event does not involve any containment barrier integrity except the tank itself and the radwaste building. The dike and walls of the radwaste building surrounding the tanks are built to Seismic Category I criteria. In the analysis of spill consequences, no credit is taken for the dike or radwaste building in recontaining the spilled liquid.

15.7.3.5 Radiological Consequences

The entire volume (700 gal) of concentrator waste tank assumed to spill with isotope inventory given in Table 11.2-1. Tritium concentration is assumed to be 0.01 µCi/ml (Environmental Report, Operating License Section 3.5.1).

The hypothetical radwaste tank failure was evaluated using conservative assumptions such as no containment in the radwaste building and unimpeded flow vertically through 50-60 ft of sand and gravel.

The following offsite concentration data for the radionuclides of interest are provided for the WNP-1/4 wells and at the Columbia River:

Radionuclide	Concentration at WNP-1/4 Wells (µCi/ml)	Concentration at Columbia River (µCi/ml)	Concentration Limit (µCi/ml)
^{3}H	1.0×10^{-7}	1.3×10^{-8}	1×10^{-3}
^{90}Sr	1.7×10^{-4}	4.2×10^{-7}	5×10^{-7}
^{137}Cs	2.2×10^{-10}	1.4×10^{-27}	1×10^{-6}

The calculations show the strontium concentration exceeding the unrestricted area limitation at the WNP-1/4 wells. These wells are a temporary water supply and are under the control of Energy Northwest. Should a spill occur there will be ample time to assess the severity and extent of contamination.

Concentration at the river bank will be diluted by the river flow. The nearest surface water users are several miles downstream.

15.7.4 FUEL HANDLING ACCIDENT

15.7.4.1 Identification of Causes and Frequency Classification

15.7.4.1.1 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 dropping a raised irradiated fuel assembly onto other fuel bundles seated in the reactor pressure vessel (RPV). The event was selected as the bounding event because it considers the maximum height and weight, while assuming a minimum water level above the damaged fuel.

15.7.4.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.7.4.2 Sequence of Events and Systems Operation

The following sequence of events is assumed in the analysis.

No fuel movement will take place in the first 24 hr following shutdown. At 24 hr post-shutdown fuel movement starts and the fuel handling equipment is assumed to fail dropping the fuel grapple and an irradiated fuel bundle onto the irradiated fuel bundles seated in the RPV. Fuel is damaged and fission products are released to the reactor coolant, then to the reactor building atmosphere, and finally to the environment over a 2-hr period. No credit is taken for holdup or mixing in the reactor building, nor is credit taken for filtration by the standby gas treatment (SGT).

15.7.4.2.1 The Effects of Single Failures and Operator Errors

No systems or operator actions are credited to mitigate a fuel handling accident.

15.7.4.3 Core and System Performance

15.7.4.3.1 Mathematical Model

Because of the complex nature of the impact and the resulting damage to fuel assembly components, a rigorous prediction of the number of failed rods is not possible. For this reason, a simplified energy approach was taken and numerous conservative assumptions were made to ensure a conservative estimate of the number of failed rods.

The kinetic energy acquired by a falling fuel assembly may be dissipated in one or more impacts. The energy absorption on successive impacts is estimated by considering a plastic impact.

The energy transferred in the dropped assembly is considered in two phases. First, the fuel assembly is expected to impact on the reactor core at a small angle from the vertical, inducing a bending mode of failure on the fuel rods of the dropped assembly. The kinetic energy of the fall is dissipated in the impact. The analysis assumes that the energy of the dropped assembly is absorbed by only the cladding and other core structures. The assumption that no energy is absorbed by the fuel material results in considerable conservatism in the mass-energy calculations. Half of the energy is dissipated in the structure of the dropped assembly, failing all the rods in the assembly. The remaining half is allocated evenly across the structural mass of the impacted assemblies. The energy dissipated by the cladding is calculated by multiplying the impacted assembly energy by the cladding mass fraction and dividing by the energy required to fail a rod (based on 1% uniform plastic deformation).

The second phase considers the kinetic energy developed by the irradiated fuel assembly and lifting mechanism tipping over and impacting the core horizontally. The kinetic energy developed is equal to the initial potential energy of the assembly relative to the top of the core. Again, half of the energy is absorbed by the dropped assembly and half by the impacted assemblies. The number of failed rods in the impacted assemblies is determined using the cladding mass fraction and the energy required to fail a rod.

15.7.4.3.2 Input Parameters and Initial Conditions

The parameters and conditions used to determine the number of failed rods are listed below:

a. The fuel assembly is dropped from a height of 34 ft. The maximum height allowed by the fuel handling equipment is less than 34 ft;

b. The dropped mass consists of a fuel assembly (586 lb bounding analyzed wet weight for GE14 10 x 10 fuel, 617 lb wet weight for GE 8 x 8 fuel and 665 lb dry weight for ATRIUM-10) and the fuel grapple (350 lb wet weight);

c. The energy required to fail a fuel rod is approximately 175 ft-lb for GE14 10 x 10 fuel, 250 ft-lb for GE 8 x 8 fuel and 205 ft-lb for ATRIUM-10; see Reference 15.7-3 for SVEA-96.

15.7.4.3.3 Results

Based on a core of GE 8 x 8 fuel, the calculation predicts 124 failed fuel rods; 62 rods in the dropped assembly, 43 rods in the first impact, and 19 additional rods in the second impact. Westinghouse analysis predicts a maximum of 123 failed rods (Reference 15.7-3) and AREVA NP calculated that up to 156 rods could fail (Reference 15.7-4). Analysis of the GE14 10 x 10 fuel estimates that a total of 151 fuel rods will fail (Reference 15.7-5).

15.7.4.4 Barrier Performance

The reactor coolant pressure boundary, primary containment and secondary containment are open at the time of the accident. However, a similar event could occur in the spent fuel pool (SFP), during spent fuel transfer from the RPV to, or handling in, the SFP. Assuming a drop height of 4 ft, the number of failed rods as a result of a GE 8 x 8 bundle (unchanneled) drop in the SFP was calculated and found to be 90 rods; the number for channeled GE 8 x 8 is less than that. The dose analysis for a drop of a bundle over the core, which assumes 250 failed rods, bounds a drop of a bundle in the SFP.

15.7.4.5 Radiological Consequences

The fission product inventory is based on a plant-specific ORIGEN 2 run for pre-power uprate basis of 3489 MW with 1000 days exposure adjusted as described in Section 15.4.9.5.1. The release is based on damage to 250 fuel rods. A 24-hr period for decay from the power condition is assumed. Figure 15.7-1 indicates the leakage flow path for this accident.

15.7.4.5.1 Design Basis Analysis

Specific values of parameters used in the evaluation are presented in Table 15.7-2. The dispersion coefficients used to determine offsite doses are presented in Table 15.0-4.

15.7.4.5.1.1 Fission Product Release From Fuel. The fission product inventory of a core average exposure fuel rod is adjusted by a peaking factor of 1.7 to establish the inventory of each damaged rod. Five percent of the noble gases inventory (10% for ^{85}Kr) and 5% of the iodine inventory (8% for ^{131}I), and 12% of the alkali metals inventory are assumed to be released to the reactor well. The activity airborne in the secondary containment is presented in Table 15.7-3.

15.7.4.5.1.2 Fission Product Transport to the Environment. The transport pathway consists of mixing in the reactor well water, migration from the reactor well to the secondary containment atmosphere, and release to the environment without passing through the SGT. All of the noble gas, 0.5% of the iodines, and 0% of the alkali metals are assumed to become airborne in the secondary containment (Reference 15.7-1).

From the activity airborne in the reactor building, 99% is released to the environment in 2 hr.

The release of activity to the environment is presented in Table 15.7-4.

15.7.4.5.1.3 Results. The calculated doses for the design basis analysis are presented in Table 15.7-5 and are within the limits of 10 CFR 50.67.

15.7.5 SPENT FUEL CASK DROP ACCIDENT

The spent fuel cask is equipped with ANSI N14.6 (Reference 15.7-2) compliant lifting lugs and a lifting yoke compatible with the reactor building crane main hook. The reactor building crane is provided with sufficient redundancy such that no credible postulated failure of any crane component required to lift, hold, and move loads, will result in the dropping of the fuel cask. Therefore, an analysis of the spent fuel cask drop is not required.

15.7.6 REFERENCES

15.7-1 Energy Northwest, "Columbia Generating Station Alternative Source Term," CGS-FTS-0168, Revision 0, August 2007.

15.7-2 "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More," ANSI N14.6-1993, June 1993.

15.7-3 ABB/Combustion Engineering, "Fuel Assembly Mechanical Design Report for WNP-2," CE NPSD-792-P, May 1996.

15.7-4 AREVA NP, "Columbia Generating Station Cycle 19 Reload Analysis," ANP-2602, Revision 0, March 2007.

15.7-5 GEH-0000-0075-4920, "GE14 Fuel Design Cycle-Independent Analyses for Energy Northwest Columbia Generating Station" (most recent version referenced in the COLR).

Table 15.7-1

Liquid Radwaste Tanks Failure – Parameters and Concentrations

Parameter			Value
I.	Data and assumptions used to estimate radioactive source		Entire volume (700 gal) of concentrator waste tank assumed to spill with isotope inventory given in Table 11.2-1. Tritium concentration assumed to be 0.01 mCi/ml from the CGS ER-OL.
II.	Data and assumptions used to estimate activity released		
	A.	Containment leak rate (%/day)	N/A
	B.	Secondary containment leak rate (%/day)	N/A
	C.	Valve movement times	N/A
	D.	Absorption and filtration efficiencies	N/A
		(1) Organic iodine	N/A
		(2) Elemental iodine	N/A
		(3) Particulate iodine	N/A
		(4) Particulate fission products	N/A
	E.	Recirculation system parameters	N/A
		(1) Flow rate	N/A
		(2) Mixing efficiency	N/A
		(3) Filter efficiency	N/A
	F.	Containment spray parameters (flow rate, drop size, etc.)	N/A
	G.	Containment volumes	N/A
	H.	Other pertinent data and assumptions	See Section 2.4.13.3
III.	Concentration data		

Radionuclide	@ Col. R (µCi/ml)	@ WNP-1/4 Wells (µCi/ml)	Conc. Limit[a] (µCi/ml)
^{3}H	1.0×10^{-7}	1.3×10^{-8}	1×10^{-3}
^{90}Sr	1.7×10^{-4}	4.2×10^{-7}	5×10^{-7}
^{137}Cs	2.2×10^{-10}	1.4×10^{-27}	1×10^{-6}

[a] From 10 CFR Part 20.

Table 15.7-2

Fuel Handling Accident Parameters Tabulated for Postulated Accident Analysis

			Parameters	Design Basis Assumptions
I.	Data and assumptions used to estimate radioactive source from postulated accidents			
	A.	Power level		3556
		Fuel decay period		24 hrs
	B.	Radial peaking factor		1.7
	C.	Assumed fuel damaged		250 rods
		Bundles in the core		764
		Rods per bundle		62
	D.	Release of activity from the gap to the reactor well water		Figure 15.7-1
	E.	Iodine species fractions released		Figure 15.7-1
		(1)	Organic	
		(2)	Elemental	
		(3)	Particulate	
	F.	Reactor coolant activity before the accident		N/A
II.	Data and assumptions used to estimate activity released			
	A.	Primary containment leak rate (%/day)		N/A
	B.	Secondary containment release rate		99% of the activity in 2 hr with a flow rate of 2.3 SC volumes per hr
	C.	Valve movement times		N/A
	D.	SGT filtration		N/A
	E.	Scrubbing by reactor well water		Figure 15.7-1
		(1)	Organic iodine	
		(2)	Elemental iodine	
		(3)	Particulate iodine	
		(4)	Particulate alkali metals	

Table 15.7-2

Fuel Handling Accident Parameters Tabulated
for Postulated Accident Analysis (Continued)

Parameters			Design Basis Assumptions
	F.	Recirculation system parameters	
		(1) Flow rate	N/A
		(2) Mixing efficiency	N/A
		(3) Filter efficiency	N/A
	G.	Containment spray parameters (flow rate, drop size, etc.)	N/A
	H.	Containment volumes	N/A
	I.	Other pertinent data and assumptions	
		(1) SGT filtration	None
		(2) CREF filtration	None
		(3) Holdup in reactor building	None
		(4) Mixing in reactor building	None
III.	Dispersion data (for duration of release, 0 – 2 hr)		
	(1)	Offsite	Table 15.0-4
	(2)	Control room	8.69E-4 sec/m^3
IV.	Dose data		
	A.	Method of dose calculation	Regulatory Guide 1.183
	B.	Dose conversion assumptions	Regulatory Guide 1.183
	C.	Peak activity concentrations in containment	N/A
	D.	Doses	Table 15.7-5

Table 15.7-3

Fuel Handling Accident
Activity Airborne in Secondary Containment (Curies)

Isotope	6 minutes	12 minutes	0.5 hr	1 hr	2 hr	4 hr	8 hr	1 day	4 days	30 days
^{131}I	2.64E+02	2.10E+02	1.05E+02	3.31E+01	3.29E+00	3.26E+00	3.22E+00	3.04E+00	2.35E+00	2.51E-01
^{132}I	1.95E-01	1.50E-01	6.87E-02	1.87E-02	1.39E-03	7.65E-04	2.33E-04	1.98E-06	9.73E-16	6.23E-20
^{133}I	1.58E+02	1.25E+02	6.19E+01	1.93E+01	1.86E+00	1.74E+00	1.52E+00	8.96E-01	8.21E-02	8.31E-11
^{134}I	1.54E-06	1.13E-06	4.44E-07	9.40E-08	4.21E-09	8.45E-10	3.40E-11	8.99E-17	6.50E-19	1.25E-36
^{135}I	2.72E+01	2.14E+01	1.04E+01	3.11E+00	2.80E-01	2.28E-01	1.51E-01	2.91E-02	1.75E-05	1.10E-22
Total iodine	4.49E+02	3.56E+02	1.77E+02	5.55E+01	5.43E+00	5.23E+00	4.90E+00	3.97E+00	2.43E+00	2.51E-01
^{83m}Kr	5.55E-01	4.25E-01	1.90E-01	4.98E-02	3.42E-03	1.62E-03	3.62E-04	9.06E-07	1.78E-18	7.63E-20
^{85m}Kr	2.10E+02	1.64E+02	7.85E+01	2.29E+01	1.95E+00	1.42E+00	7.57E-01	6.04E-02	6.91E-07	5.85E-21
^{85}Kr	1.06E+03	8.42E+02	4.22E+02	1.33E+02	1.33E+01	1.33E+01	1.33E+01	1.33E+01	1.33E+01	1.32E+01
^{87}Kr	3.24E-02	2.43E-02	1.03E-02	2.49E-03	1.44E-04	4.81E-05	5.39E-06	8.50E-10	9.87E-22	1.21E-30
^{88}Kr	6.26E+01	4.85E+01	2.26E+01	6.30E+00	4.91E-01	2.99E-01	1.11E-01	2.11E-03	3.80E-11	4.03E-22
^{89}Kr	6.66E-19	1.43E-19	1.54E-21	1.49E-23	1.56E-24	2.38E-20	3.88E-19	4.10E-20	1.19E-19	4.03E-51
^{131m}Xe	3.39E+02	2.70E+02	1.35E+02	4.26E+01	4.24E+00	4.22E+00	4.18E+00	4.02E+00	3.38E+00	7.55E-01
^{133m}Xe	1.58E+03	1.25E+03	6.26E+02	1.97E+02	1.94E+01	1.89E+01	1.80E+01	1.47E+01	5.95E+00	2.34E-03
^{133}Xe	6.73E+04	5.34E+04	2.68E+04	8.50E+03	9.24E+02	1.30E+03	1.98E+03	3.78E+03	4.60E+03	1.62E+02
^{135m}Xe	1.92E-20	1.17E-20	2.64E-21	2.22E-22	1.88E-24	1.85E-26	4.61E-28	1.38E-20	1.72E-21	7.32E-53
^{135}Xe	1.67E+04	1.32E+04	6.58E+03	2.14E+03	3.64E+02	1.02E+03	1.71E+03	1.39E+03	1.22E+01	5.37E-20
^{137}Xe	6.42E-21	1.76E-21	3.71E-23	2.48E-25	6.80E-27	4.31E-24	1.39E-19	1.11E-20	1.01E-20	8.74E-51
^{138}Xe	1.73E-20	1.07E-20	2.58E-21	2.42E-22	2.50E-24	2.92E-26	4.76E-28	1.83E-21	2.69E-22	3.34E-52
Total noble gases	8.73E+04	6.92E+04	3.47E+04	1.10E+04	1.33E+03	2.36E+03	3.73E+03	5.20E+03	4.63E+03	1.76E+02

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Table 15.7-4

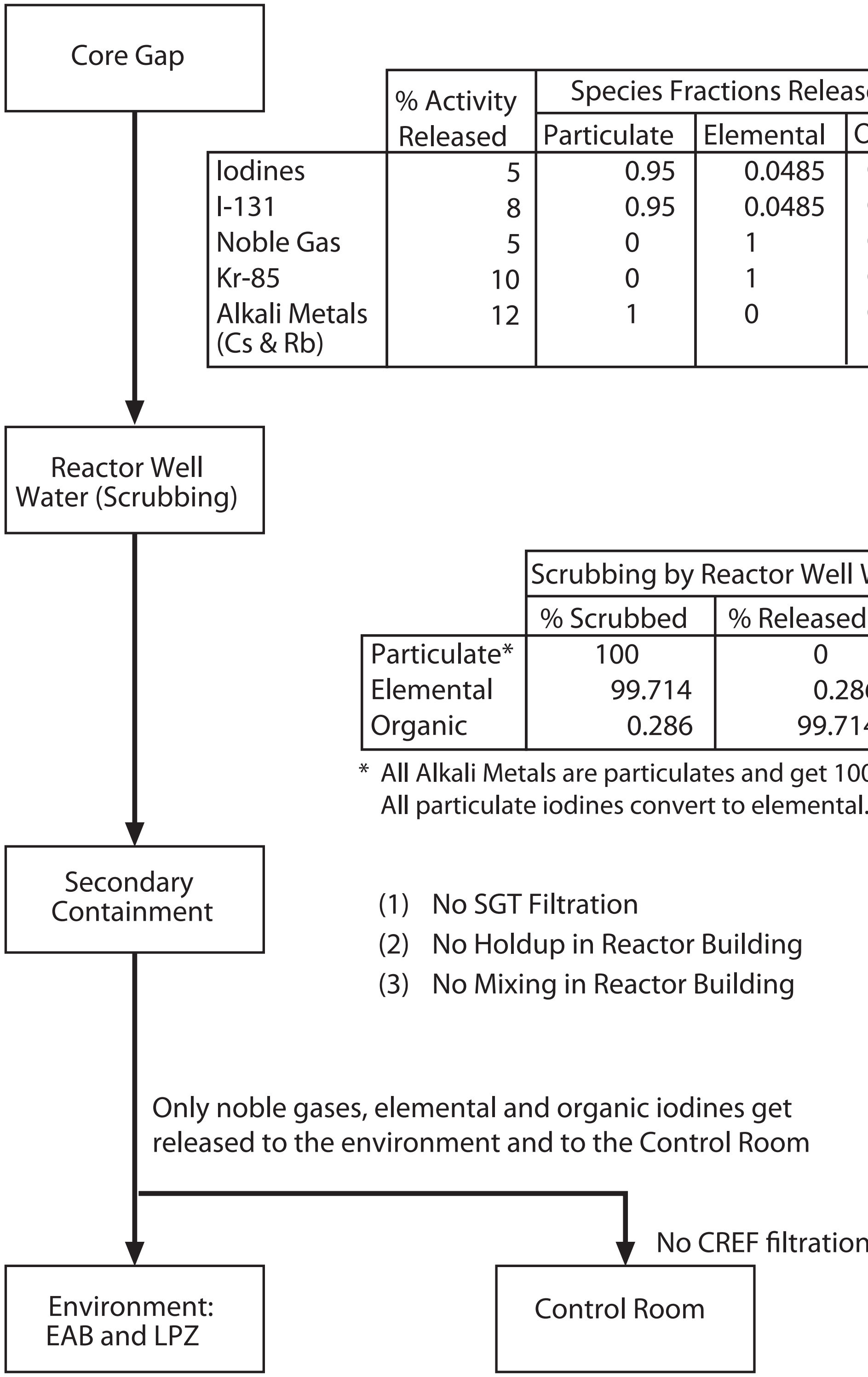
Fuel Handling Accident
Activity Released to the Environment (Curies)

Isotope	6 minutes	12 minutes	0.5 hr	1 hr	2 hr	4 hr	8 hr	1 day	4 days	30 days
^{131}I	6.84E+01	1.23E+02	2.28E+02	2.98E+02	3.28E+02	3.28E+02	3.29E+02	3.29E+02	3.29E+02	3.29E+02
^{132}I	5.12E-02	9.07E-02	1.63E-01	2.07E-01	2.23E-01	2.23E-01	2.23E-01	2.23E-01	2.23E-01	2.23E-01
^{133}I	4.09E+01	7.33E+01	1.35E+02	1.77E+02	1.95E+02	1.95E+02	1.95E+02	1.95E+02	1.95E+02	1.95E+02
^{134}I	4.16E-07	7.21E-07	1.23E-06	1.49E-06	1.56E-06	1.56E-06	1.56E-06	1.56E-06	1.56E-06	1.56E-06
^{135}I	7.07E+00	1.26E+01	2.31E+01	3.01E+01	3.28E+01	3.28E+01	3.28E+01	3.28E+01	3.28E+01	3.28E+01
Total iodine	1.16E+02	2.09E+02	3.86E+02	5.06E+02	5.56E+02	5.56E+02	5.57E+02	5.57E+02	5.57E+02	5.57E+02
^{83m}Kr	1.47E-01	2.59E-01	4.61E-01	5.81E-01	6.21E-01	6.21E-01	6.21E-01	6.21E-01	6.21E-01	6.21E-01
^{85m}Kr	5.49E+01	9.78E+01	1.78E+02	2.30E+02	2.50E+02	2.50E+02	2.50E+02	2.50E+02	2.50E+02	2.50E+02
^{85}Kr	2.75E+02	4.93E+02	9.13E+02	1.20E+03	1.32E+03	1.32E+03	1.32E+03	1.32E+03	1.32E+03	1.32E+03
^{87}Kr	8.63E-03	1.51E-02	2.64E-02	3.28E-02	3.47E-02	3.47E-02	3.47E-02	3.47E-02	3.47E-02	3.47E-02
^{88}Kr	1.64E+01	2.92E+01	5.26E+01	6.73E+01	7.26E+01	7.26E+01	7.26E+01	7.26E+01	7.26E+01	7.26E+01
^{89}Kr	3.64E-19	4.43E-19	4.64E-19	4.64E-19	4.64E-19	4.64E-19	4.65E-19	4.78E-19	4.38E-16	1.27E-13
^{131m}Xe	8.80E+01	1.58E+02	2.92E+02	3.85E+02	4.23E+02	4.23E+02	4.23E+02	4.23E+02	4.23E+02	4.23E+02
^{133m}Xe	4.10E+02	7.35E+02	1.36E+03	1.79E+03	1.96E+03	1.96E+03	1.96E+03	1.96E+03	1.96E+03	1.96E+03
^{133}Xe	1.74E+04	3.13E+04	5.79E+04	7.63E+04	8.40E+04	8.40E+04	8.40E+04	8.40E+04	8.40E+04	8.40E+04
^{135m}Xe	5.74E-21	9.23E-21	1.34E-20	1.46E-20	1.47E-20	1.47E-20	1.47E-20	1.47E-20	9.44E-18	1.02E-13
^{135}Xe	4.34E+03	7.78E+03	1.44E+04	1.89E+04	2.10E+04	2.10E+04	2.10E+04	2.10E+04	2.10E+04	2.10E+04
^{137}Xe	3.03E-21	3.86E-21	4.16E-21	4.17E-21	4.17E-21	4.17E-21	4.31E-21	2.95E-20	8.44E-16	1.27E-13
^{138}Xe	5.10E-21	8.26E-21	1.22E-20	1.34E-20	1.35E-20	1.35E-20	1.35E-20	1.35E-20	5.69E-18	1.05E-13
Total noble gases	2.26E+04	4.06E+04	7.51E+04	9.89E+04	1.09E+05	1.09E+05	1.09E+05	1.09E+05	1.09E+05	1.09E+05

Table 15.7-5

Fuel Handling Accident (Design Basis Analysis)
Radiological Effects

Area	TEDE (rem)
Exclusion area (1950 m) (2 hr)	1.0
Low population zone (4827 m) (30 day)	0.3
Control room (30 day)	3.7



	% Activity Released	Species Fractions Released		
		Particulate	Elemental	Organic
Iodines	5	0.95	0.0485	0.0015
I-131	8	0.95	0.0485	0.0015
Noble Gas	5	0	1	0
Kr-85	10	0	1	0
Alkali Metals (Cs & Rb)	12	1	0	0

	Scrubbing by Reactor Well Water		
	% Scrubbed	% Released	DF
Particulate*	100	0	∞
Elemental	99.714	0.286	350
Organic	0.286	99.714	1

* All Alkali Metals are particulates and get 100% scrubbed, All particulate iodines convert to elemental.

(1) No SGT Filtration
(2) No Holdup in Reactor Building
(3) No Mixing in Reactor Building

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

15.8.0 CAPABILITIES OF PRESENT DESIGN TO ACCOMMODATE ANTICIPATED TRANSIENTS WITHOUT SCRAM

The anticipated transients without scram (ATWS) events described in this section are not design basis events for Columbia Generating Station (CGS). A proposed method for minimizing the effects of failure-to-scram is described in References 15.8-1 and 15.8-2.

The recirculation pump trip (RPT), alternate rod insertion (ARI), and two pump standby liquid control (SLC) system operation features are utilized at CGS to provide protection against failure to scram. Due to the CGS design feature utilizing SLC system injection through the high-pressure core spray (HPCS) header, a plant-unique analysis was performed to demonstrate ATWS protection and mitigation at pre-power uprate conditions.

The ATWS acceptance criteria are established in Reference 15.8-3 as:

a. The reactor coolant pressure boundary (RCPB) remains below emergency pressure limits,

b. The containment pressure remains below design limits. The suppression pool temperature remains below local saturation temperature limits as defined in Reference 15.8-3,

c. A coolable geometry is maintained,

d. Radiological releases are maintained within 10 CFR 100 allowable limits. With implementation of Alternate Source Term (AST), the radiological release acceptance criterion becomes 10 CFR 50.67, and

e. Equipment necessary to mitigate the postulated ATWS event are evaluated to provide a high degree of assurance (assurance of function) that it will function in the environment (pressure, temperature, humidity, and radiation) predicated to occur as a result of the ATWS event.

The ATWS analysis, performed in conformance with NEDE-24222, did not include a SLCS pump suction valve delay in the SLCS injection time. To determine the impact of the 35 sec opening time for the suction valves upstream of the SLCS pumps, the limiting ATWS event for peak suppression pool temperature (i.e., MSIVC) was analyzed with the 35 sec delay in SLC system injection time. The results presented for the hot shutdown time, peak suppression pool temperature, and peak containment pressure for MSIVC in Table 15.8-3 include the effects of the 35 sec delay.

Section 15.8.9 shows that for the ATWS event with the most severe heat flux transient, fuel related applicable limits were met with considerable margins. In addition, Reference 15.8-3 concludes that maximum peak cladding temperature will not exceed 2200°F and the maximum local oxidation will be much less than 17%. Thus, criteria 3 and 4 are shown to be satisfied by the plant specific and generic analyses. Sections 15.8.7 and 15.8.9 show that resulting primary system pressures will be less than emergency pressure limit and that suppression pool temperature increase and peak pressure are within design limits. Reference 15.8-3 concludes that the safety/relief valve (SRV) air clearing loads will be bounded by the design loads. Thus criteria 1 and 2 are satisfied. In Reference 15.8-5, Energy Northwest concluded that ATWS equipment had been determined to be qualified by (a) materials analysis of agreeable components including test reports when available, (b) existing qualification to other accident profiles (LOCA, HELB) that encompass the ATWS profile, or (c) location in a mild environment that is not affected by the ATWS accident environment. This satisfies criterion 5.

Power Uprate Evaluation

The ATWS events were analyzed at power uprate operating conditions to demonstrate protection and mitigation of the consequences of these events. These analyses were performed at 3629 MWt power level and bound operation at uprate power level of 3486 MWt. The selection of critical events which were analyzed were guided by Reference 15.8-3.

For power uprate evaluation, it was conservatively assumed that ARI has failed, thus, requiring SLC system injection to achieve reactor shutdown.

The analysis presented herein are applicable to application of flow control valve (FCV) or adjustable speed drive to reactor recirculation system (RRC). A summary of ATWS results are shown in Table 15.8-3. The analysis results presented in this section are based on a representative reload core at the time of the analysis (Cycle 8). The Power Uprate ATWS Evaluation was confirmed for the introduction of GE14 (Reference 15.8-8).

15.8.1 INADVERTENT CONTROL ROD WITHDRAWAL

This transient is bounded by assumptions in the GE licensing topical reports and the other transients analyzed in this section.

15.8.2 LOSS OF FEEDWATER

15.8.2.1 Identification of Causes and Frequency Classification

15.8.2.1.1 Identification of Causes

Section 15.2.7 provides identification of causes for loss of feedwater event. The loss of feedwater event with failure to scram will initiate an ATWS event.

15.8.2.1.2 Frequency Classification

This event is of extremely low probability and is categorized as a limiting fault.

15.8.2.2 Sequence of Events and System Operation

15.8.2.2.1 Sequence of Events

Table 15.8-4 lists the sequence of events for Figure 15.8-1.

15.8.2.2.1.1 Identification of Operator Actions. For the simulation purpose, the following operator actions have been assumed.

a. Allow automatic operation of the HPCS and reactor core isolation cooling (RCIC),

b. Begin boron injection at two minutes following ATWS high-pressure trip or at boron injection initiation temperature (BIIT), whichever is later, and

c. Switch residual heat removal (RHR) to suppression pool cooling mode 11 minutes following initiation of the transient.

The emergency operating procedures provide operator actions for an ATWS event.

15.8.2.2.2 System Operation

For the loss of feedwater ATWS event, a complete failure to scram is postulated to occur for all reactor protection system (RPS) scram signals. All other plant control systems maintain normal operation. The relief valves are all assumed to function at the specified setpoints. 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 initiation of HPCS and RCIC on Level 2. For this event, a complete failure of ARI is postulated. The operator must manually initiate SLC system to inject boron into the reactor vessel for reactor shutdown.

15.8.2.2.3 The Effect of Single Failure and Operator Errors

This ATWS event is based on the assumed complete failure of all control rods to scram. This is a multiple equipment failure. For the conservative assumption of failure of the ARI system, the ATWS event is terminated by boron injection through operator activation of the SLC system. This event is less limiting compared with other ATWS events analyzed at power uprate condition.

15.8.2.3 Core and System Performance

15.8.2.3.1 Mathematical Model

Reference 15.8-7 describes the generic evaluation methodology for the ATWS event evaluated at uprated power conditions. Additional plant specific analyses were performed for a bounding 10% power uprate using the same methodology.

15.8.2.3.2 Input Parameters and Initial Conditions

The initial operating conditions and equipment performance characteristics are given in Tables 15.8-1 and 15.8-2, respectively. MSIV closure occurs on low-low-low water level (L1) but is analyzed based on low-low water level (L2), conservatively overpredicting suppression pool heatup. The HPCS/RCIC flow rates are conservatively high and water level setpoints represent nominal values. The ATWS high pressure setpoint was set at the upper analytical limit. The SRV setpoints were set using a statistical spread of the analytical setpoint limits for the first opening of each value and reset to a statistical spread of the nominal setpoints for all remaining SRV openings during the transient event.

15.8.2.3.3 Results

The results of this ATWS event simulation are shown in Figure 15.8-1. Feedwater pump trip is assumed to occur at the onset of the event. Upon the loss of the feedwater flow, reactor pressure, water level, and neutron flux begin to fall. Once reactor water level reaches low-low water level (L2), the protection system trips the recirculation pumps, initiates HPCS and RCIC and signals closure of main steam line isolation valves (MSIVs). Reactor pressure begins to rise due to closure of MSIVs. The relief valves begin to open due to reactor pressure increase. It is conservatively assumed the operator manually initiates SLC system 2 minutes after the ATWS setpoint has been reached.

15.8.2.3.4 Consideration of Uncertainties

Uncertainties in these analyses involve protection system setpoints, system capacities, and system response times. For ATWS transient analyses, best estimated values are used when possible. Examples of conservative bounding values which were used to cover uncertainties are as follows:

a. For conservatism, the analysis assumed the highest probable ATWS high-pressure trip setpoint, and

b. Boron injection is the later time of BIIT or 2 minutes following ATWS high-pressure trip.

15.8.2.4 Barrier Performance

The calculated peak vessel bottom head pressure is 1202 psig, which is below the American Society of Mechanical Engineers (ASME) Code Limit of 1375 psig for the RCPB and well below the ASME service level C of 1500 psig. The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed. Therefore, barrier integrity and function is maintained.

15.8.2.5 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 by means of SRV operation. Since this activity is contained in the primary containment, there will be no uncontrolled release to the environment.

15.8.3 LOSS OF ALTERNATE CURRENT POWER

This transient is bounded by the other transients analyzed in this section.

15.8.4 LOSS OF ELECTRICAL LOAD

This transient is bounded by assumptions in the GE licensing topical reports and the other transients analyzed in this section.

15.8.5 LOSS OF CONDENSER VACUUM

This transient is bounded by assumptions in the GE licensing topical reports and the other transients analyzed in this section.

15.8.6 TURBINE TRIP

This event was analyzed at pre-power uprate condition for low power and full power (corresponding to 3323 MWt) operation. At uprated conditions, the event is bounded by the other transients analyzed in this section. The selection of critical events were guided by Reference 15.8-3.

15.8.7 CLOSURE OF MAIN STEAM LINE ISOLATION VALVES

15.8.7.1 Identification of Causes and Frequency Classification

15.8.7.1.1 Identification of Causes

Various steam line and nuclear system malfunctions, or operator actions, can initiate MSIV closure. These are detailed in Section 15.2.4. The MSIV closure event with failure to scram will initiate an ATWS event.

15.8.7.1.2 Frequency Classification

This event is of extremely low probability and is categorized as a limiting fault.

15.8.7.2 Sequence of Events and System Operation

15.8.7.2.1 Sequence of Events

Table 15.8-5 lists the sequence of events for Figure 15.8-2.

15.8.7.2.1.1 Identification of Operator Actions. For the simulation purpose, the following operator actions have been assumed:

a. Allow automatic operation of the HPCS and RCIC,

b. Begin boron injection at 2 minutes following ATWS high-pressure trip or at BIIT, whichever is later, and

c. Switch RHR to suppression pool cooling mode 11 minutes following initiation of the transient.

Emergency Operating Procedures provide operator actions for an ATWS event.

15.8.7.2.2 System Operation

For the MSIV closure ATWS event, a complete failure to scram is postulated to occur for all RPS scram signals. All other plant control systems maintain normal operation. The relief valves are all assumed to function at the specified setpoints. The RPT occurs at the ATWS high pressure trip setpoint. For this event, a complete failure of ARI is postulated. The operator must manually initiate SLC system to inject boron into the reactor vessel for reactor shutdown.

15.8.7.2.3 The Effect of Single Failures and Operator Errors

For the conservative assumption of failure of ARI system, the ATWS event is terminated by boron injection through operator activation of the SLC system. Relief valves operate to limit system pressure. All of these aspects are designed to single failure criterion. Two sensitivity analysis were performed. One to determine the impact of four SRVs inoperable and the other to determine the impact of delay in SLC system injection time.

15.8.7.3 Core and System Performance

15.8.7.3.1 Mathematical Model

Reference 15.8-7 describes the generic evaluation methodology for the ATWS event evaluated at uprated power conditions. Additional plant specific analyses were performed for a bounding 10% power uprate using the same methodology.

15.8.7.3.2 Input Parameters and Initial Conditions

The initial operating conditions and equipment performance characteristics are given in Tables 15.8-1 and 15.8-2, respectively. The HPCS/RCIC flow rates are conservatively high and water level setpoints represent nominal values. The ATWS high pressure setpoint was set at the upper analytical limit. The SRV setpoints were set using a statistical spread of the analytical setpoint limits for the first opening of each value and reset to a statistical spread of the nominal setpoints for all remaining SRV openings during the transient event.

15.8.7.3.3 Results

The results of this ATWS event simulation are shown in Figure 15.8-2. The MSIVs close within a nominal 4 sec stroke time. Once the MSIVs reach the 85% open position, a reactor scram is initiated. The scram was assumed to fail to insert any control rods. The rapid increase in reactor pressure generates rapid increase in reactor core power due to collapsing core voids. The relief valves begin to open responding to reactor pressure rise. Upon reaching the ATWS high-pressure setpoint, the RPT occurs and reduces core power. It is conservatively assumed the operator manually initiates SLC system 2 minutes after the ATWS setpoint has been reached.

The MSIV closure (MSIVC) ATWS event was analyzed with four SRVs inoperable for maximum reactor vessel pressure determination. The MSIVC and PREGO event described in Section 15.8.9 were selected based upon previous ATWS analyses that indicated these two events are most limiting with respect to vessel pressure. The results of an ATWS event simulation with four SRVs inoperable are shown in Figure 15.8-3. The sequence of events for this event are shown in Table 15.8-6. The peak calculated vessel bottom head pressure is 1467 psig, which is below the ASME Service Level C limit of 1500 psig.

The calculated peak suppression pool temperature of the event with delayed SLC system injection time is less than 175°F, which is below the containment design limit. The calculated peak containment pressure for this event is 8.46 psig, which is also below the containment design limit.

15.8.7.3.4 Consideration of Uncertainties

Uncertainties in these analyses involve protection system setpoints, system capacities, and system response times. For ATWS transient analyses, best estimated values are used when possible. Examples of conservative bounding values which were used to cover uncertainties are as follows:

a. For conservatism, the analyses assumed the highest probable ATWS high-pressure trip setpoint, and

b. Boron injection is the later time of BIIT or 2 minutes following ATWS high-pressure trip.

c. A 35 sec opening time for the suction values upstream to the SLCS pumps was modeled increasing the SLCS initiation time.

15.8.7.4 <u>Barrier Performance</u>

The calculated peak vessel bottom head pressure is 1310 psig and 1467 psig for MSIV with full complement of SRVs and with four SRVs inoperable respectively. The calculated peak vessel bottom head pressure is below the ASME Service Level C limit of 1500 psig, thus, meeting criterion 1. 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 containments are designed. Therefore, these barrier integrity and function is maintained.

15.8.7.5 <u>Radiological Consequences</u>

While this event does not result in fuel failure it does result in the discharge of normal coolant activity to the suppression pool by means of SRV operation. Since this activity is contained in the primary containment, there will be no uncontrolled release to the environment.

15.8.8 INADVERTENT OPENING OF RELIEF VALVE

15.8.8.1 Identification of Causes and Frequency Classification

15.8.8.1.1 Identification of Causes

This event assumes that a SRV may "open" and stick in the "open" position. These events are detailed in Section 15.1.4. The inadvertent opening of relief valve (IORV) event with failure to scram will initiate an ATWS event.

15.8.8.1.2 Frequency Classification

This event is of extremely low probability and is categorized as a limiting fault.

15.8.8.2 Sequence of Events and System Operation

15.8.8.2.1 Sequence of Events

The analysis has not been updated for the change in MSIV isolation setpoint from Level 2 to Level 1 because the analysis is bounding and conclusions of the analysis are not affected (Reference 15.8-9).

Table 15.8-7 lists the sequence of events for Figure 15.8-4.

15.8.8.2.1.1 Identification of Operator Actions. For the simulation purpose, the following operator actions have been assumed:

a. Initiate boron injection 2 minutes after BIIT,
b. Disable HPCS, RCIC, and low level MSIV closure,
c. Use feedwater to manually control the water level at the top of active fuel, and
d. Manually trip the recirculation pumps.

15.8.8.2.1.2 System Operation. For the IORV ATWS event, a complete failure to scram is postulated to occur for all RPS scram signals. All other plant control systems maintain normal operation. For this event, a complete failure of ARI is also postulated. The operator must manually initiate SLC system to inject boron into the reactor vessel for reactor shutdown.

15.8.8.2.2 The Effect of Single Failures and Operator Errors

For the conservative assumption of failure of ARI system, the ATWS event is terminated by boron injection through operator activation of the SLC system. This is a multiple equipment failure event. All of these aspects are designed to single failure criterion.

The instrumentation, which detects and audibly alarms the resulting suppression pool temperature rise, and the RHR containment heat removal system are designed to meet the single failure criteria. The operator must, however, manually initiate suppression pool cooling.

15.8.8.3 Core and System Performance

15.8.8.3.1 Mathematical Model

Reference 15.8-7 describes the generic evaluation methodology for the ATWS event evaluated at uprated power conditions. Additional plant specific analyses were performed for a bounding 10% power uprate using the same methodology.

15.8.8.3.2 Input Parameters and Initial Conditions

The initial operating conditions and equipment performance characteristics are given in Tables 15.8-1 and 15.8-2, respectively. The HPCS/RCIC flow rates are conservatively high and water level setpoints represent nominal values. The ATWS high pressure setpoint was set at the upper analytical limit. The SRV setpoints were set using a statistical spread of the analytical setpoint limits for the first opening of each value and reset to a statistical spread of the nominal setpoints for all remaining SRV openings during the transient event.

15.8.8.3.3 Results

The results of this ATWS event simulation are shown in Figure 15.8-4. The opening of a SRV allow steam to be discharged into the suppression pool. The sudden increase in the rate of steam flow leaving the reactor vessel causes a mild depressurization transient.

Discharge of steam into the suppression pool increases the suppression pool temperature. The operator initiates SLC system 2 minutes after the suppression pool temperature reaches 110°F, trips the recirculation pumps, and initiates feedwater runback to lower the reactor water level to top of active fuel (TAF). Suppression pool cooling begins 11 minutes after the initiation of the event. The operator disables HPCS and RCIC level 2 initiation. The MSIV Level 2 closure is also disabled. Turbine steam flow is terminated upon closure of the MSIVs due to low steam line pressure.

15.8.8.3.4 Consideration of Uncertainties

Uncertainties in these analyses involve protection system setpoints, system capacities, and system response times. For ATWS transient analyses, best estimated values are used when possible. Examples of conservative bounding values which were used to cover uncertainties are as follows:

a. For conservatism, the analysis assumed the highest probable ATWS high-pressure trip setpoint, and

b. Boron injection is the later time of BIIT or 2 minutes following ATWS high-pressure trip.

15.8.8.4 Barrier Performance

The IORV ATWS event is a mild depressurization which has no significant effect on RCPB. During the event, the suppression pool is continually heated due to SRV discharge. The peak suppression pool temperature and pressure are within the design criteria of the containment.

15.8.8.5 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 by means of SRV operation. Since this activity is contained in the primary containment, there will be no uncontrolled release to the environment.

15.8.9 PRESSURE REGULATOR FAILURE - OPEN (PREGO)

15.8.9.1 Identification of Causes and Frequency Classification

15.8.9.1.1 Identification of Causes

The causes for this event is detailed in Section 15.1.3. The PREGO event with failure to scram will initiate an ATWS event.

15.8.9.1.2 Frequency Classification

This event is of extremely low probability and is categorized as a limiting fault.

15.8.9.2 Sequence of Events and System Operation

15.8.9.2.1 Sequence of Events

Table 15.8-8 lists the sequence of events for Figure 15.8-5.

15.8.9.2.1.1 Identification of Operator Actions. For the simulation purpose, the following operator actions have been assumed.

a. Allow automatic operation of the HPCS and RCIC,

b. Begin boron injection at 2 minutes following ATWS high-pressure trip or at BIIT, whichever is later, and

c. Switch RHR to suppression pool cooling mode 11 minutes following initiation of the transient.

The emergency operating procedures provide operator actions for an ATWS event.

15.8.9.2.1.2 System Operation. For the PREGO ATWS event, a complete failure to scram is postulated to occur for all RPS scram signals. All other plant control systems maintain normal operation. For this event, a complete failure of ARI is also postulated. The operator must manually initiate SLC system to inject boron into the reactor vessel for reactor shutdown.

15.8.9.2.3 The Effect of Single Failures and Operator Errors

For the conservative assumption of failure of ARI system, the ATWS event is terminated by boron injection through operator activation of the SLC system. This is a multiple equipment failure event. All of these aspects are designed to single failure criterion.

The instrumentation, which detects and audibly alarms the resulting suppression pool temperature rise, and the RHR containment heat removal system are designed to meet the single failure criteria.

15.8.9.3 Core and System Performance

15.8.9.3.1 Mathematical Model

Reference 15.8-7 describes the generic evaluation methodology for the ATWS event evaluated at uprated power conditions. Additional plant specific analyses were performed for a bounding 10% power uprate using the same methodology.

15.8.9.3.2 Input Parameters and Initial Conditions

The initial operating conditions and equipment performance characteristics are given in Tables 15.8-1 and 15.8-2, respectively. The HPCS/RCIC flow rates are conservatively high and water level setpoints represent nominal values. The ATWS high pressure setpoint was set at the upper analytical limit. The SRV setpoints were set using a statistical spread of the

analytical setpoint limits for the first opening of each value and reset to a statistical spread of the nominal setpoints for all remaining SRV openings during the transient event.

15.8.9.3.3 Results

The results of this ATWS event simulation are shown in Figure 15.8-5. The DEH control system failure with 130% steam flow demand signal is assumed to occur. Ensuing reactor depressurization results in formation of voids in the reactor coolant and causes a decrease in reactor power almost immediately. The MSIV closure occurs due to trip signal from low steam line pressure. Reactor pressure rises to the relief setpoints and the recirculation pumps trip on the high pressure ATWS setpoint.

Discharge of steam into the suppression pool increases the suppression pool temperature. The operator initiates feedwater runback to lower the reactor water level to TAF after the suppression pool temperature reaches 110°F. The HPCS and RCIC systems are initiated at low reactor water level. The SLC system is manually initiated 2 minutes after the ATWS high pressure setpoint was reached.

The PREGO ATWS event was also analyzed with four SRVs inoperable for maximum reactor vessel pressure determination. The PREGO and MSIVC event described in Section 15.8.7 were selected based upon previous ATWS analyses that indicated these two events are most limiting with respect to vessel pressure. The analysis performed in Reference 15.8-6 determined the peak vessel pressure for the PREGO ATWS event is bounded by the results for the MSIVC event.

15.8.9.3.4 Consideration of Uncertainties

Uncertainties in these analyses involve protection system setpoints, system capacities, and system response times. For ATWS transient analyses, best estimated values are used when possible. Examples of conservative bounding values which were used to cover uncertainties are as follows:

a. For conservatism, the analysis assumed the highest probable ATWS high-pressure trip setpoint, and

b. Boron injection is the later time of BIIT or 2 minutes following ATWS high-pressure trip.

15.8.9.4 <u>Barrier Performance</u>

The calculated peak vessel bottom head pressure is 1306.5 psig, which is below ASME Code limit of 1375 psig for the RCPB and well below the ASME Service Level C of 1500 psig. The consequences of this event do not result in any temperature or pressure transient in excess

of the criteria for which the fuel pressure vessel or containment are designed. Therefore, barrier integrity and function is maintained.

The calculated peak suppression pool temperature for this event is less than 173°F, which is below the containment design limit. The calculated peak containment pressure for this event is 8.10 psig, which is also below the containment design limit.

15.8.9.5 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 by means of SRV operation. Since this activity is contained in the primary containment, there will be no uncontrolled release to the environment.

15.8.10 SINGLE REACTOR RECIRCULATION SYSTEM PUMP OPERATION

The following discussion is based on pre-uprate power level of 3323 MWt. Thus, the 100% rod line corresponds to 3323 MWt power at rated core flow.

For pre-uprate condition, it was shown that operation of the plant with only a single RRC pump and the resulting transient conditions which could occur while in this mode are bounded by the other transients analyzed in this section, and the parametric studies performed in Reference 15.8-4. This conclusion was based on evaluating the effects of power, void reactivity worth and doppler worth at both 100% conditions, and at conditions present under single RRC pump operation.

Sensitivity studies presented in Reference 15.8-4 compare the turbine trip at 100% power condition with the turbine trip at lower power conditions, such as one would have under single RRC pump.

The rod line for single RRC pump operation for pre-uprate condition was normally maintained between 100% and 104.25% power level. At less than 100% power the average void in the core was slightly higher for operation on the 104.25% rod line than for operation on the 100% rod line. In addition, the doppler worth at lower power conditions is higher than at 100% power. The presence of higher voids and the increased doppler worth when operating at the 104.15% rod line is bounded by the parametric analyses in Reference 15.8-4. These parametric analyses determined the sensitivity of plant response between the MSIV closure at 100% power and the MSIV closure with higher reactivity coefficients at 100% power. The void worth assumed in the higher reactivity coefficient case gives a much higher effect than the increased average void present in the single RRC pump operation mode at the 104.25% rod line, which bounds this case. The doppler reactivity worth used in the MSIV closure with higher reactivity coefficients is representative of the doppler reactivity worth found at lower power conditions such as those present in the single RRC pump operation mode.

15.8.11 EXTENDED LOAD LINE LIMIT ANALYSIS OPERATION

Power uprate ATWS analysis were performed with Extended Load Line Limit Analysis (ELLLA) operating conditions. These analyses show that performance at the power uprate condition is within vessel maximum pressure, fuel temperature, and containment pressure limit for the most severe ATWS transients (Reference 15.8-6).

15.8.12 REFERENCES

15.8-1 Hatch Unit 1 FSAR. Amendment 10, Appendix L, "Failure-to-Scram Analysis," October 27, 1971.

15.8-2 Michelotti, L. A., "Analysis of Anticipated Transients Without Scram," NEDO-10349.

15.8-3 NEDE-24222, "Assessment of BWR Mitigation of ATWS, Volume II (NUREG-0460 Alternate No. 3)."

15.8-4 EI International, Inc., "Final Report, Anticipated Transients Without Scram Analysis for the WNP-2 Nuclear Power Plant," SA-JAD-087-90, December 1989.

15.8-5 Supply System Letter G02-90-116, G. C. Sorensen (Supply System) to NRC, "Nuclear Plant No. 2. Operating License (NPF-21 Resolution of Anticipated Transient Without Scram (ATWS) for WNP- 2," dated June 29, 1990.

15.8-6 GE Nuclear Energy, "WNP-2 Power Uprate Project NSSS Engineering Report," GE-NE-208-17-0993, Revision 1, December 1994.

15.8-7 GE Nuclear Energy, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31984P, July 1991 and Supplements 1 & 2.

15.8-8 GEH-0000-0075-4920, "GE14 Fuel Design Cycle-Independent Analyses for Energy Northwest Columbia Generating Station" (most recent version referenced in the COLR).

15.8-9 GE Hitachi Nuclear Energy, "License Amendment Request for Proposed Changes to Columbia Technical Specifications: Changing Group 1 Isolation Valves' Low Reactor Water Level Isolation Signal from the Current Level 2 to Level 1," 0000-0081-6730-R1, July 2008.

Table 15.8-1

Anticipated Transients Without Scram Analysis
Initial Conditions

Parameters	Value
Reactor dome pressure (psig)	1020
Vessel core flow (Mlb/hr)	108.5
Vessel steam flow (Mlb/hr)	15.728
Reactor thermal power (MWt)	3629
Initial vessel and recirculation piping inventory (lbm)	609,600
Narrow range sensed initial water level (ft above separator skirt)	4.13
Initial core average void fraction (%)	41.8
Void reactivity coefficient (¢/%)	-12.937
Doppler coefficient (%/F)	-0.31087
Feedwater enthalpy (Btu/lb)	403.1
Sodium penetaborate solution concentration (% by weight)	13.6
Suppression pool liquid volume (ft^3)	112,197
Suppression pool temperature (°F)	90
Service water temperature (°F)	90

Table 15.8-2

Anticipated Transients Without Scram Analysis
Equipment Performance Characteristics

Parameter	Value
Main steam line isolation valve nominal closure time (sec)	4
Relief valve system capacity (% of current NBR steam flow at 1144 psia)	100.18
Number of SRVs	18
Relief valve and sensor time delay (sec)	0.4
Relief valve opening time (sec)	0.15
Relief valve closure time delay (sec)	0.3
Standby liquid control system injection rate (gpm)	86.0
High-pressure core spray/RCIC low water level initiation nominal setpoint (ft above separator skirt)	-3.04 (L2)
High-pressure core spray/RCIC high water level shutoff setpoint (ft above separator skirt)	5.667 (L8)
High-pressure core spray flow rate (gpm at 1035 psia)	3875
Reactor core isolation cooling flow rate (gpm)	600
Anticipated transients without scram high pressure UAL setpoint (psia)	1186
Anticipated transients without scram dome pressure sensor and logic time delay (sec)	0.53
Total bypass capacity (Mlb/hr)	35.5
Total bypass capacity (% of uprate steam flow)	22.7
Pump inertia constant (sec)	5.4729
Residual heat removal pool cooling capacity (Btu/sec-°F)	578

Table 15.8-3

Summary of Anticipated Transients
Without Scram Results

Parameter	ATWS Event			
	PREGO	MSIVC	LOFW	IORV
Maximum neutron flux (%)	421.22	683.2	282.07	116.6
Time (sec)	22.59	4.02	22.36	7.96
Maximum average fuel heat flux (%)	170.99	151.83	102.9	103.14
Time (sec)	24.41	5.11	0.49	0.69
Maximum bottom pressure (psig)	1306.5	1310.1	1202.2	1061.4
Time (sec)	27.62	8.28	23.62	0.19
Peak suppression pool temperature (°F)	172.79	173.40	161.06	165.29
Peak containment pressure (psig)	8.10	8.20	5.97	6.87
Time (sec)	4600	4500	8400	6600
Peak cladding temperature (°F)	1473.67	N/A	N/A	N/A
Time (sec)	83.5			
Min. water level (ft above sep. skirt)	-10.47	-10.27	-11.24	-10.66
Time (sec)	211.18	197.29	91.38	975.4
Time of hot shutdown[a] (sec)	984.6	962.6	977.0	1524.6
Time of reaching ATWS setpoint (sec)	24.09	4.73	17.5	N/A
Time of BIIT (sec)	67.9	54.6	170.0	554.0

[a] Hot shutdown is defined as generated power remaining below 1% NBR.

Table 15.8-4

Sequence of Events for Loss of Feedwater

Time	Event
0 sec	Feedwater pump trip.
17.5 sec	High-pressure core spray and RCIC initiated on Level 2.
17.5 sec	Main steam line isolation valve closure on Level 2 (see Section 15.8.2.3.2) - scram fails.
17.5 sec	Recirculation pump tripped on Level 2 (ATWS setpoint reached, ARI fails).
22.9 sec	Relief valves lift.
23.6 sec	Vessel pressure peaks.
2 minutes 18 sec	Operator initiates SLCS (2 minutes after ATWS setpoint reached).
3 minutes 3 sec	Liquid control flow enters the core.
16 minutes	Hot shutdown achieved.
140 minutes	Suppression pool temperature and containment pressure peak.

Table 15.8-5

Sequence of Events for Main Steam Line
Isolation Valve Closure
(Long Term Transient)

Time	Event
0 sec	Nominal 4-sec MSIV closure - scram fails.
4.37 sec	Relief valves lift.
4.73 sec	Recirculation pump trip on high pressure (ATWS setpoint reached, ARI fails).
8.28 sec	Vessel pressure peaks.
54.60 sec	Operator initiates feedwater runback (suppression pool at 110°F).
1 minute 15 sec	High-pressure core spray and RCIC initiated on Level 2.
2 minutes 4 sec	Operator initiates SLC system 2 minutes after ATWS setpoint reached.
2 minutes 51 sec	Liquid control flow enters the core.
11 minutes	Suppression pool cooling begins.
16 minutes	Hot shutdown achieved.
75 minutes	Suppression pool temperature and containment pressure peak.

Table 15.8-6

Sequence of Events for Main Steam Line Isolation Valve
Closure with Four Safety/Relief Valves Out-Of-Service

Time (sec)	Event
0	Nominal 4-sec MSIV closure - scram fails.
4.47	Relief valves lift.
4.73	Recirculation pump trip on high pressure (ATWS setpoint reached).
12.30	Vessel pressure peaks.

Table 15.8-7

Sequence of Events for Inadvertent Open Relief Valve

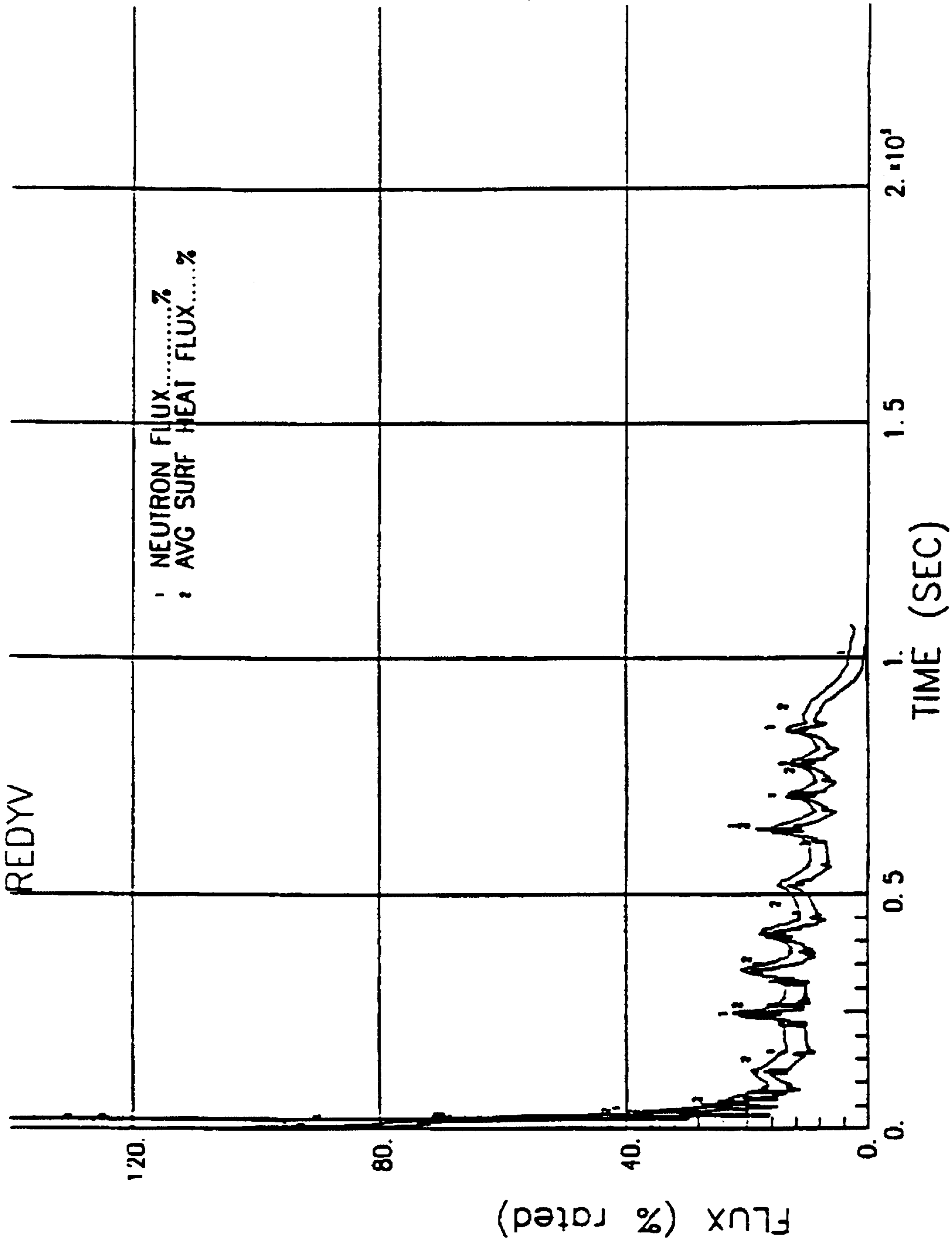
Time	Event
0 sec	Relief valve with the lowest opening setpoint opens.
9 minutes 14 sec	Operator initiates SLCS 2 minutes after suppression pool temperature = 110°F (scram and ARI fail).
9 minutes 14 sec	Operator trips recirculation pumps.
9 minutes 14 sec	Operator initiates feedwater runback to bring level to TAF.
11 minutes	Suppression pool cooling begins.
13 minutes	Operator disables HPCS, RCIC Level 2 initiation and MSIV Level 2 closure[a].
14 minutes	Liquid control flow enters the core.
25 minutes	Hot shutdown achieved.
29 minutes	Main steam line isolation valve closure on low pressure.
110 minutes	Suppression pool temperature and containment pressure peak.

[a] The analysis has not been updated for the change in MSIV isolation setpoint from Level 2 to Level 1 because the analysis is bounding and conclusions of the analysis are not affected (Reference 15.8-9).

Table 15.8-8

Sequence of Events for Pressure Regulator
Failure Open (Long Term Transient)

Time	Event
0 sec	Pressure regulator fails to maximum demand.
15.3 sec	Main steam line isolation valve closure on low steam line pressure - scram fails.
23.7 sec	Relief valves lift.
24.1 sec	Recirculation pump trip on high pressure (ATWS setpoint reached, ARI fails).
27.6 sec	Vessel pressure peaks.
1 minute 8 sec	Operator initiates feedwater runback (Suppression pool at 110°F).
1 minute 30 sec	High-pressure core spray and RCIC initiated on Level 2.
2 minutes 24 sec	Operator initiates SLC system 2 minutes after ATWS setpoint is reached.
3 minutes 9 sec	Liquid control flow enters the core.
11 minutes	Suppression pool cooling begins.
16 minutes	Hot shutdown achieved.
77 minutes	Suppression pool temperature and containment pressure peak.



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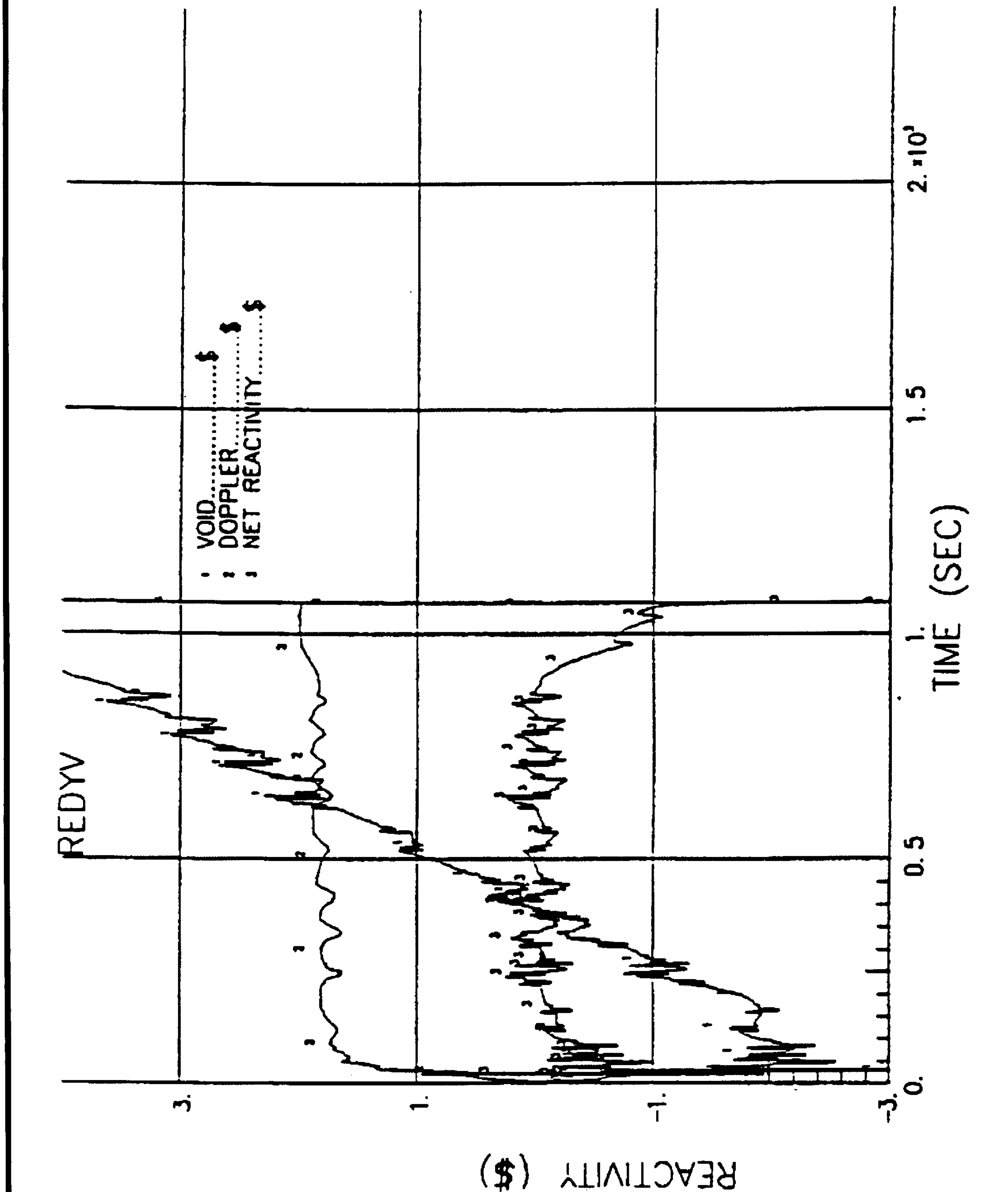
Loss of Feedwater Event

Draw. No. 020002.29	Rev.	Figure 15.8-1.1



Columbia Generating Station Final Safety Analysis Report	Loss of Feedwater Event	
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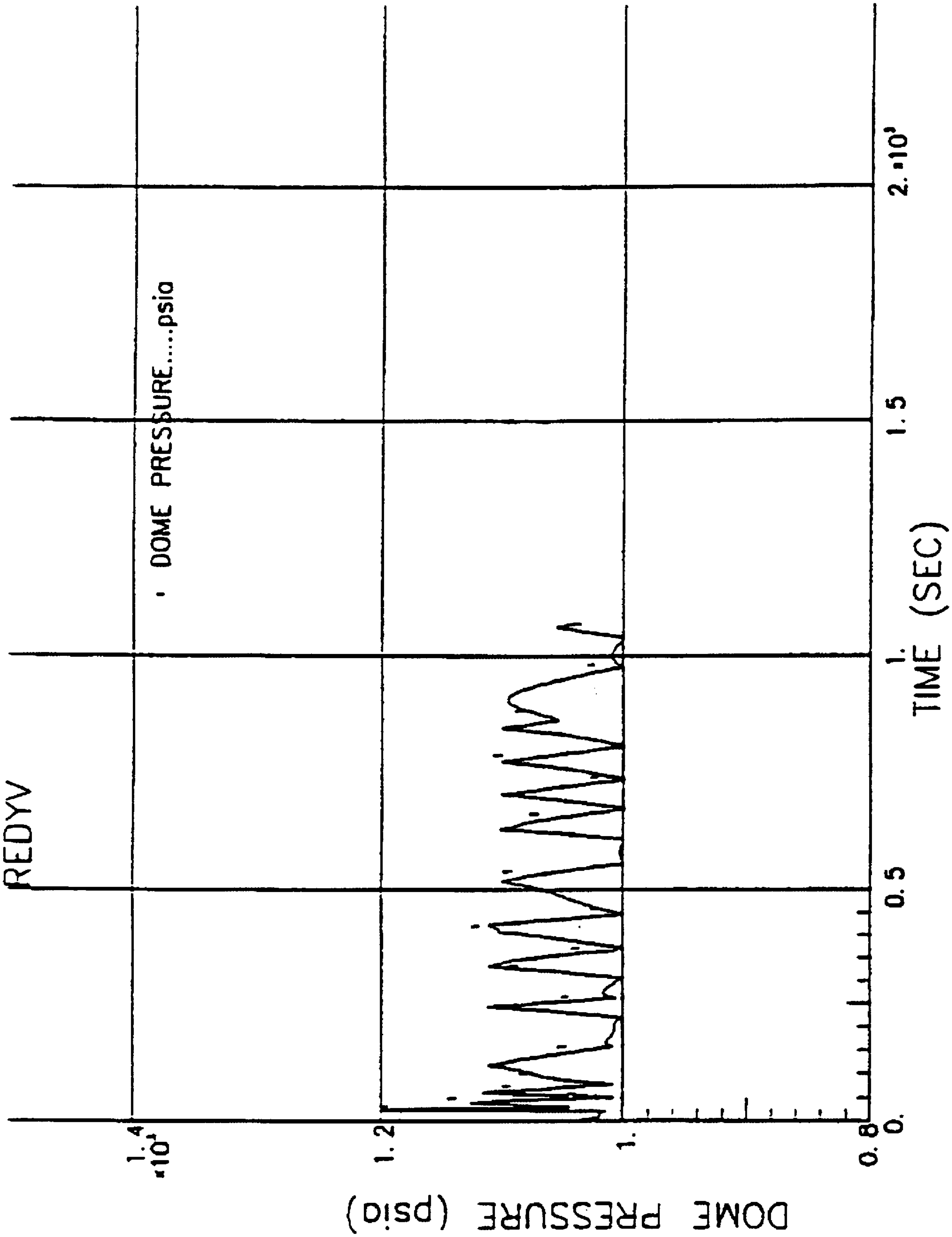

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Loss of Feedwater Event

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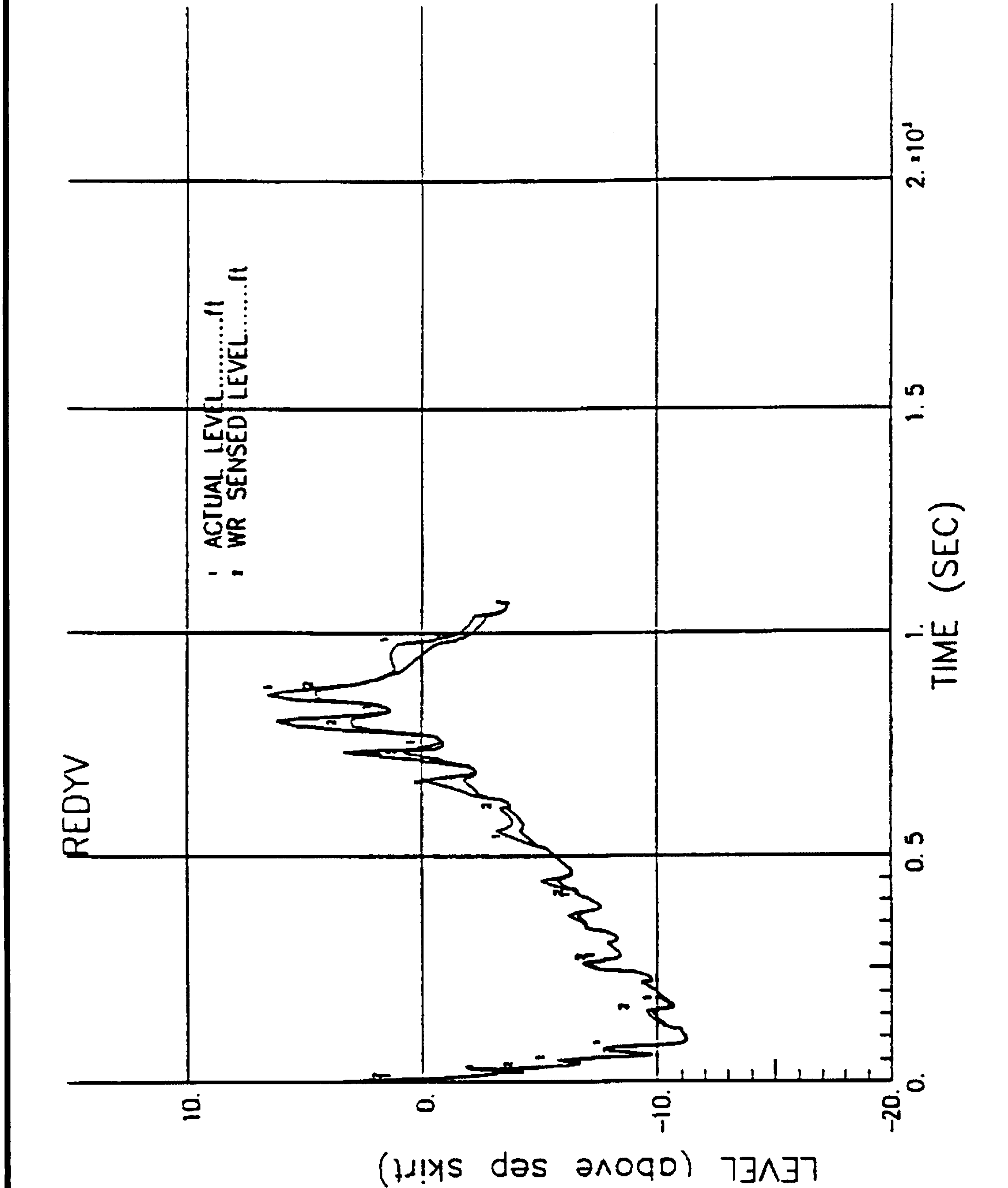

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Loss of Feedwater Event

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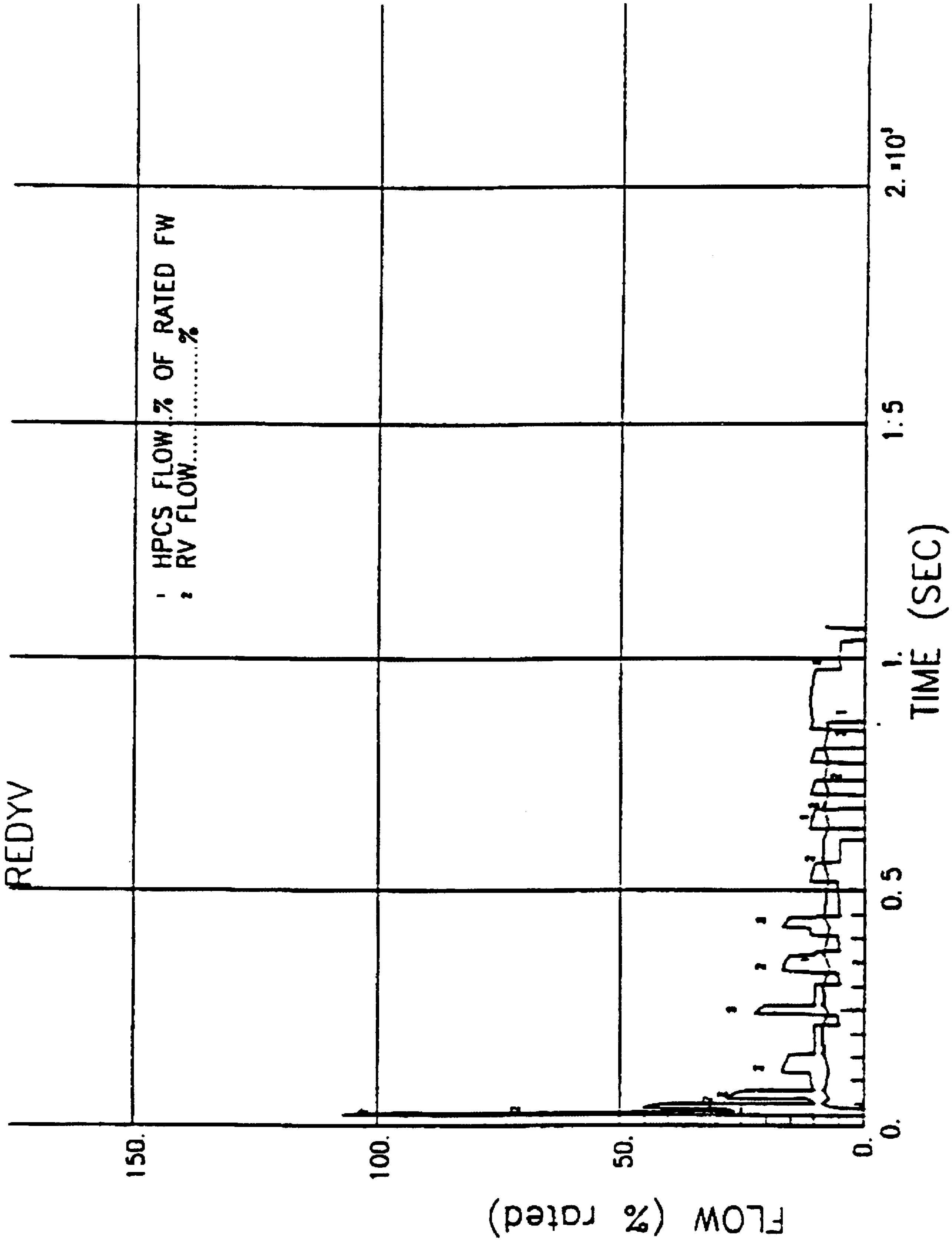
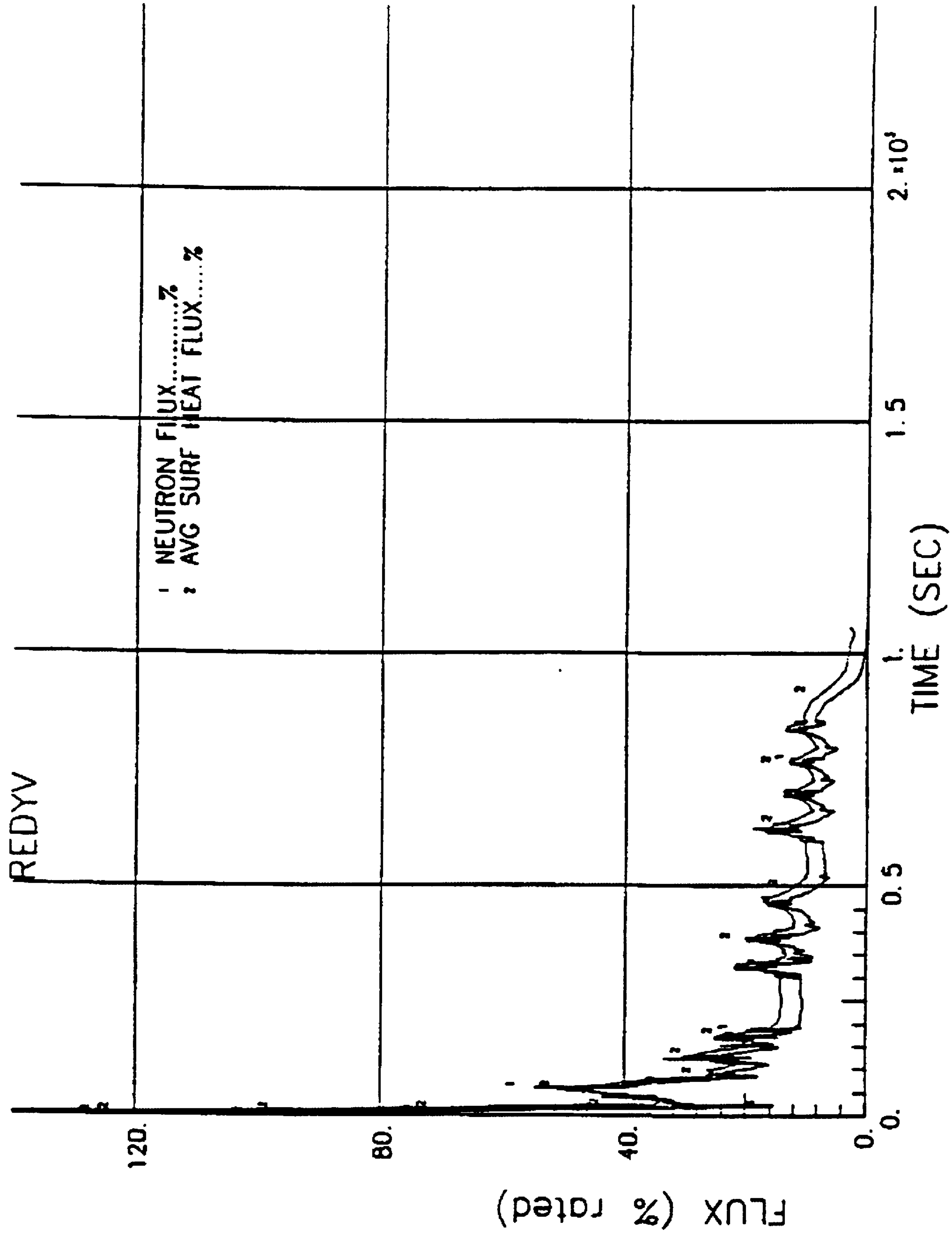
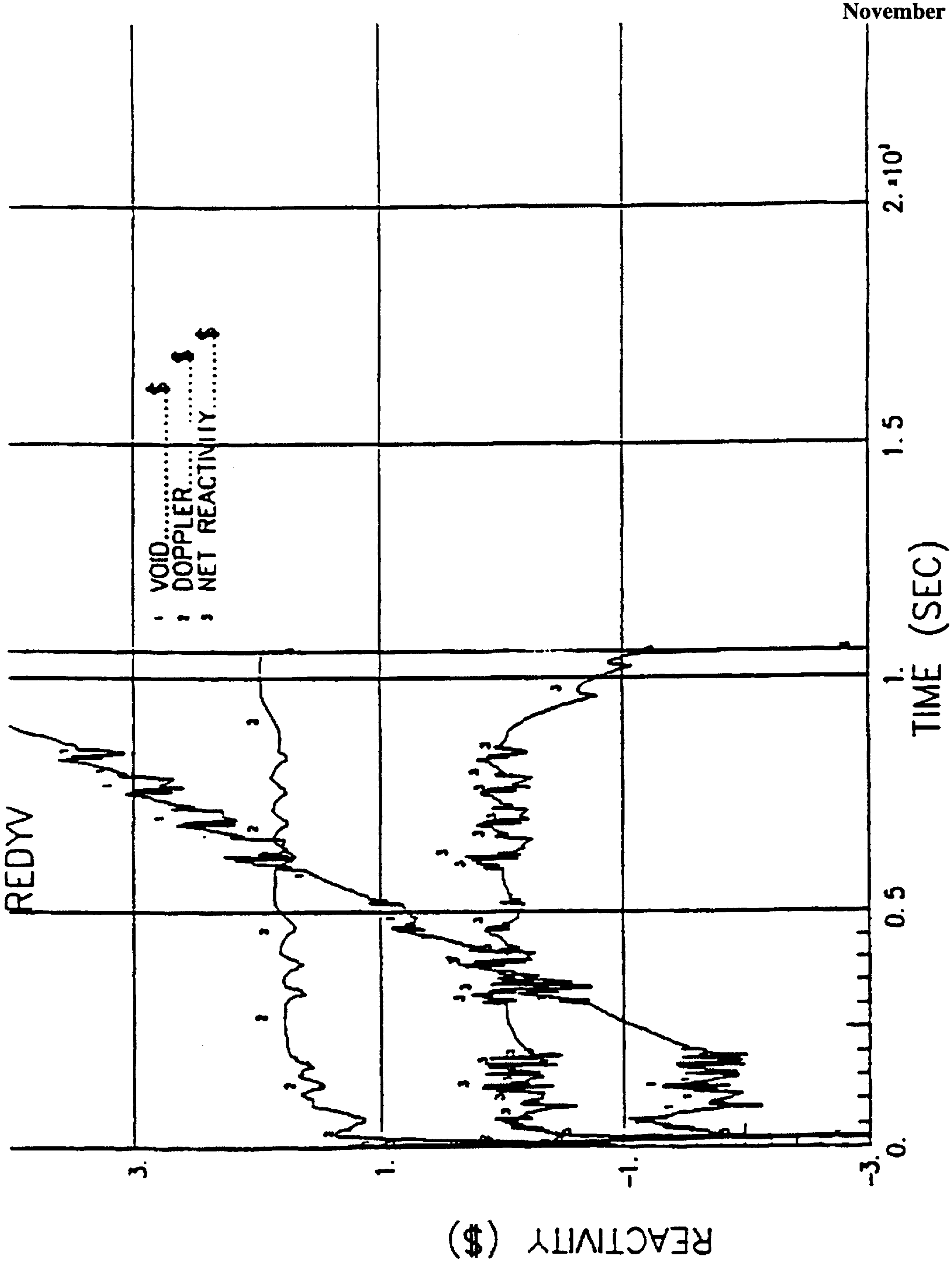


Loss of Feedwater Event

Draw. No. 020002.33	Rev.	Figure 15.8-1.5



Columbia Generating Station Final Safety Analysis Report	Main Steam Isolation Valve Closure Event		
	Draw. No. 020002.34	Rev.	Figure 15.8-2.1



Main Steam Isolation Valve Closure Event

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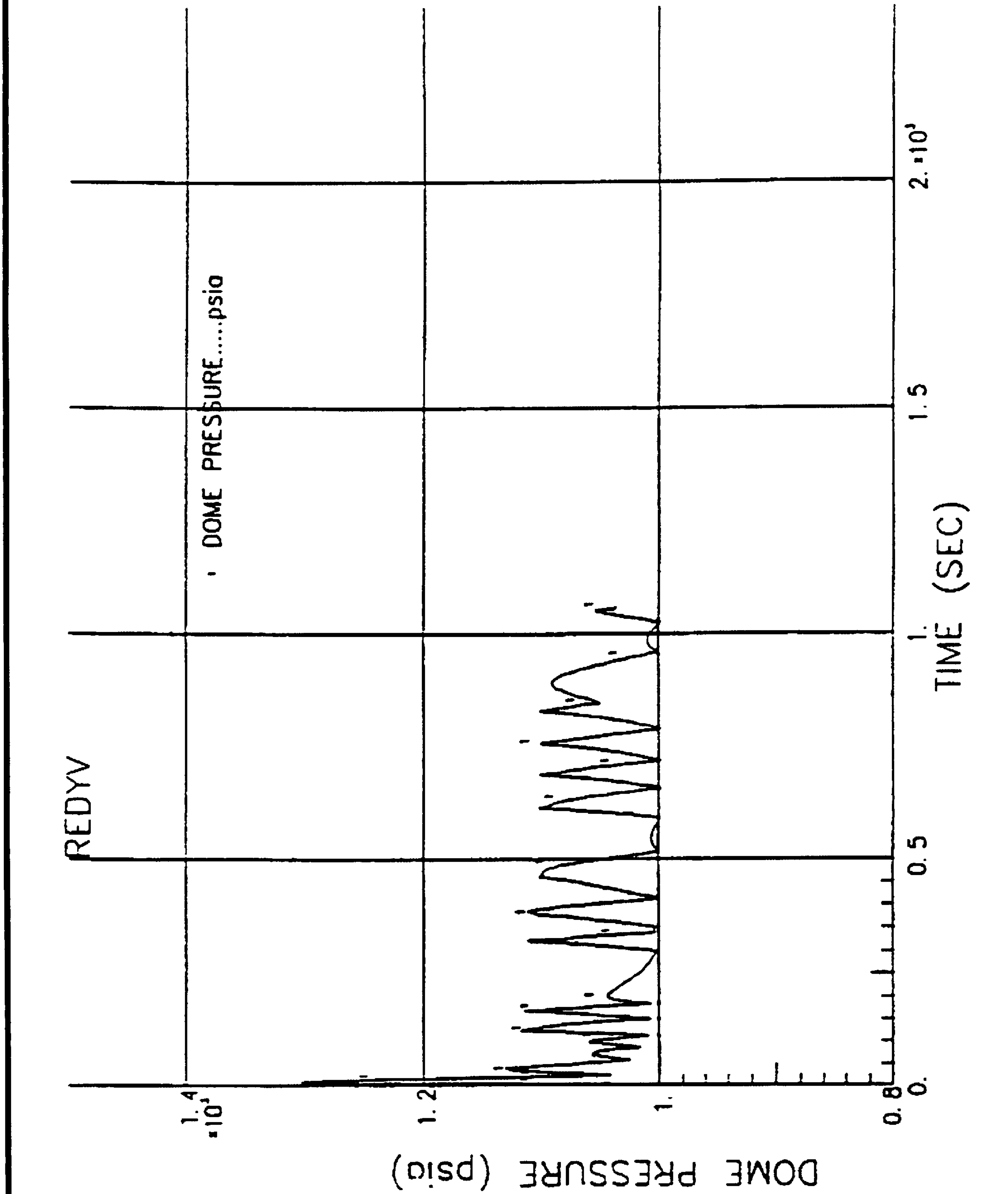
Draw. No. 020002.35	Rev.	Figure 15.8-2.2



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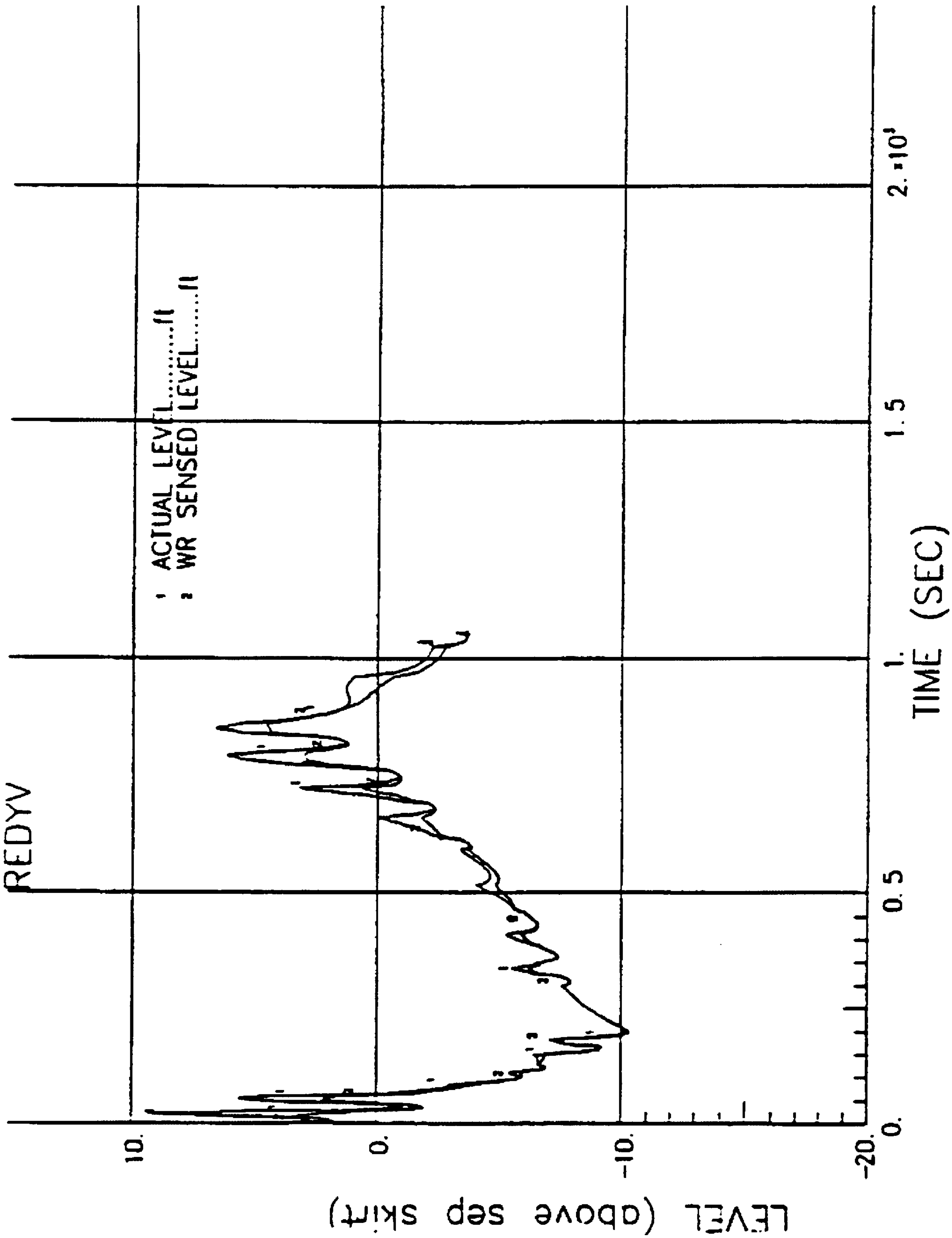
Main Steam Isolation Valve Closure Event

Draw. No. 020002.36	Rev.	Figure 15.8-2.3



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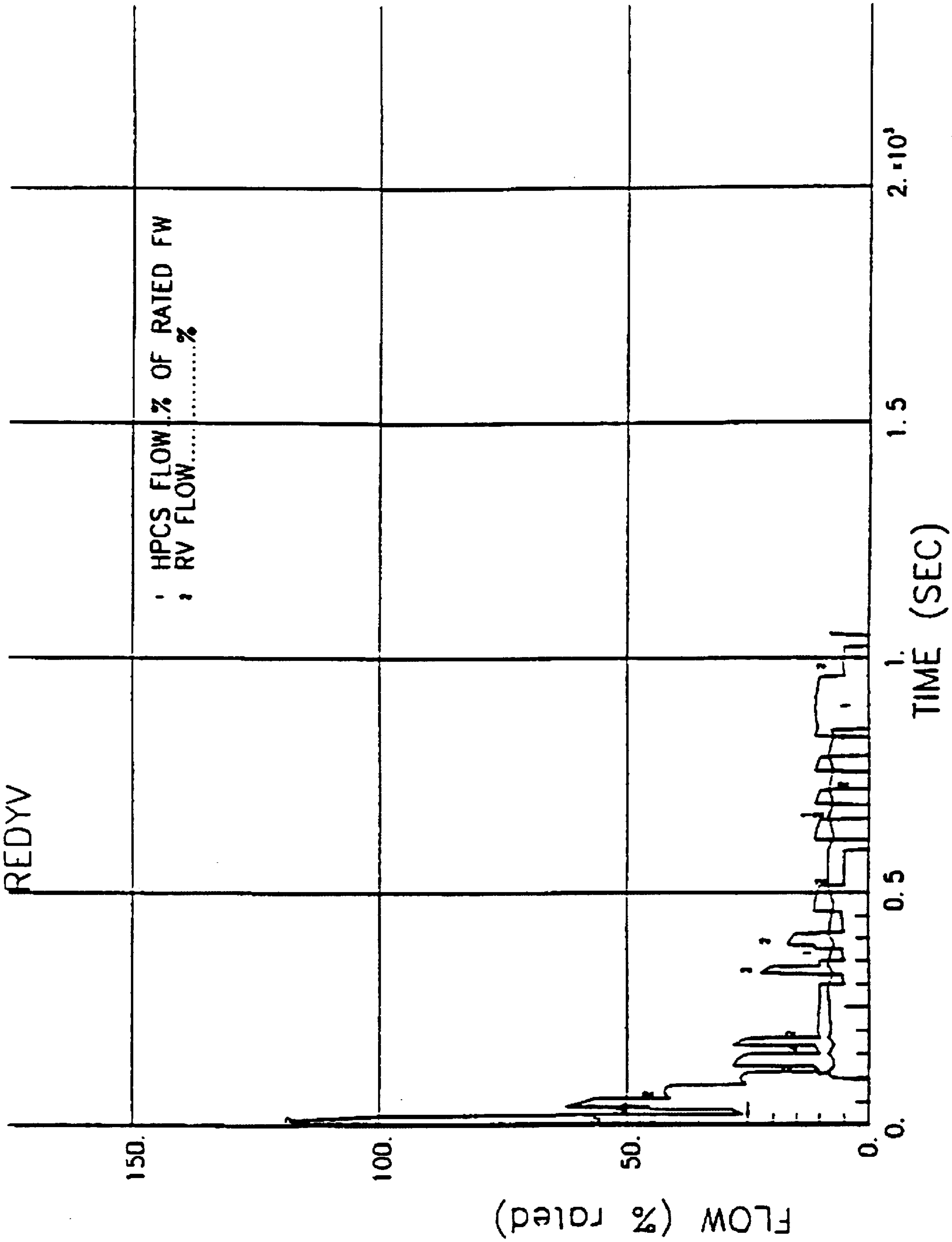
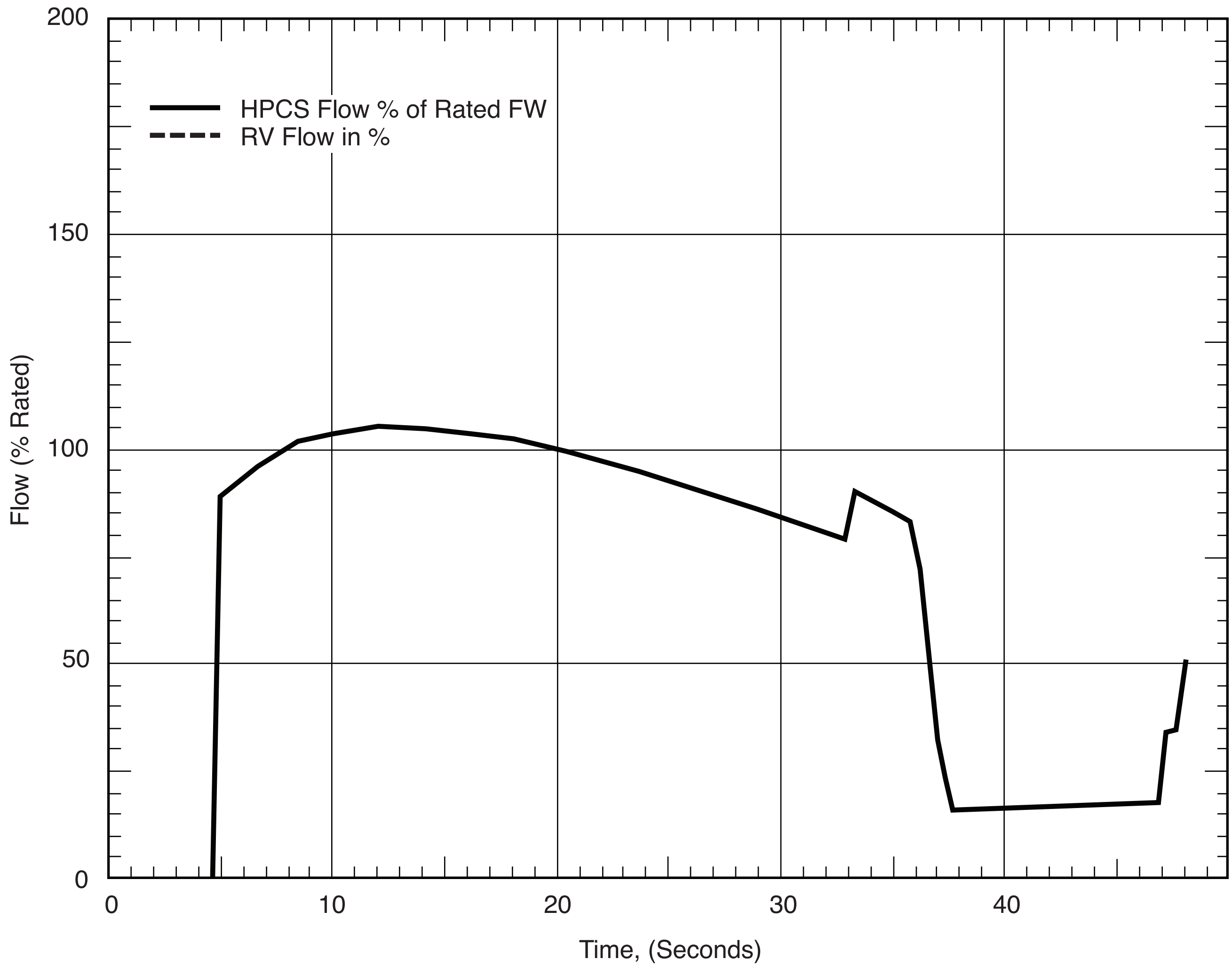
Main Steam Isolation Valve Closure Event

Draw. No. 020002.37	Rev.	Figure 15.8-2.4



Main Steam Isolation Valve Closure Event

Draw. No. 020002.38	Rev.	Figure 15.8-2.5



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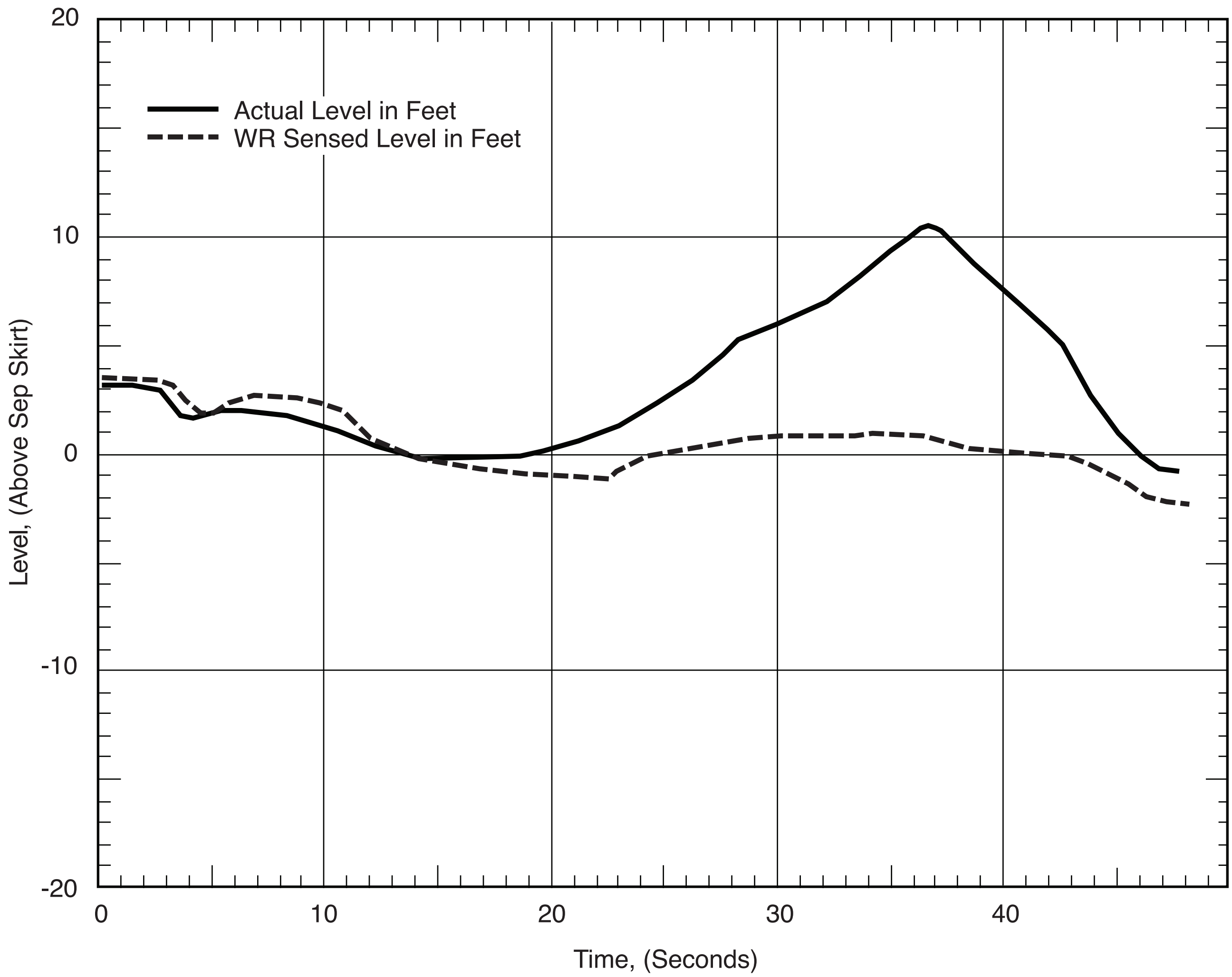
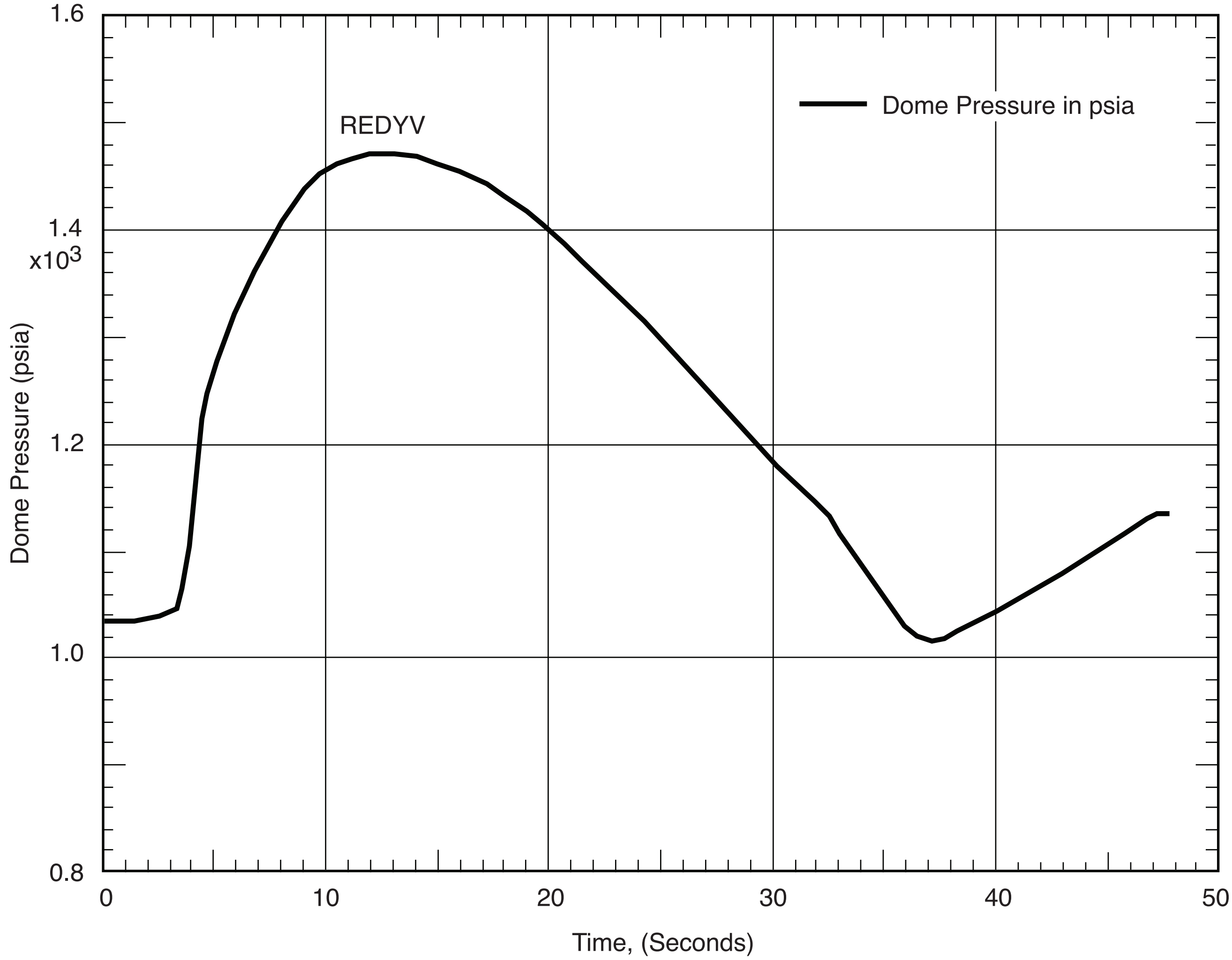
Main Steam Isolation Valve Closure Event with 4 SRVs Out-of-Service

Draw. No. 960222.40 | Rev. | Figure 15.8-3.1



Main Steam Isolation Valve Closure Event with 4 SRVs Out-of-Service

Draw. No. 960222.41 | Rev. | Figure 15.8-3.2



Main Steam Isolation Valve Closure Event with 4 SRVs Out-of-Service

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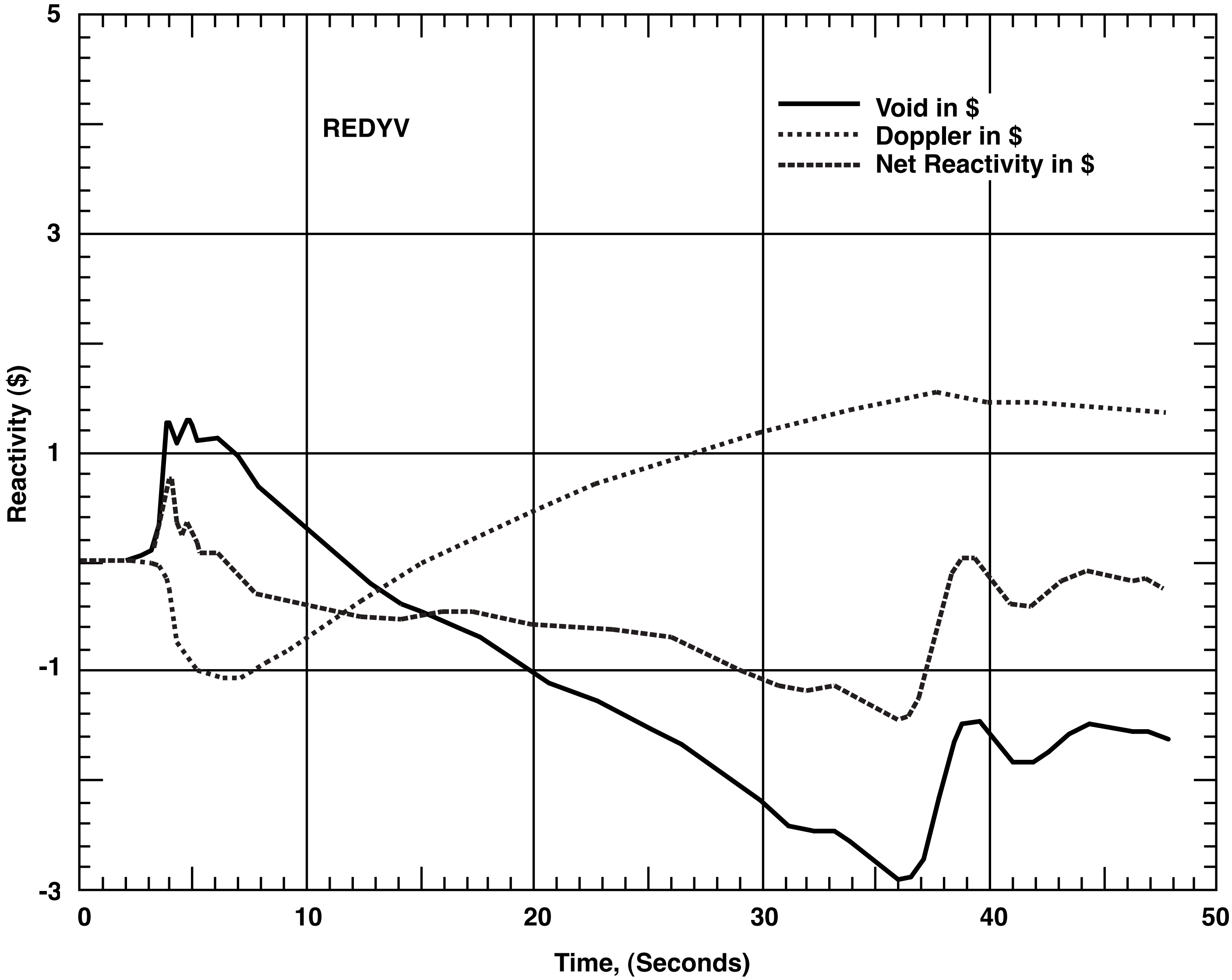

Main Steam Isolation Valve Closure Event with 4 SRVs Out-of-Service

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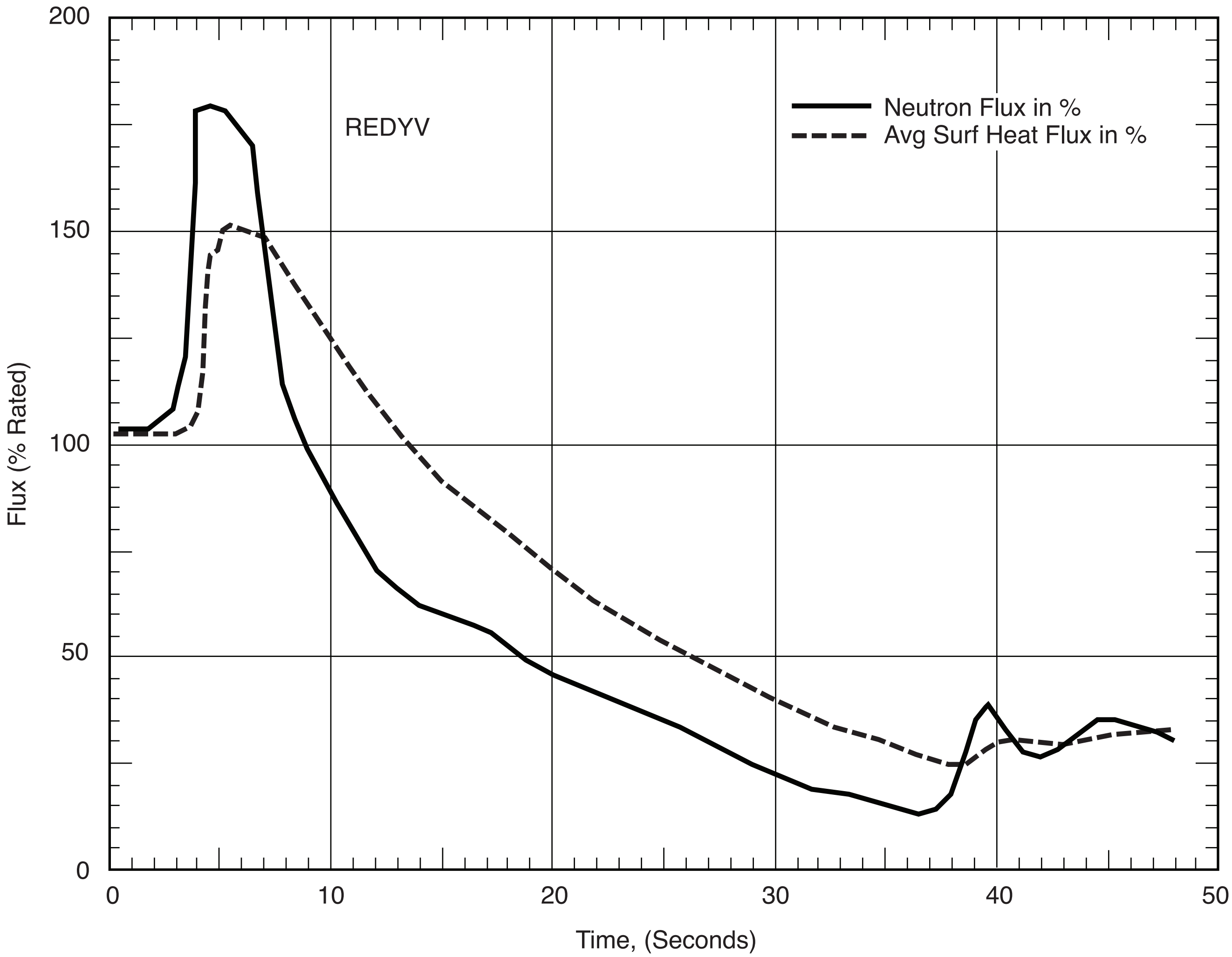

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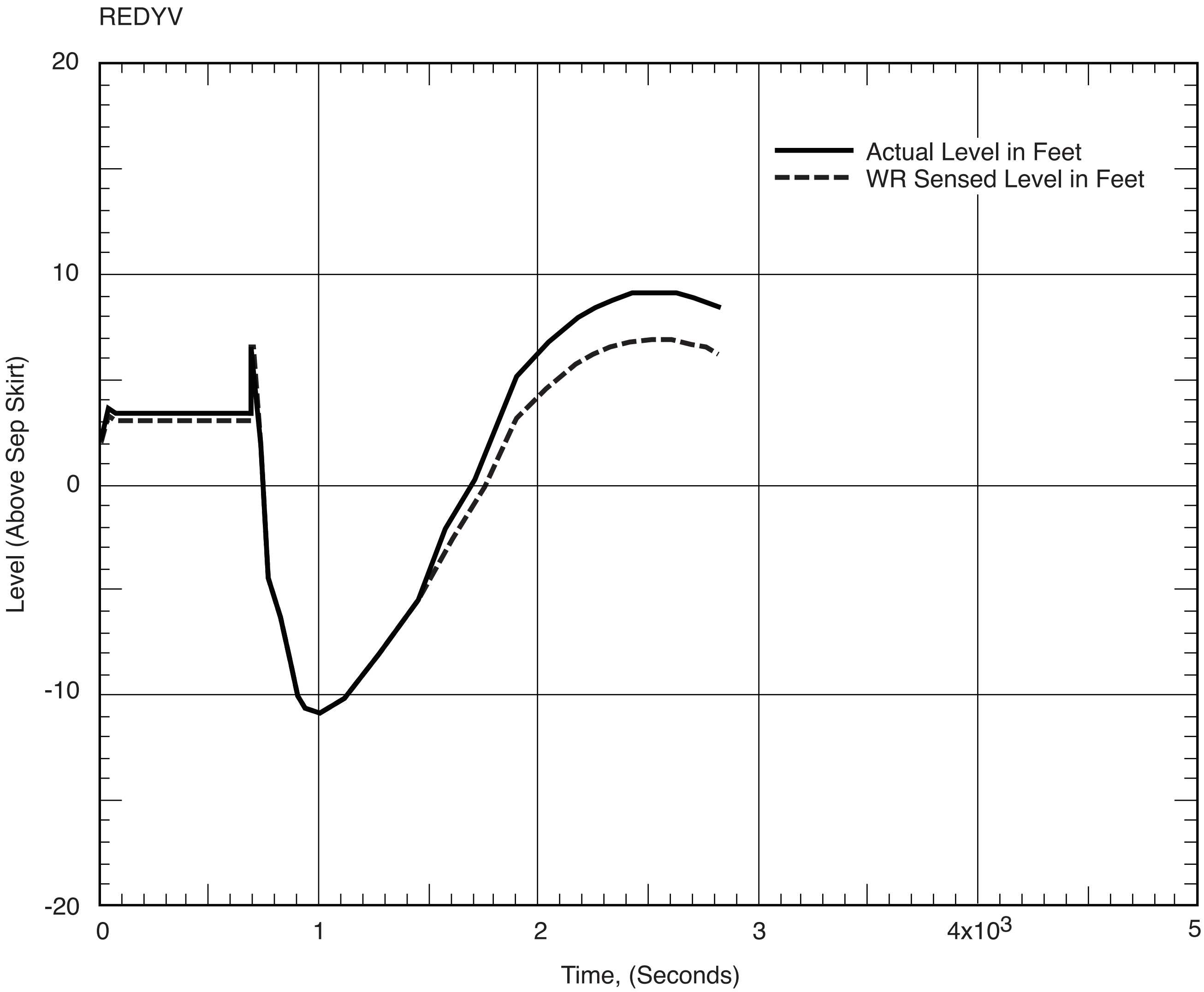
Main Steam Isolation Valve Closure Event with 4 SRVs Out-of-Service

Draw. No. 960222.44 | Rev. | Figure 15.8-3.5



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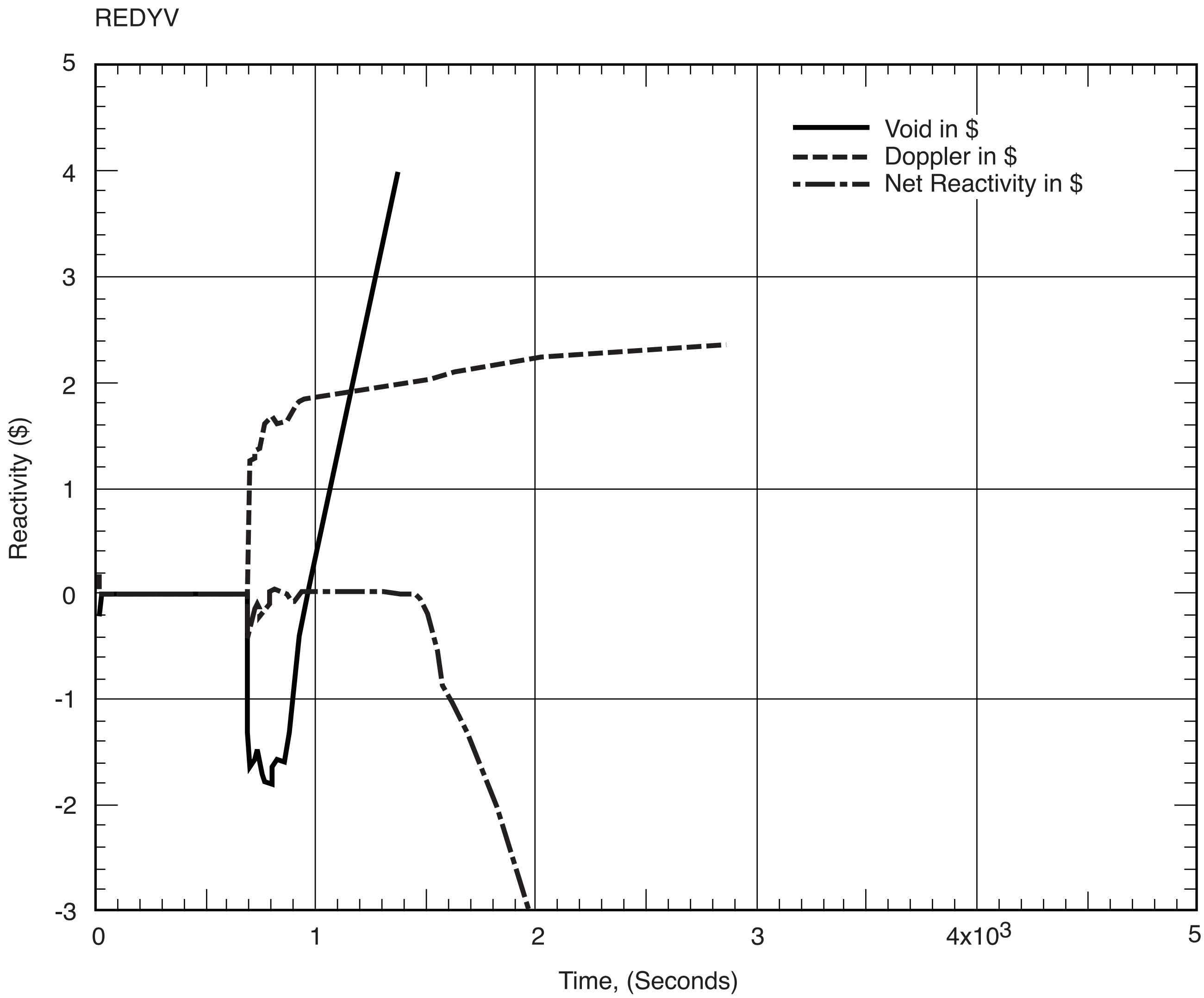
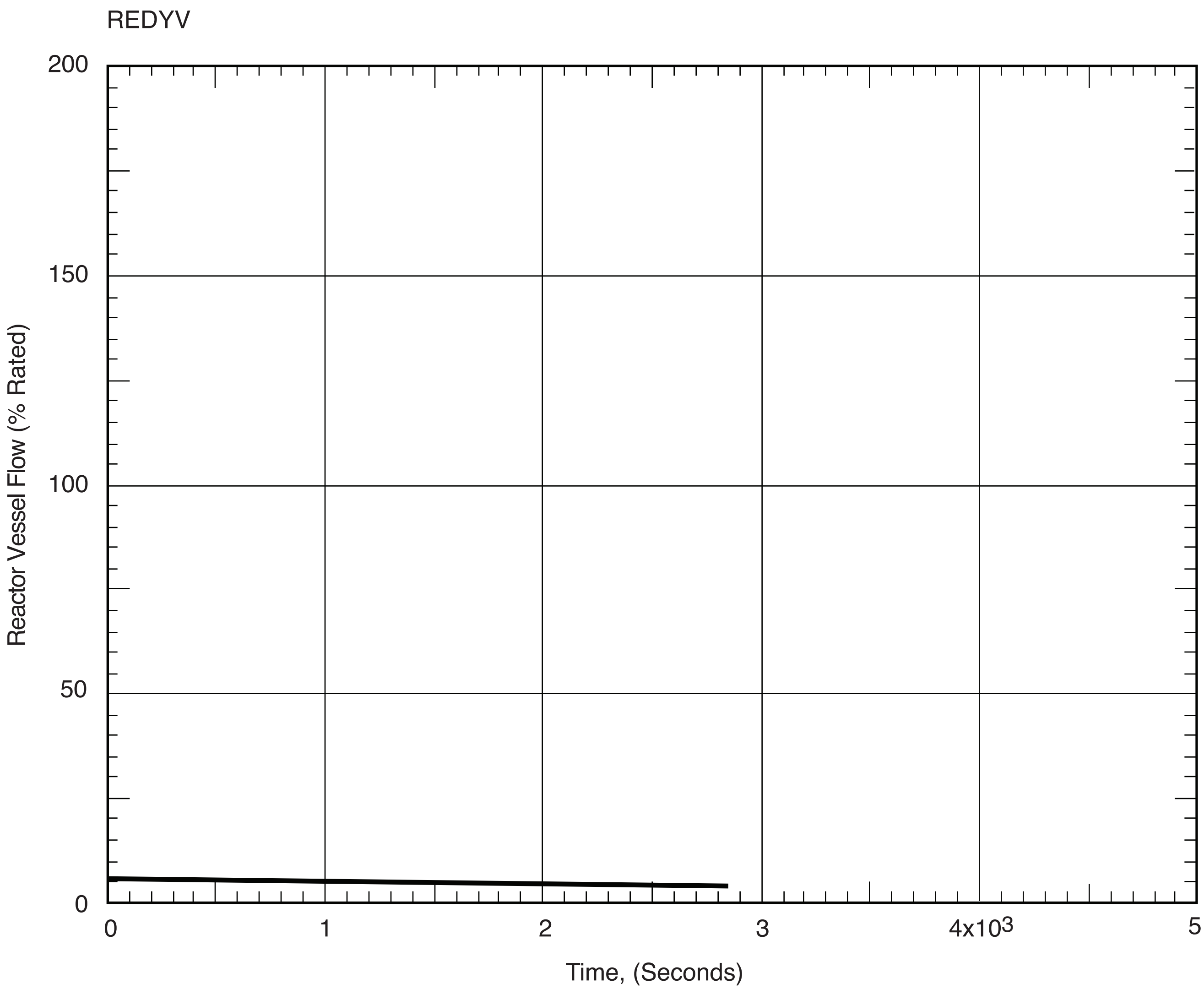
Inadvertent Opening of Relief Valve Event

Draw. No. 960222.45 | Rev. | Figure 15.8-4.1



Inadvertent Opening of Relief Valve Event

Draw. No. 960222.48 | Rev. | Figure 15.8-4.2



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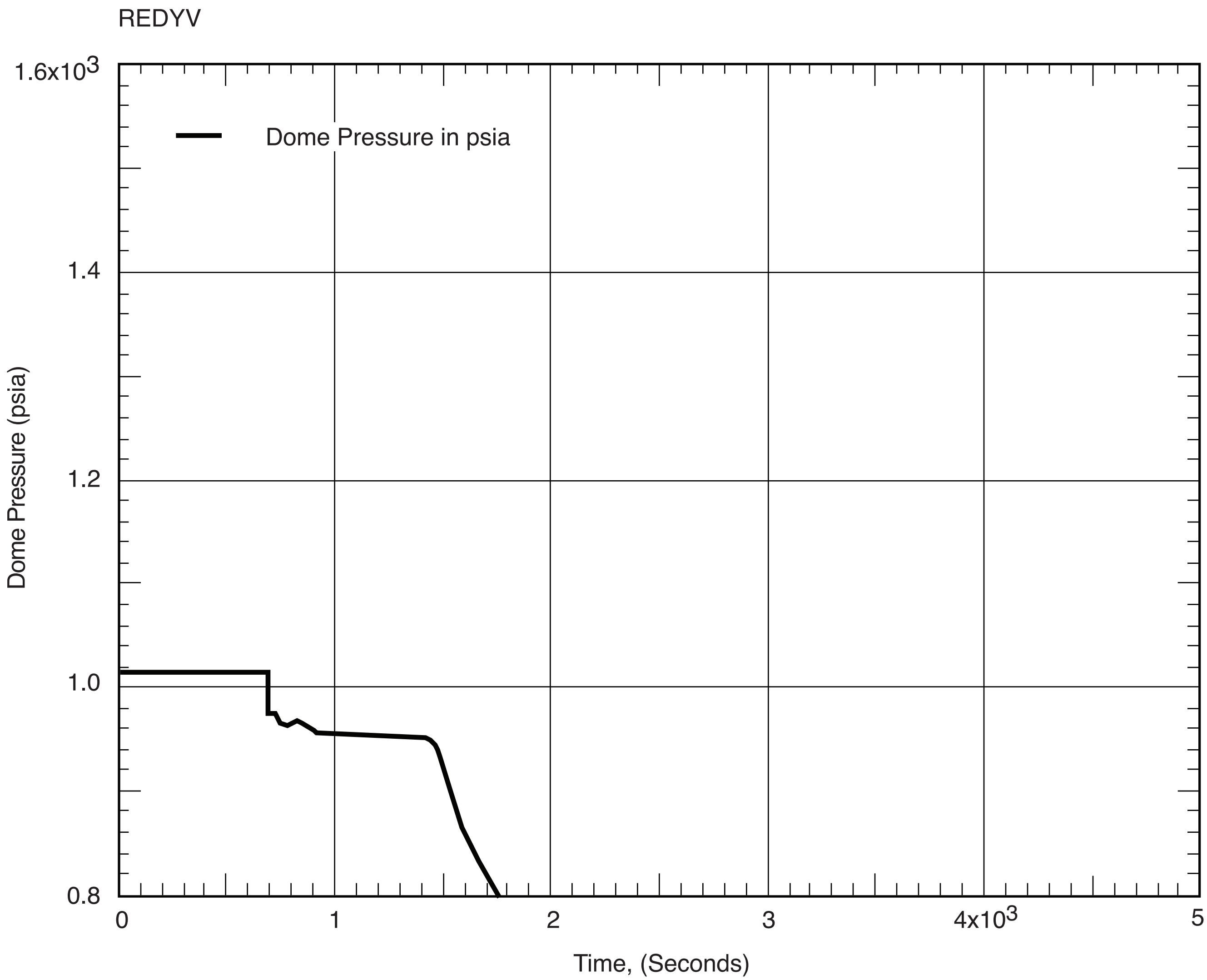
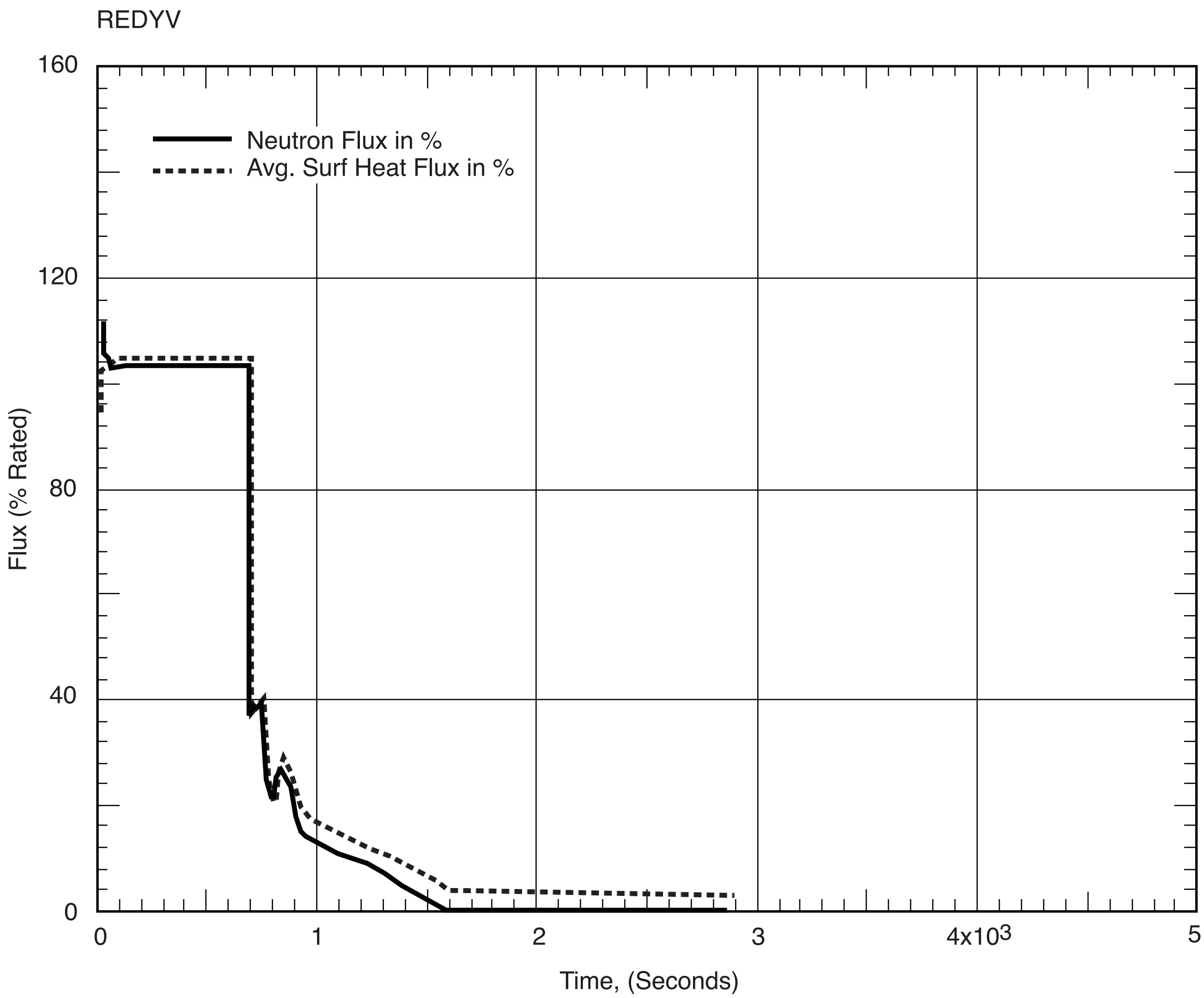
Inadvertent Opening of Relief Valve Event

Draw. No. **960222.46** Rev. Figure **15.8-4.3**



Inadvertent Opening of Relief Valve Event

Draw. No. 960222.47 | Rev. | Figure 15.8-4.4



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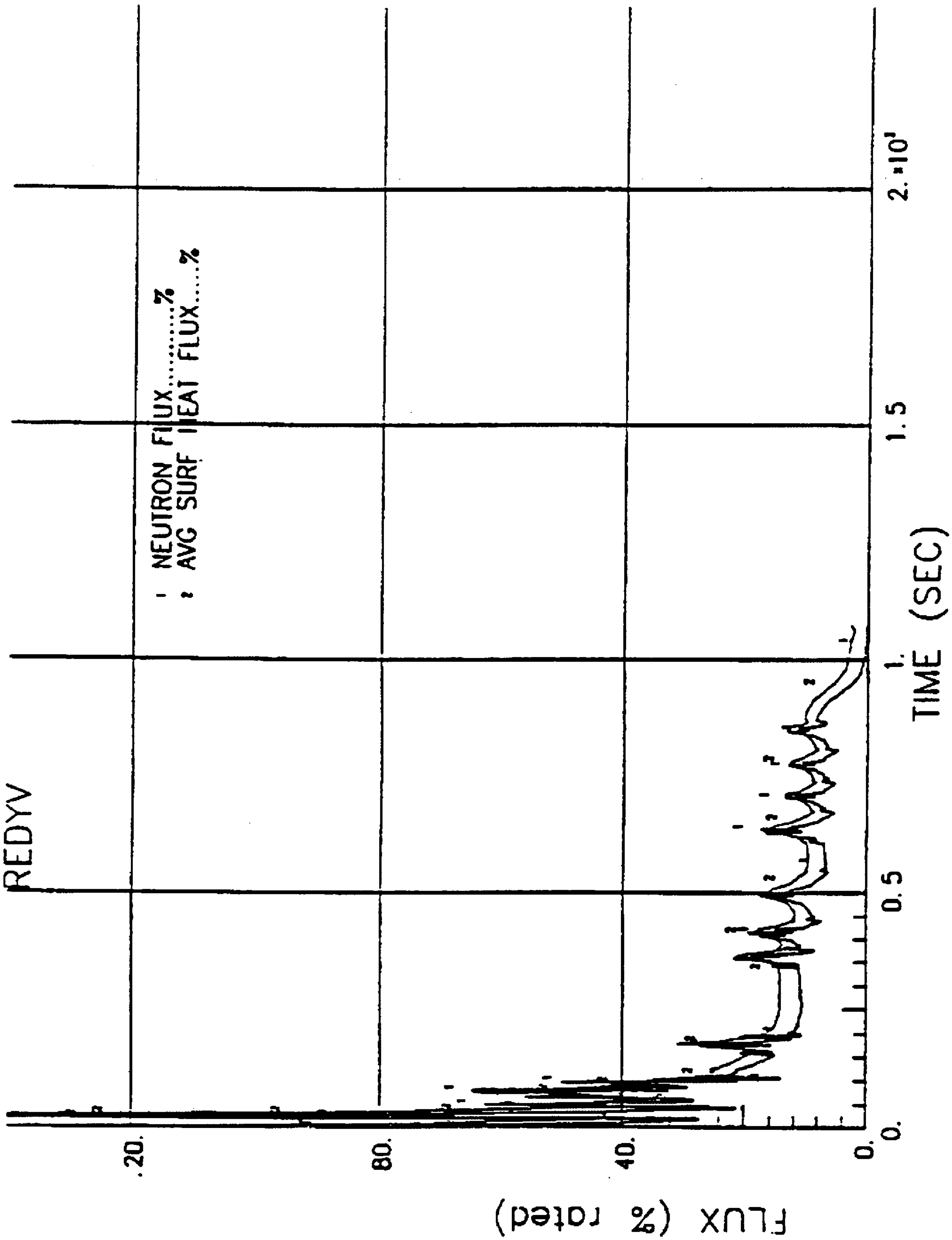
Final Safety Analysis Report

Inadvertent Opening of Relief Valve Event

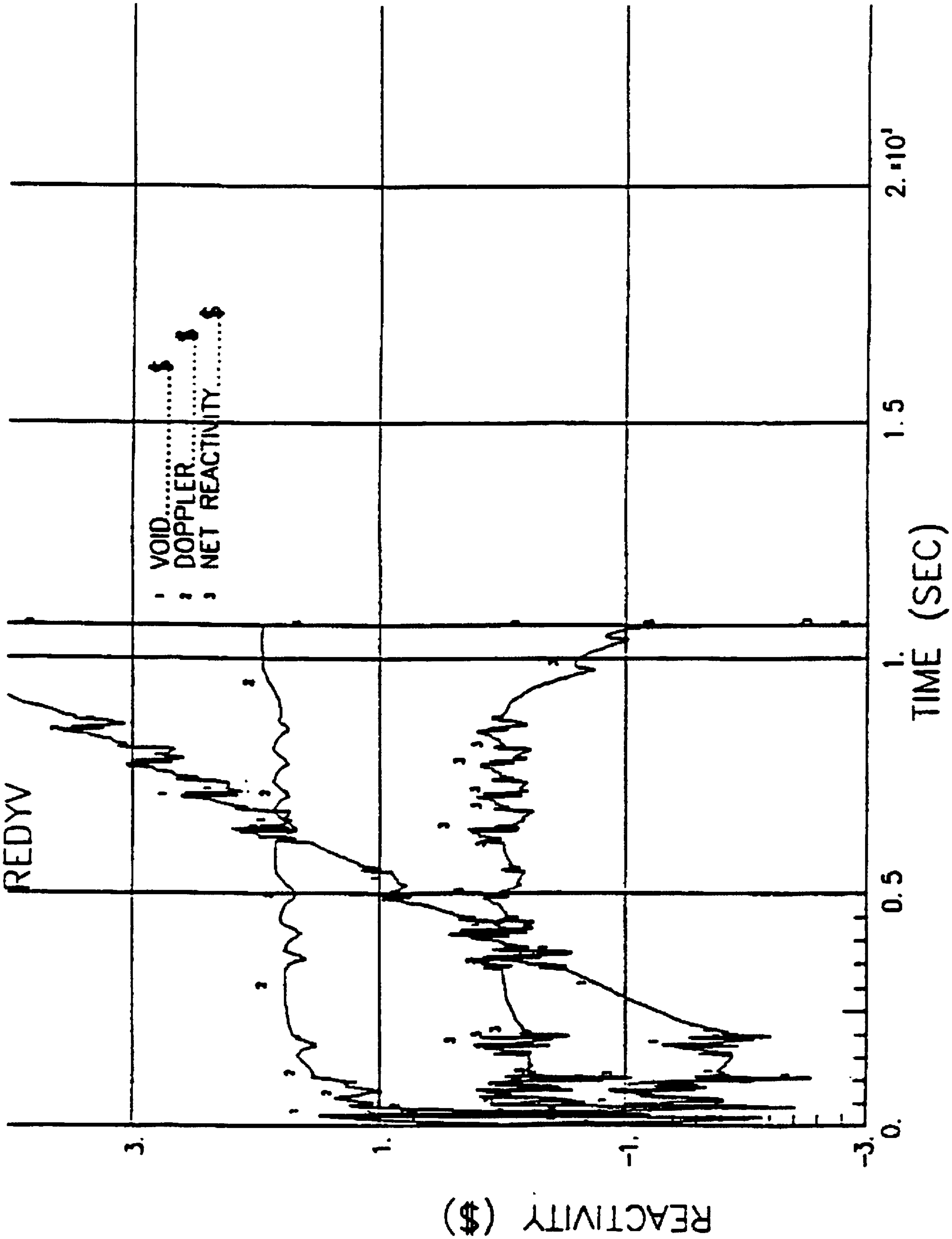
Draw. No. 960222.49	Rev.	Figure 15.8-4.5



Columbia Generating Station
Final Safety Analysis Report

Pressure Regulator Failure - Open Event

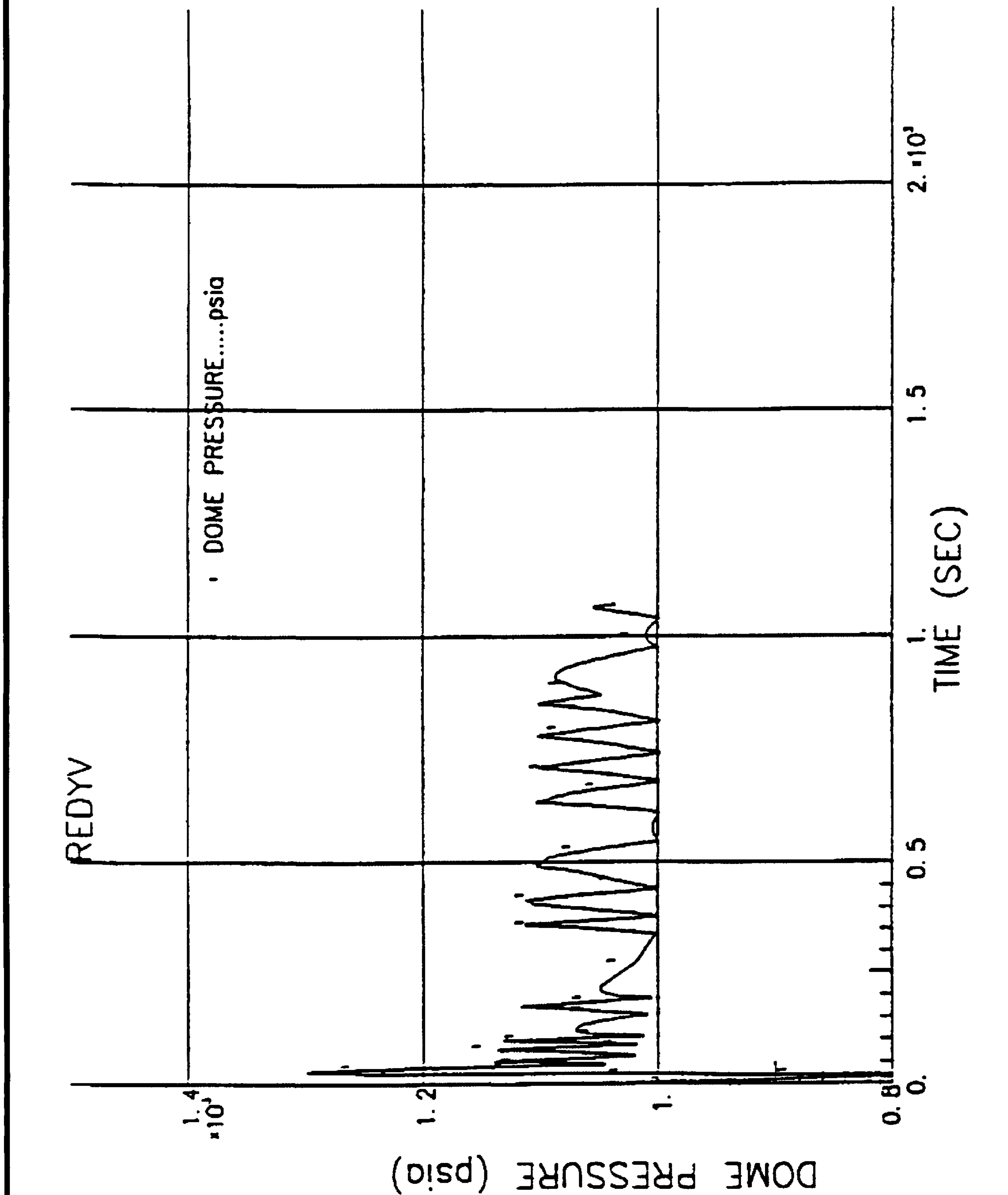
Draw. No. 020002.39	Rev.	Figure 15.8-5.1



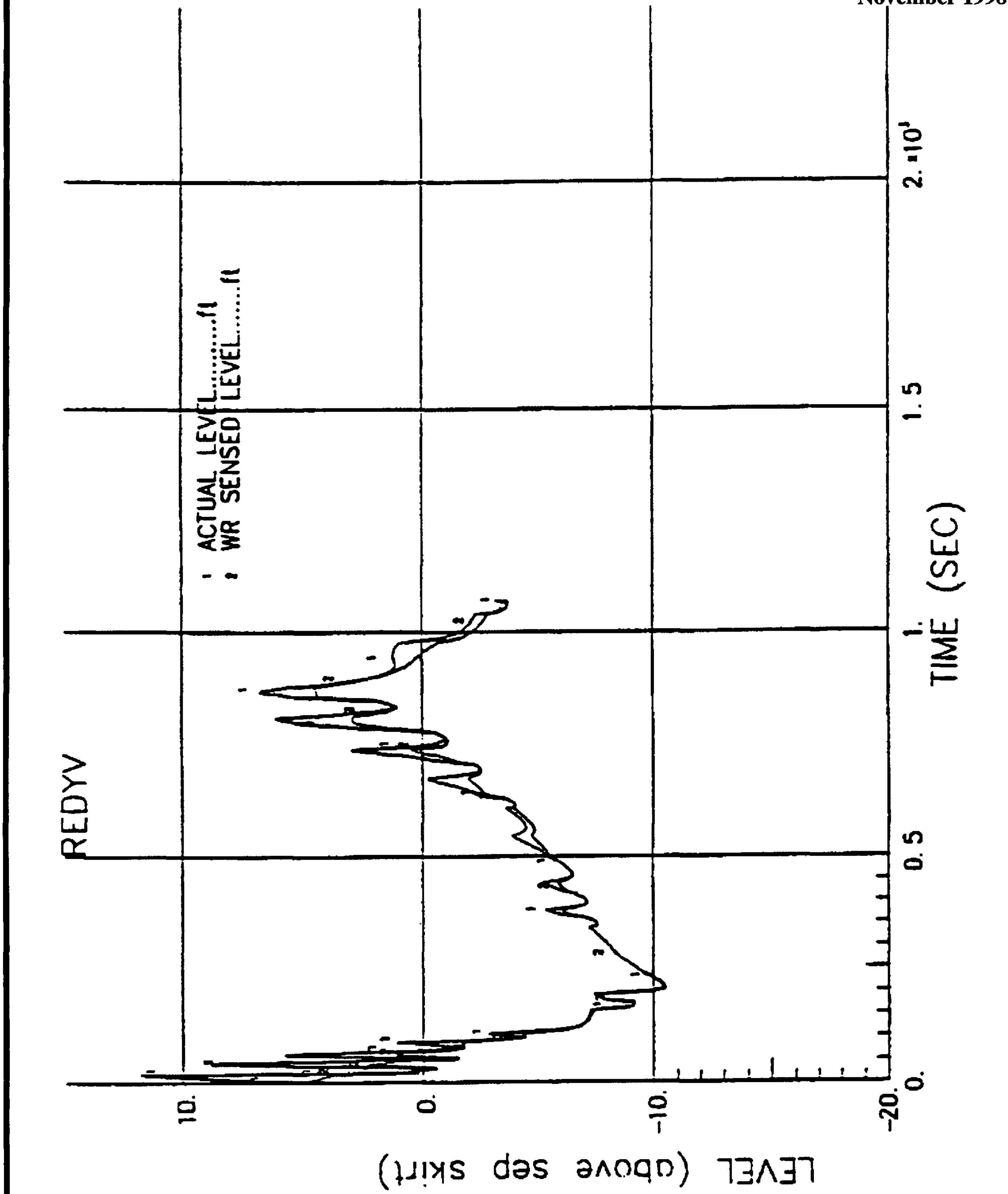
Columbia Generating Station
Final Safety Analysis Report

Pressure Regulator Failure - Open Event

Draw. No. 020002.40	Rev.	Figure 15.8-5.2



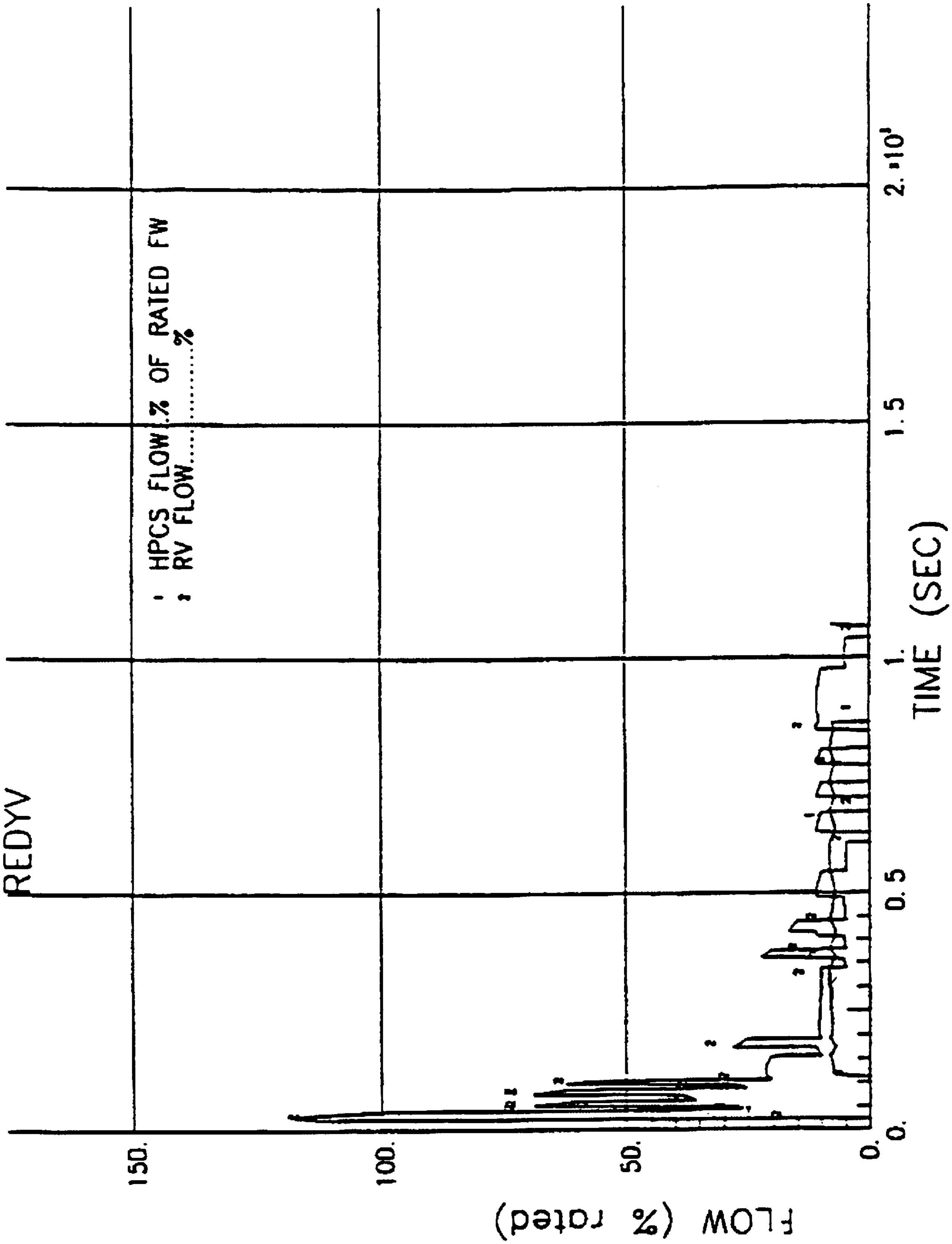
Columbia Generating Station Final Safety Analysis Report	Pressure Regulator Failure - Open Event		
	Draw. No. 020002.41	Rev.	Figure 15.8-5.3



Columbia Generating Station
Final Safety Analysis Report

Pressure Regulator Failure - Open Event

Draw. No. 020002.42	Rev.	Figure 15.8-5.4



Columbia Generating Station
Final Safety Analysis Report

Pressure Regulator Failure - Open Event

Draw. No. 020002.43	Rev.	Figure 15.8-5.5